Facility: 2016 TP Exam Date of Exam: 8/22/16

Tier	Group	RO K/A Category Points K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	SRO-Only Points A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18	3	3	6
	2	2	1	2				1	2			1	9	2	2	4
	Tier Totals	5	4	5				4	5			4	27	5	5	10
2. Plant Systems	1	3	2	2	2	3	3	3	3	1	3	3	28	3	2	5
	2	1	0	1	1	1	1	1	1	1	1	1	10	2	1	3
	Tier Totals	4	2	3	3	4	4	4	4	2	4	4	38		3	8

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

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 Radiation Control K/A is allowed if the K/A is replaced by a K/A from another Tier 3 Category).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. *The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to section D.1.b of ES-401 for the applicable KAs.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics= importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note # 1 does not apply). Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43..

KA	NAME / SAFETY FUNCTION:	IR		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
		RO	SRO												
007EG2.4.34	Reactor Trip - Stabilization - Recovery / 1	4.2	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects
008AA1.05	Pressurizer Vapor Space Accident / 3	3.4	3.3	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	LPI System
009EK1.02	Small Break LOCA / 3	3.5	4.2	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Use of steam tables
011EK3.03	Large Break LOCA / 3	4.1	4.3	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Starting auxiliary feed pumps and flow, ED/G, and service water pumps
015AK2.08	RCP Malfunctions / 4	2.6	2.6	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	CCWS
022AK1.03	Loss of Rx Coolant Makeup / 2	3	3.4	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Relationship between charging flow and PZR level
025AA1.09	Loss of RHR System / 4	3.2	3.1	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	LPI pump switches, ammeter, discharge pressure gauge, flow meter, and indicators
027AA1.03	Pressurizer Pressure Control System Malfunction / 3	3.6	3.5	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Pressure control when on a steam bubble
029EK2.06	ATWS / 1	2.9	3.1	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Breakers, relays, and disconnects.
038EA2.13	Steam Gen. Tube Rupture / 3	3.1	3.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Magnitude of rupture
054AA2.08	Loss of Main Feedwater / 4	2.9	3.3	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Steam flow-feed trend recorder

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
055EG2.1.19	Station Blackout / 6	3.9	3.8											✔	Ability to use plant computer to evaluate system or component status.
056AK1.04	Loss of Off-site Power / 6	3.1	3.2	✔											Definition of saturation conditions implication for the systems
057AK3.01	Loss of Vital AC Inst. Bus / 6	4.1	4.4			✔									Actions contained in EOP for loss of vital ac electrical instrument bus
062AK3.02	Loss of Nuclear Svc Water / 4	3.6	3.9			✔									The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS
065AA2.06	Loss of Instrument Air / 8	3.6	4.2								✔				When to trip reactor if instrument air pressure is decreasing
WE05EK2.2	Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	3.9	4.2		✔										Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.
we12EG2.2.4	Steam Line Rupture - Excessive Heat Transfer / 4	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:
003AK1.02	Dropped Control Rod / 1	3.1	3.4	✔											Effects of turbine-reactor power mismatch on rod control
005AA2.01	Inoperable/Stuck Control Rod / 1	3.3	4.1								✔				Stuck or inoperable rod from in-core and ex-core NIS, in-core or loop temperature measurements
024AA2.06	Emergency Boration / 1	3.6	3.7								✔				When boron dilution is taking place
036AK3.01	Fuel Handling Accident / 8	3.1	3.7			✔									Different inputs that will cause a reactor building evacuation
037AG2.4.20	Steam Generator Tube Leak / 3	3.8	4.3											✔	Knowledge of operational implications of EOP warnings, cautions and notes.
067AA1.06	Plant Fire On-site / 8	3.5	3.7							✔					Fire alarm
WE03EK2.2	LOCA Cooldown - Depress. / 4	3.7	4.0		✔										Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.
WE06EK1.3	Degraded Core Cooling / 4	3.7	3.9	✔											Annunciators and conditions indicating signals, and remedial actions associated with the (Degraded Core Cooling).
WE15EK3.2	Containment Flooding / 5	2.8	3.1			✔									Normal, abnormal and emergency operating procedures associated with (Containment Flooding).

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
003A2.02	Reactor Coolant Pump	3.7	3.9								✔				Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP
003K3.04	Reactor Coolant Pump	3.9	4.2			✔									RPS
004K6.09	Chemical and Volume Control	2.8	3.1						✔						Purpose of VCT divert valve
004K6.22	Chemical and Volume Control	2.6	2.9						✔						Design minimum and maximum flow rates for letdown system.
005K2.03	Residual Heat Removal	2.7	2.8		✔										RCS pressure boundary motor-operated valves
006A4.08	Emergency Core Cooling	4.2	4.3										✔		ESF system, including reset
007A4.10	Pressurizer Relief/Quench Tank	3.6	3.8										✔		Recognition of leaking PORV/code safety
008K4.01	Component Cooling Water	3.1	3.3				✔								Automatic start of standby pump
010K2.02	Pressurizer Pressure Control	2.5	2.7		✔										Controller for PZR spray valve
012K1.04	Reactor Protection	3.2	3.3	✔											RPIS
013K5.01	Engineered Safety Features Actuation	2.8	3.2					✔							Definitions of safety train and ESF channel

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
022G2.2.12	Containment Cooling	3.7	4.1											✔	Knowledge of surveillance procedures.
022K1.01	Containment Cooling	3.5	3.7	✔											SWS/cooling system
026G2.4.4	Containment Spray	4.5	4.7											✔	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.
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
059A1.03	Main Feedwater	2.7	2.9							✔					Power level restrictions for operation of MFW pumps and valves.
059K4.19	Main Feedwater	3.2	3.4				✔								Automatic feedwater isolation of MFW
061K1.10	Auxiliary/Emergency Feedwater	2.6	2.7	✔											Diesel fuel oil
061K5.01	Auxiliary/Emergency Feedwater	3.6	3.9					✔							Relationship between AFW flow and RCS heat transfer
062A1.01	AC Electrical Distribution	3.4	3.8							✔					Significance of D/G load limits
063A1.01	DC Electrical Distribution	2.5	3.3							✔					Battery capacity as it is affected by discharge rate

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
063A4.01	DC Electrical Distribution	2.8	3.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	Major breakers and control power fuses
064K6.07	Emergency Diesel Generator	2.7	2.9	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	Air receivers
073K3.01	Process Radiation Monitoring	3.6	4.2	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Radioactive effluent releases
076A2.02	Service Water	2.7	3.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Service water header pressure
078G2.2.44	Instrument Air	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
103A2.04	Containment	3.5	3.6	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Containment evacuation (including recognition of the alarm)

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
002K5.14	Reactor Coolant	3.8	4.2					✔							Consequences of forced circulation loss
011K4.05	Pressurizer Level Control	3.7	4.1				✔								PZR level inputs to RPS
014A1.02	Rod Position Indication	3.2	3.6							✔					Control rod position indication on control room panels
017K6.01	In-core Temperature Monitor	2.7	3.0						✔						Sensors and detectors
029A4.01	Containment Purge	2.5	2.5										✔		Containment purge flow rate
033A2.01	Spent Fuel Pool Cooling	3.0	3.5								✔				Inadequate SDM
035G2.4.6	Steam Generator	3.7	4.7											✔	Knowledge symptom based EOP mitigation strategies.
055K3.01	Condenser Air Removal	2.5	2.7			✔									Main condenser
068A3.02	Liquid Radwaste	3.6	3.6									✔			Automatic isolation
079K1.01	Station Air	3.0	3.1	✔											IAS

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
G2.1.36	Conduct of operations	3.0	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of procedures and limitations involved in core alterations
G2.1.45	Conduct of operations	4.3	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to identify and interpret diverse indications to validate the response of another indication
G2.1.8	Conduct of operations	3.4	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to coordinate personnel activities outside the control room.
G2.2.12	Equipment Control	3.7	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of surveillance procedures.
G2.2.2	Equipment Control	4.6	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.
G2.2.41	Equipment Control	3.5	3.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to obtain and interpret station electrical and mechanical drawings
G2.3.15	Radiation Control	2.9	3.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiation monitoring systems
G2.4.3	Emergency Procedures/Plans	3.7	3.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to identify post-accident instrumentation.
G2.4.37	Emergency Procedures/Plans	3.0	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the lines of authority during implamentation of an emergency plan.
G2.4.5	Emergency Procedures/Plans	3.7	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the organization of the operating procedures network for normal, abnormal and emergency evolutions.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
007EA2.02	Reactor Trip - Stabilization - Recovery / 1	4.3	4.6	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Proper actions to be taken if the automatic safety func- tions have not taken place
015AG2.4.31	RCP Malfunctions / 4	4.2	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of annunciators alarms, indications or response procedures
026AA2.03	Loss of Component Cooling Water / 8	2.6	2.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	The valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition
062AG2.4.2	Loss of Nuclear Svc Water / 4	4.5	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.
065AG2.4.30	Loss of Instrument Air / 8	2.7	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of events related to system operations/status that must be reported to internal orginizations or outside agencies.
WE04EA2.1	LOCA Outside Containment / 3	3.4	4.3	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	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:
060AG2.2.40	Accidental Gaseous Radwaste Rel. / 9	3.4	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to apply technical specifications for a system.
061AA2.06	ARM System Alarms / 7	3.2	4.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Required actions if alarm channel is out of service
069AA2.01	Loss of CTMT Integrity / 5	3.7	4.3	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Loss of containment integrity
we03EG2.4.20	LOCA Cooldown - Depress. / 4	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:
004G2.1.30	Chemical and Volume Control	4.4	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to locate and operate components, including local controls.
013A2.04	Engineered Safety Features Actuation	3.6	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Loss of instrument bus
061A2.04	Auxiliary/Emergency Feedwater	3.4	3.8	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	pump failure or improper operation
064G2.4.9	Emergency Diesel Generator	3.8	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies.
073A2.02	Process Radiation Monitoring	2.7	3.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Detector failure

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
001A2.17	Control Rod Drive	3.3	3.8	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Rod-misalignment alarm
045A2.08	Main Turbine Generator	2.8	3.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Steam dumps are not cycling properly at low load or stick open at higher load (isolate and use atmospheric reliefs when necessary)
017G2.4.30	In-core temperature Monitors	2.7	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
G2.1.4	Conduct of operations	3.3	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license statur, 10CFR55 etc.
G2.2.19	Equipment Control	2.3	3.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of maintenance work order requirements.
G2.2.43	Equipment Control	3.0	3.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process used to track inoperable alarms
G2.1.36	Conduct of operations	3.0	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of procedures and limitations involved in core alterations.
G2.3.14	Radiation Control	3.4	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities
G2.4.26	Emergency Procedures/Plans	3.1	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage.
G2.4.8	Emergency Procedures/Plans	3.8	4.5	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Facility: **Turkey Point Units 3 & 4** Date of Examination: **08/22/2016**

Examination Level: **RO** ☒ **SRO** ☐ Operating Test Number: **2016-301**

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, M	Calculate a Manual Makeup to the VCT 2.1.25 (3.9): Ability to interpret reference materials, such as graphs, curves, tables, etc.
Conduct of Operations	R, D	Determine Heatup of the RCS 2.1.20 (4.6): Ability to interpret and execute procedure steps.
Equipment Control	R, D	Review an ECO for the B AFW Pump 2.2.13 (4.1): Knowledge of tagging and clearance procedures.
Radiation Control	R, D, P	Evaluate Conditions for Restart of Refueling Pre-shuffle in the Spent Fuel Pit 2.3.12 (3.2): 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.
Emergency Procedures/Plan		NOT SELECTED FOR RO EXAM

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

* 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: **Turkey Point Units 3 & 4** Date of Examination: **08/22/2016**

Examination Level: **RO** ☐ **SRO** ☒ Operating Test Number: **2016-301**

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, M	Calculate a Manual Makeup to the VCT 2.1.25 (4.2): Ability to interpret reference materials, such as graphs, curves, tables, etc.
Conduct of Operations	R, D	Determine Heatup of the RCS 2.1.20 (4.6): Ability to interpret and execute procedure steps.
Equipment Control	R, D	Evaluate TS Conditions While Performing a Valve Operability Test 2.2.40 (4.7): Ability to apply Technical Specifications for a system.
Radiation Control	R, D	Authorize Emergency Exposure Limits 2.3.4 (3.7): Knowledge of radiation exposure limits under normal or emergency conditions.
Emergency Procedures/Plan	R, D	Classify Event and Complete SNF 2.4.41 (4.6): Knowledge of the emergency action level thresholds and classifications.

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

* 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)

JPM SUMMARY STATEMENTS

A.1.a Calculate a Manual Makeup to the VCT – Unit is at 100% power, with a VCT level of 20%. Examinee is given a desired VCT level and boric acid flow rate and is directed to calculate the primary water flow rate, boric acid and primary water volumes, and controller potentiometer settings for the manual makeup. This is a modified bank JPM.

A.1.b Determine Heatup of the RCS – Unit has undergone a heatup to 380°F and relevant data is provided on Attachment 2, Heatup Data Sheet, of 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification. Examinee must complete the procedure and record any discrepancies, subsequent procedural actions, and/or Technical Specification actions that apply. This is a bank JPM.

A.2 Evaluate TS Conditions While Performing a Valve Operability Test – Unit is in Mode 3 and 4-OSP-047.1E, Letdown Line Isolation Valve Test, is in progress. Given Control Room indications, examinee must determine if any actions are required with regard to ESFAS instrumentation, accident monitoring instrumentation, or containment isolation valves. This is a bank JPM.

A.3 Authorize Emergency Exposure Limits – A General Emergency has been declared and an Owner Controlled Area evacuation is in progress. A rescue of an unconscious person in a high-dose area is to be performed. Examinee must select two individuals from a list of available rescuers and determine whether they should be issued potassium iodide. This is a bank JPM.

A.4 Classify Event and Complete SNF – Unit is in Mode 5, when it experiences a loss of all AC power. Plant and meteorological conditions are provided and examinee must classify the event using 0-EPIP-20101, Duties of Emergency Coordinator, and issue protective action recommendations using 0-EPIP-20134, Offsite Notifications and Protective Action Recommendations. This is a bank JPM.

JPM SUMMARY STATEMENTS

A.1.a Calculate a Manual Makeup to the VCT – Unit is at 100% power, with a VCT level of 20%. Examinee is given a desired VCT level and boric acid flow rate and is directed to calculate the primary water flow rate, boric acid and primary water volumes, and controller potentiometer settings for the manual makeup. This is a modified bank JPM.

A.1.b Determine Heatup of the RCS – Unit has undergone a heatup to 380°F and relevant data is provided on Attachment 2, Heatup Data Sheet, of 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification. Examinee must complete the procedure and record any discrepancies and subsequent procedural actions that apply. This is a bank JPM.

A.2 Review an ECO for the B AFW Pump – Maintenance requests that the B AFW Pump's turbine be disabled from starting. Examinee is directed to review the prepared ECO for completeness and accuracy (with eSOMS NOT available) and identify any items that do not meet the requirements of OP-AA-101-1000, Clearance and Tagging. This is a bank JPM.

A.3 Evaluate Conditions for Restart of Refueling Pre-shuffle in the Spent Fuel Pit – Refueling pre-shuffle activities in the SFP were interrupted and management desires to resume the shuffle. Examinee is provided a list of plant conditions and inoperable equipment and must determine whether recommencement may occur in accordance with 3-NOP-040.03, Fuel Handling and Insert Shuffle in the Spent Fuel Pit. Examinee will use Attachment 2, Restart Minimum Equipment Checklist, and must identify four items that preclude recommencement. This is a bank JPM, previously used on the 2013 NRC exam.

A.4 NOT SELECTED FOR RO EXAM

Facility: **Turkey Point Units 3 & 4** Date of Examination: **08/22/2016**
Exam Level: **RO** ☒ **SRO-I** ☐ **SRO-U** ☐ Operating Test Number: **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. 001 Control Rod Drive System (A4.14, 3.0) / Respond to Control Bank D Demanded Past 230 Steps	A, D, P, S	1
b. 004 Chemical and Volume Control System (A4.06, 3.6) / Place Excess Letdown In Service	A, D, S	2
c. EPE 038 Steam Generator Tube Rupture (EA1.04, 4.3) / Establish Auxiliary Pressurizer Spray per 3-EOP-E-3	A, N, S	3
d. APE 025 Residual Heat Removal System (AA1.03, 3.4) / Respond to a Loss of RHR	L, D, P, S	4P
e. 026 Containment Spray System (A3.01, 4.3) / Manually Initiate Containment Spray	D, EN, S	5
f. EPE 055 Station Blackout (EA1.07, 4.3) / Restore Power to the 3A 4kV Bus	A, N, S	6
g. 015 Nuclear Instrumentation System (A4.02, 3.9) / Place N-3-42 Power Range Drawer in Service	D, S	7
h. APE 068 Control Room Evacuation (AA1.23, 4.3) / Respond to Control Room Evacuation Condition – Unit 3 RO	D, S	8
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. EPE 029 Anticipated Transient Without Scram (EA1.12, 4.1) / Locally Trip the Reactor and Turbine	D, E	1
j. APE 054 Loss of Main Feedwater (AA1.01, 4.5) / Control Steam Generator Level Locally with Auxiliary Feedwater Control Valve	A, D, E	4S
k. APE 026 Loss of Component Cooling Water (AA1.03, 3.6) / Align Emergency Service Water to the Charging Pumps	D, E, R	8

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

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

JPM SUMMARY STATEMENTS

a. Respond to Control Bank D Demanded Past 230 Steps – Examinee enters 3-ONOP-028, Reactor Control System Malfunction, to restore the Rod Control System to normal configuration with Bank D at 229 steps withdrawn. When examinee places the Rod Control Selector Switch in automatic, rods begin inserting at fast speed, and examinee must respond by placing the Rod Control Selector Switch in manual. This is a bank alternate-path JPM, previously used on the 2013 NRC exam.

b. Place Excess Letdown In Service – Examinee uses 3-OP-047, CVCS Charging and Letdown, to place excess letdown in service. When CV-3-387 (Excess Letdown Isolation Valve) is opened, examinee must recognize that RV-3-304 has failed open and provides a direct path to the containment sump. Examinee must either close CV-3-387 or transition to 3-ONOP-041.3, Excessive Reactor Coolant System Leakage, to start a charging pump and maintain pressurizer level. This is a bank alternate-path JPM.

c. Establish Auxiliary Pressurizer Spray per 3-EOP-E-3 – The unit has experienced a SGTR. The ruptured SG has been isolated, the RCS has been cooled down, and the examinee is directed to depressurize the RCS to minimize break flow and refill the pressurizer. Examinee must recognize that the PORVs can NOT be opened and, alternatively, will establish auxiliary pressurizer spray using Attachment 4 of 3-EOP-E-3, Steam Generator Tube Rupture. This is a new alternate-path JPM.

d. Respond to a Loss of RHR – The unit is on RHR cooling, when MOV-3-750 (RHR Pump Suction from RCS) inadvertently closes and the running RHR pump’s shaft shears. To mitigate, examinee enters 3-ONOP-050 (Loss of RHR) and re-opens the suction valve, secures the damaged pump, realigns RHR, starts the standby pump, and reinitiates cooling flow. This is a bank shutdown JPM, previously used on the 2015 NRC exam.

e. Manually Initiate Containment Spray – The unit has tripped and safety injection/phase-A containment isolation have actuated. Examinee is performing prompt action verifications in Attachment 3 of 3-EOP-E-0 (Reactor Trip or Safety Injection) and must recognize that containment spray/phase-B containment isolation have NOT actuated; examinee will manually initiate at least one train of containment spray, actuate a phase-B containment isolation and manually close phase-B valves that fail to reposition, secure RCPs, and secure the Unit 4 HHSI pumps. This is a bank engineered-safeguards JPM.

f. Restore Power to the 3A 4kV Bus – The unit has experienced a loss of all AC power. The 3A EDG did NOT start. The 3B EDG started but did not energize the 3B 4kv Bus. The Examinee is directed to restore power with a priority on the 3B EDG. The Examinee will discover the 3B 4KV Bus is locked out will restore power to the 3A 4kV Bus via the SBO tie line. This is a new alternate-path and time critical JPM.

g. Place N-3-42 Power Range Drawer in Service – The unit is at 100% power and examinee is directed to place the N-3-42 power range drawer in service using 3-OSP-059.4, Power Range Nuclear Instrumentation Analog Channel Operational Test. This is a bank JPM.

h. Respond to Control Room Evacuation Condition (Unit 3 RO) – Due to a fire, examinee responds as the unit RO per Attachment 14 of 0-ONOP-105, Control Room Evacuation, and trips the reactor/ turbine, closes the MSIVs, trips the main feedwater pumps, closes the atmospheric steam dumps, closes the PORVs/block valves, and trips the RCPs. This is a time-critical bank JPM.

i. Locally Trip the Reactor and Turbine – The unit has experienced an ATWS and the examinee is directed to trip the reactor and turbine locally. Examinee will proceed to the 3B MCC Room, open all reactor trip, bypass, and MG set breakers, and then trip the turbine at the turbine’s front standard. This is a bank JPM.

j. Control Steam Generator Level Locally with Auxiliary Feedwater Control Valve – The unit has tripped. AFW flow is required to the 3C SG. Examinee is directed to investigate and locally restore AFW flow to the 3C SG per 3-ONOP-075, Auxiliary Feedwater System Malfunction. Examinee will discover that train-2 flow to the 3C SG is NOT available and train-1 flow control valve will NOT open manually. Examinee will transition to Attachment 3 of the ONOP to locally manipulate valves and restore feedwater flow to the 3C SG. This is a bank alternate-path/emergency JPM.

k. Align Emergency Service Water to the Charging Pumps – Level can NOT be maintained in the CCW Surge Tank and a loss of cooling to the charging pumps is imminent. Examinee will use Attachment 1 of 3-ONOP-030, Component Cooling Water Malfunction, to locally establish emergency cooling water to these pumps. This is a bank RCA/emergency JPM.

Facility: **Turkey Point Units 3 & 4** Date of Examination: **08/22/2016**
Exam Level: **RO** ☐ **SRO-I** ☒ **SRO-U** ☐ Operating Test Number: **2016-301**

System / JPM Title	Type Code*	Safety Function
Control Room Systems:* 8 for RO; 7 for SRO-I; 2 or 3 for SRO-U		
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d. APE 025 Residual Heat Removal System (AA1.03, 3.3) / Respond to a Loss of RHR	L, D, P, S	4P
e. 026 Containment Spray System (A3.01, 4.5) / Manually Initiate Containment Spray	D, EN, S	5
f. EPE 055 Station Blackout (EA1.07, 4.5) / Restore Power to the 3A 4kV Bus	A, N, S	6
g. 015 Nuclear Instrumentation System (A4.02, 3.9) / Place N-3-42 Power Range Drawer in Service	D, S	7
h. NOT SELECTED FOR SRO EXAM		
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. EPE 029 Anticipated Transient Without Scram (EA1.12, 4.0) / Locally Trip the Reactor and Turbine	D, E	1
j. APE 054 Loss of Main Feedwater (AA1.01, 4.4) / Control Steam Generator Level Locally with Auxiliary Feedwater Control Valve	A, D, E	4S
k. APE 026 Loss of Component Cooling Water (AA1.03, 3.6) / Align Emergency Service Water to the Charging Pumps	D, E, R	8

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

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(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

JPM SUMMARY STATEMENTS

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c. Establish Auxiliary Pressurizer Spray per 3-EOP-E-3 – The unit has experienced a SGTR. The ruptured SG has been isolated, the RCS has been cooled down, and the examinee is directed to depressurize the RCS to minimize break flow and refill the pressurizer. Examinee must recognize that the PORVs can NOT be opened and, alternatively, will establish auxiliary pressurizer spray using Attachment 4 of 3-EOP-E-3, Steam Generator Tube Rupture. This is a new alternate-path JPM.

d. Respond to a Loss of RHR – The unit is on RHR cooling, when MOV-3-750 (RHR Pump Suction from RCS) inadvertently closes and the running RHR pump's shaft shears. To mitigate, examinee enters 3-ONOP-050 (Loss of RHR) and re-opens the suction valve, secures the damaged pump, realigns RHR, starts the standby pump, and reinitiates cooling flow. This is a bank shutdown JPM, previously used on the 2015 NRC exam.

e. Manually Initiate Containment Spray – The unit has tripped and safety injection/phase-A containment isolation have actuated. Examinee is performing prompt action verifications in Attachment 3 of 3-EOP-E-0 (Reactor Trip or Safety Injection) and must recognize that containment spray/phase-B containment isolation have NOT actuated; examinee will manually initiate at least one train of containment spray, actuate a phase-B containment isolation and manually close phase-B valves that fail to reposition, secure RCPs, and secure the Unit 4 HHSI pumps. This is a bank engineered-safeguards JPM.

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g. Place N-3-42 Power Range Drawer in Service – The unit is at 100% power and examinee is directed to place the N-3-42 power range drawer in service using 3-OSP-059.4, Power Range Nuclear Instrumentation Analog Channel Operational Test. This is a bank JPM.

h. NOT SELECTED FOR SRO EXAM

i. Locally Trip the Reactor and Turbine – The unit has experienced an ATWS and the examinee is directed to trip the reactor and turbine locally. Examinee will proceed to the 3B MCC Room, open all reactor trip, bypass, and MG set breakers, and then trip the turbine at the turbine's front standard. This is a bank JPM.

j. Control Steam Generator Level Locally with Auxiliary Feedwater Control Valve – The unit has tripped. AFW flow is required to the 3C SG. Examinee is directed to investigate and locally restore AFW flow to the 3C SG per 3-ONOP-075, Auxiliary Feedwater System Malfunction. Examinee will discover that train-2 flow to the 3C SG is NOT available and train-1 flow control valve will NOT open manually. Examinee will transition to Attachment 3 of the ONOP to locally manipulate valves and restore feedwater flow to the 3C SG. This is a bank alternate-path/emergency JPM.

k. Align Emergency Service Water to the Charging Pumps – Level can NOT be maintained in the CCW Surge Tank and a loss of cooling to the charging pumps is imminent. Examinee will use Attachment 1 of 3-ONOP-030, Component Cooling Water Malfunction, to locally establish emergency cooling water to these pumps. This is a bank RCA/emergency JPM.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	007		2.4.34
	Importance Rating	4.2		

Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: RO Question # 1

Given the following conditions:

- Unit 3 and Unit 4 are at 100% power.
- A fire is reported in the control room electrical chase.
- Units are tripped in accordance with 0-ONOP-105, Control Room Evacuation.

Which one of the following completes the statements below?

In accordance with 0-ONOP-105, the Third Licensed Reactor Operator will mechanically trip the ___(1)___ AFW pump in the AFW cage.

When ASP transfer switches are placed to LOCAL, a ___(1)___ AFW pump may occur.

A.	(1) A (2) trip of a running
B.	(1) A (2) start of a non-running
C.	(1) C (2) trip of a running
D	(1) C (2) start of a non-running

Proposed Answer: A

A.	Correct. Part 1 is correct. The Train 1 AFW pump (AFWP A) is not protected and has no control from either ASP so it will be mechanically tripped. B and C AFWPs are the Appendix R pumps and remain running. Part 2 is correct. Placing the ASP switches to LOCAL may trip the running pumps. This question requires knowledge of Appendix R equipment operated from outside the Control Room and the effects of taking local remote switches to LOCAL at the ASP.
B.	Incorrect. First part is correct but second part is incorrect. Second part is plausible because a start of a non-running AFW pump is possible given that safe shutdown modifications complies with Appendix R criteria, which has caused a loss of or potential loss of or unreliable Control Room controls and instrumentation (spurious starting or tripping of equipment). For example, the candidate may believe that a steam supply MOV can spuriously open causing the steam turbine AFW pump to start. It is incorrect in this case because placing the Appendix R remote switches in LOCAL may trip a running pump in accordance with the cautions in 0-ONOP-105. This will require the Appendix R pump to be restarted. This question statement is independent of part 1. Part 1 requires knowledge of which AFW pumps are Appendix R qualified. Part 2 requires knowledge of an ONOP-specific caution.
C.	Incorrect. Only AFW pump A is tripped. The B and C pumps are verified running or they are reset and started. B and C AFW pumps are the Appendix R pumps. Plausible because the B and C AFW pumps may trip when their T&T valves are placed in local control. Also, the candidate must know which pumps are Appendix R qualified. The second part is correct.
D.	Incorrect. Plausibility for both parts is described in the analysis for Options B and C

Technical Reference(s)	0-ONOP-105 Att 16		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902252 Obj 8		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		

10 CFR Part 55 Content:	55.41	10
	55.43	
Administrative, normal, abnormal, and emergency operating procedures for the facility.		
Comments:		

ATTACHMENT 16
Third Licensed Reactor Operator
(Page 2 of 18)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

At least one AFW pump should be in operation within 20 minutes following a Unit TRIP.

3. **PROCEED** to AFW Pump Cage.

NOTE

- AFW Pump B T&T Valve is controlled from Unit 4 ASP.
- AFW Pump C T&T Valve is controlled from Unit 3 ASP.
- AFW Pumps B and C are Alternate Shutdown protected and normally aligned to Train 2.
- AFW Pump A is normally aligned to Train 1.
- Only Train 2 Auxiliary Feedwater Flow is controllable from the ASPs, therefore A AFW Pump is TRIPPED to ensure zero Train 1 Auxiliary Feedwater Flow.
- AFW Pump A and Train 1 are **NOT** Alternate Shutdown protected, therefore should only be operated under close supervision.

4. Mechanically **TRIP** A AFW Pump.

Step	Action/Expected Response	Response Not Obtained
5.	**DETERMINE** if AFW is REQUIRED: * Unit 3 RHR System was **NOT** IN SERVICE prior to Control Room evacuation * Unit 4 RHR System was **NOT** IN SERVICE prior to Control Room evacuation	**PERFORM** the following: A. **ENSURE** B and C AFW Pumps STOPPED. B. **GO TO** Attachment 16, Step 8.

ATTACHMENT 16
<u>Third Licensed Reactor Operator</u>
(Page 3 of 18)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

<u>NOTE</u>

When Unit ROs place respective T&T Valve Transfer Switches to LOCAL, the associated AFW Pump may TRIP and require a restart.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**CHECK** B and C AFW Pumps **NOT** TRIPPED.	Locally **RESET** and **START** AFW Pump(s) as follows: **A. PLACE** Keylock Switch on applicable AFW Pump Local Control Panel to LOCAL. **B. RESET** mechanical trip by moving trip linkage towards MOV. **C. PRESS** OPEN T&T Pushbutton for applicable AFW Pump.
7.	**CHECK** B and C AFW Pumps <u>both</u> RUNNING.	Locally **OPEN** MOV-3 (4)-1403, STM GEN 3A (4A) SUPPLY TO AFW PUMPS.

<u>NOTE</u>

Attachment 16, Step 8 and Attachment 16, Step 9 shall be completed <u>prior</u> to leaving the AFW Pump area.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	Using Alternate Shutdown Communication System, **CHECK** AFW Pump T&T Valve Transfer Switches on ASP in LOCAL on <u>both</u> Units.	**RETURN TO** Attachment 16, Step 5.
9.	**CHECK** Keylock Switches for B and C AFW Pumps in REMOTE.	**PLACE** applicable Keylock Switch(s) to REMOTE.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	008		AA1.05
	Importance Rating	3.4		

Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: LPI System

Proposed Question: RO Question # 2

Given the following initial conditions:

- Unit 3 trips from full power.
- RCS pressure is 1200 psig and lowering.
- PRZ level is 80% and rising.

Subsequently:

- The crew transitions to 3-EOP-E-1, Loss of Reactor or Secondary Coolant.
- Containment temperature is 160°F.
- RCS pressure stabilizes at 590 psig.

Which one of the following completes the statements below?

The crew is responding to a break on the PRZ ___(1)___ line.

RHR stop criteria ___(2)___ met.

A.	(1) surge (2) is NOT
B.	(1) surge (2) is
C.	(1) safety (2) is NOT
D.	(1) safety (2) is

Proposed Answer:	D		
A.	Incorrect. Plausible because all indications, with the exception of PRZ level, are indications of a break in the PRZ surge line / RCS leg. Additionally, the candidate must understand that 3-EOP-E-1 provides guidance to stop RHR if RHR flow is <1100 gpm and RCS pressure is >275 psig.		
B.	Incorrect. First part plausible for same reason as option A and second part is correct.		
C.	Incorrect. First part is correct and second part is plausible for same reason as in option A		
D.	Correct. Indications of a steam-space LOCA are present. 3-EOP-E-1 provides guidance to stop RHR if RHR flow is <1100 gpm and RCS pressure is >275 psig, in this case RHR pumps are at shut-off head and will be secured.		
Technical Reference(s)	3-EOP-E-1, step 13		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902327 Obj 10		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

High-Head SI flow and RCS subcooling are required to be monitored. If either High-Head SI flow increases or RCS subcooling decreases in an uncontrolled manner, the RHR Pumps must be manually restarted to supply water to the RCS.

➔ **13. Check If RHR Pumps Should Be Stopped**

a. SI System – ALIGNED IN THE RWST INJECTION MODE

a. IF SI System has already been aligned for Cold OR Hot Leg Recirculation, THEN go to Step 15.

b. RCS pressure – GREATER THAN 275 PSIG[575 PSIG]

b. IF RHR flow greater than 1100 gpm, THEN go to Step 15.

c. RHR flow – LESS THAN 1100 GPM

c. Go to Step 14.

d. SI – RESET

e. Stop RHR Pumps and place in standby

14. Check RCS And S/G Pressures

- Pressure in all S/Gs – STABLE OR INCREASING
- RCS pressure – STABLE OR DECREASING

Observe NOTE prior to Step 1 and return to Step 1.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	009		EK1.02
	Importance Rating	3.5		

Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Use of steam tables

Proposed Question: RO Question # 3

Given the following conditions:

- 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization, is in progress.
- PRZ narrow range pressure is 1500 psig.
- RCS wide range pressure is 1400 psig.
- CET temperatures are 500°F.

Which one of the following completes the statements below?

Subcooling based on CETs is ___(1)___ .

Subcooling is monitored on the foldout page to determine if ___(2)___ .

A.	(1) 88°F (2) SI re-initiation is required
B.	(1) 88°F (2) voiding will occur during depressurization
C.	(1) 97°F (2) SI re-initiation is required
D.	(1) 97°F (2) voiding will occur during depressurization

Proposed Answer: A

A.	Correct. Per ES-1.2 Foldout Page criteria, RCS subcooling is monitored for SI re-initiation criteria. Current saturation temperature for lowest pressure is 588°F. CET temperature of 500°F equals 88°F subcooling. Wide range RCS pressure is used to calculate subcooling. Note that PRZ pressure bottoms out at 1500 psig.
B.	Incorrect. Correct subcooling but incorrect reason. Voiding concern is plausible and the procedure cautions that it could occur; however, there is no subcooling limit given for this concern in the foldout page, but the procedure gives subcooling values in the SI flow reduction steps.
C.	Incorrect. Reason given is correct but subcooling is incorrect. Subcooling value is plausible if the non-conservative RCS temperature/pressure are used in the calculation (i.e., 597°F for 1500 psig).
D.	Incorrect. Both subcooling and reason are incorrect. See A and B for explanations.

Technical Reference(s)	3-EOP-ES-1.2, Foldout Page Item 3 Steam Tables		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	15669	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2013	Callaway
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

FOLDOUT PAGE
For Procedure 3-EOP-ES-1.2

1. **ADVERSE CONTAINMENT CONDITIONS**
   A. IF either condition listed below occurs, THEN use [Adverse Containment Setpoints]:
   Containment atmosphere temperature ≥ 180°F
   OR
   Containment radiation levels ≥ $1.3x10^5$ R/hr
   B. WHEN Containment atmosphere temperature returns to less than 180°F, THEN Normal Setpoints can again be used.
   C. WHEN Containment radiation levels return to less than $1.3x10^5$ R/hr, THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.
2. **SI TERMINATION CRITERIA**
   IF all conditions listed below occur, THEN go to 3-EOP-ES-1.1, SI TERMINATION, Step 1:
   A. RCS Subcooling based on Core Exit TCs – GREATER THAN 19°F[GREATER THAN ADVERSE VALUE IN TABLE BELOW]

SI TERMINATION ADVERSE SUBCOOLING VALUE	
RCS PRESSURE (PSIG)	**ADVERSE SUBCOOLING VALUE**
< 2485 AND ≥ 2000	35 °F
< 2000 AND ≥ 1500	45 °F
< 1500 AND ≥ 1000	55 °F
< 1000 AND ≥ 500	110 °F
< 500	160 °F

   B. Total feed flow to intact S/Gs – GREATER THAN 400 GPM OR Narrow Range Level in at least one intact S/G – GREATER THAN 7%[27%]
   C. RCS pressure – GREATER THAN 1625 PSIG[1950 PSIG] AND STABLE OR INCREASING
   D. PRZ level – GREATER THAN 7%[48%]
   E. Charging Capability – AVAILABLE
3. **SI RE-INITIATION CRITERIA**
   IF either condition listed below occurs following SI reduction,
   THEN manually start SI pumps as necessary to restore RCS subcooling and PRZ level:
   * RCS subcooling based on Core Exit TCs – LESS THAN 19°F[73°F]
   OR
   * PRZ level – CAN **NOT** BE MAINTAINED GREATER THAN 7%[48%]
4. **SECONDARY INTEGRITY CRITERIA**
   IF any S/G pressure is decreasing in an uncontrolled manner OR has completely depressurized, AND that S/G has **NOT** been isolated, THEN go to 3-EOP-E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.
5. **E-3 TRANSITION CRITERIA**
   IF any S/G level increases in an uncontrolled manner OR any S/G has abnormal radiation,
   THEN manually start SI Pumps and go to 3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.
6. **COLD LEG RECIRCULATION SWITCHOVER CRITERIA**
   IF RWST level decreases to less than 155,000 gallons,
   THEN go to 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1.
7. **CST MAKEUP WATER CRITERIA**
   IF CST level decreases to less than 12%,
   THEN add makeup to CST using 3-NOP-018.01, CONDENSATE STORAGE TANK (CST).
8. **LOSS OF OFFSITE POWER OR SI ON OTHER UNIT**
   IF SI has been reset AND subsequently either offsite power is lost OR SI actuates on the other unit,
   THEN restore safeguards equipment, and at least one Computer Room Chiller to required configuration.
   Refer to Attachment 2 for essential loads.
9. **LOSS OF CHARGING CRITERIA**
   IF charging capability has been lost, AND High-Head SI Pumps are running at shutoff head,
   THEN rotate High-Head SI Pumps as necessary to maintain continuous run time of any pump less than 30 minutes while maintaining at least one High-Head SI Pump running.

Facility: WTSI Corporate

Question 3 original

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following plant conditions:

- A Reactor Trip and Safety Injection have occurred due to a small break Loss of Coolant Accident (LOCA)

- ES-1.2, Post LOCA Cooldown and Depressurization, is in progress

- Current Reactor Coolant System (RCS) conditions are as indicated below:

  - BB PI-455A, RCS Narrow Range Pressure 1700 psig
  - BB PI-456, RCS Narrow Range Pressure 1700 psig
  - BB PI-403, RCS Wide Range Pressure 1535 psig
  - Highest Core Exit Thermocouple 530F
  - Highest RCS Hot Leg Temperature 510F

Which one of the following choices correctly completes the statement below? Subcooling is monitored on the Foldout Page to (1) AND the current value of subcooling is (2).

A. (1) ensure SI reinitiation, if required
(2) 70F

B. (1) prevent voiding during depressurization
(2) 70F

C. (1) ensure SI reinitiation, if required
(2) 104F

D. (1) prevent voiding during depressurization

(2) 104F

Proposed Answer: A

Explanation (Optional):

A. Correct. Per ES-1.2.Foldout Page criteria, RCS subcooling is monitored for SI reinitiation criteria. Current subcooling for given conditions is: saturated temperature for lowest pressure is 600F 6 highest temperature of 530F equals 70F.

B. Incorrect. Correct subcooling but incorrect reason. Voiding concern is plausible as the procedure caution that it could occur; however there is no subcooling limits given for this concern.

C. Incorrect. Reason given is correct but subcooling is incorrect. Subcooling value is plausible if the non-conservative RCS temperature is used in the calculation.

D. Incorrect. Both subcooling and reason are incorrect. See A and B for explanations.

Techical Reference(s): ES-1.2, Post LOCA Cooldown and Depressurization ERG Executive Volume-Generic Issue Foldout Page Items Steam Tables (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: YES

Learning Objective: T61.003D, LP D-10, Obj E, Describe the criteria and the basis for information as stated on the ES-1.2, Post LOCA Cooldown and Depressurization, Foldout Page. (As available)

Question Source: Bank 15669

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2013 Callaway

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

References to be provided to applicants during examination: Steam Tables

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	011		EK3.03
	Importance Rating	4.1		

Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: Starting auxiliary feed pumps and flow, ED/G, and service water pumps

Proposed Question: RO Question # 4

Given the following initial conditions:

- Unit 3 experiences a reactor trip due to a LOOP from 100% power.
- 3B EDG fails to start.

Subsequently:

- A Large Break LOCA occurs.
- Containment pressure is 28 psig.

Which one of the following completes the statements below?

The AFW System will start and ___(1)___ required to support core decay heat removal during this accident.

During the injection phase of the accident, ICW pumps will be loaded on the 3A EDG for ___(2)___ .

A.	(1) is NOT (2) containment heat removal via ECCs
B.	(1) is NOT (2) core decay heat removal via RHR HXs
C.	(1) is (2) containment heat removal via ECCs
D.	(1) is (2) core decay heat removal via RHR HXs

Proposed Answer: A	
A.	Correct. Part 1- AFW pumps will not be required during a LBLOCA where accumulators and safety injection pumps will provide the inventory for decay heat removal. Part 2- ICW pumps will be loaded onto an EDG in accordance with Attachment 3 of 3-EOP-E-0 to support containment heat removal during a LBLOCA. Candidate must know reasons why components start during a LBLOCA.
B.	Incorrect. Part1- same as A. Part 2- Plausible if candidate confuses the inventory provided by RHR with the heat exchange method during a LBLOCA (accumulator and RWST to the core and out the break during the injection phase). The candidate may believe that the RHR HXs play a vital role during the injection phase. Incorrect because RWST water flows through the RHR pumps and RHR HXs but is not cooled by them, as is the case during the recirculation phase of a LBLOCA.
C.	Incorrect. Part 1- Plausible if candidate believes that AFW will provide a secondary heat sink as in a SBLOCA. Part 2 is correct.
D.	Incorrect. Plausible for reasons as stated in options B and C

Technical Reference(s)	LP 6902163		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902163 obj 8		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	7
	55.43	
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		
Comments:		

ATTACHMENT 3
Prompt Action Verifications
(Page 5 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**Verify Proper ICW System Operation**	
	a. Verify ICW Pumps – AT LEAST TWO RUNNING	**a.** Start ICW Pump(s) to establish at least two running.
	b. Verify ICW To TPCW Heat Exchanger – ISOLATED: • POV-3-4882 – CLOSED • POV-3-4883 – CLOSED	**b.** Manually close valve(s). IF valve(s) can **NOT** be closed, THEN locally close the following valves: * 3-50-319 for POV-3-4882 * 3-50-339 for POV-3-4883
	c. Check ICW Headers – TIED TOGETHER	**c.** IF both ICW headers are intact, THEN direct operator to tie headers together.
8.	**Verify Containment Cooling**	
	a. Check Emergency Containment Coolers – ONLY TWO RUNNING	**a.** Manually start or stop Emergency Containment Coolers to establish only two running.
9.	**Verify Containment Ventilation Isolation**	
	a. Unit 3 Containment Purge Exhaust And Supply Fans – OFF	**a.** Manually stop fans.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	015		AK2.08
	Importance Rating	2.6		

Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions and the following: CCWS

Proposed Question: RO Question # 5

Given the following conditions:

- Unit 3 is at 100% power.
- The crew enters 3-ONOP-030, Component Cooling Water Malfunction.
- 3B RCP lower guide bearing reaches 215°F.
- Component Cooling Water flow can NOT be established in either header.

Which one of the following completes the statements below?

The crew will trip the reactor and immediately secure ___(1)___ .

The crew will operate the charging pumps at ___(2)___ speed.

A.	(1) only the 3B RCP (2) maximum
B.	(1) only the 3B RCP (2) minimum
C.	(1) all RCPs (2) maximum
D.	(1) all RCPs (2) minimum

Proposed Answer: C

A.	Incorrect. Plausible because RCP 3B exceeds the bearing temperature required for RCP trip. Second half is correct.
B.	Plausible for same reason as option A, but second half is incorrect. When charging pump is run at slow speed it creates more heat, so it is run at max speed until cooling water can be realigned.
C.	Correct. All RCPs will be secured IAW the foldout page of 3-ONOP-030 given CCW flow can NOT be established. The charging pumps will be run at maximum speed.
D.	Incorrect. Plausible because first half is correct and for same reason as option B.

Technical Reference(s)	3-ONOP-030		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902229		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.2 Subsequent Operator Actions (continued)

3.	**CHECK** flow normal in both Component Cooling Water headers. • FI-3-613A, FLOW IND FOR CCW LOOP A • FI-3-613B, FLOW IND FOR CCW LOOP B	IF CCW flow to RCPs can **NOT** be established, THEN: **A.** Manually **TRIP** the reactor. **B.** **ENSURE** reactor trip per EOP. **C.** **STOP** all RCPs. **D.** **ISOLATE** Letdown and Excess Letdown. **E.** IF any Charging Pump is operating, THEN **OPERATE** at maximum speed until Attachment 1, Control of Emergency Cooling Water to Charging Pumps is COMPLETE. **F.** **ESTABLISH** emergency cooling water to desired Charging Pump(s) per Attachment 1, Control of Emergency Cooling Water to Charging Pumps.

FOLDOUT PAGE
For Procedure 3-ONOP-030

TOTAL LOSS OF CCW FLOW

1) Manually **TRIP** the reactor.
2) **CONFIRM** reactor trip using the EOP network.
3) **STOP** all RCPs.
4) **ISOLATE** Letdown and Excess Letdown.
5) **ESTABLISH** one Charging Pump running at maximum speed, and **DISPATCH** operator to establish emergency cooling water to one of the remaining two Charging Pumps per Attachment 1.
6) **MONITOR** RCS pressure closely while running Charging Pump at maximum speed.
7) WHEN Attachment 1 is COMPLETE, THEN **OPERATE** Charging Pump supplied with emergency cooling to maintain RCP seal cooling.

LOSS OF CCW TO ANY COMPONENT
IF Component Cooling Water flow to any component cooled by CCW is lost, THEN **SHUT DOWN** the affected component.

CHARGING PUMP EMERGENCY COOLING CRITERIA
IF Cooling Water is **NOT** available to Charging Pumps, THEN **OPERATE** Charging Pump at maximum speed until cooling is restored from CCW System or per Attachment 1.

CCW PUMP STOPPING CRITERIA
IF any Component Cooling Water Pump is cavitating, THEN **STOP** the affected Component Cooling Water Pumps, and **PLACE** in PULL-TO-LOCK.

REACTOR TRIP CRITERIA
IF tripping a RCP is required, THEN manually **TRIP** the reactor prior to STOPPING the RCP.

RCP STOPPING CRITERIA
IF any RCP bearing temperature annunciator alarm actuates AND its associated motor bearing temperature is greater than 195°F, THEN **TRIP** reactor and **STOP** the affected RCPs.

CCW PUMPS, HEAT EXCHANGERS, AND FLOWS/LOADS
CCW System operation once CCW System Header has been restored shall be within the operating restrictions of 3-NOP-030 summarized as follows:

- N-1 CCW Pumps (where N = number of CCW HXs aligned to CCW)
- All CCW HXs in service when RHR in service
- With only two CCW HXs in service AND both RHR HXs aligned to CCW, **PLACE** two CCW Pumps in PULL-TO-LOCK.
- Maximum five out of six CCW Heat Loads.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	022		AK1.03
	Importance Rating	3.0		

Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Relationship between charging flow and PRZ level

Proposed Question: RO Question # 6

Given the following initial conditions:

- Unit 3 trips from full power.
- The crew enters 3-EOP-ES-0.1, Reactor Trip Response.
- RCS temperature stabilizes at 547°F.
- PRZ level stabilizes on program.

Subsequently:

- A loss of all charging occurs.
- PRZ level is lowering by 1% every four minutes.

Which one of the following identifies the MINIMUM time to restore charging flow BEFORE letdown is automatically isolated (assuming no operator actions)?

A.	2 - 10 minutes
B.	30 - 40 minutes
C.	80 - 90 minutes
D.	170 - 180 minutes

Proposed Answer: B

A.	Incorrect. Plausible if candidate miscalculates by taking the difference of 8% between actual level and isolation and dividing this by 4 to give 2 minutes.

B.	Correct. No load PRZ level is approximately 22% and letdown isolation occurs at 14%. 22% - 14% = 8%. If PRZ level is lowering 1% every 4 minutes, then it should take approximately 32 minutes to reach letdown isolation.
C.	Incorrect. Plausible because the candidate may calculate 22% to 0% (off-scale low). At a rate of 1% every 4 minutes, it will take 88 minutes to reach this value.
D.	Incorrect. Plausible because full power PRZ level is 57% and the candidate may inadvertently consider this the PRZ level number to begin at. 57% - 14% is 43%. At 1% every 4 minutes, it would take 172 minutes to reach letdown isolation.

Technical Reference(s)	Drawing 5610-T-D-15	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:	N

Learning Objective:	6902109 Obj 7.b	(As available)

Question Source:	Bank		
	Modified Bank	X	(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	2008	VC Summer

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

10 CFR Part 55 Content:	55.41	7
	55.43	

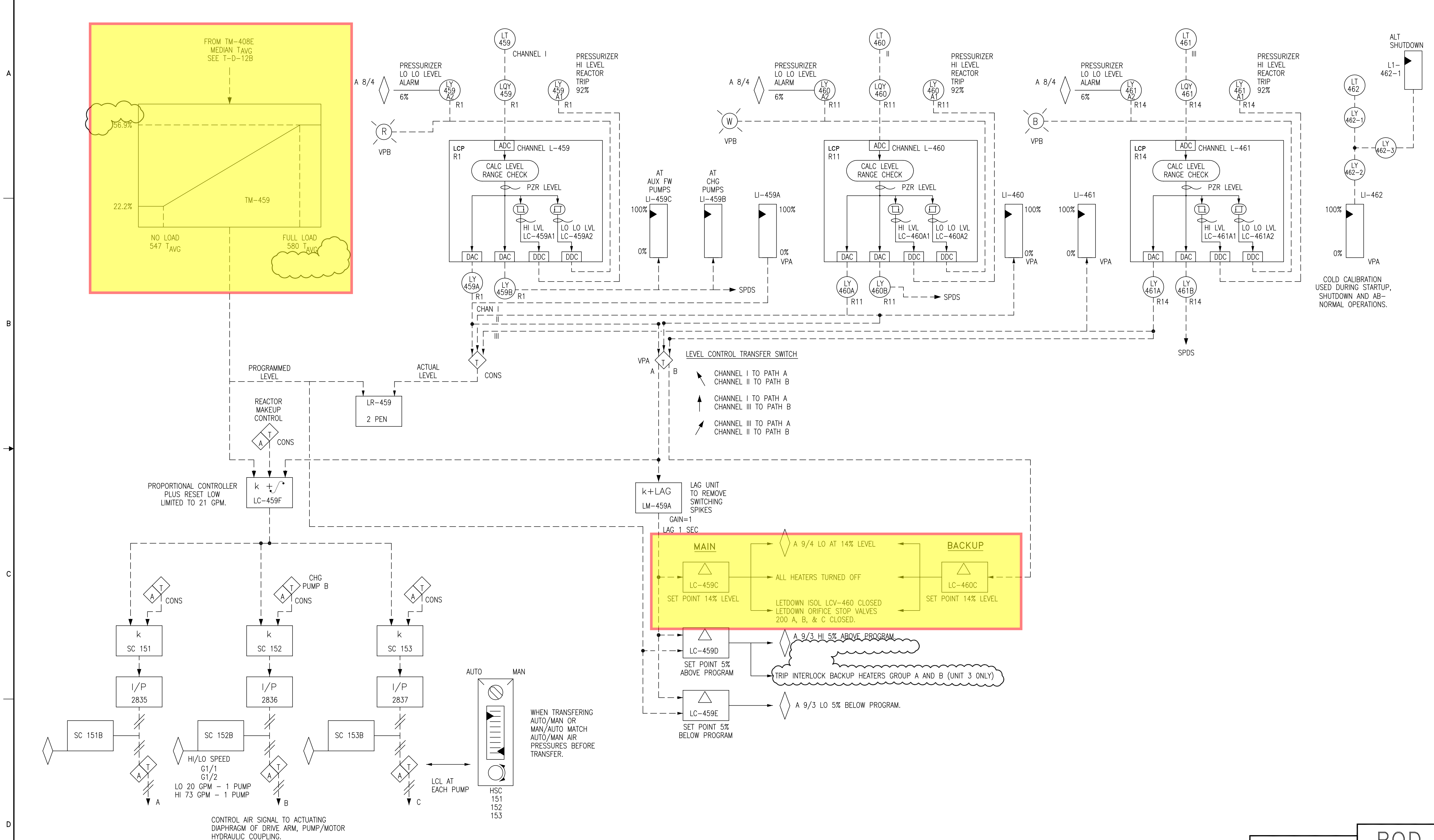
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Question was modified by changing conditions to solicit letdown isolation instead of SI actuation, used plant specific information and changed correct answer.

DWG NO 5610-T-D-15 SHT 1

PRESSURIZER LEVEL



REV	DATE	REVISION	BY	CH	APP	APP
24	07-10-13	ISSUED AS-BUILT PER EC'S 247049, 246978, 246977; INCORP. CRN'S 002, 017 AND 023 (ITOP'S 13-07-008, 015 & 017).	RV	BB	–	PMB
23	1-5-13	ISSUED AS-BUILT PER CRN-017 (EC 246977) ITOP 13-01-013.	AFG	BB	–	PMB
22	7-29-12	ISSUED AS-BUILT PER EC 247048 (PC/M 10-039).	RH	BB	TK	JD
21	6/26/09	ISSUED AS-BUILT PER CRN I-5048 (PC/M 09-034).	RV	RH	–	TS
20	10-21-97	ISSUED AS-BUILT PER CRN-I-3593(PC/M 97-028).	RV	RH	BAP	TS
19	5/11/95	ISSUED AS-BUILT FOR PC/M 94-120	RV	RH	BAW	JCW
18	3/2/93	AS-BUILT PER DCR TPE-92-316	RH	RV	ET	JCW
17	5/8/92	AS-BUILT PER DCR TPI-91-463	RH	RV	ET	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

ELECTRICAL

PRESSURIZER LEVEL CONTROL AND PROTECTION CHARGING PUMP CONTROL

QUALITY RELATED | POD

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-D-15 | SYS

REV 24

SHEET 1 OF 1

5610-T-D-15_SH 1.DWG

Question 6 original

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following plant conditions:

- The plant has tripped from 100% power.
- The crew has entered EOP-1.1, *Reactor Trip Recovery*.
- RCS Temperature is STABLE at 557F.
- PZR level is at no-load Tavg program level.

Subsequently,

- A loss of all Charging occurs.
- PZR level continues to decrease at 1%/every 3 minutes.
- Attempts are continuing to restore Charging to operation.

Assuming conditions do NOT change (including the rate of PZR level decrease), which ONE (1) of the following identifies how much time the operator has to restore charging BEFORE a manual Safety Injection actuation will be required?

A. 15 minutes

B. 21 minutes

C. 39 minutes

D. 60 minutes

Proposed Answer: C

Explanation (Optional):

A. Incorrect. 25% - 20% (low deviation alarm) = 5% x 3 min/% = 15 min.. According to the EOP Reference Page, SI will NOT need to be actuated until PZR Level drops to 12%

B. Incorrect. 25% - 18% (old Reference Page value) = 7% x 3 min/% = 21 minutes

C. With RCS temperature not changing the PZR level decrease is associated with the inventory loss which is occurring at a rate of 1% of PZR level every 3 minutes. The EOP-1.1 Reference Page (Rev 15) requires that Safety Injection be manually actuated if PZR Level cannot be maintained > 12%. With PZR level at 25%, and the present rate of PZR Level decrease continuing, this criteria will be met in 39 minutes. 25% - 12% = 13% x 3 min/% = 39 min

D. Incorrect. 25% - 5% (values in AOP-101.1 & 112.2 for manual SI based on VCT level) = 20% x 3min/% = 60 min

Techical Reference(s): IC3, p37, Rev 9
EOP-1.1 Rev 15 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: (As available)

Question Source: Bank 11219
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2008 VC Summer

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

The KA is met because it requires the operator to have knowledge of the manual ESFAS actuation requirements (operational implications) during a situation where a loss of Charging is causing a significant loss of inventory (as evidenced by a decreasing PZR level)

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	025		AA1.09
	Importance Rating	3.2		

Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: LPI pump switches, ammeter, discharge pressure gauge, flow meter, and indicators

Proposed Question: RO Question # 7

Given the following initial conditions:

- Unit 3 is in Mode 5.
- PRZ cold cal level is 25% and stable.
- 3A RHR Pump is in service.
- RHR flow is 3100 gpm.

Subsequently:

- The 3A RHR Pump's motor amps and flow lower and become erratic.
- The crew stops the 3A RHR Pump in accordance with 3-ONOP-050, Loss of RHR.

Which one of the following completes the statements below?

The crew will first attempt to start the ___(1)___ RHR Pump.

When RCS temperature is subsequently stabilized, FI-3-605, RHR flow indicator, will measure flow through ___(2)___ .

A.	(1) 3A (2) both the RHR HXs and the bypass flow line
B.	(1) 3A (2) the RHR HXs only
C.	(1) 3B (2) the RHR HXs and the bypass flow line

D.	(1) 3B (2) the RHR HXs only			
Proposed Answer: A				
A.	Correct. Part 1- The ONOP requires a start of the previously running RHR pump. Part 2- HCV-3-758 will be controlled in manual and FCV-3-605 will be controlled in auto to return flow on FI-3-605 to 3000 - 3750 gpm.			
B.	Incorrect. Part 1- correct. Part 2- plausible if candidate believes FCV-3-605 RHR bypass flow controller is not used when restoring RCS temperature.			
C.	Incorrect. Part 1- plausible because candidate may believe that the 3A RHR Pump is not available and the procedure requires a start of unaffected pump.			
D.	Incorrect. Plausible for same reasons as options B and C.			
Technical Reference(s)	3-ONOP-050		(Attach if not previously provided)	
Proposed Reference to be provided to applicants during examination:			N	
Learning Objective:	6902266 Obj 4		(As available)	
Question Source:	Bank			
	Modified Bank	X	(Note changes or attach parent)	
	New			
Question History:	Last NRC Exam:	2012	Sequoyah	
Question Cognitive Level:	Memory or Fundamental Knowledge		X	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41		10	
	55.43			
Administrative, normal, abnormal, and emergency operating procedures for the facility.				
Comments:				
Modified stem and distractors. Only used concept.				

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If leakage from the RHR system is discovered, the leak should be isolated using 3-ONOP-041.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE.

NOTES

- *Oscillations in flow or motor amps may be indicative of RHR pump cavitation.*
- *If loss of RHR is due to a loss of off-site power capability, power and RHR flow should be restored utilizing 3-ONOP-004, LOSS OF OFFSITE POWER or 3-EOP-ECA-0.0, LOSS OF ALL AC. During a loss of power, this procedure should be used to establish containment closure and alternate cooling if RHR flow remains unavailable.*
- *The foldout page shall be monitored during the performance of this procedure.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check If RHR Pumps Should Be Stopped**	
	a. RCS level - GREATER THAN 10% PRESSURIZER COLD CAL	a. **IF** RCS Draindown Level Instrumentation is not available or RCS draindown level is LESS than 23%, **THEN** stop the running RHR pump **AND** go to 3-ONOP-041.8, Shutdown LOCA (Mode 5 or 6).
	b. RHR pumps - ANY RUNNING	b. Go to Step 2.
	c. RHR pumps - NOT CAVITATING • Amps Stable at normal value • Flow Stable at normal value	c. Stop RHR pumps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Interrupt feature for MOV-3-750 and MOV-3-751 is functional only with OMS in LO PRESS OPS.

2 **Check Loop 3C RHR Pump Suction Stop Valves – OPEN**

- MOV-3-750
- MOV-3-751

Perform the following:

a. Stop RHR pumps.

b. **IF** a momentary pressure spike has caused either or both valves to start closing, **THEN** perform the following at the Pushbutton Interrupt switches:

1) Determine affected valve(s).
   - Yellow light – ON

2) Verify over pressure signal **NOT** present:
   - Blue light – ON

3) Push Interrupt Pushbutton for affected valve(s).

4) Verify yellow light - DE-ENERGIZES.

5) **WHEN** blue light DE-ENERGIZES, **THEN** verify affected valve(s) - OPEN.

6) **IF** both valves are open, **THEN** go to Step 3.

c. **IF** RCS pressure GREATER THAN 525 psig, **THEN** perform the following:

1) Stop the charging pump(s).

2) Reduce RCS pressure to 425 psig.

d. **IF** MOV-3-750 and MOV-3-751 were **NOT** closed to isolate system leakage, **THEN** reopen MOV-3-750 and MOV–3-751. **IF** either valve can **NOT** be opened, **THEN** direct an operator to locally reopen MOV-3-750 and MOV-3-751.

e. **IF** BOTH valves can **NOT** be reopened, **THEN** monitor RCS Heatup Rate using Step 4 **AND** go to Step 11.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Dispatch An Operator To Monitor RHR Pumps**	
	a. Monitor RHR pump locally	
	b. Maintain communication with Control Room	

NOTE

RCS heatup rate is required to be monitored by the Shift Technical Advisor or any available operator until RHR cooling has been re-established. |

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Monitor RCS Heatup Rate**	
	a. Plot core exit temperature every minute for 5 minutes	a. **IF** core exit temperatures are **NOT** available, **THEN** perform the following: 1) Assume a 12°F per minute heatup rate unless the refueling cavity is flooded. **IF** the refueling cavity is flooded, **THEN** use 4°F per minute. 2) Go to Step 5.
	b. Calculate RCS heatup rate	
	c. Determine time required to reach saturation in RCS	
	d. Report results to Unit Reactor Operator and the Shift Manager	
	e. Repeat this step every 15 minutes until RHR cooling is restored	

.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Verify RHR Discharge To Cold Leg Isolation Valves – OPEN** • MOV-3-744A • MOV-3-744B	**<u>IF</u>** RHR Discharge To Cold Leg Isolation valve(s) were **<u>NOT</u>** closed to isolate system leakage, **<u>THEN</u>** perform the following: a. Reopen RHR discharge valve(s). b. **<u>IF</u>** at least one valve can **<u>NOT</u>** be opened, **<u>THEN</u>** perform the following: 1) Stop RHR pump(s). 2) Direct operators to locally reopen RHR Discharge To Cold Leg Isolation Valve(s). 3) Go to Step 11.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C A U T I O N

RCS Cooldown Rate shall be maintained LESS than 90 degrees per hour.

6 Establish Conditions For Restarting An RHR Pump

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	a. RHR pumps – BOTH STOPPED	a. Go to Step 7.
	b. Close RHR Heat Exchanger Outlet Flow valve, HCV-3-758	
	c. Close RHR Heat exchanger Bypass Flow valve, FCV-3-605	
	d. Verify MOV-3-750 and MOV-3-751 – OPEN	d. Go to Step 11.
	e. Start the previously running RHR pump	e. Start the Standby RHR pump. 1) **IF** neither RHR pump can be restarted, **THEN** perform the following: a) Direct appropriate personnel to restore at least one RHR pump to operable status. b) Go to Step 11.
	f. Return RHR Heat Exchanger Bypass Flow valve, FCV-3-605, to AUTOMATIC operation increasing flow in increments of 500 gpm until desired flow is established	**Both valves are open; hence, flow is measured both through and around the HXs**
	g. Open RHR Heat Exchanger Outlet Flow valve, HCV-3-758, as necessary to maintain desired RCS temperature	

Question 7 original

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following:

- Unit 1 is in Mode 5 with RCS at Mid-Loop.
- 1A-A RHR pump in service.
- RHR flow was indicating 2400 gpm.
- The 1A-A RHR pump motor amps and flow suddenly become very erratic.

In accordance with AOP-R.03, 3RHR System Malfunction,u which ONE of the following indentifies:

(1) the RHR flow rate the operators are to maintain
and
(2) the action(s) that will be required if the 1A-A RHR pump motor amps continue to be erratic?

A. (1) between 1500 gpm and 2000 gpm
(2) make up to the RCS immediately by opening valve 1-FCV-63-1, RHR suction from RWST.

B. (1) between 1000 gpm and 1500 gpm
(2) make up to the RCS immediately by opening valve 1-FCV-63-1, RHR suction from RWST.

C. (1) between 1000 gpm and 1500 gpm
(2) stop the 1A-A RHR pump, increase the level in the RCS and attempt to restore RHR cooling.

D. (1) between 1500 gpm and 2000 gpm
(2) stop the 1A-A RHR pump, increase the level in the RCS and attempt to restore

RHR cooling.

Proposed Answer: C

Explanation (Optional):

A. Incorrect.

B. Incorrect.

C. Correct.

D. Incorrect.

Techical Reference(s): - (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: - (As available)

Question Source: Bank 13940
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2012 Sequoyah

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	027		AA1.03
	Importance Rating	3.6		

Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure control when on a steam bubble

Proposed Question: RO Question # 8

Given the following conditions:

- Unit 3 is in Mode 2.
- PRZ pressure control is in automatic.
- Pressurizer pressure is 2150 psig and lowering.
- 3-ONOP-041.5, Pressurizer Pressure Control Malfunction, is entered.

Which one of the following completes the statements below?

The operator will manually ___(1)___ the output on PC-3-444J, PZR PRESS CONTROL, to stabilize pressure.

When adjusting the output on the pressurizer pressure controller, PCV-3-455C, PZR PORV, ___(2)___ expected to automatically open if the controlling PZR pressure instrument reaches 2335 psig.

A.	(1) raise (2) is
B.	(1) lower (2) is
C.	(1) raise (2) is NOT
D.	(1) lower (2) is NOT

Proposed Answer: D

A.	Incorrect. Both incorrect, plausible per B and C explanations.
B.	Incorrect. Part 1- correct. Part 2- Incorrect. Plausible if candidate believes that PCV-3-455C operates like PCV-3-456.
C.	Incorrect. Part 1- Raising controller output will de-energize heaters/open spray valves, lowering RCS pressure. Plausible if candidate confuses this with another controller that is direct acting (there are direct and reverse acting controllers on the RCS part of the console). Part 2- correct.
D.	Correct. Part 1- Lowering controller output will energize heaters and raise pressure. Part 2- PCV-3-455C does not open at 2335 psig (as does PCV-3-456), but when demanded output is raised to 92.5%.

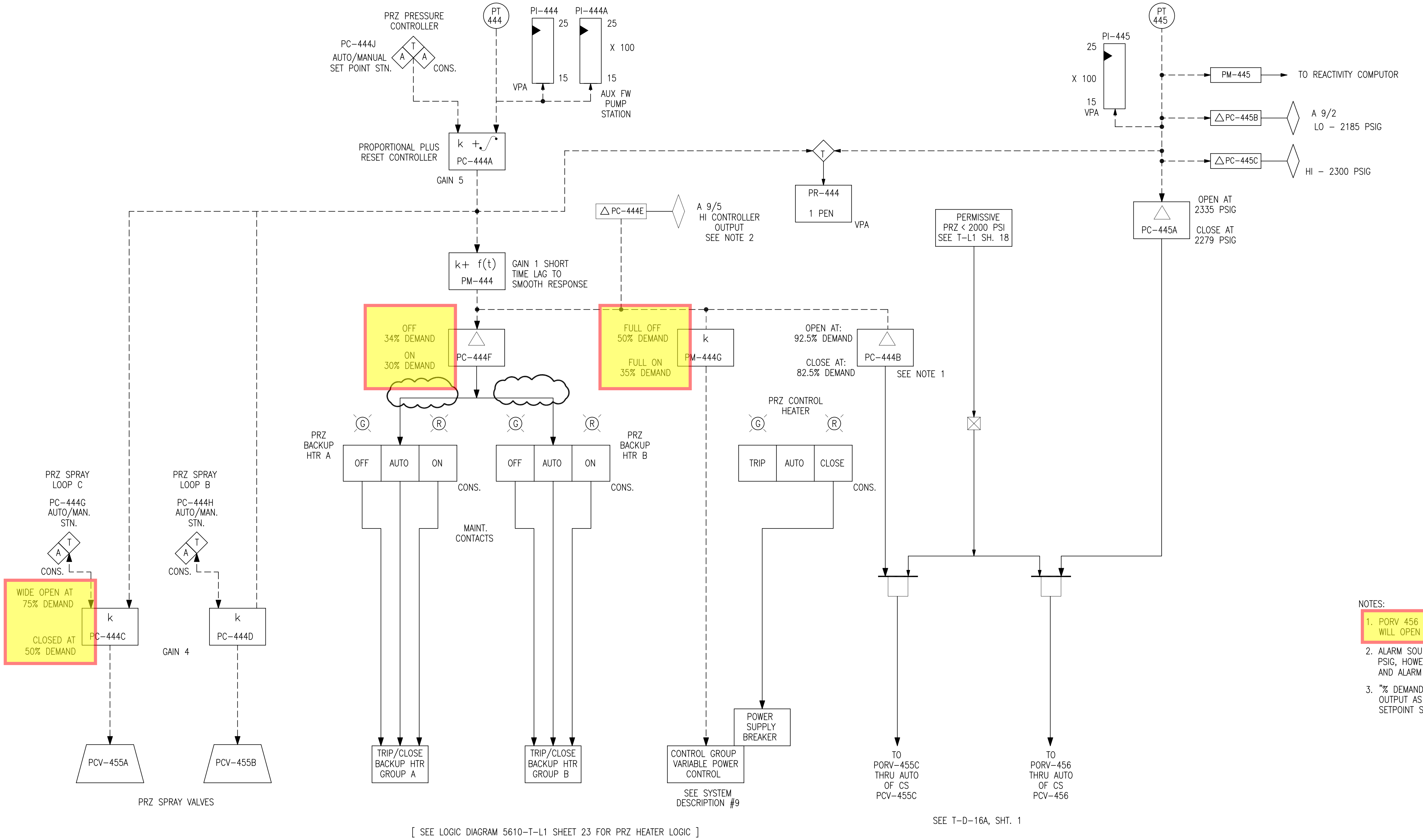
Technical Reference(s)	Drawing 5610-T-D-16B		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902109 Obj		(As available)
Question Source:	Bank		
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2009	Harris
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Question modified to change failure and change correct answer. Meets KA because PRZ is on a bubble and question solicits information on how it will be controlled.

PRESSURIZER PRESSURE



Lowering demand will close the spray valves and energize heaters, thereby raising pressure

UNIT 3 & 4 SET POINTS PRZ PRESSURE – PSIG

2465	SAFETY VALVES OPEN
2385	REACTOR TRIP
2335	PORV'S OPEN
2300	SPRAYS WIDE OPEN
	HIGH PRESSURE ALARM
2250	SPRAYS START TO OPEN
2250	CONTROL HEATERS FULL OFF
2235	NORMAL OPERATION
2220	CONTROL GP FULL ON
>2210	BACKUP HEATERS OFF (REF. ONLY)
2210	BACKUP HEATERS ON
2185	LO PRESSURE ALARM
2000	PERMISSIVE TO OPEN PORV'S
1835	REACTOR TRIP
1730	SAFETY INJECTION

NOTES:

1. PORV 456 WILL OPEN AT 2335 PSIG, PORV 455C WILL OPEN AT 92.5% DEMAND OUTPUT OF PC-444A.
2. ALARM SOUNDS AT 79% DEMAND OUTPUT OF PC-444A OR ABOUT 2300 PSIG, HOWEVER PC-444A IS A PROPORTIONAL PLUS RESET CONTROLLER AND ALARM MAY SOUND PRIOR TO 2300 PSIG.
3. "% DEMAND" REFERS TO CONTROLLER PC-444A OUTPUT AS DISPLAYED ON THE AUTO/MANUAL SETPOINT STATION.

SAFETY RELATED | POD

REV	DATE	REVISION	BY	CH	APP	APP
18	08-01-13	ISSUED AS-BUILT PER CRN-013 (EC 246978) (ITOP 13-08-002).	RV	RH	–	PMB
17	07-22-13	ISSUED AS-BUILT PER EC 246978 AND INCORP. CRN-012 (ITOP 13-07-008).	RV	RH	–	PMB
16	1-5-13	ISSUED AS-BUILT PER CRN-019 (EC 246977) ITOP 13-01-013.	AFG	BB	–	PMB
15	8-14-12	ISSUED AS-BUILT PER EC 246924 (PC/M 08-175). (PARTIAL)	RV	BB	–	JD
14	7/30/12	ISSUED AS-BUILT PER EC 246977 (PC/M 09-083).	RV	BB	–	PMB
13	7-1-12	ISSUED AS-BUILT PER EC 246924 (PARTIAL).	AFG	SB	–	BP
12	12-8-06	ISSUED AS-BUILT PER CRN-I-4500 (PC/M 06-073).	RV	BB	–	TS
11	8-5-97	ISSUED AS-BUILT PER CRN-I-3578(PC/M 97-019)	RV	RH	BP	TS
10	5-18-95	INCORPORATED CRN-I-3176 FOR PC/M 95-042.	RV	RH	JJN	JCW
9	3-8-94	INCORPORATED CRN-I-1142 FOR PC/M 93-221.	RV	RH	RAS	JCW
8	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	MSP	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4
CONTROL SYSTEM DIAGRAM
PRESSURIZER PRESSURE CONTROL

FLORIDA POWER & LIGHT
DRAWING NUMBER 5610-T-D-16B
SYS 041
REV 18
SHEET: 1

Question 8 original

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following:
• The plant is operating at 100% power.
• A failure of the PT-444 input to PK-444A, Pressurizer Pressure Master Controller, caused actual pressurizer pressure to decrease to 2190 psig.
• PK-444A has been placed in MANUAL.
Which ONE of the following describes the action required to return pressure to 2235 psig?

A. Decrease the controller output.

B. Increase the controller output.

C. Lower the pressure setpoint adjustment.

D. Raise the pressure setpoint adjustment.

Proposed Answer: A

Explanation (Optional):

A. *correct. Decreasing controller output will energize heaters and raise pressure.*

B. *incorrect. Increasing controller output will de-energize heaters/open spray valves lowering pressure.*

C. *incorrect, once in manual adjusting the setpoint will have no effect. Plausible if applicant believes setpoint is still in the control circuitry while in manual.*

D. *incorrect, once in manual adjusting the setpoint will have no effect. Plausible if applicant believes setpoint is still in the control circuitry while in manual.*

Techical Reference(s): AOP-019, PPCS Text (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: (As available)

Question Source: Bank 11196

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2008 Harris

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	029		EK2.06
	Importance Rating	2.9		

Knowledge of the interrelations between the following and ATWS: Breakers, relays, and disconnects

Proposed Question: RO Question # 9

Which one of the following describes how the AMSAC (ATWS Mitigating System Actuation Circuit) trips the reactor?

A.	Energizes both Control Rod MG set input breaker trip coils.
B.	Energizes both Control Rod MG set output breaker trip coils.
C.	Energizes the Shunt Trip Coils on both Reactor Trip Breakers and Bypass Breakers.
D	Deenergizes the Undervoltage Trip Coils on both Reactor Trip Breakers and Bypass Breakers.

Proposed Answer: B

A.	Incorrect. This is incorrect because AMSAC does not affect the Control Rod MG set input breakers. This is plausible because, according to EOP-FR-S.1, Step 7, the A/B MG set motor input breakers are used to locally trip the reactor in the event that the reactor will not trip form the Control Room. The operator may incorrectly believe that AMSAC operates the input breakers rather than the output breakers.
B.	Correct. According to Drawing 5614-T-L1 Sheet 33A, when AMSAC actuates the reactor is tripped by energizing the control rod MG set Output Breaker Trip Coils for both the A and B Control Rod MG sets.

C.	Incorrect. This is incorrect because AMSAC does not affect the Shunt Trip Coils on both Reactor Trip Breakers. This is plausible because the Reactor Protection System uses UV Coils and Shunt Trip Coils for the Reactor Trip and Bypass Breakers to trip the reactor, rather than tripping breakers associated with the Control Rod MG sets. The operator may incorrectly believe that the AMSAC trips the reactor via a subset of the RPS.
D.	Incorrect. This is incorrect because AMSAC does not affect the Undervoltage Trip Coils on the Reactor Trip Breakers and Bypass Breakers. This is plausible because the Reactor Protection System uses UV Coils and Shunt Trip Coils for the Reactor Trip and Bypass Breakers to trip the reactor, rather than tripping breakers associated with the Control Rod MG sets. The operator may incorrect believe that the AMSAC trips the reactor via a subset of the RPS.

Technical Reference(s)	Drawing 5614-T-L1 Sheet 33A	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:	N

Learning Objective:	6902163 Objectives 5.c, 5.d and 9	(As available)

Question Source:	Bank	12998	
	Modified Bank		(Note changes or attach parent)
	New		

Question History:	Last NRC Exam:	2013	Turkey Point

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

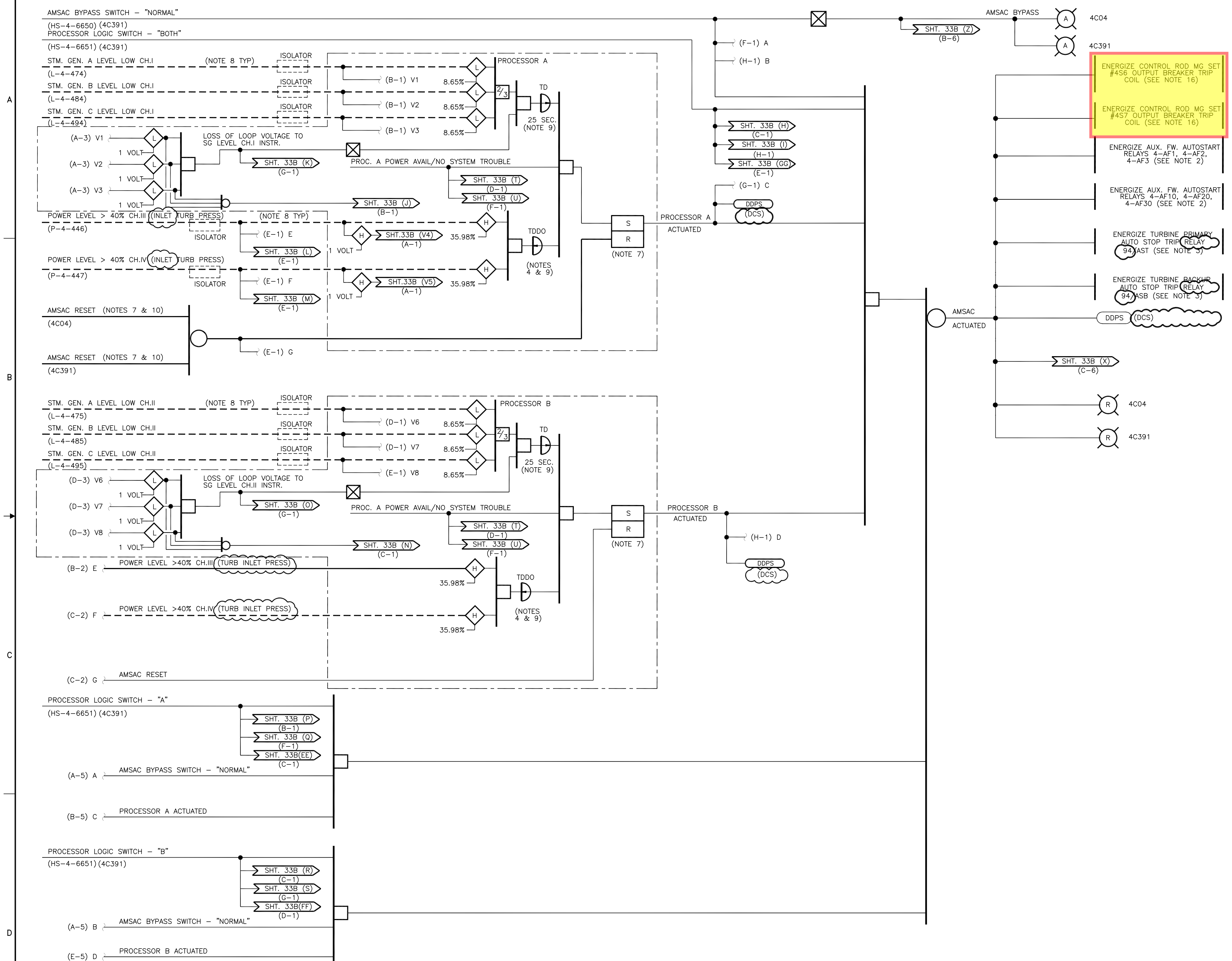
10 CFR Part 55 Content:	55.41	7
	55.43	

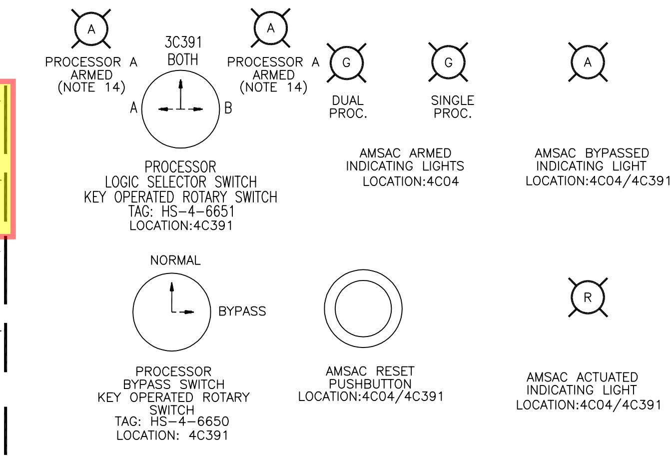
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

The KA is matched because the operator must demonstrate knowledge of the interrelations between breakers (Reactor Trip Breakers), relays (Shunt, UV Coils), and disconnects (MG Set breakers) used in the ATWS event. This is accomplished by requiring the operator to identify how a circuit (AMSAC) designed specifically for protecting the reactor against the ATWS event will trip the reactor when it is actuated.

The question is at the Memory (1F) cognitive level because the operator must recall bits of information (how the AMSAC trips the reactor) to answer the question correctly.





NOTES:

1. 4C391 IS THE AMSAC CABINET TAG NUMBER. THE CABINET WILL BE LOCATED IN THE CABLE SPREADING ROOM. 4C04 IS THE EXISTING VERTICAL CONTROL PANEL LOCATED IN THE MAIN CONTROL ROOM.
2. THE ADDITIONAL MITIGATING FUNCTIONS OF BLOWDOWN RECOVERY ISOLATION AND BLOWDOWN SAMPLING ISOLATION ARE INITIATED BY THE AUXILIARY FEEDWATER AUTO START RELAYS AND DO NOT REQUIRE SEPARATE OUTPUT SIGNALS. THE SIX AUTO START RELAYS OPEN STEAM SUPPLY VALVES MOV-4-1403, MOV-4-1404, MOV-4-1405, CLOSE STEAM GENERATOR WATER SAMPLING ISOLATION VALVES MOV-4-1425, MOV-4-1426, & MOV-4-1427, AND STEAM GENERATOR BLOWDOWN RECOVERY ISOLATION VALVES SV-4-6275A-1,B-1 & C-1 AND CV-4-6275A, B, & C.
3. THE AMSAC ACTUATION OUTPUT SIGNAL WILL TRIP THE TURBINE BY ENERGIZING TURBINE TRIP RELAYS 94/AST AND 94/ASB.
4. FOLLOWING A TURBINE TRIP, AMSAC SHALL REMAIN ARMED FOR 360 SECONDS AFTER POWER HAS BEEN REDUCED TO BELOW 40% TO ENSURE THAT AMSAC CAN ACTUATE IF AN ANTICIPATED TRANSIENT HAS OCCURRED. THE 360 SECOND TIME DELAY SHALL NOT PREVENT THE OPERATORS FROM RESETTING THE SYSTEM AFTER ACTUATION.
5. THIS DRAWING IS NOT AN EXACT PHYSICAL REPRESENTATION FOR THE LOCATION OF ALL THE LOGIC SHOWN. THE TERMINOLOGY USED HEREIN FOR THE PROCESSOR IS MEANT TO BE GENERAL IN NATURE.
6. REFER TO 5610-T-L1, SH.1 FOR STANDARD SYMBOLS.
7. THE RESET SIGNAL SHALL REMOVE THE OUTPUT ONLY IF THE SET INPUT IS NO LONGER PRESENT.
8. EACH AMSAC INPUT IS ISOLATED FROM ITS RESPECTIVE HAGAN LOOP INPUT BY AN ELECTRICAL ISOLATOR. EACH AMSAC LEVEL OR PRESSURE INPUT SHALL BE PROVIDED WITH VISIBLE INDICATION THAT THE HAGAN LOOP BISTABLE DEVICE HAS TRIPPED. ALSO, THE INDIVIDUAL PROCESSOR UNITS SHALL HAVE LIGHTS TO INDICATE PROCESSOR TROUBLE.
9. TIME DELAY SETTINGS SHALL BE ADJUSTABLE.
10. BOTH THE CONTROL ROOM (4C04) RESET PUSHBUTTON AND THE LOCAL PANEL (4C391) RESET PUSHBUTTON SHALL HAVE SIMILAR FUNCTIONS. HOWEVER, THE LOCAL PANEL RESET PUSHBUTTON SHALL HAVE THE ADDITIONAL CAPACITY TO RESET THE LOCAL (4C391) INDICATING LIGHTS.
11. HS-4-6651 SHALL REMAIN IN THE "BOTH" (DUAL PROCESSOR) POSITION UNLESS ONE OF THE PROCESSORS IS TAKEN OUT OF SERVICE FOR MAINTENANCE OR TESTING. IN THIS CASE, HS-4-6651 SHALL BE SWITCHED TO OPERATING PROCESSOR, AND AMSAC MAY REMAIN FUNCTIONAL IN THE SINGLE PROCESSOR MODE.
12. PLACING HS-4-6650 IN "BYPASS" SHALL INHIBIT THE OUTPUTS FROM BOTH PROCESSORS, THEREBY DISABLING AMSAC.
13. THE PROCESSOR AUTO TEST WILL BE ABORTED IF THE PROCESSOR RUNNING THE TEST SEES CONDITIONS REQUIRING ACTUATION.
14. THE PROCESSOR SELECT SWITCH (HS-4-6651) ON PANEL 4C391 HAS TWO LIGHTS FOR INDICATING WHEN AMSAC IS ARMED. HAVING BOTH ARMED INDICATING LIGHTS ON WILL INDICATE THAT AMSAC IS ARMED IN THE DUAL PROCESSOR MODE. WHEN AMSAC IS IN THE SINGLE PROCESSOR MODE HAVING ONE ARMED INDICATING LIGHT ON FOR THAT OPERATING PROCESSOR WILL INDICATE THAT AMSAC IS ARMED IN THE SINGLE PROCESSOR MODE.
15. AMSAC INPUT/OUTPUT TESTING TOGGLE SWITCHES AND LOGIC ARE NOT SHOWN. FOR DETAILS OF TESTING LOGIC SEE REFERENCE DRAWINGS (later).
16. THE AMSAC ACTUATION OUTPUT SIGNAL WILL TRIP THE REACTOR BY TRIPPING CONTROL ROD MG SET OUTPUT BREAKERS.
17. THE INPUT ISOLATORS ARE POWERED FROM 1 VITAL CLASS 1E POWER SUPPLY BUT HAVE SEPARATE INTERNAL FUSES. THEREFORE TO PROVIDE CONTROL ROOM ANNUNCIATION WHEN AN ISOLATOR (NOT BYPASSED) LOSES POWER THE VOLTAGE LEVEL OF EACH ISOLATOR IS MONITORED BY ITS PROCESSOR. A VOLTAGE LEVEL OF LESS THAN 1 VOLT (EXCLUDING LOOP UNCERTAINTIES) WHICH IS MONITORED FROM EACH ISOLATORS INPUT TO THE PROCESSOR WILL INDICATE A LOSS OF VOLTAGE AND WILL LIGHT UP THE AMSAC TROUBLE/ACTIVATED ANNUNCIATOR.

REFERENCE DRAWINGS

1. 5614-M-430-254, LOOP DIAGRAM
2. 5614-M-430-255, LOOP DIAGRAM
3. 5614-M-430-256, LOOP DIAGRAM
4. 5614-M-430-257, LOOP DIAGRAM
5. 5614-M-430-258, LOOP DIAGRAM
6. 5614-M-430-259, LOOP DIAGRAM
7. 5614-M-430-267, LOOP DIAGRAM
8. 5614-M-430-274, LOOP DIAGRAM

POD

REV	DATE	REVISION	BY	CH	APP	APP
3	5/1/13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-04-155)	RV	BB	–	PMB
2	3-16-09	ISSUED AS-BUILT PER CRN-I-4982 (PC/M 09-005).	RV	BB	–	LG
1	12-1-08	ISSUED AS-BUILT PER CRN-I-4914 (PC/M 08-071).	RV	RH	–	PMB
0	6-20-91	ISSUED AS POD FOR PC/M 89-169	RH	RV	LRM	JG

FPL

TURKEY POINT NUCLEAR UNIT 4

LOGIC DIAGRAM

OPERATING DIAGRAM
ATWS MITIGATING SYSTEM ACTUATION CIRCUITRY (AMSAC)

FLORIDA POWER & LIGHT

DRAWING NUMBER 5614-T-L1 | SYS – | REV 3 | SHEET 33A

5614-T-L1_SH 33A.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	038		EA2.13
	Importance Rating	3.1		

Ability to determine or interpret the following as they apply to a SGTR: Magnitude of rupture

Proposed Question: RO Question # 10

Given the following conditions:

- The crew is performing 4-ONOP-071.2, Steam Generator Tube Leakage.
- Reactor power is 50%.
- Tavg and Tref are matched.
- Pressurizer level starts lowering.
- All charging pumps are running with individual controllers in AUTOMATIC.

Which one of the following completes the statement below?

The crew is required to manually initiate safety injection when the master charging pump controller's demand is ___(1)___ in MANUAL with letdown isolated OR when PRZ level is ___(2)___ and lowering.

A.	(1) 100% (2) 40.5%
B.	(1) 100% (2) 29.5%
C.	(1) 0% (2) 40.5%
D.	(1) 0% (2) 29.5%

Proposed Answer: B

A.	Incorrect – Part 1- correct, max charging is at 100% demand. Part 2- incorrect, plausible when candidate believes SI is required when PRZ level is deviating by more than 5% from program (PZR HI/LO alarm setpoint) or if candidate makes an error in calculating current PRZ program. For example: candidate incorrectly calculates program PRZ level as (57%) / 2 = 28.5%. Current calculated PZR program level is actually 39.5% [(57-22% / 2) +22%= 39.5%]. IN ACCORDANCE WITH ONOP, operator must trip and SI when PZR level is +/- 10% from program.
B.	Correct. Part 1- correct, max charging is at 100% demand. Part 2- correct, Candidate will determine leakage is greater than charging pump capacity with letdown isolated (satisfies item 1). IAW 4-ONOP-071.2, SI is required IF any of the following limits are reached: 1) RCS Leakage greater than Charging Pump capacity AND letdown isolated or 2) PRZ Level can NOT be maintained within 10% of program
C.	Incorrect - Part 1- incorrect, lowering controller demand will minimize charging. Plausible given candidate confuses this with another controller which is reverse acting (there are direct and reverse acting controllers on the RCS part of the console). Part 2- incorrect.
D.	Incorrect – Part 1 incorrect, same as C. Part 2- correct.

Technical Reference(s)	4-ONOP-071.2, Foldout Page		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6900236 EO7		(As available)
Question Source:	Bank		
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

Changed power level in plant conditions. Changed distractor values.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *Foldout Page shall be monitored throughout this procedure.*
- *Step 1 is a Continuous Action step and should be monitored throughout the performance of this procedure in the event the Steam Generator leak worsens.*
- *ATTACHMENT 12 contains a listing of TCS Trip Setpoints*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Monitor Affected Plant Parameters**	
	a. Check PRZ level – STABLE OR INCREASING	a. Perform the following: 1) Start additional charging pumps as required. 2) Reduce letdown flow as necessary.
	b. Maintain PRZ level – MAINTAIN STABLE OR INCREASING	b. **IF** PRZ level can **NOT** be maintained, **THEN**, manually trip the reactor **AND** go to 4-EOP–E-0, REACTOR TRIP OR SAFETY INJECTION.
2	Check High Alarm ON For The Following PRMS Channels	
	a. Check R-15 High Alarm light – ON	a. Go to Step 2c.
	b. Go to Step 3	
	c. Check R-19 High Alarm light - ON	c. Go to Step 5.
	d. Go to Step 4	

FOLDOUT PAGE

1. 4-EOP-E-0 TRANSITION CRITERIA

a. **IF** RCS Tavg GREATER THAN Tref by 6°F, **THEN** Trip the Reactor and Turbine **AND** go to 4-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

b. **IF** any of the following limits are reached,

- RCS Leakage greater than Charging Pump capacity **AND** letdown isolated
- PZR Level can **NOT** be maintained within 10% of program

THEN perform the following:

1) Manually Trip the reactor **AND** perform 4-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.
2) **WHEN** power is verified to Emergency 4KV Buses, **THEN** initiate Safety Injection and Phase A.

2. CONTROL ROOM VENTILATION MANUAL ISOLATION CRITERIA

IF a reactor trip occurs **AND** any PRMS channels listed below is in the alarm state, **THEN** manually align Control Room ventilation for Emergency Recirculation mode with 30 minutes of the alarm:
[Commitment Step 3.4.1]
* R-15 Condenser Air Ejector Monitor
* R-19 Steam Generator Blowdown Monitor
* R-20 CVCS Letdown Line Radioactivity Monitor

3. TURBINE LOAD WITHIN 10% OF TARGET POWER LEVEL

WHEN turbine load is within 10% of end target load, **THEN** stop boration by performing the following:

1) Place the Reactor Makeup Selector Switch to Auto.
2) Set FC-4-113A, Boric Acid Flow Controller pot setting as desired.
3) Place the RCS Makeup Control Switch to Start

4. BLOWDOWN RELEASE PATH ISOLATION

IF PRMS R-19 Count Rate is increasing **OR** High Alarm is present, **THEN** verify the following:

a) Steam Generator Blowdown Flow Control Valves are Closed.
   - FCV-4-6278A
   - FCV-4-6278B
   - FCV-4-6278C
b) Blowdown Tank to Canal Level Control Valve, LCV-4-6265B is Closed.
c) **WHEN** R-19 High Alarm is present, **THEN** verify NO FLOW on S/G Sample Flow Indicators at the Cold Chem Lab. (Ensures Sample Valves SV-4-2800, SV-4-2801, SV-4-2802 are Closed.)

5. AFW STEAM SUPPLY RELEASE PATH ISOLATION

WHEN the affected Steam Generator is identified, **THEN** perform the following**:**

a) Verify Steam Supply aligned to both trains of AFW from the Intact Steam Generators.
b) Verify Closed **AND** De-Energize the affected Steam Generator AFW Steam Supply MOV using Attachment 4.

Facility:	WTSI Corporate		Question 10 original	
Vendor	WEC			
Exam Date:				
Exam Type:				

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

The crew is performing 4-ONOP-071.2, Steam Generator Tube Leakage.
The crew reduced power to less than 5% and has just tripped the reactor.

In accordance with 4-ONOP-071.2, which ONE of the following identifies a subsequent plant condition that requires the crew to manually initiate safety injection?

A. Pressurizer level steady at 14% with charging at maximum and letdown automatically isolated

B. Pressurizer level at 19% and decreasing with charging at maximum and letdown isolated

C. With makeup in automatic, Charging Pump suction swaps to the RWST due to low level in the VCT

D. STA performs 4-OSP-041.1, RCS Leak Rate Calculation, and reports a RCS leak rate of 150 gpm

Proposed Answer: B

Explanation (Optional):

A. Incorrect - PZR level not decreasing and within 10% of program. Plausible - maximum charging within letdown isolated and near the 10% deviation from

program level

B. Correct IAW above discussion

C. Incorrect - the maximum blended flow is approximately 150 gpm; this is less than maximum available charging. Plausible - need to realize maximum blended makeup flow and charging pump capacity

D. Incorrect - 200 gpm does not require maximum charging. Plausible - 4-OSP-041.1 directed by ONOP-071.2 and need to realize capacity of charging pumps

Techical Reference(s):
1. 4-ONOP-071.2 FOP item 1.b rev. 6/28/01
2. 4-OSP-047.1 pp. 55, 60, 65 rev. 3/10.9
3. 5610-T-D-15 sheet 1 rev. 21
SI required if S/G tube leak greater than charging pump capacity with letdown isolated or cannot maintain PZR level within 10% of program. Program level for no load is 22.2%. Acceptable range for each Charging Pump is approximately 75 gpm. 3 X 75 = 225; 225 - 9 (RCP seal return) = 216 gpm

(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: 6900236 EO7 (As available)

Question Source: Bank 11032
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2010 Turkey Point

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

10 CFR Part 55 Content: 55.41
55.43

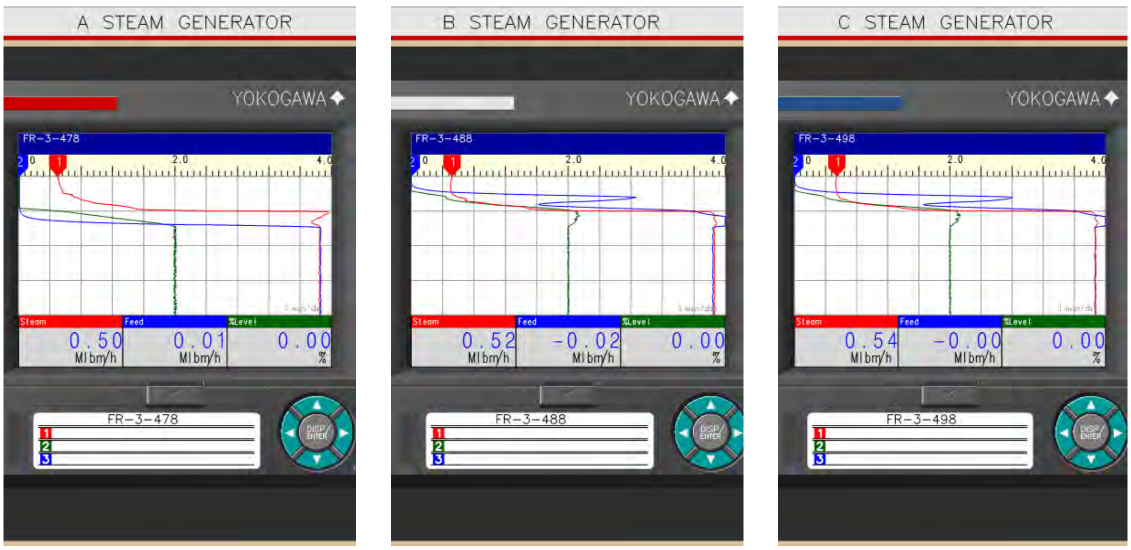
Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	054		AA2.08
	Importance Rating	2.9		

Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Steam flow-feed trend recorder

Proposed Question: RO Question # 11

Post trip steam generator trends on Unit 3 are as follows:



TRENDS ALSO PROVIDED AS REFERENCE IN LARGER FORMAT

Which one of the following identifies the initiating event?

A.	One main feed regulation valve failed closed.
B.	A feedwater isolation occurred.
C.	One steam dump to condenser failed open.
D.	A steamline isolation occurred.

Proposed Answer: A

A.	Correct. Indications are for a loss of feed water to only the 3A SG.		
B.	Incorrect, but plausible when candidate sees Feedwater flow is 0 on all recorders and determines Feedwater isolation occurred.		
C.	Incorrect, but plausible when candidate assumes SDTC valves		
D.	Incorrect, but plausible when candidate observes steam flow is minimal (steam dumps opened for Tavg control), therefore Main Steam Isolation occurred.		
Technical Reference(s)	Simulator trends		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y-DCS SG LVL trends
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	4	
	55.43		
Secondary coolant and auxiliary systems that affect the facility.			
Comments:			

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	055		2.1.19
	Importance Rating	3.9		

Conduct of Operations: Ability to use plant computers to evaluate system or component status.

Proposed Question: RO Question # 12

Given the following initial conditions:

- Unit 3 experienced a safety injection at full power.

Subsequently:

- SG NR levels are 4% and rising.
- AFW flow is 375 gpm.
- CET temperatures are 670°F and rising.
- CET subcooling is 0°F.
- Containment temperature is 160°F and rising.
- A station blackout occurs on Unit 3.
- 3-EOP-ECA-0.0, Loss of All AC Power, is entered.

Which one of the following completes the statements below?

QSPDS ___(1)___ available for monitoring.

If power is immediately restored, the crew will transition to ___(2)___ .

A.	(1) is (2) 3-EOP-FR-H.1
B.	(1) is (2) 3-EOP-FR-C.2
C.	(1) is NOT (2) 3-EOP-FR-H.1

D	(1) is NOT (2) 3-EOP-FR-C.2

Proposed Answer: A

A.	Correct. Part 1 is correct. QSPDS is available during a station blackout because it is powered from batteries/inverters. Part 2 is correct. H.1 conditions are present with <400 gpm AFW flow and <7% SG NR level.
B.	Incorrect. Part 1 is correct. Part 2 is incorrect. C.2 conditions are only met when the RCS has lost subcooling (true in this case) and CETs >700°F (not true). Plausible if the candidate forgets the C.2 CET setpoints.
C.	Incorrect. Part 1 is incorrect. Plausible because candidate may believe that QSPDS de-energizes due to loss of vital load centers during a station blackout. Part 2 is correct.
D.	Incorrect. Plausible per B and C explanations.

Technical Reference(s)	3-EOP-F-0 Drawing 5610-T-E-1592, Sheet 1	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:	N

Learning Objective:		(As available)

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	

Question History:	Last NRC Exam:		

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

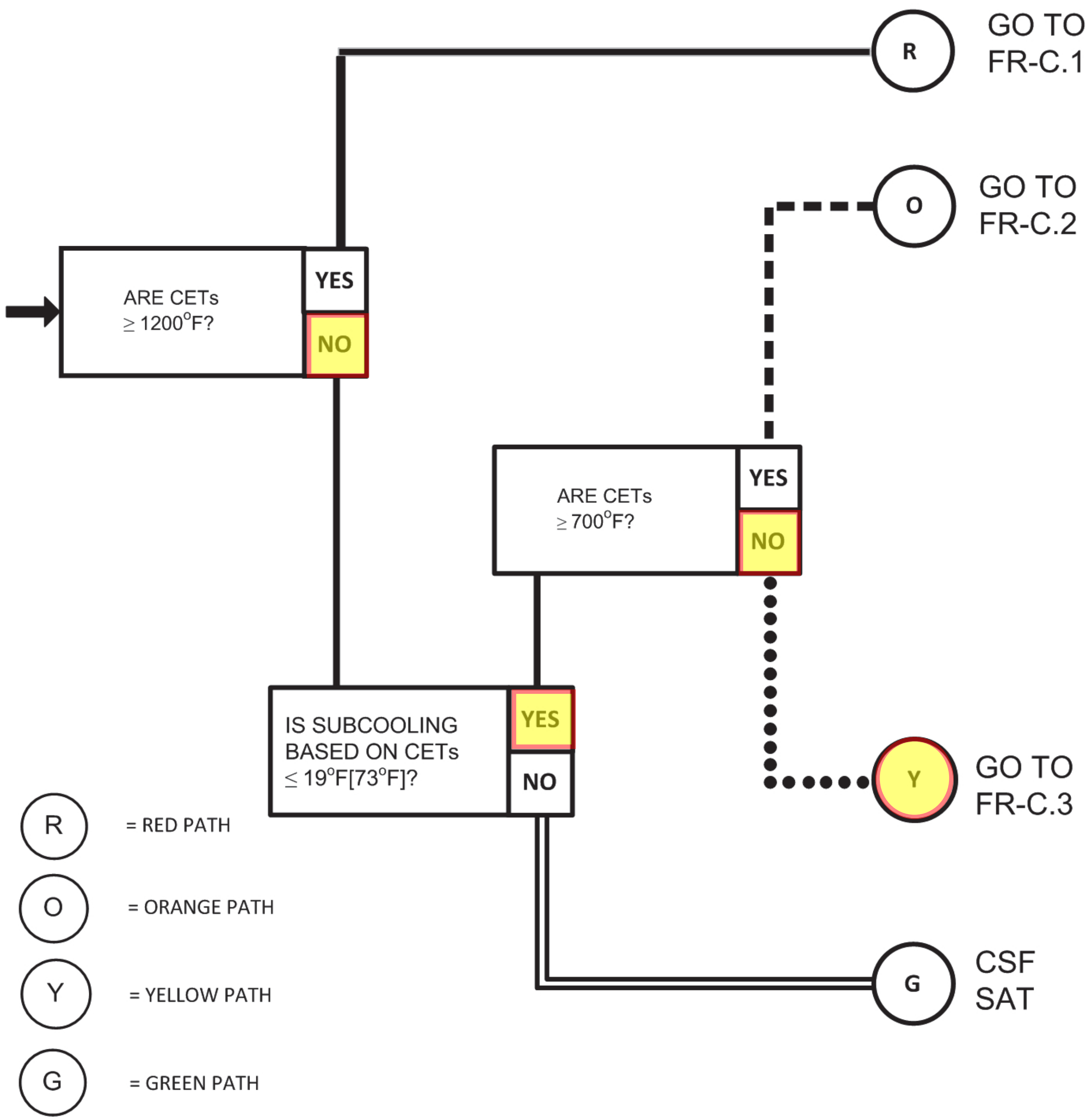
10 CFR Part 55 Content:	55.41	10
	55.43	

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

ENCLOSURE 2
<u>CSF F-0.2 Core Cooling</u>
(Page 1 of 1)

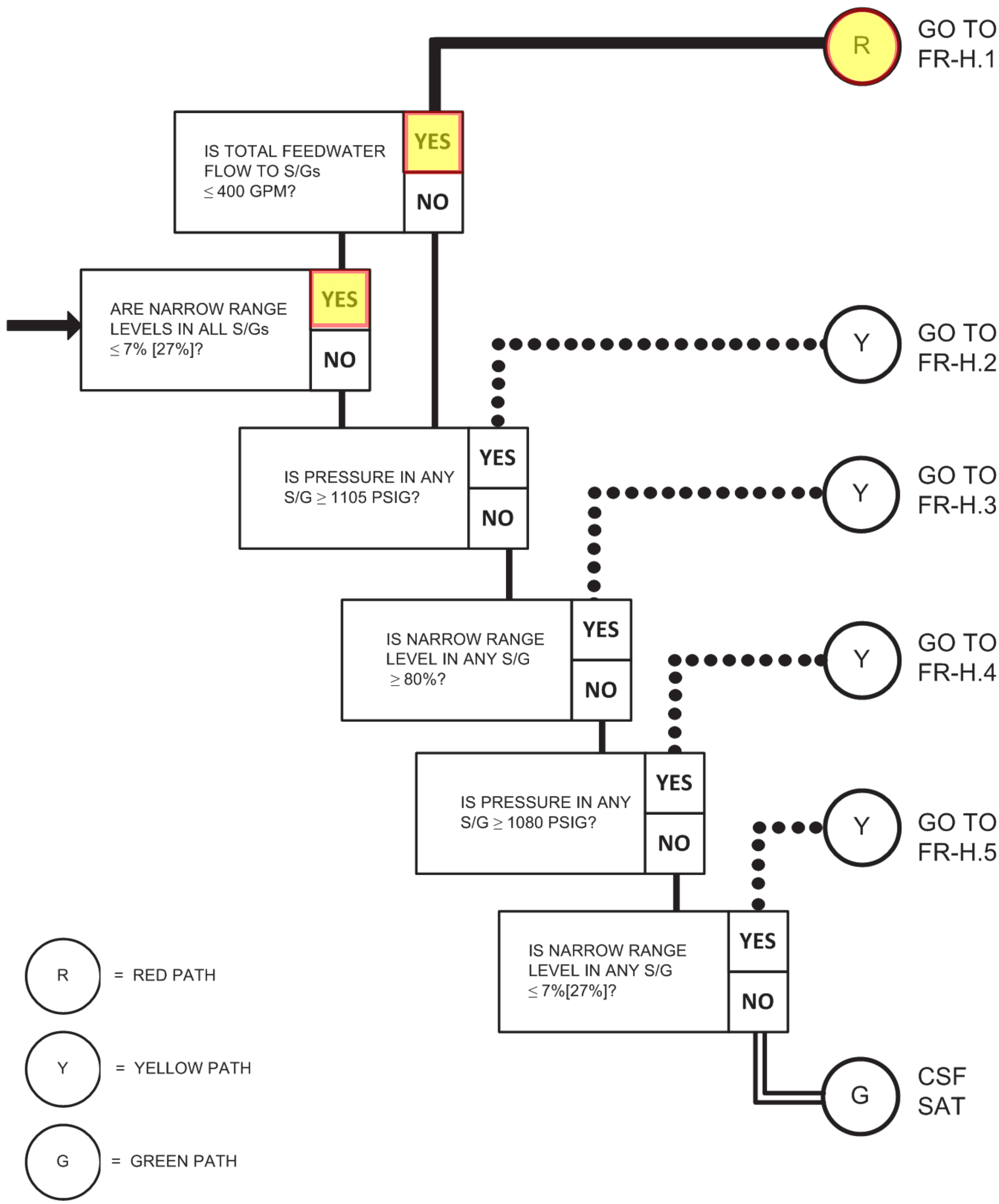
<u>NOTE</u>
Obtain core exit temperature using at least five of the hottest Core Exit Thermocouples.

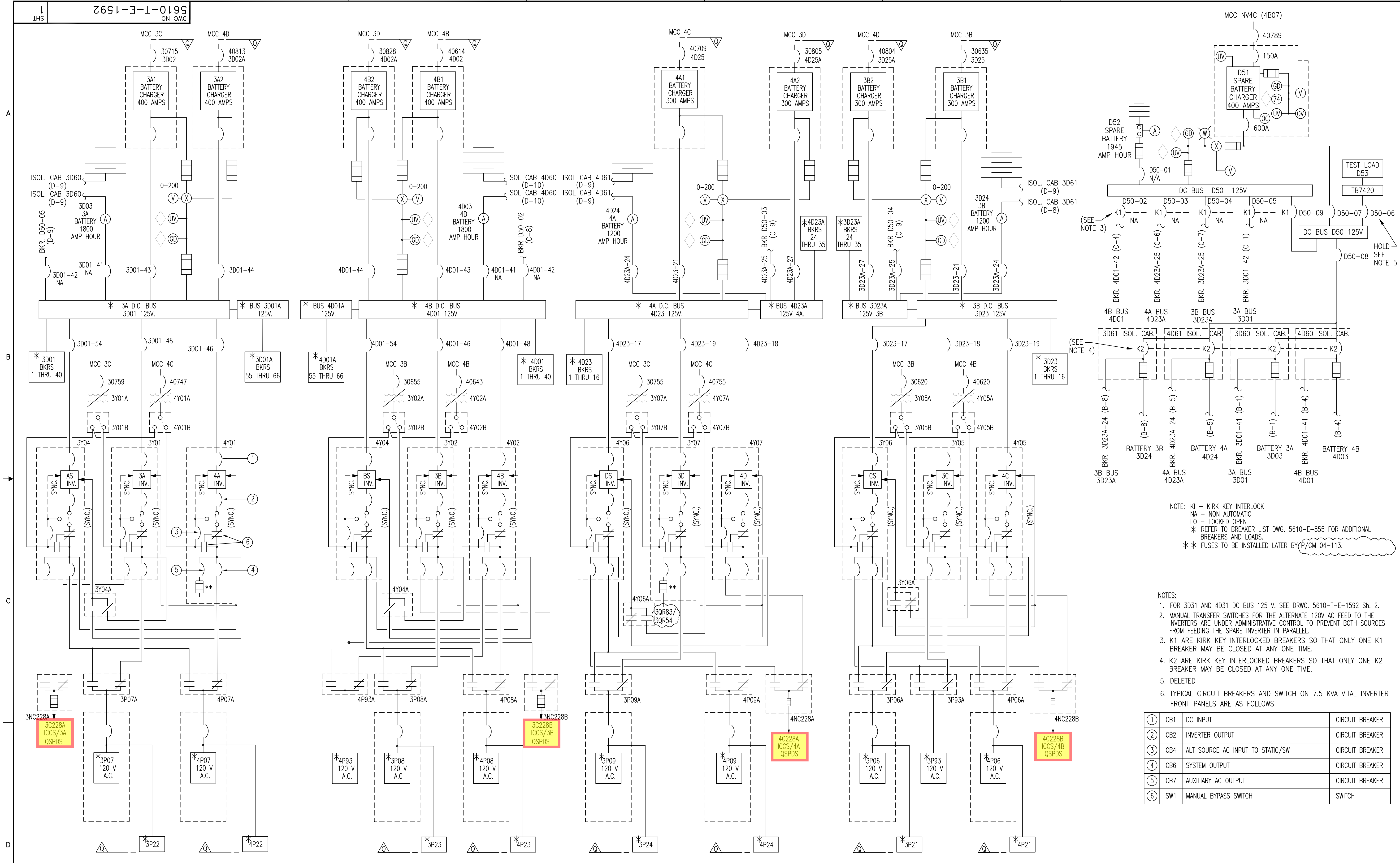


ENCLOSURE 3

CSF F-0.3 Heat Sink

(Page 1 of 1)





NOTE: KI – KIRK KEY INTERLOCK
NA – NON AUTOMATIC
LO – LOCKED OPEN
* REFER TO BREAKER LIST DWG. 5610-E-855 FOR ADDITIONAL BREAKERS AND LOADS.
** FUSES TO BE INSTALLED LATER BY P/CM 04-113.

NOTES:

1. FOR 3D31 AND 4D31 DC BUS 125 V. SEE DRWG. 5610-T-E-1592 Sh. 2.
2. MANUAL TRANSFER SWITCHES FOR THE ALTERNATE 120V AC FEED TO THE INVERTERS ARE UNDER ADMINISTRATIVE CONTROL TO PREVENT BOTH SOURCES FROM FEEDING THE SPARE INVERTER IN PARALLEL.
3. K1 ARE KIRK KEY INTERLOCKED BREAKERS SO THAT ONLY ONE K1 BREAKER MAY BE CLOSED AT ANY ONE TIME.
4. K2 ARE KIRK KEY INTERLOCKED BREAKERS SO THAT ONLY ONE K2 BREAKER MAY BE CLOSED AT ANY ONE TIME.
5. DELETED
6. TYPICAL CIRCUIT BREAKERS AND SWITCH ON 7.5 KVA VITAL INVERTER FRONT PANELS ARE AS FOLLOWS.

1	CB1	DC INPUT	CIRCUIT BREAKER
2	CB2	INVERTER OUTPUT	CIRCUIT BREAKER
3	CB4	ALT SOURCE AC INPUT TO STATIC/SW	CIRCUIT BREAKER
4	CB6	SYSTEM OUTPUT	CIRCUIT BREAKER
5	CB7	AUXILIARY AC OUTPUT	CIRCUIT BREAKER
6	SW1	MANUAL BYPASS SWITCH	SWITCH

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
45	12-18-12	ISSUED AS-BUILT PER EC 242033. (ITOP 12-12-020)	RV	BB	–	BAP	40	07-13-04	ISSUED AS-BUILT PER CRN-E-16624 (PC/M 03-106).	RH	RV	FJK	TS
44	12/22/08	ISSUED AS-BUILT PER PC/M 04-048.	RV	BB	–	PMB	39	07-31-01	ISSUED AS-BUILT PER CRN-E-16236 (PC/M 00-016).	RV	RH	MCP	TAB
43	11/24/08	ISSUED AS-BUILT PER PC/M 04-048. (PARTIAL)	RV	DAR	–	PMB	38	10-21-97	ISSUED AS-BUILT PER CRN'S E-15712 & E-15708(PC/M 97-028).	RV	RH	FJK	RB
42	7/25/07	ISSUED AS-BUILT PER PC/M 03-109.	RV	RH	–	PDS	37	4-25-94	INCORP. CRN-E-13300 FOR PC/M 93-047.	RV	RH	BAW	JCW
41	5/3/06	ISSUED AS-BUILT PER PC/M 03-135.	RV	RH	FJK	PMB	36	7/29/91	ISSUED AS-BUILT FOR PC/M's 83-06 & 87-265.	RH	LH	–	JG

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

OPERATING DIAGRAM

125V D.C. & 120V INSTRUMENT A.C. ELECTRICAL DISTRIBUTION

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-E-1592

SYS –

SHEET: 1

REV 45

5610-T-E-1592_SH 1.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	056		AK1.04
	Importance Rating	3.1		

Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Definition of saturation conditions, implication for the systems

Proposed Question: RO Question # 13

Given the following initial conditions:

- Unit 3 is operating at 100% power.
- A loss of offsite power occurs.
- The crew enters 3-EOP-ES-0.1, Reactor Trip Response.

While performing Attachment 1, Natural Circulation Indications, the following plant conditions are observed:

- CET temperatures = 595°F and stable
- RCS pressure = 1900 psig and stable
- All SG pressures = 985 psig and stable
- RCS hot leg temperatures = 585°F and stable
- RCS cold leg temperatures = 540°F and stable

Which one of the following (1) identifies the status of natural circulation and (2) the action required to enhance or establish natural circulation in accordance with 3-EOP-ES-0.1?

A.	(1) Established (2) Raise steam flow through steam dumps to condenser
B.	(1) Established (2) Raise steam flow through steam dumps to atmosphere
C.	(1) NOT established (2) Raise steam flow through steam dumps to condenser
D.	(1) NOT established (2) Raise steam flow through steam dumps to atmosphere

Proposed Answer: B

A.	Incorrect. Plausible because first part is correct; but with loss of offsite power, condenser steam dumps will not be available due to loss of circ water pumps. EOPs priority is SDTCs, then SDTAs.
B.	Correct. RCS subcooling is >19°F (630°F - 595°F = 35°F), CETs and loop T_h are stable, SG pressure is stable, and loop T_c (540°F) is within 30°F of saturation for the SGs (545°F), therefore natural circulation is occurring. SDTAs will be used, since circ water pumps are unavailable.
C.	Incorrect. Plausible because conditions are stable and not decreasing, which is an assumption the candidate might make when considering factors for natural circulation. In accordance with Attachment 1 of 3-EOP-ES-0.1, if conditions are stable, natural circulation will occur as long as the other parameters are met. Part 2 is plausible when candidate believes the SDTCs are available. EOPs priority is SDTCs, then SDTAs.
D.	Incorrect. Plausible for same reason as C, and second part is correct.

Technical Reference(s)	Steam tables 3-EOP-ES-0.1 Attachment 1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y-steam tables
Learning Objective:	6902323 obj 10		(As available)
Question Source:	Bank	13948	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2012	Sequoyah
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		14
	55.43		
Principles of heat transfer, thermodynamics and fluid mechanics.			
Comments:			
Changed answer, although conditions remained unchanged			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	**(continued)**	
		f. Check diesel capacity adequate to run <u>one</u> Computer Room Chiller <u>AND</u> at least <u>one</u> Normal Containment Cooler. <u>IF</u> adequate diesel capacity is **NOT** available, <u>THEN</u> shed **non**-essential loads. Refer to Attachment 3 for component KW load rating. **1)** Verify at least <u>one</u> Computer Room Chiller running. **2)** Reset and start at least <u>one</u> Normal Containment Cooler.
		g. Verify Battery Room Air Conditioner E16D (3D MCC) is running.
		h. <u>IF</u> 3B 4KV Bus is energized, <u>THEN</u> verify Battery Room Air Conditioner E16E (30609) is running.
		i. Verify Plant Page System (30824) is available.
9.	**Establish S/G Pressure Control Using <u>Either</u> Method Below:** * Set S/G Steam Dump To Atmosphere Valve controllers to maintain desired S/G pressure * Set Steam Dump To Condenser to maintain desired S/G pressure	
	1) Check Condenser – AVAILABLE	**1)** Use Steam Dump To Atmosphere Valves.
	2) Align Condenser Steam Dumps using Attachment 9.	

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

CAUTION

CCW System load requirements of 3-NOP-030, COMPONENT COOLING WATER SYSTEM, shall **NOT** be exceeded.

NOTE

RCPs should be run in order of priority (3B then 3C) to provide normal PRZ Spray.

10. Check 3B RCP – RUNNING

Perform one of the following:

* IF **NO** RCPs are running, THEN go to Step 10.b.
* IF any RCPs are running, THEN go to Step 10.t.

a. Go to Step 11

b. Check Startup Transformer – ENERGIZED

b. Perform the following:

1) Verify Natural Circulation using Attachment 1.

2) IF Natural Circulation can **NOT** be verified, THEN dump more steam.

3) Go to Step 11.

c. Establish RCP support conditions using Attachment 11

d. Check Auxiliary Spray – **NOT** IN SERVICE

d. Perform the following:

1) Terminate Auxiliary Spray using Attachment 7.

2) Close PCV-3-455B, Pressurizer Spray Loop B.

3) Close PCV-3-455A, Pressurizer Spray Loop C.

e. Check 3B RCP – SUPPORT CONDITIONS ESTABLISHED

e. Go to Step 10.j.

ATTACHMENT 1

Natural Circulation Indications

(Page 1 of 1)

The following conditions support or indicate Natural Circulation flow:

- RCS Subcooling based on Core Exit TCs – GREATER THAN 19°F
- S/G pressures – STABLE OR DECREASING
- RCS Hot Leg temperatures – STABLE OR DECREASING
- Core Exit TCs – STABLE OR DECREASING
- RCS Cold Leg temperatures – WITHIN 30°F OF SATURATION TEMPERATURE FOR S/G PRESSURE

End of Attachment 1

Question 13 original

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following plant conditions,

- Unit #1 is operating at 100% power
- The Turbine Driven AFW pump is removed from service for maintenance.
- A loss of Offsite power occurs on Unit #1.
- The crew has entered ES-0.1, 3Reactor Trip Response.4

While performing EA-68-6, 3Monitoring Natural Circulation Conditions,4 the following plant conditions are observed:

- Highest Core Exit Thermocouple = 595F and stable
- RCS pressure = 1900 psig and stable
- All SIG pressures = 1040 psig and stable
- RCS hot leg temperatures (all loops) = 585F and stable
- RCS cold leg temperatures = 540F and stable

Which ONE of the following indentifies the status of Natural Circulation and any action that would be required to be taken in accordance with ES-0.1?

	Natural Circulation	Action
A.	Established	Raise steam flow through Steam Dumps
B.	Established	Raise steam flow through SG PORV5
C.	NOT Established	Raise steam flow through SG PORVs
D.	NOT Established	Raise steam flow through Steam Dumps

Proposed Answer: C

Explanation (Optional):

A. Incorrect.

B. Incorrect.

C. Correct.

D. Incorrect.

Techical Reference(s): - (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: - (As available)

Question Source: Bank 13948

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2012 Sequoyah

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	057		AK3.01
	Importance Rating	4.1		

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

Proposed Question: RO Question # 14

Given the following conditions:

- Unit 3 experiences a LOOP-LOCA.
- 3P07, Vital Instrument AC bus, de-energizes.

Which one of the following identifies why safeguards equipment must be manually started in accordance with Attachment 3, Prompt Action Verifications of 3-EOP-E-0?

A.	3A train Emergency Sequencer loses power.
B.	3A train Safety Injection bistables fail to actuate.
C.	3B train Emergency Sequencer loses power.
D.	3B train Safety Injection bistables fail to actuate.

Proposed Answer: A

A.	Correct. 3A train sequencer loses power. See references.
B.	Incorrect, but plausible if candidate believes bistables go dim on loss of instrument power as is the case for various p-panel failures. Candidate may believe that bistables will fail to actuate for 3A train.
C.	Incorrect, but plausible if candidate believes the sequencer is effected which is true however the 3B train is not effected. Candidate may chose B train erroneously due to confusing procedure response / system knowledge- loss of 3P08 causes 3B sequencer failure.

D.	Incorrect, but plausible if candidate believes bistables go dim on loss of instrument power as is the case for various p-panel failures. Candidate may believe that bistables will fail to actuate for 3B train which would also be the wrong train- loss of 3P08 causes 3B sequencer failure.		
Technical Reference(s)	3-EOP-E-0, 5610-T-E-1592 sht 1 3-ONOP-003.7, Step 3 and NOTE prior to Step 3 BD-ONOP-003.7, Step 3.a		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	10439	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2009	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

ENCLOSURE 1
(Page 1 of 5)

CONTROL ROOM FUNCTIONS AND INDICATIONS
LOST ON LOSS OF 3P06

FUNCTIONS, Operating

Lock up of Pressurizer Pressure Controllers causing spray valves to stay as is
FCV-3-478, A Feedwater Control Valve - On Backup Controller
Lose Auto and Manual 3A Charging Pump Control causing Auto Lock-up
Lose Auto Speed Control of 3B and 3C Charging Pumps
Lose the Auto Makeup Control to the Volume Control Tank
Lose power to Control Relay from MOV-3-115C which opens LCV-3-115B
Letdown Isolation
Pressurizer heaters de-energize
Lose Auto and Manual control of PCV-3-145, Letdown Pressure Controller
Loss of 3B Diesel Load Sequencer, 3C23B-1 deenergized
Lose AMSAC A Processor
Lose the Ability to Block the Source Range Trip
Lose Feedwater Isolation signal (Reactor Trip with Tavg ≤554°F)
Loss of power to hand/auto station for CV-3-1607 which fails closed

NOTES

- *The following conditions exist which affect Pressurizer Pressure control:*
  - *Pressurizer Pressure Controller PC-444J - AUTO LOCKUP*
  - *PZR Spray Valve Controllers - AUTO LOCKUP*
  - *PZR heaters deenergized*
  - *Letdown isolation*
  - *3A charging pump - AUTO LOCKUP*
  - *3B AND 3C Charging pump loss of auto speed control*

ENCLOSURE 1
(Page 2 of 5)

CONTROL ROOM FUNCTIONS AND INDICATIONS
LOST ON LOSS OF 3P06

NOTES

- *With vital panel 3P06 deenergized, 3B bus sequencer is out of service resulting in the following Tech Spec implications:*

  1. *AFW actuation from bus stripping on 3B 4KV bus will NOT be generated, placing the unit in a shutdown action statement (Tech Spec 3.3.2, Table 3.3-2, functional unit 6.d action 23 invokes Tech Spec 3.0.3.)*
  2. *Loss of Power signals are lost via the 3B bus sequencer, placing the unit in a shutdown action statement (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 7a, b and c)*
  3. *Bus stripping will NOT automatically occur, 3B EDG will NOT automatically close in on the bus and is out service; actions of Tech Spec 3.8.1.1 apply.*

INDICATORS

TI-3-401	RX Vessel Leak of Temp
TI-3-133	Seal Water Return Temp
TI-3-139	Excess LTDN HX Temp
PI-3-121	Charging Pumps Disch Press
TI-3-123	Regen Hx Outlet Temp
TI-3-141	LTDN Relief To PRT Temp
TI-3-143	Non-Regen HX LTDN Temp
FI-3-150	Low Pressure Letdown Flow Indication
FR-3-154A	RCP CBO Flow Recorder
PR-3-154B	RCP P3 Pressure Recorder
PI-3-156B	3A RCP P2 Pressure
PI-3-155B	3B RCP P2 Pressure
PI-3-154B	3C RCP P2 Pressure
PI-3-156C	3A RCP P3 Pressure
PI-3-155C	3B RCP P3 Pressure
PI-3-154C	3C RCP P3 Pressure
PI-3-128A	B RCP Thermal Barrier ΔP
PI-3-402	RCS Press NR
PI-3-403	RCS Press WR
TI-3-465	Pzr Safety Valve Temp
TI-3-467	Pzr Safety Valve Temp
TI-3-469	Pzr Safety Valve Temp
TI-3-463	PZR Relief Temp
TI-3-452	PZR Spray Loop B Temp
TI-3-451	PZR Spray Loop C Temp
TI-3-412B	A Loop Ovpwr ΔT
TI-3-412A	A Loop ΔT
TI-3-412C	A Loop Ovtemp ΔT

ENCLOSURE 1
(Page 1 of 3)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON FAILURE OF VITAL INSTRUMENT PANEL 3P07

The following controls/functions are affected and the applicable controls should be returned to auto when directed by procedure:

- Loss of Auto Control of 3B Feedwater Control Valve, FCV-3-488
- 3B Charging Pump Controller Locks Up as is
- Auto VCT makeup will occur due to LT-3-115 failure.
- Failure Closed of Train 1 AFW Flow Control Valves: (CV-3-2816, 2817, 2818)
- HCV-3-121 fails full open
- Loss of Diesel 3A Load Sequencer, 3C23A-1 deenergized
- QSPDS Channel A (If 3A inverter and CVT are lost)
- Lose AMSAC Processor B
- ANN B 9/2
- ANN B 9/3
- Loss of power to hand/auto station for CV-3-1606 which fails closed
- Loss of primary power to Rod Deviation/Axial Flux Mon Rack, 3QR64. (Redundant power is supplied from Inverter 3Y111, Panel 3P31A, Ckt. 9.)
- Primary water flow controller shifts to manual (FC-3-114A)
- Boric Acid flow controller shifts to manual (FC-3-113A)

NOTES

3A bus sequencer is out of service, due to Vital Panel 3P07 deenergized, resulting in the following Tech Spec implications:

1) *AFW actuation signals from bus stripping on 3A 4KV bus will* ***NOT*** *be generated, placing the unit in Tech Spec 3.0.3 (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 6d action 23 invokes Tech Spec 3.0.3.)*

2) *Loss of Power signals are lost via the 3A bus sequencer, placing the unit in Tech Spec 3.0.3 (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 7a, b, and c.)*

3) *Bus stripping will* ***NOT*** *automatically occur, 3A EDG will* ***NOT*** *automatically close in on the bus and is out of service (actions of Tech Spec 3.8.1.1 apply).*

INDICATORS

LI-3-115	VCT Level
LI-3-106	A Boric Acid Tank Level
PI-3-155	B RCP P2 Seal Pressure
FR-3-154A	B RCP CBO Flow Recorder (Blue Pen)
PI-3-125A	C RCP Thermal Barrier ΔP
PR-3-154B	B RCP P3 Seal Pressure Recorder (Blue Pen)
TI-3-453	Pzr Liquid Temp
TI-3-450	Pzr Surge Line Temp
TI-3-454	Pzr Vapor Temp

Facility: WTSI Corporate

Question 14 original

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Unit 3 is at 50% power.

- The Pressurizer Level Control Transfer Switch on VPA is in Position 1.
- Power is lost to 120V Vital Instrument Panel 3P07.

Which ONE of the following identifies the required operator action in accordance with 3-ONOP-003.7, Loss of 120V Vital Instrument Panel 3P07, and the reasons for that action?

Place the Pressurizer Level Control Transfer Switch in:

A. Position 2 to allow letdown to be reestablished and pressurizer heaters to be reenergized.

B. Position 2 to allow letdown to be reestablished and to allow automatic charging pump operation

C. Position 3 to allow letdown to be reestablished and pressurizer heaters to be reenergized

D. Position 3 to allow letdown to be reestablished and to allow automatic charging pump operation

Proposed Answer: A

Explanation (Optional):

A. Correct per the references and discussion above. Letdown cannot be re-established and pressurizer heaters cannot be energized until the transfer switch is placed in Position 2

B. Incorrect because a loss of 3PO7 will not affect charging pump operation because LT-460 does not control charging pumps. Plausible because the transfer switch should be placed in Position 2 allowing letdown to be reestablished

C. Incorrect because placing the transfer switch in Position 3 will not eliminate the failed LT-460 and a continuous signal still exists to isolate letdown and to deenergize pressurizer heaters. Plausible if the operator does not know that Position 3 does not eliminate the failed channel. Note that this would be the correct action if the failure had been a loss of 3P08

D. Incorrect because placing the transfer switch in Position 3 will not eliminate the failed LT-460 and a continuous signal still exists to isolate letdown. Plausible if the operator does not know that Position 3 does not eliminate the failed channel. Note that this would be the correct action if the failure had been a loss of 3P08

Techical Reference(s): 3-ONOP-003.7, Step 3 and NOTE prior to Step 3
BD-ONOP-003.7, Step 3.a
5610-T-D-15, Sheet 1
(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: (As available)

Question Source: Bank 10439
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2009 Turkey Point

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Level 2 because the operator must recall that 3P07 powers Channel II instruments. With the transfer switch in Position 1, a loss of 3P07 results in pressurizer level transmitter, LT-460, failed low. This causes a continuous trip signal to the pressurizer heaters and a continuous close signal to the letdown isolation valves. These signals are removed when the transfer switch is placed in Position 2, eliminating the failed channel. Note that LT-460 (unlike LT-459 and 461) is never an input to charging pump control so there is no effect on charging pump speed

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	062		AK3.02
	Importance Rating	3.6		

Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS

Proposed Question: RO Question # 15

Which one of the following (1) identifies the signal that automatically closes ICW/TPCW Isolation Valves POV-3-4882 and POV-3-4883, and (2) the reason for the automatic closure?

A.	(1) High ICW flow (2) Prevents ICW pump runout
B.	(1) High ICW flow (2) Ensures ICW is dedicated to CCW
C.	(1) Safety Injection (2) Prevents ICW pump runout
D.	(1) Safety Injection (2) Ensures ICW is dedicated to CCW

Proposed Answer: D

A.	Incorrect. Plausible because some valves at PTN close on high flow for system protection (e.g. pump run-out / integrity). Candidate would chose this combination thinking an excess flow condition would close the POVs.
B.	Incorrect. Same reason as Option A. Part 2 is correct. Plausible to protect CCW components.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect. This combination is plausible if candidate believes the reason the POVs close on SI is for pump runout given the sequencer fails (1 active failure) and only one ICW pump is left running at runout conditions.

D.	Correct. POV-3-4882/4883 close on SI to ensure ICW is dedicated to CCW.		
Technical Reference(s)	3-NOP-019 3-ONOP-019 BD-ONOP-019 LP 6902154		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902154 obj 7		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

BASIS DOCUMENT

7. This step ensures proper lineup of ICW to the turbine plant cooling water heat exchangers. Normally both ICW to TPCW Heat Exchanger valves are open. If SI is in progress, ICW to TPCW is isolated to ensure adequate ICW flow to the CCW System. In this case both ICW to TPCW Heat Exchanger valves are verified closed and the operator skips the steps associated with verifying proper cooling to TPCW.

_C A U T I O N_

This caution stresses the importance of restoring ICW flow to the TPCW heat exchangers.

8. This step verifies proper cooling for the TPCW system. Opening the ICW valves for the TPCW heat exchanger may be necessary to compensate for leakage or blockage in the ICW System.

_C A U T I O N_

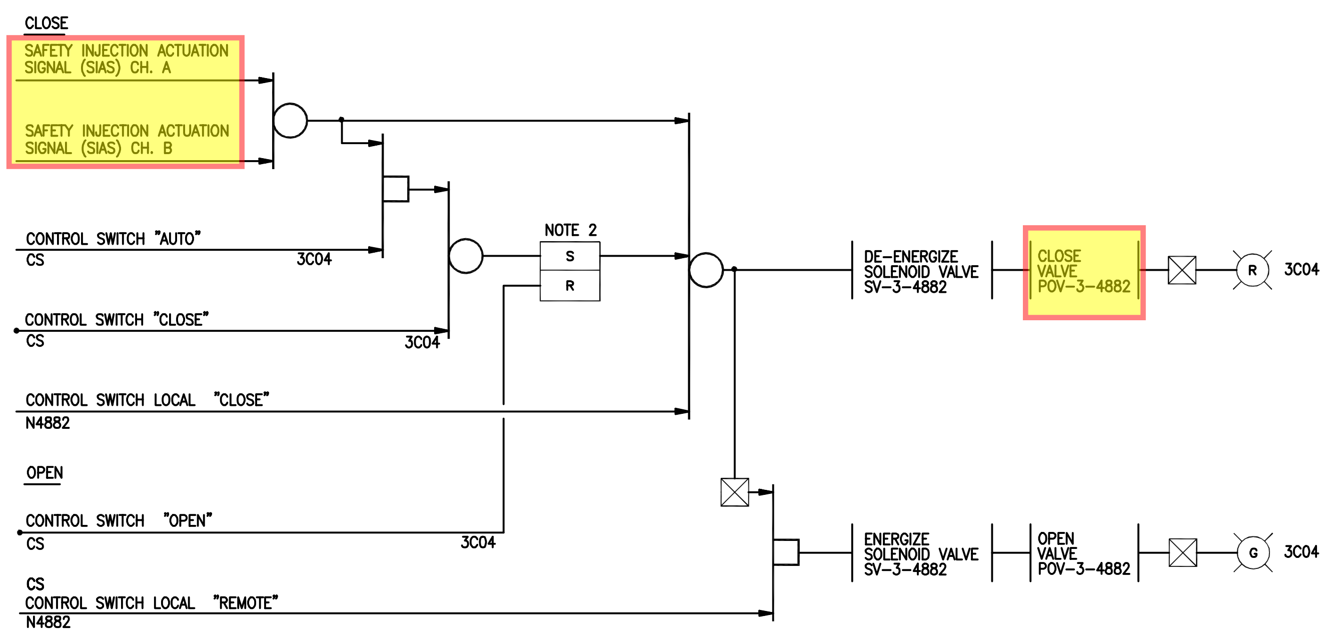
This caution stresses the importance of restoring ICW flow to the CCW heat exchangers.

9. This step verifies proper cooling for the CCW system. Opening the ICW valves for the CCW heat exchanger may be necessary to compensate for leakage or blockage in the ICW System.

10. Previous steps should have identified and corrected any problems with supplying cooling for the TPCW System. If TPCW temperatures can not be maintained, the unit must be shutdown and components cooled by TPCW should be stopped.

11. Previous steps should have identified and corrected any problems with supplying cooling for the CCW System. If CCW temperatures can not be maintained, the loss of CCW procedure should be used to address the inability to cool components supplied by the CCW System.

DWG NO 5610-T-L1 136C



CONTROL SWITCH
KEYLOCK-KEY
REMOVABLE IN
REMOTE POSITION
MAINTAINED CONTACTS
TAG: N4882

CONTROL SWITCH
SPRING RETURN
TO AUTO
TAG: CS

NOTES:
1. REFER TO 5610-T-L1 SH.1 FOR STANDARD SYMBOLS.
2. A RESET SIGNAL WILL REMOVE THE OUTPUT ONLY IF THE INPUT IS NO LONGER PRESENT.

REFERENCE DRAWINGS:

5610-M-4/88-345	P & ID
5613-E-26 SH.29A/88-345	ELEMENTARY
5610-T-L1 SH.12A	SEQUENCER
5610-M-311	INSTRUMENT INDEX
5610-T-E-4065 SH.2	OPERATING DIAGRAM

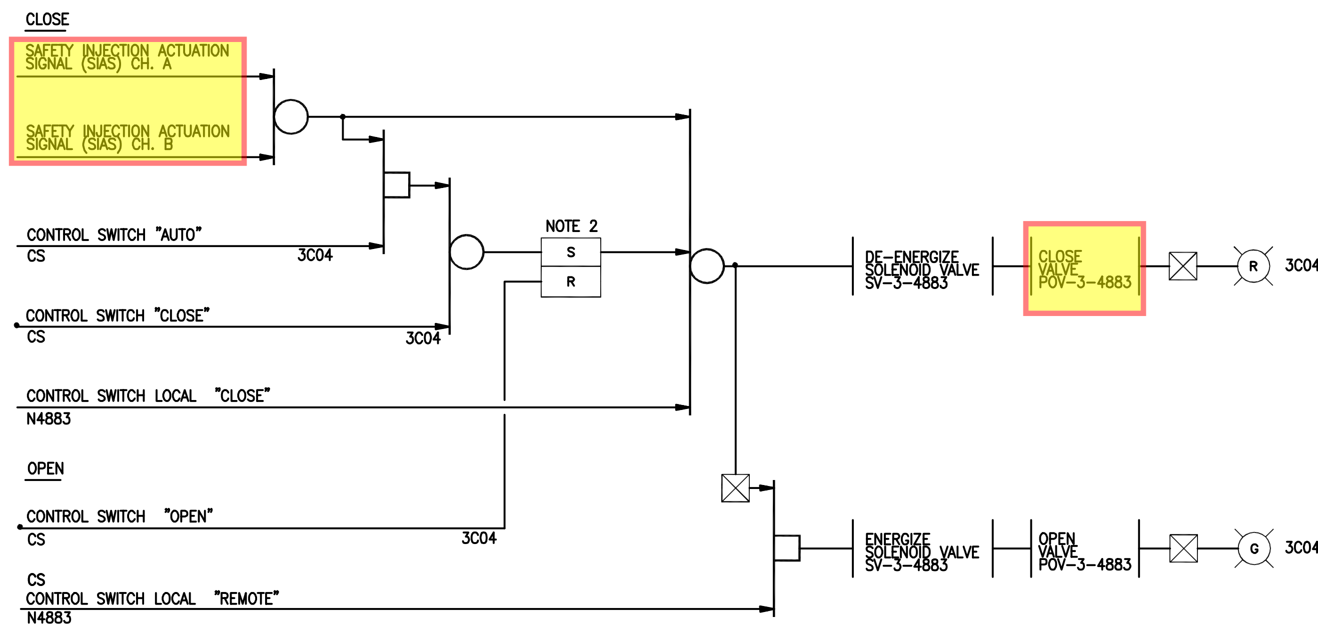
REV	DATE	REVISION	BY	CH	APP	APP
1	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	GM	RAS	JCW

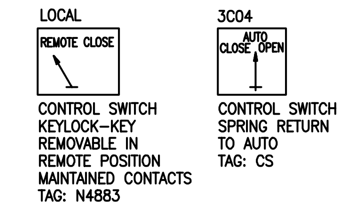
FPL

TURKEY POINT NUCLEAR UNIT 3
LOGIC DIAGRAM
TPCW/ICW ISOLATION VALVE
POV-3-4882
CHANNEL A

POD
DRAWING NUMBER 5610-T-L1
SYS 019
REV 1
SHEET 136C

136D 5610-T-L1 DWG NO





NOTES:

1. REFER TO 5610-T-L1 SH.1 FOR STANDARD SYMBOLS.
2. A RESET SIGNAL WILL REMOVE THE OUTPUT ONLY IF THE INPUT IS NO LONGER PRESENT.

REFERENCE DRAWINGS:

5610-M-4/88-345	P & ID
5613-E-26 SH.29B/88-345	ELEMENTARY
5610-T-L1 SH.12A	SEQUENCER
5610-M-311	INSTRUMENT INDEX
5610-T-E-4065 SH.2	OPERATING DIAGRAM

REV	DATE	REVISION	BY	CH	APP	APP
1	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	GM	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNIT 3

LOGIC DIAGRAM

TPCW/ICW ISOLATION VALVE POV-3-4883 CHANNEL B

POD

DRAWING NUMBER 5610-T-L1

SYS 19

REV 1

SHEET 136D

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	065		AA2.05
	Importance Rating	3.4		

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air pressure is decreasing

Proposed Question: RO Question # 16

Given the following conditions:

- Unit 3 is at 100% power.
- PRZ level is 60%.
- A loss of instrument air is in progress.
- The crew has entered 3-ONOP-013, Loss of Instrument Air.
- Instrument Air header pressure on PI-3-1444 (VPA) is 65 psig and lowering slowly.
- Pressure drop across the operating air dryer is 8 psid.
- All available Instrument Air Compressors are running.

Which one of the following completes the statements below?

In accordance with 3-ONOP-013, the crew is required to ___(1)___ .

Once the plant is shutdown, PRZ level band is maintained by cycling ___(1)___ as necessary.

A.	(1) perform a fast load reduction (2) PRZ heaters
B.	(1) perform a fast load reduction (2) charging pumps
C.	(1) trip the plant (2) PRZ heaters
D.	(1) trip the plant (2) charging pumps

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible when candidate assumes a load reduction would be called for as it is other procedures (e.g. 3-ONOP-071.2, 3-ONOP-041.4, 3-ONOP-028.3…). Part 2 is incorrect, but plausible when candidate confuses guidance in 3-ONOP-013 with guidance from 3-EOP-E-3 for example. In 3-EOP-E-3, PRZ heaters are cycled as needed to raise / lower PRZ level. Also plausible, due to recent plant OE and recent PCR to ONOP which added the guidance to cycle charging pumps to maintain PRZ level.		
B.	Incorrect. Part 1 is incorrect, but plausible per discussion above. Part 2 is correct.		
C.	Incorrect. Part 1 is correct. Part 2 is incorrect per discussion above.		
D.	Correct. Less than 75 psig on header, letdown throttles closed and charging pumps are going to max speed causing PRZ level to rise. Recent plant OE.		

Technical Reference(s)	3-ONOP-013		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**CHECK** instrument air pressure is greater than 65 psig on PI-3-1444, INSTRUMENT AIR PRESSURE, on VPA**.**	**TRIP** Unit 3 AND **ENTER** 3-EOP-E-0, Reactor Trip or Safety Injection, while continuing with efforts to restore instrument air pressure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.2 Subsequent Operator Actions

NOTE

- Foldout Pages shall be monitored throughout this procedure.
- Procedure use shall be continued until loss of Instrument Air is corrected and returned to normal configuration per 0-NOP-013, Instrument Air System, to facilitate recovery actions following the event.

CAUTION

Due to the close proximity of the automatic crosstie valves, instrument air pressure low enough to cause the auto closure of CV-3-1605, will normally CLOSE CV-4-1605 also.

1. **DISPATCH** Operator to check at least one cross tie valve CLOSED:

A. CV-3-1605, UNIT 3 INSTRUMENT AIR CROSSTIE ISOLATION CONTROL VALVE, is CLOSED,

OR

B. CV-4-1605, UNIT 4 INSTRUMENT AIR CROSSTIE ISOLATION CONTROL VALVE, is CLOSED.

Response Not Obtained:

IF IA pressure on Unit 3 is less than 80 psig, THEN **DIRECT** Unit 3 Turbine Operator to isolate Instrument Air Headers:

- **CLOSE** 3-40-308, INSTRUMENT AIR CROSSTIE HEADER UNIT 3 ISOLATION VALVE, to isolate CV-3-1605,

  OR

- IF IA pressure on Unit 4 is less than 80 psig, THEN **CLOSE** 4-40-408, INSTRUMENT AIR CROSSTIE HEADER UNIT 4 ISOLATION VALVE, to isolate CV-4-1605.

FOLDOUT PAGE
For Procedure 3-ONOP-013

1.0 UNIT TRIP CRITERIA

1. IF Instrument Air System is less than 65 psig, THEN **TRIP** Unit 3 AND **ENTER** 3-EOP-E-0, Reactor Trip or Safety Injection, while continuing efforts to restore IA pressure:

2.0 PLANT STABILIZATION

1. IF PZR LEVEL can **NOT** be maintained due to loss of IA, THEN **MAINTAIN** PZR level 22-50% by performing the following: (Section 5.1.3, Management Directive 1.A)

   A. **ENSURE** OPEN MOV-3-626, RCP THERMAL BARRIER CCW OUTLET.

   B. **ENSURE** HCV-3-121 is FULL OPEN.

   C. **START** AND **STOP** Charging Pump(s), as necessary.

   D. **ADJUST** HCV-3-121, CHARGING FLOW TO REGEN HX CONTROL VALVE, to maintain Positive Thermal Barrier Pressure.

2. IF Pressurizer OR RCS pressure can **NOT** be maintained due to loss of IA, THEN **USE** Pressurizer Heaters, as necessary, to maintain RCS pressure less than 2335 psig. (Section 5.1.3, Management Directive 1.B)

3. IF AFW is in operation, THEN:

   A. **OPERATE** Unit 3 Train 2 FCV in MANUAL.

   B. **OPERATE** Unit 4 Train 1 FCV in MANUAL.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	E05		EK2.2
	Importance Rating	3.9		

Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: Facility 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.

Proposed Question: RO Question # 17

Given the following conditions:

- The crew enters 3-EOP-FR-H.1, Loss of Secondary Heat Sink.
- Bleed and feed criteria is met.

Which one of the following describes (1) the MINIMUM number of pressurizer PORV(s) opened in order to establish bleed and feed and (2) the reason for taking this action?

A.	(1) One (2) Prevents RCS pressure from rising to the pressurizer safety valve setpoint, leading to further loss of coolant inventory
B.	(1) One (2) Allows SI flow to inject when the RCS is depressurized, leading to core decay heat removal
C.	(1) Two (2) Prevents RCS pressure from rising to the pressurizer safety valve setpoint, leading to further loss of coolant inventory
D.	(1) Two (2) Allows SI flow to inject when the RCS is depressurized, leading to core decay heat removal

Proposed Answer: D

A.	Incorrect but plausible. Having only one PORV open would reduce the ability to depressurize the RCS to ensure adequate SI flow however if one PORV is opened RCS pressure would decrease and not continue to rise. Candidate may believe this depress should be performed slowly/controlled by opening one PORV at a time as is done in ¾-EOP-E-3, SGTR.
B.	Incorrect but plausible; Second part reason is correct for 2 PORVs
C.	Incorrect but plausible because both PORVs are used, and because it is logical that there would be a concern about a pressure rise when energy isn't being removed from the system. Also because the pressure reduction is what ensures adequate SI flow
D.	Correct. Per the Westinghouse Background Document for FR-H.1 If both PRZ PORVs are not maintained open, the RCS may not depressurize sufficiently to permit adequate feed of subcooled SI flow to remove core decay heat. If core decay heat exceeds RCS bleed and feed heat removal capability the RCS will re-pressurize rapidly, further reducing the feed of subcooled SI flow and resulting in a rapid decrease in RCS inventory.

Technical Reference(s)	FR-H.1, Response to Loss of Secondary Heat Sink.		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Seabrook
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			
Changed context and wording. Changed 3 distractors and made 2X2.			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Foldout page is required to be monitored throughout this procedure.

➔ **2. Check If Bleed And Feed Is Required**

a.	Two S/G Wide Range Levels – LESS THAN 10% [Narrow Range Level in all S/Gs – LESS THAN 27%]	**a.** Observe CAUTION prior to Step 3, and go to Step 3.
b.	Stop all RCPs	
c.	Observe CAUTION prior to Step 13, and go to Step 13	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.	**(continued)**	
	h. Establish low pressure feedwater flow as directed by TSC staff	**h.** IF low pressure feedwater flow can **NOT** be established to any S/G, THEN go to Step 12.
	i. Control Steam Dump and Feed Flow to maintain Core Exit TCs stable between 420°F and 547°F	**i.** IF core exit TCs can **NOT** be maintained below 547°F, THEN observe CAUTION prior to Step 13, and go to Step 13.
12.	**Check For Loss Of Secondary Heat Sink**	Observe CAUTIONS prior to Step 1 and return to Step 1.
	a. Wide Range S/G level in two S/Gs – LESS THAN 10% [Narrow Range Level in all S/Gs – LESS THAN 27%]	

CAUTION

Step 13 through Step 17 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

13. Actuate SI And Containment Isolation Phase A

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

CAUTION

If SI is reset AND either offsite power is lost OR SI actuation occurs on the other unit, manual action may be required to restore safeguards equipment and at least one Computer Room Chiller to the required configuration.

14. Verify RCS Feed Path

a. Establish maximum Charging flow

1) Check power supply to all Charging pumps – ALIGNED TO OFFSITE POWER

Response Not Obtained:

1) Check diesel capacity adequate to run three Charging pumps.

IF adequate capacity **NOT** available, THEN shed non-essential loads.

Refer to Attachment 2 for component KW load rating.

2) Check status of Charging pumps prior to SI actuation in Step 13 – ANY RUNNING

Response Not Obtained:

2) IF CCW flow to RCP(s) Thermal Barrier is lost, THEN perform the following:

a) Locally isolate Seal Injection to affected RCP(s) before starting Charging pumps:

* 3-297A for RCP A
* 3-297B for RCP B
* 3-297C for RCP C

b) WHEN Seal Injection is isolated, THEN continue with Step 14.a.3).

3) Reset SI

4) Start all available Charging pumps

Response Not Obtained:

4) IF **NO** Charging pumps can be started, THEN continue attempts to start Charging pumps.

Observe CAUTION prior to Step 3, and return to Step 3.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. a.	**(continued)**	
	5) Adjust Charging pump speed controllers to establish maximum charging flow	**5)** IF Charging flow can **NOT** be established, THEN continue attempts to establish Charging flow. Observe CAUTION prior to Step 3, and return to Step 3.
	6) Adjust HCV-3-121, Charging Flow To Regen Heat Exchanger, to maintain proper Seal Injection flow	
	7) Place RCS Makeup Control Switch in STOP	
	8) Check Charging Pump Suction – ALIGNED TO RWST	**8)** Verify charging pump suction auto transfers to RWST while continuing with Step 14.b.
b.	Check SI pumps status – AT LEAST TWO RUNNING	**b.** Perform the following: **1)** Manually start SI pumps as necessary. **2)** IF **NO** High-Head SI pumps running, THEN continue attempts to start High-Head SI pumps. Observe CAUTION prior to Step 3, and return to Step 3

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

14. (continued)

c. Verify SI valve amber lights on VPB – ALL BRIGHT

c. Perform the following:

1) Align valves as necessary to establish RCS feed path.

2) IF at least one High-Head SI pump is running with RCS feed path established, THEN go to Step 15.

3) IF **NO** RCS feed path established, THEN continue attempts to align valves.

Observe CAUTION prior to Step 3, and return to Step 3.

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

CAUTION

If Low PRZ Pressure SI signal is **NOT** blocked prior to PRZ pressure decreasing below 1730 psig, Charging Pumps started in previous step will trip.

NOTE

- PRZ pressure must be less than 1987 psig for permissive to block the Low PRZ Pressure SI signal.
- Step 15 should be reviewed in advance to ensure timely performance.

15. Establish RCS Bleed Path

Action/Expected Response	Response Not Obtained
a. Verify power to PRZ PORV Block valves – AVAILABLE	**a.** Restore power to Block valves.
b. Verify PRZ PORV Block valves – BOTH OPEN	**b.** Open both Block valves.
c. Check BLOCK LOW PRZ PRESS. S.I. status light – ON	**c.** Go to Step 15.h.
d. Momentarily place both Safety Injection Block switches to BLOCK and return to NEUTRAL	
e. Verify LOW PRZ PRESS. S.I. BLOCKED status light – ON	
f. Open both PRZ PORVs	
g. Go to Step 16	
h. Open one PRZ PORV	
i. Check BLOCK LOW PRZ PRESS. S.I. status light – ON	**i.** WHEN BLOCK LOW PRZ PRESS. S.I. status light is ON, THEN perform Step 15.j.

BASIS DOCUMENT

WOG Procedure Step 17 **PTN Procedure Step** 17

Verify Adequate RCS Bleed Path

BASIS:

After manually opening the pressurizer PORVs, the operator should check that both pressurizer PORVs, and both PORV block valves, are maintained in the open position. If both valves are maintained open, sufficient RCS bleed flow exists to permit RCS heat removal.

If both PRZ PORVs are not maintained open, the RCS may not depressurize sufficiently to permit adequate feed of subcooled SI flow to remove core decay heat. If core decay heat exceeds RCS bleed and feed heat removal capability, the RCS will repressurize rapidly, further reducing the feed of subcooled SI flow and resulting in a rapid decrease of RCS inventory.

Although only one open PRZ PORV may not be sufficient to maintain adequate RCS bleed flow, the operator should maintain one PRZ PORV open, if possible, and open all RCS high point vents to provide additional bleed path capability.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

9 The word "all" was changed to "both" to avoid confusion, because Turkey Point only has two PRZ PORVs, and two PRZ block valves.

9 The RNO column was reindexed to improve readability.

3 The WOG step does not continue to try to open PRZ PORVs and block valves. This was added to the RNO to emphasize the need to establish adequate RCS vents.

2 At Turkey Point the fuses are normally removed for all RCS head vent valves. The RNO was modified to restore power to these valves so that they can be opened.

8 The words "high point vents" were changed to "RCS vents" to conform with plant specific terminology.

7 A list of head vent valves was provided as required by the WOG guideline.

PLANT SPECIFIC SETPOINTS:

N/A

Facility:	WTSI Corporate		Question 17 original
Vendor	WEC		
Exam Date:			
Exam Type:			

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

What is the consequence of having only one PORV open during implementation of the bleed and feed steps in FR-H.1 3Response to Loss of Secondary Heat Sink?4

A. Reactor coolant system pressure will continue to rise to the pressurizer safety valve setpoint leading to further loss of coolant inventory.

B. Insufficient bleed flow will inhibit mixing of Safety Injection flow leading to localized pressurized thermal shock conditions.

C. The reactor coolant system will not depressurize enough to allow for adequate reflux cooling between the loop hot legs and the steam generators.

D. The reactor coolant system will not depressurize enough to allow for adequate feed of subcooled SI flow to adequately remove core decay heat.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. Having only one PORV open would reduce the ability to depressurize the RCS to ensure adequate SI flow however if one PORV is opened RCS pressure would decrease and not 3continue to rise4.

B. Incorrect but plausible. Insufficient bleed flow is a valid concern however localized thermal shock is not a concern addressed by the FR-H.1 background

document.

C. Incorrect but plausible. FR-H.1 includes the strategy of stopping all reactor coolant pumps to remove pump heat input into the RCS so reflux cooling may be considered a possible condition. Reflux cooling is not a condition addressed by the FR-H.1 background document.

D. Correct. Per the Westinghouse Background Document for FR-H.1 3If both PRZR PORV2s are not maintained open, the RCS may not depressurize sufficiently to permit adequate feed of subcooled SI flow to remove core decay heat. If core decay heat exceeds RCS bleed and feed heat removal capability the RCS will repressurize rapidly, further reducing the feed of subcooled SI flow and resulting in a rapid decrease in RCS inventory.

Techical Reference(s): FR-H.1, Response to Loss of Secondary Heat Sink. (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: L1211I03 (As available)

Question Source:	Bank	13183	
	Modified Bank		(Note changes or attach parent)
	New		

Question History: Last NRC Exam: 2010 Seabrook

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	Topic and K/A #	E12		2.2.44
	Importance Rating	4.2		

Equipment Control: 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.

Proposed Question: RO Question # 18

Given the following conditions:

- The crew enters 3-EOP-ECA-2.1, Uncontrolled Depressurization of All Steam Generators.
- RCS cooldown rate is 65°F/hr.
- AFW is unavailable.
- A standby steam generator feed water pump is in service.
- All SG NR levels are off-scale low.

Which one of the following completes the statements below?

A minimum feed flow of 50 gpm ___(1)___ required to be maintained to each SG.

Low range flow indication ___(2)___ available when using standby feed.

A.	(1) is (2) is NOT
B.	(1) is (2) is
C.	(1) is NOT (2) is NOT
D.	(1) is NOT (2) is

Proposed Answer: A

A.	Correct. See CAUTION 3-EOP-ECA-2.1.
B.	Incorrect. Part 1 is correct. Part 2 is incorrect. Low range flow indication is **NOT** available when using main feedwater instrumentation and an alternate source of feedwater. Changes in RCS temperature and S/G level can be used to control feedwater flow. Plausible that low ranges of feedwater flow such as that supplied by the startup feed pump could be monitored on low range indicators. Candidate also assumes that flow indicators on VPA read in gpm as other indications do- 100 gpm would not register.
C.	Incorrect. Part 1 is incorrect but plausible because flow must be reduced to just 50 gpm / SG if the RCS cooled down at more than 100°F/hr. Also if candidate assumes all flow should be secured to the faulted SGs forgetting the SG level requirement. SGs must remain in a “wet” condition. Part 2 is correct.
D.	Incorrect. Plausible as described in options B and C

Technical Reference(s)	3-EOP-ECA-2.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	10203	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2009	Wolf Creek
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			
Changes in distractors to enhance plausibility. Otherwise question is intact.			

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

CAUTION

- A minimum feed flow of 50 gpm is required to be maintained to each S/G with a Narrow Range Level less than 7%[27%].
- Low range flow indication is **NOT** available when using Main Feedwater instrumentation and an alternate source of feedwater. Changes in RCS temperature and S/G level can be used to control feedwater flow.
- Feed flow is required to be initiated slowly to avoid excessive RCS cooldown and to limit thermal stress in S/Gs.

NOTE

Shutdown Margin is required to be monitored during RCS cooldown.

2. Control Feed Flow To Minimize RCS Cooldown

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check cooldown rate in RCS Cold Legs – LESS THAN 100°F/HR	**a.** Decrease feed flow to 50 gpm to each S/G. Go to Step 2.c.
b. Check Narrow Range Level in <u>all</u> S/Gs – LESS THAN 50%	**b.** Control feed flow to maintain Narrow Range Level less than 50% in all S/Gs.
c. Check RCS Hot Leg temperatures – STABLE <u>OR</u> DECREASING	**c.** Control feed flow <u>or</u> dump steam to stabilize RCS Hot Leg temperatures. <u>IF</u> adequate feed flow to stabilize Hot Leg temperatures <u>OR</u> 400 gpm is **NOT** available, <u>THEN</u> go to 3-EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, Step 1

Facility: WTSI Corporate

Vendor WEC

Question 18 original

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

During the performance of EMG C-21, Uncontrolled Depressurization of All Steam Generators, the following conditions exist:
• RCS cooldown rate is determined to be 165F/Hr.
• All SG NR levels are off-scale low.
• Total AFW flow is 300,000 lbm/hr.
Which ONE of the following describes how the crew is directed to control AFW flow?

A. Flow is reduced to 30,000 lbm/hr to each SG, and Thot is monitored to ensure secondary heat sink is maintained.

B. Flow is terminated to all but a single SG, which is fed at 30,000 lbm/hr, and Tcold is
monitored for conditions that may result in Pressurized Thermal Shock.

C. Total flow is maintained >270,000 lbm/hr until ANY SG narrow range level is >6%, and
Thot is monitored to ensure secondary heat sink is maintained.

D. Total flow is maintained >270,000 lbm/hr until ALL SG narrow range levels are >6%,
and Tcold is monitored for conditions that may result in Pressurized Thermal Shock.

Proposed Answer: A

Explanation (Optional):

A. Correct. See EMG C-21 step 5 and basis

B. Incorrect. Plausible because flow is initiated to only 1 SG in EMG FR-H1.

C. Incorrect. Plausible because this flow is maintained under these conditions in EMG E-0
or in EMG C-21 if RCS cooldown rate is <100F/Hr.

D. Incorrect. Plausible because second half is true, but with RCS cooldown rate >100F/hr, AFW flow is minimized.

Techical Reference(s): EMG C-21, Rev 17, Step 5 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: LO1732334 R2, R3 (As available)

Question Source: Bank 10203
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2009 Wolf Creek

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Administrative, normal, abnormal, and emergency operating procedures for the facility. WTSI 52615 - From VC Summer Audit 2006. There are items in our bank with similar context.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	003		AK1.02
	Importance Rating	3.1		

Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: Effects of turbine-reactor power mismatch on rod control

Proposed Question: RO Question # 19

Given the following conditions:

- A downpower is in progress in accordance with 3-GOP-103, Power Operation to Hot Standby.
- TCS is in MW control.
- Rods are in automatic.
- T_{avg} is 571.2°F.
- T_{ref} is 570.0°F.
- TCS is placed in hold.
- Control Bank D rod M8 drops fully into the core.

Which one of the following completes the statements below?

(Assuming no operator action)

TCS ___(1)___ automatically adjust to maintain MW output.

Control rods ___(2)___ insert to close the given T_{avg}-T_{ref} mismatch.

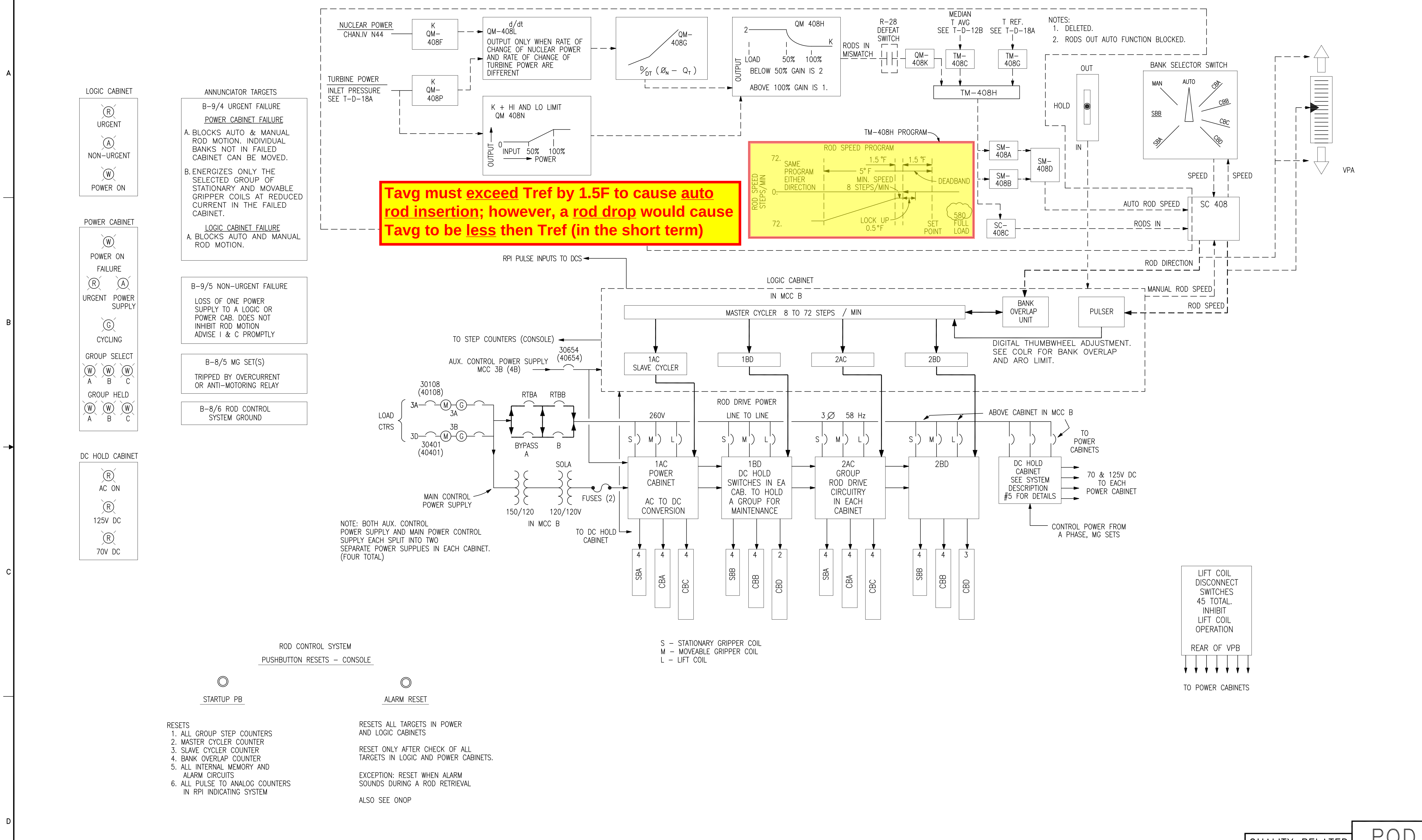
A.	(1) will (2) will NOT
B.	(1) will (2) will
C.	(1) will NOT (2) will NOT
D.	(1) will NOT (2) will

Proposed Answer: A			
A.	Correct. TCS is in hold and will respond to a change in load. Reactor power and steam pressure will lower due to the dropped rod. On a dropped rod, reactor power immediately lowers and Tavg immediately lowers. Although PTN has disabled auto rods out, a demand signal will be created in the power mismatch circuit because of the rate of change of reactor power compared to turbine power. In this case, the Tavg and Tref mismatch isn't great enough before the rod drop and power lowers and Tavg lowers after the rod drop, therefore, rods will not move. Candidate may also believe the that a rod drop has a greater effect on Tref vs Tavg.		
B.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate believes either the Tavg-Tref mismatch is exceeded or the rate of change circuit causes rods to move.		
C.	Incorrect. Part 1 is incorrect, but plausible if candidate believes that taking TCS to hold will prevent TCS from reacting to lowering turbine load. Common misperception when TCS was first implemented.		
D.	Incorrect. Same as B and C		

Technical Reference(s)	LP 6900105, Full Length Rod Control 3-ONOP-28.3 TS Basis 0-ADM-536		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	LP 6900105, Obj. 10		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			

Comments:

Tavg must exceed Tref by 1.5F to cause auto rod insertion; however, a rod drop would cause Tavg to be less then Tref (in the short term)



REV	DATE	REVISION	BY	CH	APP	APP
16	1-20-13	ISSUED AS-BUILT PER CRN-031 (EC 247048) (ITOP 13-01-028)	AFG	BB	–	JAD
15	9/19/11	ISSUED AS-BUILT PER EC-DCR 273667.	RV	RH	–	LG
14	12/17/07	ISSUED AS-BUILT PER PC/M 07-019.	RV	BB	–	RJT
13	6-2-06	ISSUED AS-BUILT PER CRN-I-4427 (PC/M 03-106).	RV	RH	–	TS
12	5-7-04	ISSUED AS-BUILT PER CRN-E-16603 (PC/M 03-106).	RV	RH	–	PDS
11	2-12-04	ISSUED AS-BUILT PER CRN-I-4131 (PC/M 03-050).	RV	RH	TEB	TS

REV	DATE	REVISION	BY	CH	APP	APP
10	9-30-97	ISSUED AS-BUILT FOR PC/M 96-013.	RV	RH	BAP	TS
9	3-21-97	ISSUED AS-BUILT FOR PC/M 95-047.	RV	RH	BAP	TS
8	5-11-95	ISSUED AS-BUILT FOR PC/M 94-120	RV	RH	BAW	JCW
18	3/10/16	ISSUED AS-BUILT PER EC-DCR 285982.	RH	AFG	–	SB
17	5/1/13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-04-155).	RV	BB	–	PMB

QUALITY RELATED

POD

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

CONTROL SYSTEM DIAGRAM

ROD CONTROL SYSTEM

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-D-12A

SYS –

SHEET 1

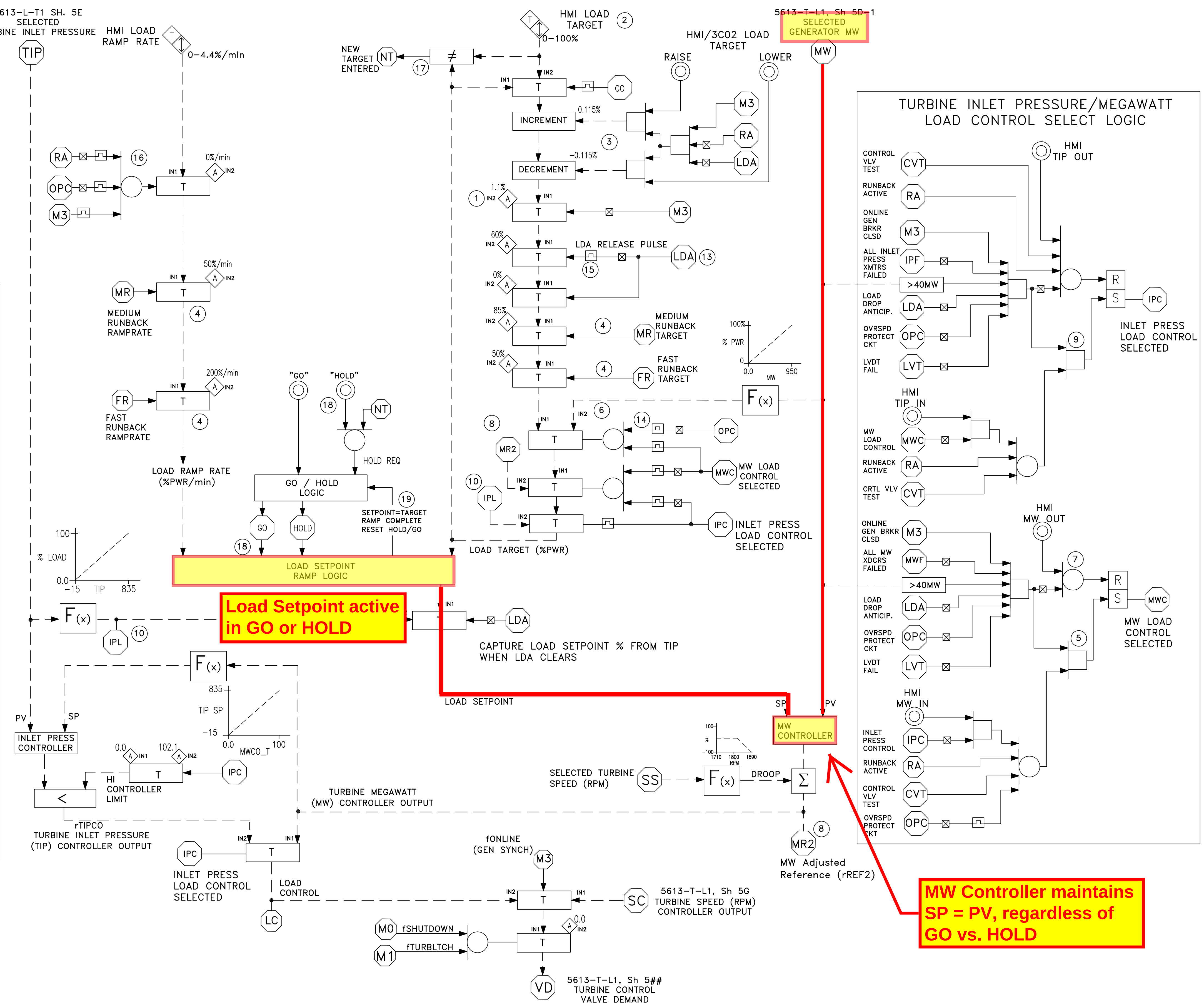
REV 18

DWG NO 5613-T-L1 SHT 5D-2

Symbol	Description
M0	TURBINE MODE 0 SHUTDOWN (fSHUTDOWN)
M1	TURBINE MODE 1 LATCHED (fTURBLTCH)
M3	TURBINE MODE 3 ONLINE-BREAKER CLOSED
MR	5613-L-T1 SH. 5E MEDIUM RUNBACK ACTIVE
FR	5613-L-T1 SH. 5E TURBINE FAST RUNBACK ACTIVE
RA	5613-L-T1 SH. 5E TURBINE RUNBACK ACTIVE
LDA	5613-L-T1 SH. 5f-1 LOAD DROP ANTICIPATORY
OPC	5613-L-T1 SH. 5F-1 OVRSPEED PROTECTIVE CKT
IPC	TURBINE INLET PRESSURE LOAD CONTROL SELECTED
MWC	MEGAWATT LOAD CONTROL SELECTED
IPF	ALL INLET PRESS XMTRS FAILED
MWF	ALL MW XDCRS FAILED
CVT	CONTROL VLV TEST
LVT	LVDT FAIL
SS	5613-T-L1, Sh 5D SELECT TURBINE SPEED (RPM)
IPL	% LOAD FROM TURBINE INLET PRESS

LOAD TARGET

1. LOAD TARGET IS INITIALLY SET TO 1.1% TURBINE POWER WHEN THE GENERATOR BREAKER CLOSES.
2. OPERATOR MAY ENTER A LOAD TARGET (0-100%) FROM THE TCS HMI.
3. OPERATOR MAY INCREMENTALLY RAISE/LOWER THE LOAD TARGET FROM THE 3C02 HANDSWITCHES OR THE TCS HMI.
4. WHEN A TURBINE RUNBACK IS ACTIVE, THE LOAD TARGET IS SET TO THE RUNBACK TARGET AND RAMPRATE IS SET TO THE RUNBACK RAMPRATE
5. MW FEEDBACK CAN BE ENABLED WHEN
   - MODE=3 (ONLINE BREAKER CLOSED)
   - >1 VALID MW TRANSDUCER SIGNAL
   - NO LDA OR OPC ACTIVE
   - GEN MW >40 MW
6. WHEN MW CONTROL IS 1ST ACTIVATED THE LOAD TARGET IS SET TO THE CURRENT GENERATOR MW OUTPUT CONVERTED TO % LOAD
7. MW CONTROL CAN BE MANUALLY DE-ACTIVATED FROM THE HMI OR WILL BE AUTOMATICALLY DE-ACTIVATED WHEN
   - NOT MODE=3 (GEN BREAKER OPEN)
   - BOTH MW TRANSDUCERS FAILED
   - LDA OR OPC ACTIVE
   - GEN MW <18 MW
8. WHEN MW CONTROL IS DE-ACTIVATED THE LOAD TARGET IS SET TO rREF2
9. TURBINE INLET PRESSURE (TIP) CONTROL CAN BE ENABLED WHEN
   - MODE=3 (ONLINE BREAKER CLOSED)
   - >1 VALID TURBINE INLET PRESSURE TRANSMITTER SIGNAL
   - NO LDA OR OPC ACTIVE
   - GEN MW >40 MW
10. WHEN TIP CONTROL IS 1ST ACTIVATED THE LOAD TARGET IS SET TO THE CURRENT TURBINE INLET PRESSURE CONVERTED TO % LOAD
11. TIP CONTROL CAN BE MANUALLY DE-ACTIVATED FROM THE HMI OR WILL BE AUTOMATICALLY DE-ACTIVATED WHEN
    - NOT MODE=3 (GEN BREAKER OPEN)
    - ALL TURBINE INLET PRESSURE TRANSMITTERS FAILED
    - LDA OR OPC ACTIVE
    - GEN MW <18 MW
12. TIP CONTROL IS DE-ACTIVATED THE LOAD TARGET IS SET TO TIP CONTROLLER ADJUSTED REFERENCE rREF3.
13. WHEN AN OPC OR LDA EVENT OCCURS THE LOAD TARGET IS SET TO 0%.
14. WHEN OPC EVENT CLEARS THE LOAD TARGED IS SET TO THE % LOAD BASED ON CURRENT MW.
15. WHEN A LDA EVENT CLEARS LOAD TARGET IS SET TO 60%
16. WHEN A RUNBACK, LDA OR OPC CLEARS THE RAMPRATE IS SET TO 0%/MIN
17. WHEN A NEW TARGET IS ENTERED, HOLD CONDITION BECOMES ACTIVE. IF GO IS SELECTED ON THE HMI, GO BECOMES ACTIVE, HOLD IS CLEARED END THE RAMPING TARGET RAMPS THE SETPOINT TO THE TARGFET AT THE SPECIFIED RAMPRATE.
18. IF "HOLD" IS SELECTED BY THE OPERATOR THE RAMP STOPS AND "GO" IS CLEARED
19. WHEN THE RAMP MATCHES THE SETPOINT TO THE TARGET BOTH "GO" AND "HOLD" ARE CLEARED



REV	DATE	REVISION	BY	CH	APP	APP
0	08-24-12	ISSUED AS-BUILT PER EC 246849 (PC/M 08-077) AND ENTERED INTO THE FPL SYSTEM.	BHS	BB	DDM	PMB

FPL

TURKEY POINT NUCLEAR UNIT 3

LOGIC DIAGRAM

TCS LOAD CONTROL

POD

DRAWING NUMBER 5613-T-L1

SYS –

SHEET 5H

REV 0

5613-T-L1_SH 5H.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	005		AA2.01
	Importance Rating	3.3		

Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Stuck or inoperable rod from in-core and ex-core NIS, in-core or loop temperature measurements

Proposed Question: RO Question # 20

Given the following conditions:

- Unit 3 is at 75% power during a plant startup.
- A rod withdrawal of 3 steps is initiated in manual.
- ANN B 9/2, AXIAL FLUX TILT alarms.
- Rod withdrawal is stopped.
- ANN B 9/3, SHUTDOWN ROD OFF TOP/ DEVIATION alarms.
- Tavg rises 0.2°F and stabilizes.

Which one of the following describes (1) the event and (2) the cause of the AXIAL FLUX TILT alarm?

A.	(1) One RCCA fully drops during withdrawal (2) Exceeding a maximum delta between any two PR channels
B.	(1) One RCCA fully drops during withdrawal (2) Exceeding a maximum delta between upper and lower detectors on any PR channel
C.	(1) One RCCA sticks during withdrawal (2) Exceeding a maximum delta between any two PR channels
D.	(1) One RCCA sticks during withdrawal (2) Exceeding a maximum delta between upper and lower detectors on a any PR channel

Proposed Answer: D

A.	Incorrect. 1st part is incorrect, the candidate focuses on one of the many indications for a dropped rod (shutdown rod off top / deviation). However, when an axial flux tilt alarm is present a fully dropped rod condition does not exist. 2nd part is incorrect, the candidate assumes 2 of 4 power range channels are required for the B 9/3 alarm. This is not true.
B.	Incorrect. 1st part is incorrect. 2nd part is correct.
C.	Incorrect. 1st part is correct. 2nd part is incorrect.
D.	Correct. The event is a stuck RCCA, given that Tavg rises, a rod drop alarm is not in and no prompt drop in reactor parameters. Axial flux alarms on delta between upper and lower detectors on any channel.

Technical Reference(s)	3-ONOP-028.1 LP 6902106		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902106 obj 2		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

2.9 Annunciators

2.9.1 B 2/2, POWER RANGE UPPER DET HI FLUX DEV/AUTO DEFEAT (Normally lit less than 50 percent power)

2.9.2 B 2/3, POWER RANGE LOWER DET HI FLUX DEV/AUTO DEFEAT (Normally lit less than 50 percent power)

2.9.3 B 6/4, POWER RANGE CHANNEL DEVIATION

2.9.4 B 9/2, AXIAL FLUX TILT

2.9.5 B 9/3, SHUTDOWN ROD OFF TOP/DEVIATION

3.0 AUTOMATIC ACTIONS

3.1 **IF** the axial flux difference exceeds the positive or negative threshold value specified in the COLR, **THEN** the OTΔT setpoint will be lowered.

4.0 IMMEDIATE ACTIONS

4.1 None

CAUSES:
1. Actual flux tilt due to control bank positioning or Xenon oscillations
2. Control rod misalignment
3. PR NI malfunction

B9/2

AXIAL FLUX TILT

DEVICE:
Software
DCS Point ANN_B18_A

SETPOINT:
Axial flux difference greater than +10% or less than -10% on any PR channel
Alarm resets when axial flux difference is less than +7.5% and greater than -7.5% on all four PR channels

LOCATION:
DCS

ALARM CONFIRMATION

1. **CHECK** controlling bank inserted into core too far.
2. **CHECK** Xenon induced power oscillations.
3. **CHECK** failed PR detector in any channel.
4. **CHECK** RPI and step counters on console for rod misalignment.

OPERATOR ACTIONS

1. IF condition is **NOT** corrected before OTΔT setpoint is reached, THEN **ENSURE** reactor trip.
2. IF reactor tripped, THEN **ENTER** 3-EOP-E-0, Reactor Trip or Safety Injection.
3. IF control rod misalignment, THEN **REFER TO** 3-ONOP-028.1, RCC Misalignment.
4. IF PR NI malfunction, THEN **REFER TO** 3-ONOP-059.8, Power Range Nuclear Instrumentation Malfunction.
5. IF approaching limitations, THEN **PERFORM** <u>any</u> of the following to prevent exceeding limitations:
   - Borate
   - Dilute
   - Move control rods
6. IF load is high, THEN **REDUCE** load.
7. **REFER TO** the following for actions on excessive axial flux difference:
   - 3-ONOP-059.4, Excessive Axial Flux Difference
   - Tech Specs 3.2.1

REFERENCES:
1. Tech Spec Section 3.2
2. PC/M 03-048, OPDT/OTDT Turbine Runback Elimination
3. PC/M 09-006 (EC 242469), Rod Position Indication System Replacement
4. SPEC-IC-066, PTN Unit 3 and 4 Rod Position Indication System
5. DCS Vendor Manual V000812, Drawing Number US000867-3FD-3042, Sheets 1-3

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	024		AA2.06
	Importance Rating	3.6		

Ability to determine and interpret the following as they apply to the Emergency Boration: When boron dilution is taking place

Proposed Question: RO Question # 21

Given the following conditions:

- Unit 3 is reducing power from 100% power for maintenance.
- B 8/2, ROD BANK A/B/C/D EXTRA LO LIMIT alarms concurrently with a rod control card issue which prevents further rod motion.
- The unit is stabilized at 75%.

Subsequently:

- Rod Control repairs are complete after one hour.
- The crew initiates an emergency boration in accordance with 3-ONOP-046.1, Emergency Boration, to restore from excessive rod motion.
- Turbine control is in manual.
- Tavg is matched with Tref while withdrawing Control Rods.

Which one of the following complete the statements below?

During the boration, the RCO ensures charging flow is a minimum of ___(1)___ on FI-3-122A in accordance with 3-ONOP-046.1.

With the same initial boration flow and after enough time elapses for the effects of the boration, a rise of reactor power will next require the RCO to ___(2)___ .

A.	(1) 45 gpm (2) start an additional Boric Acid Transfer Pump
B.	(1) 40 gpm (2) start an additional Boric Acid Transfer Pump
C.	(1) 45 gpm (2) verify FCV-3-114A, Primary Water to Blender, is closed

D.	(1) 40 gpm (2) verify FCV-3-114A, Primary Water to Blender, is closed
Proposed Answer: C	
A.	Incorrect. Part 1 is correct. 45 gpm is the minimum required charging flow on FI-3-122A IAW 3-ONOP-046.1. Part 2 is incorrect. A dilution is observed by the rise in reactor power. However, starting another Boric Acid Transfer Pump to raise flow is not the required action.
B.	Incorrect. Part 1 is incorrect. The misconception is the charging flow minimum is 40 gpm, which is the minimum required on FI-3-113 when using 3-356 manual emergency boration valve. Part 2 is incorrect. A dilution is observed by the rise in reactor power. However, starting another Boric Acid Transfer Pump to raise flow is not the required action.
C.	Correct. Part 1 is correct. 45 gpm is the minimum required charging flow on FI-3-122A IAW 3-ONOP-046.1. Part 2 is correct. The RCO in step 5 of 3-ONOP-046.1 checks for a dilution and verifies (ensures) FCV-3-114A is closed.
D.	Incorrect. Part 1 is incorrect. The misconception is the charging flow minimum is 40 gpm, which is the minimum required on FI-3-113 when using 3-356 manual emergency boration valve. Part 2 is correct. The RCO in step 5 of 3-ONOP-046.1 checks for a dilution and verifies (ensures) FCV-3-114A is closed.

Technical Reference(s)	3-ONOP-046.1 entry conditions		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X

10 CFR Part 55 Content:	55.41	5
	55.43	
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.		
Comments:		

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

If Emergency Operating Procedures are in effect, this procedure shall be used only as directed by the EOPs.

1 **Initiate Emergency Boration Of The RCS**

a. Verify charging pumps - AT LEAST ONE RUNNING

b. Turn RCS Makeup Control Switch to STOP

c. Manually start Boric Acid Pump 3A or 3B

RESPONSE NOT OBTAINED:

c. Perform the following:

1) Align charging pump suction to the RWST.

2) Hold closed LCV-3-115C.

3) Direct an operator to open Breaker 30669 for LCV-3-115C.

4) **WHEN** 30669 is open, **THEN** release LCV-3-115C Control Switch.

5) Go to Step 1e.

d. Open Emergency Boration Valve, MOV-3-350

RESPONSE NOT OBTAINED:

d. Perform the following:

1) Open Boric Acid to Blender, FCV-3-113A.

2) Open Blender Flow to Charging Pump, FCV-3-113B.

3) Locally open Manual Emergency Boration Valve 3-356.

4) **WHEN** Valve 3-356 is open, **THEN** close FCV-3-113B.

5) Continue with Step 1e.

e. Open Charging Flow to Regen Heat Exchanger, HCV-3-121

f. Verify Loop A Charging Isolation, CV-3-310A - OPEN

RESPONSE NOT OBTAINED:

f. Open Loop C Charging Isolation, CV-3-310B.

g. **IF** using MOV-3-350 for boration, **THEN** establish emergency boration flow

- FI-3-110 - GREATER THAN 60 GPM
- FI-3-122A - GREATER THAN 45 GPM

RESPONSE NOT OBTAINED:

g. Start additional charging pumps and align valves as necessary to establish emergency boration flow.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Initiate Emergency Boration Of The RCS (Cont'd)**	
	h. **IF** using 3-356 for boration, **THEN** establish emergency boration flow • FI-3-113 - INDICATING 40 GPM • FI-3-122A,- GREATER THAN 45 GPM	h. Start additional charging pumps and align valves as necessary to establish emergency boration flow.
	i. **IF** using RWST for boration, **THEN** establish emergency boration flow • FI-3-122A,- GREATER THAN 45 GPM	i. Start additional charging pumps and align valves as necessary to establish emergency boration flow.
2	**Stop Any Charging Pump Operating On Full Flow Recirculation**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Control Banks shall not remain below their rod insertion limit for more than 2 hours when the Reactor is critical (Tech Spec 3.1.3.6).

3 Respond To Reactivity Increase At Power

a. Check Reactor Critical

RNO: a. Go to Step 4.

b. Check control rods - Annunciator B 8/2 ROD BANK A/B/C/D EXTRA LO LIMIT - CLEAR

RNO: b. Perform the following:

1) Place rods in manual, stop rod insertion.

2) Verify Emergency Boration is having desired effect:

* Reactor Power - DECREASING

OR

* Tavg - DECREASING

3) **WHEN** Emergency Boration is having desired effect, **THEN** withdraw control rods to above bank insertion limits.

4) Continue with Step 3e.

c. Check Tavg - MORE THAN 3°F GREATER THAN Tref

RNO: c. Go to Step 3e.

d. Insert control rods

RNO: d. **IF** control rods do **NOT** move in Automatic or Manual, **THEN** concurrently perform 3-ONOP-028, ROD CONTROL SYSTEM MALFUNCTION **AND** continue with Step 3e.

e. Maintain Tavg - WITHIN 3°F OF Tref

1) Continue Emergency Boration

OR

2) Adjust control rods to maintain desired Tavg

OR

3) Adjust turbine load as directed by the Shift Manager

f. Go to Step 5.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Determine Stuck Rod Criteria Following A Reactor Trip**	
	a. Check this procedure entered following a reactor trip	a. Go to Step 5.
	b. Check control rods - ANY STUCK OUT	b. Go to Step 5.
	c. Continue boration	
	* 50 minutes for each rod not fully inserted using BAST water at 60 GPM through MOV-3-350. 75 minutes for each rod not fully inserted using BAST water at 40 GPM through 3-356.	
	* 226 minutes for each rod not fully inserted using RWST water at 60 GPM	
	d. Go to step 6.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Stop Dilution**	
	a. Verify FCV-3-114A control switch – CLOSE	
	b. Check dilution - STOPPED	b. **IF** dilution is continuing, **THEN** dispatch an operator to perform the following: 1) Stop both primary water pumps 2) Verify the following valves are closed: • FCV-3-114A • 10-563, PW System Header Tie Valve • 3-353A, PW to Charging Pump Suction • 3-359A, PW to Chem Mix Tank • 3-246, PW to Demin A • 3-232C, PW to Demin B • 3-232B, PW to Demin C • 3-236A, PW to Demin D • 3-234A, PW to Demin E • 3-233, PW to Demin D and E 3) Report any deviations of valve alignments to the Shift Manager.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	036		AK3.01
	Importance Rating	3.1		

Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents: Different inputs that will cause a reactor building evacuation

Proposed Question: RO Question # 22

Given the following conditions:

- Unit 3 is in Mode 6.
- A fuel assembly drops into the core during fuel movement.
- ANN B 4/1, SOURCE RANGE HIGH FLUX AT SHUTDOWN, alarms.

Which one of the following completes the statements below?

For the event in progress, the SOURCE RANGE HIGH FLUX AT SHUTDOWN alarm, ___(1)___the ONLY clad damage indication.

SOURCE RANGE HIGH FLUX AT SHUTDOWN ___(2)___ automatically cause a containment evacuation alarm to occur

A.	(1) is (2) will
B.	(1) is (2) will NOT
C.	(1) is NOT (2) will
D.	(1) is NOT (2) will NOT

Proposed Answer: C

A.	Incorrect. Part 1 is incorrect. The candidate does not recall other indications such as the gas located inside the cladding releasing or take into account localized boiling. Part 2 is correct. 3-ARP-097-CR.B 4/1 describes the automatic Containment evacuation.
B.	Incorrect. Part 1 is incorrect. The candidate does not recall other indications such as the gas located inside the cladding releasing or take into account localized boiling. Part 2 is incorrect. Manual actions are performed such as a plant announcement for Containment evacuation. However, 3-ARP-097-CR.B 4/1 describes the automatic Containment evacuation.
C.	Correct. Part 1 is correct. The candidate recalls other indications such the gas located inside the cladding releasing or localized boiling. Part 2 is correct. 3-ARP-097-CR.B 4/1 describes the automatic Containment evacuation.
D.	Incorrect. Part 1 is correct. The candidate recalls other indications such the gas located inside the cladding releasing or localized boiling. Part 2 is incorrect. Manual actions are performed such as a plant announcement for Containment evacuation. However, 3-ARP-097-CR.B 4/1 describes the automatic Containment evacuation.

Technical Reference(s)	3-ARP-097-CR.B 4/1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902168 obj 2		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		6
	55.43		
Design, components, and function of reactivity control mechanisms and instrumentation.			
Comments:			

CAUSES:
1. Neutron flux in reactor increased to alarm setpoint
2. SR NI malfunction

If the condition persists for >2 seconds, the Containment Evacuation Alarm will actuate automatically

B4/1

SOURCE RANGE HI FLUX AT SHUTDOWN

DEVICE:	SETPOINT:	LOCATION:
Source Range detectors: • N-31 • N-32	Half decade above count rate at shutdown. Variable and resets at each shutdown	N/A

NOTE

If the annunciator is in alarm from a spike on either N-31 or N-32, there is a two second time delay in the Source Range High Flux at Shutdown circuitry to prevent the actuation of the Containment Evacuation Alarm.

ALARM CONFIRMATION

1. **CHECK** count trend on NI level recorder on console.
2. **CHECK** both source range indicators for increase since shutdown.

OPERATOR ACTIONS

1. IF in Mode 6, THEN **PLACE** any or both of the two PRIMARY SR NI HI FLUX AT SHUTDOWN BLOCK SWITCHES to the BLOCK position to eliminate nuisance B4/1 AND Containment evacuation alarms caused by spiking.
   A. WHEN spiking is no longer present, THEN **PLACE** HI FLUX AT SHUTDOWN BLOCK SWITCH to NORMAL.
2. IF a startup is **NOT** in progress, THEN **ENSURE** actuation of Containment Evacuation alarm.
3. **ANNOUNCE** containment evacuation over the Page System.
4. IF a startup is in progress, THEN **BLOCK** the Containment Evacuation alarm.
5. IF count rate has changed due to a planned change in plant condition such as, heatup, boron concentration change, etc., THEN **ADJUST** the High Flux at Shutdown alarm setpoint using 3-OSP-059.6, High Flux at Shutdown, to maintain a one-half decade above indicated source range count rate.
6. IF rods are withdrawn AND count rates have changed due to changing plant conditions that were **NOT** planned, THEN **TRIP** the reactor.
7. IF flux continues to increase, THEN **BORATE** using 3-ONOP-046.1, Emergency Boration.
8. **INVESTIGATE** for possible dilution/cooldown of RCS.
9. IF SR NI malfunction, THEN **PERFORM** 3-ONOP-059.5, Source Range Nuclear Instrumentation Malfunction.

REFERENCES: Tech Spec Sections 3.3.1 and 3.9.2

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	037		2.4.20
	Importance Rating	3.8		

Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes.

Proposed Question: RO Question # 23

Given the following conditions:

- Unit 3 trips due to a LOOP.
- Secondary radiation levels rise.
- 3C SG is identified as being ruptured and is isolated.
- The cooldown to required Core Exit Temperature is complete.

Which one of the following describes a concern during the initial RCS depressurization of 3-EOP-E-3, Steam Generator Tube Rupture and how the depressurization will be performed?

A.	Voiding in the reactor vessel upper head when using Auxiliary Spray.
B.	Loss of RCS subcooling when using Auxiliary Spray.
C.	Voiding in the reactor vessel upper head when using a PRZ PORV.
D.	Loss of RCS subcooling when using RCS Vent Valves.

Proposed Answer: C

A.	Incorrect. Plausible because cooldown and depressurization are the 2 factors affecting RCS subcooling. Initial cooldown is performed at a higher RCS pressure than subsequently in E-3, so voiding is unlikely. Candidate may believe that with no RCPs running, voiding is inevitable. Candidate may also believe that due to loss of RCPs Aux Spray has priority over PORVs.
B.	Incorrect. Plausible because depressurization will lower subcooling, but Aux Spray supports a controlled depressurization, making loss of subcooling unlikely in this situation. Aux spray is not the preferred depress method.
C.	Correct, IAW 3-EOP-E-3 DB.

D.	Incorrect. At this procedure step, a loss of RCS subcooling should not be a concern.		
Technical Reference(s)	3-EOP-E-3, Caution prior to step 18 E-3 BD step 18		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	X	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			
Bank question from a Robinson Exam about 10-12 years ago.			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	**Check If RCS Cooldown Should Be Stopped**	
	a. Check Core Exit TCs – LESS THAN REQUIRED TEMPERATURE FROM Step 6	**a.** WHEN Core Exit TCs are less than required temperature from Step 6, THEN go to Step 14.b. Do **NOT** continue until cooldown is stopped.
	b. Stop RCS cooldown	
	c. Maintain Core Exit TCs – LESS THAN REQUIRED TEMPERATURE FROM Step 6	
15.	**Check Ruptured S/G(s) Pressure – STABLE OR INCREASING**	IF pressure continues to decrease to less than 250 psig above the pressure of the intact S/G(s) used for cooldown, THEN go to 3-EOP-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED, Step 1.
16.	**Check RCS Subcooling Based On Core Exit TCs – GREATER THAN 39°F[93°F]**	Go to 3-EOP-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED, Step 1.

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

17. Depressurize RCS To Minimize Break Flow And Refill PRZ

a. Normal PRZ Spray – AVAILABLE

a. Observe CAUTIONS and NOTE prior to Step 18 and go to Step 18.

Normal spray unavailable, due to LOOP/loss of RCPs

b. Spray PRZ with maximum available spray until any of the following conditions satisfied using Attachment 6 as reference:

* Both of the following:
  - RCS pressure – LESS THAN RUPTURED S/G(s) PRESSURE
  - PRZ level – GREATER THAN 7%[48%]

  OR

* Both of the following:
  - RCS pressure – WITHIN 300 PSI OF RUPTURED S/G(s) PRESSURE
  - PRZ level – GREATER THAN 37%[50%]

  OR

* PRZ level – GREATER THAN 73%[60%]

  OR

* RCS subcooling based on Core Exit TCs – LESS THAN 19°F[73°F]

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17.	**(continued)**	
	c. Stop depressurization by closing Spray Valve(s)	
	1) IF Normal Spray in service, THEN close Normal Spray valves	**1)** Stop RCP(s) as necessary to stop spray flow.
	2) IF Auxiliary Spray in service, THEN reduce Auxiliary Spray flow to minimum by performing the following:	
	a) Fully open PCV-3-455A, Pressurizer Spray Loop C, and PCV-3-455B, Pressurizer Spray Loop B	
	d. Observe CAUTION prior to Step 20 and go to Step 20	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

- If a PRZ PORV is used to depressurize the RCS, the PRT rupture disk may rupture. This may result in abnormal Containment conditions.
- Cycling of the PRZ PORV shall be minimized.

NOTE

If RCPs are **NOT** running, the upper head region may void during RCS depressurization. This will result in a rapidly increasing PRZ level.

18. Depressurize RCS Using PRZ PORV To Minimize Break Flow And Refill PRZ

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a.	Check PRZ PORV – AT LEAST ONE AVAILABLE	**a.** Establish Auxiliary Spray using Attachment 4 and return to Step 17.b. **1)** IF Auxiliary Spray can **NOT** be established, THEN continue to disregard any false Integrity Status Tree indication caused by ruptured loop T-cold, and go to 3-EOP-ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL, Step 1.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	067		AA1.06
	Importance Rating	3.5		

Ability to operate and / or monitor the following as they apply to the Plant Fire on Site: Fire alarm

Proposed Question: RO Question # 24

Which one of the following is the correct responses after acknowledgement of a Fire Alarm Operator Workstation C41 alarm?

Dispatch ___(1)___ to inspect the alarming zone(s) for fire or smoke and be ready for further response.

The crew will reset the alarming fire detector ___(2)___ .

A.	(1) the fire brigade (2) locally at the detector
B.	(1) an operator (2) locally at the detector
C.	(1) the fire brigade (2) on Fire Alarm Operator Workstation C41
D.	(1) an operator (2) on Fire Alarm Operator Workstation C41

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect. An operator is dispatched to ensure alarm validity. The further response portion of this situation is the confirmation of the alarm and call Control Room. Part 2 is incorrect. Some fire detectors allow local resets. In this condition, an operator will reset after report/confirmation IAW ONOP-016.10 from Fire Alarm Operator Workstation C41 panel.

B.	Incorrect. Part 1 is correct. An operator is dispatched to ensure alarm validity. The further response portion of this situation is the confirmation of the alarm and call Control Room. Part 2 is incorrect. Some fire detectors allow local resets. In this condition, an operator will reset after report/confirmation IAW ONOP-016.10 from Fire Alarm Operator Workstation C41 panel.
C.	Incorrect. Part 1 is incorrect. An operator is dispatched to ensure alarm validity. The further response portion of this situation is the confirmation of the alarm and call Control Room. Part 2 is correct. In this condition, an operator will reset after report/confirmation IAW ONOP-016.10 from Fire Alarm Operator Workstation C41 panel.
D.	Correct. Part 1 is correct. An operator is dispatched to ensure alarm validity. The further response portion of this situation is the confirmation of the alarm and call Control Room. Part 2 is correct. In this condition, an operator will reset after report/confirmation IAW ONOP-016.10 from Fire Alarm Operator Workstation C41 panel.

Technical Reference(s)	0-ONOP-016.8 0-ONOP-016.10		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902261 obj 9		(As available)
Question Source:	Bank		
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			
PTN Bank 69022610901. Reworded distractors and changed context of distractors for discriminatory value.			

1.0 **PURPOSE**

1.1 This procedure provides instructions to be followed after the receipt of an Alarm or Trouble signal from the Fire and Smoke Detection System, or when corrective action is directed by 3/4-ARP-097.CR, Window I 6/6, XFMR / HYDROGEN SEAL OIL DELUGE OPERATING.

2.0 **SYMPTOMS**

2.1 Fire Alarm Operator Workstation C41 (FAOWS C41)

2.2 Control Room Alarm I 6/6, XFMR / HYDROGEN SEAL OIL DELUGE OPERATING

3.0 **AUTOMATIC ACTIONS**

3.1 Water Suppression Systems:

3.1.1 **IF** the Alarm indicator comes on at Alarm Point 37, **THEN** its associated water suppression system will actuate.

3.1.2 **IF** the Alarm signal comes from any Heat Detector (DET45T-1, DET45T-2, DET45T-3, DET45T-4) **AND** a Supervisory Signal from Pressure Switch PSL-4-45L-1 is also present, **THEN** Fire Zone 45 Unit 4 Charging Pump Room water suppression system will Actuate.

3.1.3 **IF** the Alarm signal comes from Heat Detector DET47AT-1, DET47AT-2, or Flame Detector DET47AIR-1, **THEN** Fire Zone 47A Component Cooling Water area suppression system will Actuate.

3.1.4 **IF** the Alarm signal comes from Heat Detector DET47BT-1, DET47BT-2, or Flame Detector DET47BIR-1, **THEN** Fire Zone 47B Component Cooling Water area suppression system will Actuate.

3.1.5 **IF** the Alarm signal comes from any Heat Detector DET52-1 through DET52-9 **OR** Flow Alarm Pressure Switch PS-3-1590, **THEN** Fire Zone 81/86 Unit 3 Lube Oil Reservoir area suppression system will Actuate.

3.1.6 **IF** the Alarm signal comes from any Heat Detector DET53-1 through DET53-9 **OR** Flow Alarm Pressure Switch PS-4-1590, **THEN** Fire Zone 76 Unit 4 Lube Oil Reservoir area suppression system will Actuate.

3.1.7 **IF** the Alarm signal comes from Heat Detector DET54AT-1, DET54AT-2, or Flame Detector DET54AIR-1, **THEN** Fire Zone 54A Component Cooling Water area suppression system will Actuate.

3.1.8 **IF** the Alarm signal comes from Heat Detector DET54BT-1, DET54BT-2, or Flame Detector DET54BIR-1, **THEN** Fire Zone 54B Component Cooling Water area suppression system will Actuate.

3.1.9 **IF** the Alarm signal comes from any Heat Detector (DET55T-1, DET55T-2, DET55T-3, DET55T-4) **AND** a Supervisory Signal from Pressure Switch PSL-3-55L-1 is also present, **THEN** Fire Zone 55 Unit 3 Charging Pump Room water suppression system will Actuate.

4.0 IMMEDIATE ACTIONS

CAUTION

All alarms are to be considered valid until proven otherwise.

4.1 Acknowledge the alarm, BUT DO NOT RESET THE ALARM.

4.2 Determine the Alarm point location of the fire from FAOWS C41.

4.3 Dispatch an operator to inspect the alarming zone(s) for indications of fire or smoke.

Item: 1.1.25.61.9.1

Question 24 original

69022610901;

Which ONE of the following is the proper response to an alarm on the Fire Alarm Operator Workstation (FAOWS) C41?

A) Acknowledge and reset the alarm. No further action is required if the alarm clears.

B) Acknowledge and reset the alarm. Immediately assemble the Fire Brigade and sound the fire alarm.

C) Acknowledge the alarm, but do NOT reset it. Determine the alarm point location of the fire. Immediately assemble the Fire Brigade. Dispatch Fire Brigade personnel to inspect the alarming zone(s) for indications of fire or smoke.

D) Acknowledge the alarm, but do NOT reset it. Determine the alarm point location fo the fire. Dispatch an operator to inspect the alarming zone(s) for indications of fire or smoke.

CORRECT or INCORRECT feedback for item: 1.1.25.61.9.1

RCO Group 19 Audit Exam
0-ONOP-016.8/016.10

Item Classification: Knowledge
Item difficulty: 0.50
Keywords: 2.4.25
Item weight: 10
Points required for mastery: 1
Correct alternative(s): D
Judging values of alternatives:
A=-1 B=-1 C=-1 D=1

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	E03		EK2.2
	Importance Rating	3.7		

Knowledge of the interrelations between the (LOCA Cooldown and Depressurization) and the following: Facilitys 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.

Proposed Question: RO Question # 25

Given the following conditions:

- 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization, is entered.
- Charging pumps become gas bound.
- Unit 3 HHSI pumps are running at shutoff head.
- Containment temperature is 195°F.
- PRZ level is 10%.

Which one of the following completes the statements below?

The pressurizer level requirement ____(1)____ met for terminating HHSI.

HHSI pumps will be rotated to limit continuous runtime of any pump to a maximum of ____(2)____ minutes.

A.	(1) is (2) 44
B.	(1) is (2) 30
C.	(1) is NOT (2) 44
D.	(1) is NOT (2) 30

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible if candidate does NOT apply termination criteria correctly. Candidate uses PRZ level setpoint of 7% to terminate vs 48% with adverse containment. Part 2 is incorrect, but plausible if candidate confuses with RHR pump operation criteria of 44 minutes.
B.	Incorrect. Part 1 is incorrect. Part 2 is correct.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect.
D.	Correct. Part 1 is correct. Termination criteria is not met. PRZ level must be 48% with adverse containment. Part 2 is correct. IAW 3-EOP-ES-1.2, IF charging capability has been lost, AND high-head SI Pumps are running at shutoff head, THEN rotate High-Head SI Pumps as necessary to maintain continuous run time of any pump less than 30 minutes while maintaining at least one High-Head SI Pump running.

Technical Reference(s)	ES-1.2, Post LOCA Cooldown and Depressurization, Step 11		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2011	Watts Bar
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		14
	55.43		
Principles of heat transfer, thermodynamics and fluid mechanics.			
Comments: 13684. Changed part 2. HHSI pp rotation requirements from foldout page.			

FOLDOUT PAGE
For Procedure 3-EOP-ES-1.2

1. **ADVERSE CONTAINMENT CONDITIONS**
   A. IF either condition listed below occurs, THEN use [Adverse Containment Setpoints]:
   Containment atmosphere temperature ≥ 180°F
   OR
   Containment radiation levels ≥ $1.3x10^5$ R/hr
   B. WHEN Containment atmosphere temperature returns to less than 180°F,
   THEN Normal Setpoints can again be used.
   C. WHEN Containment radiation levels return to less than $1.3x10^5$ R/hr,
   THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.
2. **SI TERMINATION CRITERIA**
   IF all conditions listed below occur, THEN go to 3-EOP-ES-1.1, SI TERMINATION, Step 1:
   A. RCS Subcooling based on Core Exit TCs – GREATER THAN 19°F[GREATER THAN ADVERSE VALUE IN TABLE BELOW]

SI TERMINATION ADVERSE SUBCOOLING VALUE	
RCS PRESSURE (PSIG)	**ADVERSE SUBCOOLING VALUE**
< 2485 AND ≥ 2000	35 °F
< 2000 AND ≥ 1500	45 °F
< 1500 AND ≥ 1000	55 °F
< 1000 AND ≥ 500	110 °F
< 500	160 °F

   B. Total feed flow to intact S/Gs – GREATER THAN 400 GPM OR Narrow Range Level in at least one intact S/G – GREATER THAN 7%[27%]
   C. RCS pressure – GREATER THAN 1625 PSIG[1950 PSIG] AND STABLE OR INCREASING
   D. PRZ level – GREATER THAN 7%[48%]
   E. Charging Capability – AVAILABLE

   Conditions not met
3. **SI RE-INITIATION CRITERIA**
   IF either condition listed below occurs following SI reduction,
   THEN manually start SI pumps as necessary to restore RCS subcooling and PRZ level:
   * RCS subcooling based on Core Exit TCs – LESS THAN 19°F[73°F]
   OR
   * PRZ level – CAN **NOT** BE MAINTAINED GREATER THAN 7%[48%]
4. **SECONDARY INTEGRITY CRITERIA**
   IF any S/G pressure is decreasing in an uncontrolled manner OR has completely depressurized, AND that S/G has **NOT** been isolated, THEN go to 3-EOP-E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.
5. **E-3 TRANSITION CRITERIA**
   IF any S/G level increases in an uncontrolled manner OR any S/G has abnormal radiation,
   THEN manually start SI Pumps and go to 3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.
6. **COLD LEG RECIRCULATION SWITCHOVER CRITERIA**
   IF RWST level decreases to less than 155,000 gallons,
   THEN go to 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1.
7. **CST MAKEUP WATER CRITERIA**
   IF CST level decreases to less than 12%,
   THEN add makeup to CST using 3-NOP-018.01, CONDENSATE STORAGE TANK (CST).
8. **LOSS OF OFFSITE POWER OR SI ON OTHER UNIT**
   IF SI has been reset AND subsequently either offsite power is lost OR SI actuates on the other unit,
   THEN restore safeguards equipment, and at least one Computer Room Chiller to required configuration.
   Refer to Attachment 2 for essential loads.
9. **LOSS OF CHARGING CRITERIA**
   IF charging capability has been lost, AND High-Head SI Pumps are running at shutoff head,
   THEN rotate High-Head SI Pumps as necessary to maintain continuous run time of any pump less than 30 minutes while maintaining at least one High-Head SI Pump running.

Facility: WTSI Corporate

Question 25 original

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following:

- A small break LOCA occurred on Unit 1.
- ES-1.2, 3Post LOCA Cooldown and Depressurization,4 is in progress.
- RCS pressure is 1420 psig and one charging pump has been stopped.
- The crew is ready to stop the first SI pump.

Which ONE of the following completes the statements below?

When the SI pump is stopped, RCS subcooling will drop (1)

The minimum RCS subcooling value required to allow the second SI pump to be stopped is (2) than the value required for stopping the first pump.

A. (1) and stabilize at a lower value due to an increase in RCS temperature with lower ECCS injection flow
(2) less

B. (1) and stabilize at a lower value due to an increase in RCS temperature with lower ECCS injection flow
(2) greater

C. (1) due to reduced ECCS injection flow and stabilize at a lower value when break flow equals ECCS injection flow.
(2) less

D. (1) due to reduced ECCS injection flow and stabilize at a lower value when break flow equals ECCS injection flow.

(2) greater

Proposed Answer: D

Explanation (Optional):

A. Incorrect Plausible because the total ECCS flow from the SI pumps will be decreased when the first SI pump is stopped but the RCS temperature rising is not the cause of RCS subcooling dropping. Also, the amount of subcooling required to stop the second SI pump does change, but more is required not less.

B. Incorrect, Plausible because the total ECCS flow from the SI pumps will be decreased when the first SI pump is stopped but the RCS temperature rising is not the cause of RCS subcooling dropping. Also, the amount of subcooling required to stop the second SI pump being higher is correct.

C. Incorrect Plausible because the subcooling value will first drop due RCS pressure dropping because of a reduction in the ECCS injection flow when the SI pump is stopped, allowing the break flow to drop due to reduce RCS pressure. Also, the amount of subcooling required to stop the second SI pump does change, but more is required not less.

D. Correct, The subcooling value will first drop due RCS pressure dropping because of a reduction in the ECCS injection flow when the SI pump is stopped. Then as the pressure in the RCS drops the break flow will drop. Eventually the RCS break flow and the ECCS injection flow will reach equilibrium at a lower pressure. The procedure does require a higher subcooling to stop the second pump.

Techical Reference(s): ES-1.2, Post LOCA Cooldown and Depressurization, Revision 0015 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: 3-OT-EOP0100
18. Analyze and explain the process that leads to a new RCS equilibrium pressure following the shutdown of
an ECCS pump during the ES-1.2 reduction sequence (As available)

Question Source: Bank 13684

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 Watts Bar

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	E06		EK1.3
	Importance Rating	3.7		

Knowledge of the operational implications of the following concepts as they apply to the (Degraded Core Cooling): Annunciators and conditions indicating signals, and remedial actions associated with the (Degraded Core Cooling).

Proposed Question: RO Question # 26

Given the following conditions:

- Unit 3 tripped from 100% power.
- A LOCA is in progress.
- Attachment 3 of 3-EOP-E-0, Reactor Trip or Safety Injection, is complete.
- The running HHSI pumps trip.
- ANN A 4/2, QSPDS INADEQUATE CORE COOLING, alarms.
- CET temperatures are 750°F and rising.
- RCPs are secured.

Which one of the following completes the statements below?

The highest priority Core Cooling CSF is a/an ___(1)___ path.

In accordance with the required Core Cooling FRP, the 1st priority is to ___(2)___ .

A.	(1) ORANGE (2) start RCPs for forced flow
B.	(1) ORANGE (2) establish HHSI flow
C.	(1) RED (2) start RCPs for forced flow
D.	(1) RED (2) establish HHSI flow

Proposed Answer: B

A.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate believes starting RCPs for forced flow takes precedence over establishing HHSI flows.
B.	Correct. CET temperature >700°F is an orange path. SI is the 1st priority in FR-C.2 .
C.	Incorrect. Part 1 is incorrect, but plausible if the candidate believes that core cooling is inadequate by forgetting that C.1 addresses the red path condition at 1200°F and C.2 addresses the orange path. Some FRPs (FR-P.1 and FR-S.1) address either an orange or red path condition. Part 2 is incorrect.
D.	Incorrect. Part 1 is incorrect. Part 2 is correct.

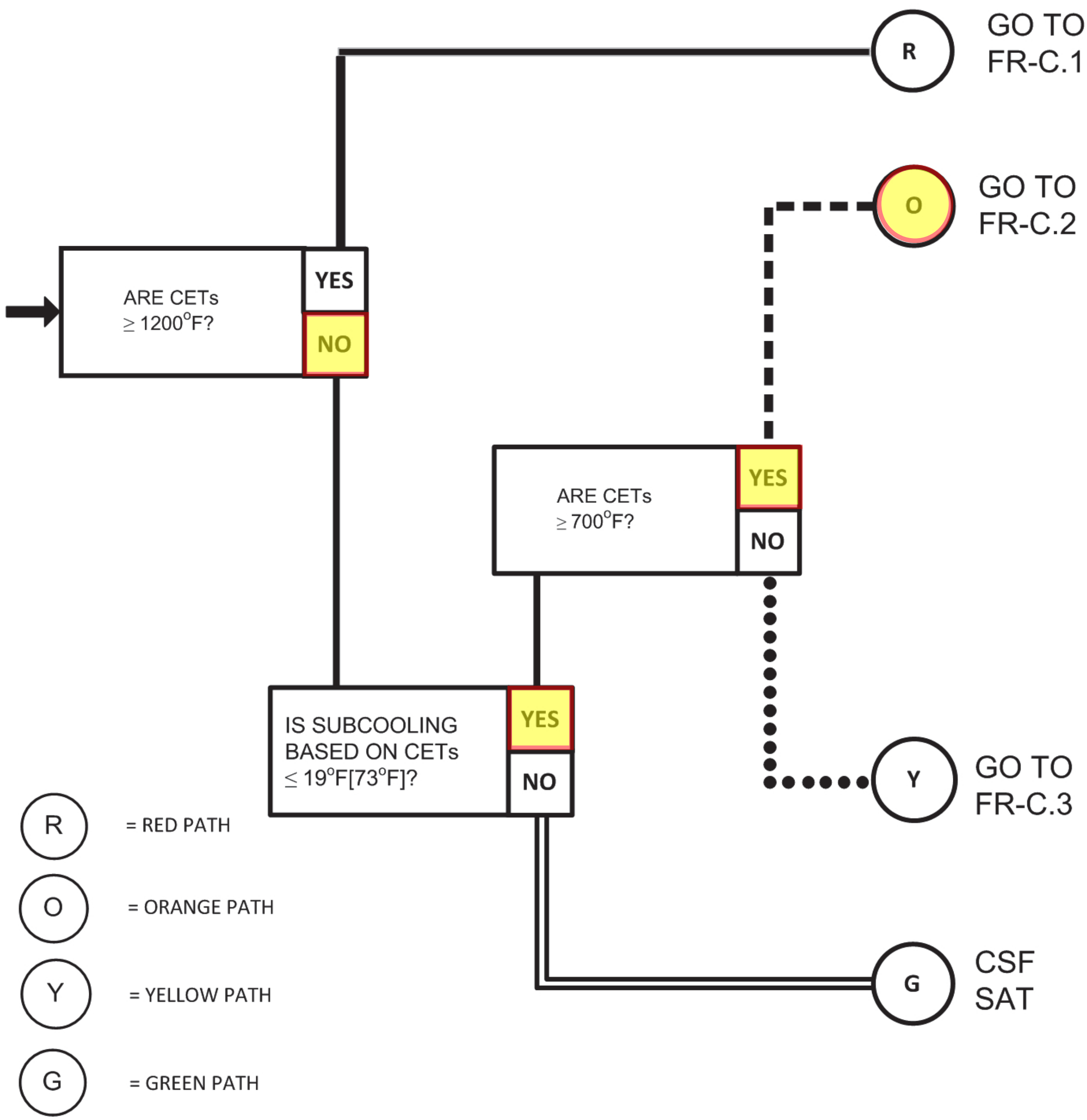
Technical Reference(s)	F-0 3-EOP-FR-C.2		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

ENCLOSURE 2

CSF F-0.2 Core Cooling

(Page 1 of 1)

NOTE
Obtain core exit temperature using at least five of the hottest Core Exit Thermocouples.



STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.0 OPERATOR ACTIONS

CAUTION

IF RWST level decreases to less than 155,000 gallons, AND SI system is in RWST injection alignment, THEN SI System shall be aligned for Cold Leg Recirculation using 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

NOTE

- Normal conditions for running RCPs are desired, but RCPs shall **NOT** be tripped if normal conditions can **NOT** be established or maintained.
- Foldout page is required to be monitored throughout this procedure.

1. Verify SI Valve Alignment – PROPER EMERGENCY ALIGNMENT

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check SI System – ALIGNED FOR RWST INJECTION	**a.** Perform the following: **1)** Refer to 3-EOP-ES-1.3 or 3-EOP-ES-1.4, as applicable, for proper SI alignment. **2)** Manually align valves as necessary to establish proper SI alignment. **3)** Go to Step 2.
b. Verify SI Valve amber lights on VPB – ALL BRIGHT	**b.** Manually align valves to establish proper SI alignment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**Verify SI Flow In All Trains**	
	a. Check SI System – ALIGNED FOR RWST INJECTION	**a.** Verify only one RHR pump running. Go to Step 2.d.
	b. RCS pressure – LESS THAN 275 PSIG[575 PSIG]	**b.** Go to Step 2.d.
	c. RHR Pump Flow Indicator – CHECK FOR FLOW	**c.** Start pumps and align valves to establish RHR flow.
	d. High-Head SI Pump Flow Indicator – CHECK FOR FLOW	**d.** Perform the following: **1)** Start pumps and align valves to establish High-Head SI flow. **2)** Try to establish any other high pressure injection as follows: **a)** Reset SI. **b)** IF offsite power is **NOT** available, THEN check diesel capacity adequate to run three Charging Pumps. IF adequate diesel capacity is **NOT** available, THEN shed non-essential loads. Refer to Attachment 1, for component KW load rating. **c)** Start Charging Pumps to deliver maximum flow.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	Topic and K/A #	E15		EK3.2
	Importance Rating	2.8		

Knowledge of the reasons for the following responses as they apply to the (Containment Flooding) Normal, abnormal and emergency operating procedures associated with (Containment Flooding).

Proposed Question: RO Question # 27

Given the following condition:

- The crew enters 4-EOP-FR-Z.2, Response to Containment Flooding.

Which one of the following identifies an unexpected source of water to containment and the reason this condition must be addressed?

A.	Water from the Accumulators may exceed the containment level design basis criterion when injected in an uncontrolled manner.
B.	Water from the RCS lower vessel may cause a thermal stress on the core when re-injected during the recirculation phase of accident.
C.	Water from the accident unit RWST may block the sump filters with contaminants if pumped below the low level setpoint.
D.	Water from the opposite unit RWST may reach critical plant components necessary for plant recovery and may be damaged.

Proposed Answer: D

A.	Incorrect. Plausible if thought the accumulator fluid occupies too much of the containment volume needed to handle accident steam exceeding containment pressure design criteria.
B.	Incorrect. Plausible if candidate believes there is colder water left standing at the bottom of the vessel which would cause a thermal stress once reinjected.

C.	Incorrect. Plausible if thought that the RWST contaminants/sludge will cause sump blockage.		
D.	Correct. This is the only condition from the list that is unexpected in FR-Z.2.		
Technical Reference(s)	FR-Z.2, Step 3 FRZ-0.2, Step 3 Basis		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.0 OPERATOR ACTIONS

1. **Try To Identify Unexpected Source Of Water To Sump:**
   - Component Cooling Water
   - Primary Makeup Water
   - Unit 4 RWST
   - Secondary coolant (steam or feedwater)

2. **Check Containment Sump Activity Level:**

   a. Direct Nuclear Chemistry to sample Containment Sumps

3. **Notify TSC Staff Of Sump Level And Activity Level To Obtain Recommended Action**

4. **Return To Procedure And Step In Effect**

End of Section 3.0

BASIS DOCUMENT

WOG Procedure Step 1 **PTN Procedure Step** 1

Try To Identify Unexpected Source Of Water To Sump

BASIS:

This step instructs the operator to try to identify the unexpected source of the water in the containment sump. Containment flooding is a concern since critical plant components necessary for plant recovery may be damaged and rendered inoperable. A water level greater than the design basis flood level provides an indication that water volumes other than those represented by the emergency stored water sources such as RWST, accumulators, etc. have been introduced into the containment sump. Typical sources which penetrate containment are component cooling water, primary makeup water, feedwater, and main steam. All possible water sources that penetrate containment should be included in this step. These systems provide large water flow rates to components inside the containment, and a major leak or break in one of these lines could introduce large quantities of water into the sump. Identification and isolation of any broken or leaking water line inside containment is essential to maintaining the water level below the design basis flood level.

STEP DEVIATIONS FROM WOG GUIDELINES:
TYPE DESCRIPTION

2 The plant specific equivalent of the service water system is intake cooling water. Since ICW does not enter containment, the step to check for ICW (service water) was deleted.

7 Primary makeup water and the opposite unit's RWST were added to the list of systems to be checked, as required by the WOG guideline.

PLANT SPECIFIC SETPOINTS:

N/A

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	003		K3.04
	Importance Rating	3.9		

Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: RPS

Proposed Question: RO Question # 28

Given the following conditions:

- Unit 3 is at 14% power.
- The 3B 4kV Bus de-energizes due to an undervoltage condition.

Which one of the following completes the statement below?

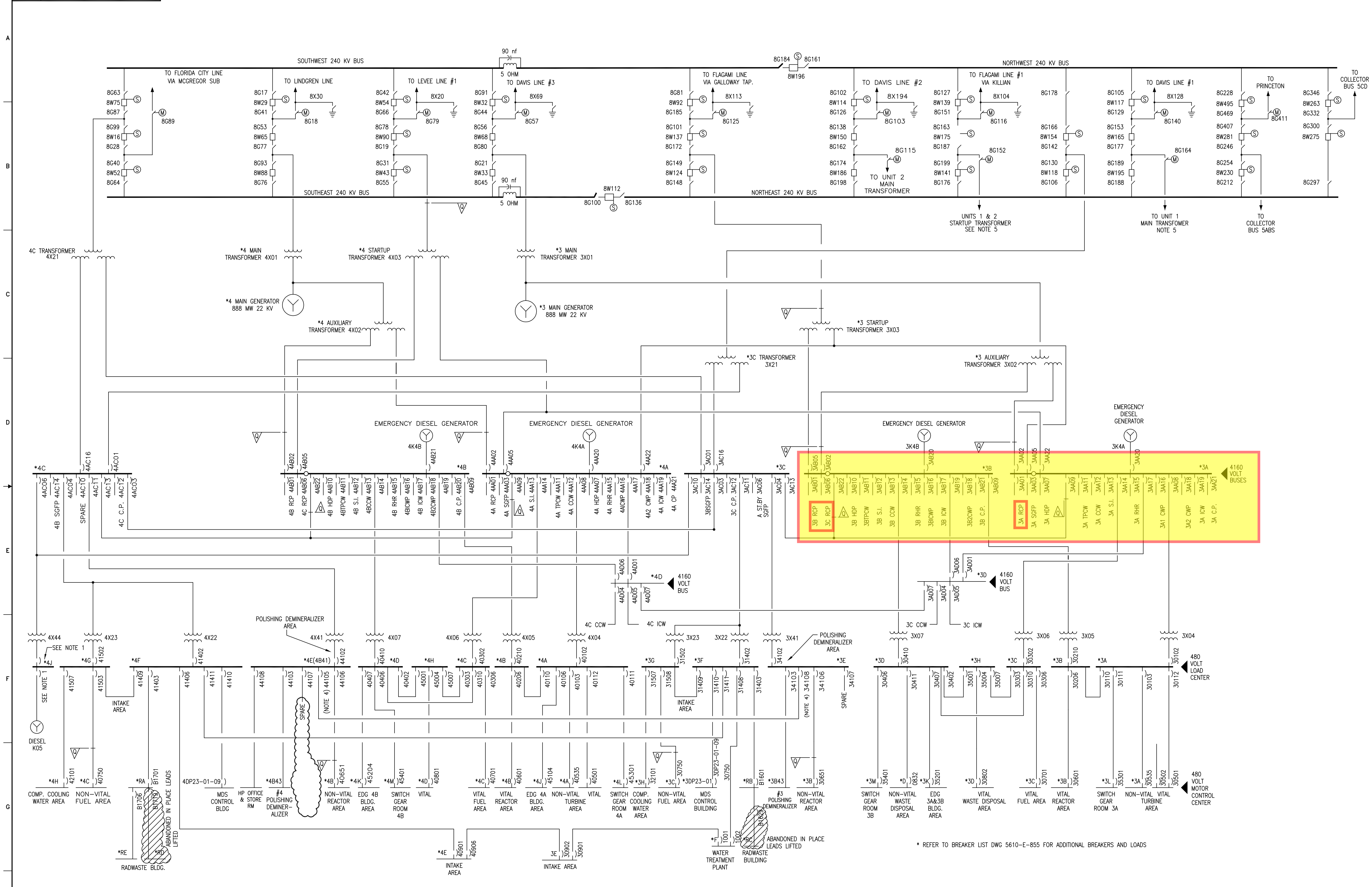
___(1)___ RCP(s) will trip and the reactor ___(2)___ automatically trip.

A.	(1) Only one (2) will
B.	(1) Only one (2) will NOT
C.	(1) Two (2) will
D.	(1) Two (2) will NOT

Proposed Answer: C

A.	Incorrect. Plausible because one RCP would trip if the condition was on bus 3A. The candidate must understand the arrangement of reactor protection system permissives P-7 and P-8. At this power the reactor would not automatically trip if one breaker were to open. Part 2 is plausible if candidate believes 2/3 RCPs tripped causes reactor trip forgetting that the plant is > P-7.

B.	Incorrect. Plausibility for one breaker tripping same as in Option A and second part is correct.		
C.	Correct. Power is between P-7 and P-8 so 2 RCPs must trip to trip the reactor, and Bus 3B is the power supply to RCPs 3B and 3C		
D.	Incorrect. Plausible because 2 RCPs trip, but incorrect because the reactor will automatically trip. Additionally, 2 RCPs tripping without a reactor trip is plausible because this condition could exist if this same event happened with reactor power less than P-7		
Technical Reference(s)	LP 6902163 LP 6902108 5610-T-E-1591 5613-T-L1 (sh 2, 20)		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902108 obj 5 6902163 obj 7		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			



NOTES:

1) SEE 5610-E-1522 FOR ADDITIONAL INFORMATION
2) DELETED
3) DELETED
4) 480V TIE-LINE BETWEEN LOAD CENTERS 3E & 4E IS A TEMPORARY MAINTENANCE TIE AND ITS USE IS CONTROLLED BY PROCEDURE 3/4-NOP-006.
5) NOTIFY NPS PRIOR TO COMMENCING WORK ON THIS SYSTEM.

REV	DATE	REVISION	BY	CH	APP	APP
71	3/19/13	ISSUED AS-BUILT PER EC 249331 (ITOP 13-03-150).	RH	BB	–	ERJ
70	02-09-12	ISSUED AS-BUILT PER EC 242437 (ITOP 13-02-034)	RH	RV	–	PMB
69	12-08-12	ISSUED AS-BUILT PER EC-DCR 277655.	RV	BB	–	MA
68	08-30-12	ISSUED AS-BUILT PER EC 246916 (PC/M 06-167) .	RV	BB	DA	JD
67	08-23-12	ISSUED AS-BUILT PER EC 275192	RV	RH	–	PMB
66	06-11-12	ISSUED AS-BUILT PER EC 275192	RV	BB	–	PDS

REV	DATE	REVISION	BY	CH	APP	APP
65	4/25/12	ISSUED AS-BUILT PER EC 249330 AND INCORP. CRN-038.	RV	BB	–	AH
64	2/23/12	ISSUED AS-BUILT PER EC 249330. (PARTIAL – BREAKER ONLY)	RV	RH	–	LG
63	1/27/12	ISSUED AS-BUILT PER EC 246989(PC/M 09-105).	RV	RH	–	LG
74	06-13-13	ISSUED AS-BUILT PER EC 246917 (ITOP 13-06-052).	RV	RH	–	PMB
73	04-02-13	ISSUED AS-BUILT PER EC-DCR 278017.	RV	BB	–	DBJ
72	3-29-13	ISSUED AS-BUILT PER EC 246904 & INCORP. CRN-017 (ITOP 13-03-168).	RH	RV	–	TK

SAFETY RELATED | POD

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

OPERATING DIAGRAM
ELECTRICAL
DISTRIBUTION

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-E-1591

SYS –

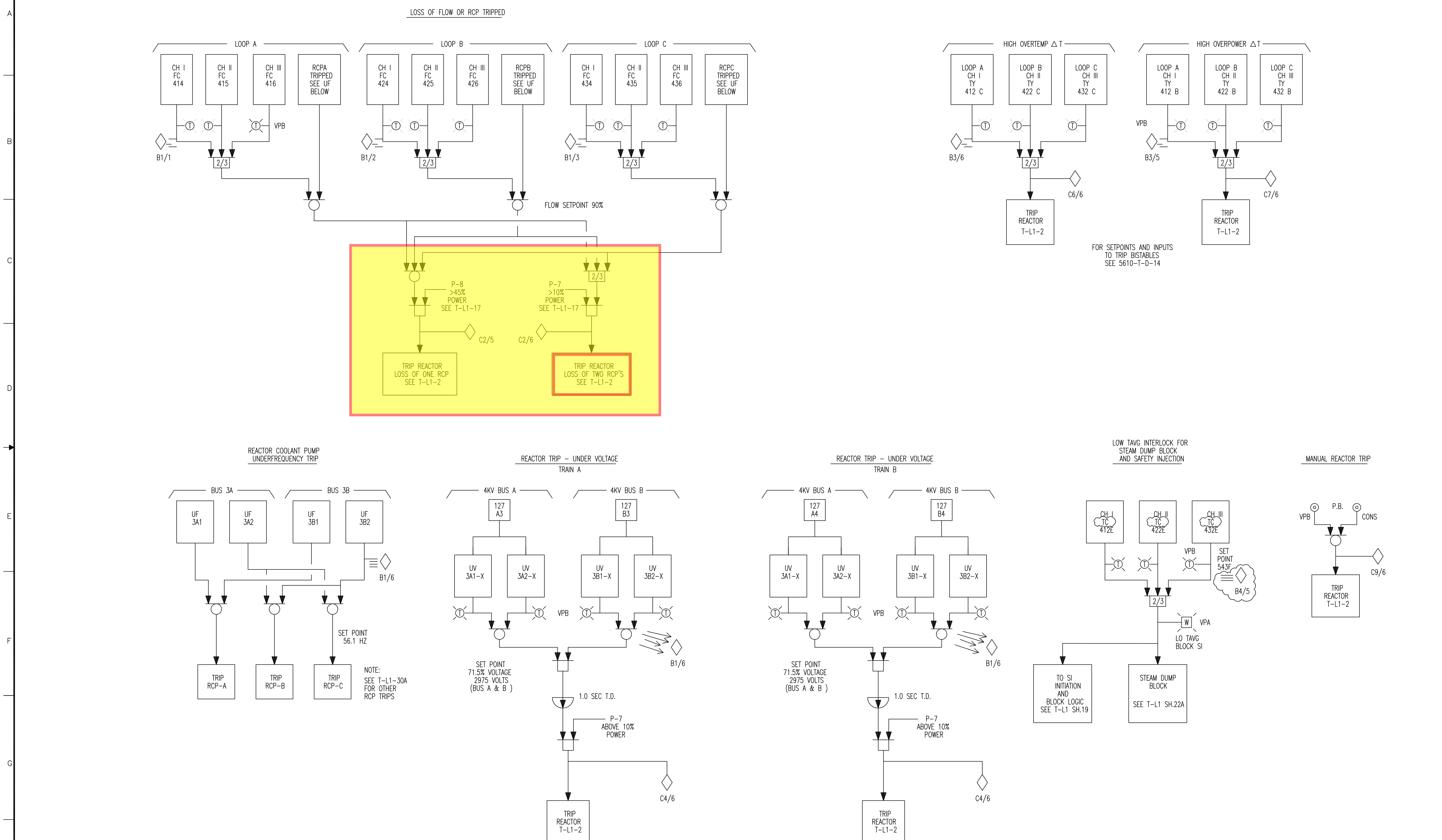
SHEET: 1

REV 74

5610-T-E-1591_SH 1.DWG

20 5610-T-L1 DWG NO

REACTOR TRIP



CAD FILE: UTE01313.DWG

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
23	4-4-06	ISSUED AS-BUILT PER PC/M 05-107.	RV	BB	TBH	TS	18	5-1-96	ISSUED AS-BUILT FOR PC/M 94-035.	RV	RH	BAP	JCW
22	4-23-05	ISSUED AS-BUILT PER CRN-I-4300 (PC/M 03-106).	RV	SB	FJK	SCD	17	10-14-94	ISSUED AS-BUILT FOR PC/M 93-207.	RV	RH	RAS	JCW
21	1-20-98	ISSUED AS-BUILT PER CRN-I-3663 (PC/M 97-042).	RV	RH	MA	TS	25	7-15-14	ISSUED AS-BUILT PER EC 281974.	RH	BB	–	AFG
20	6-18-96	INCORP. CRN-E-15470 FOR PC/M 96-023	RV	RH	BAP	JCW	24	5-7-08	ISSUED AS-BUILT FOR PC/M 05-137	BB	SB	–	PDS
19	5-14-96	ISSUED AS-BUILT FOR PC/M 95-100.	RV	RH	JJN	JCW	14	1-20-94	REDRAWN AND ISSUED AS-BUILT PER DCR-TPI-93-515.	SP	MBP	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

LOGIC DIAGRAM

PRIMARY COOLANT SYSTEM
REACTOR TRIPS AND TAVG INTERLOCK

POD

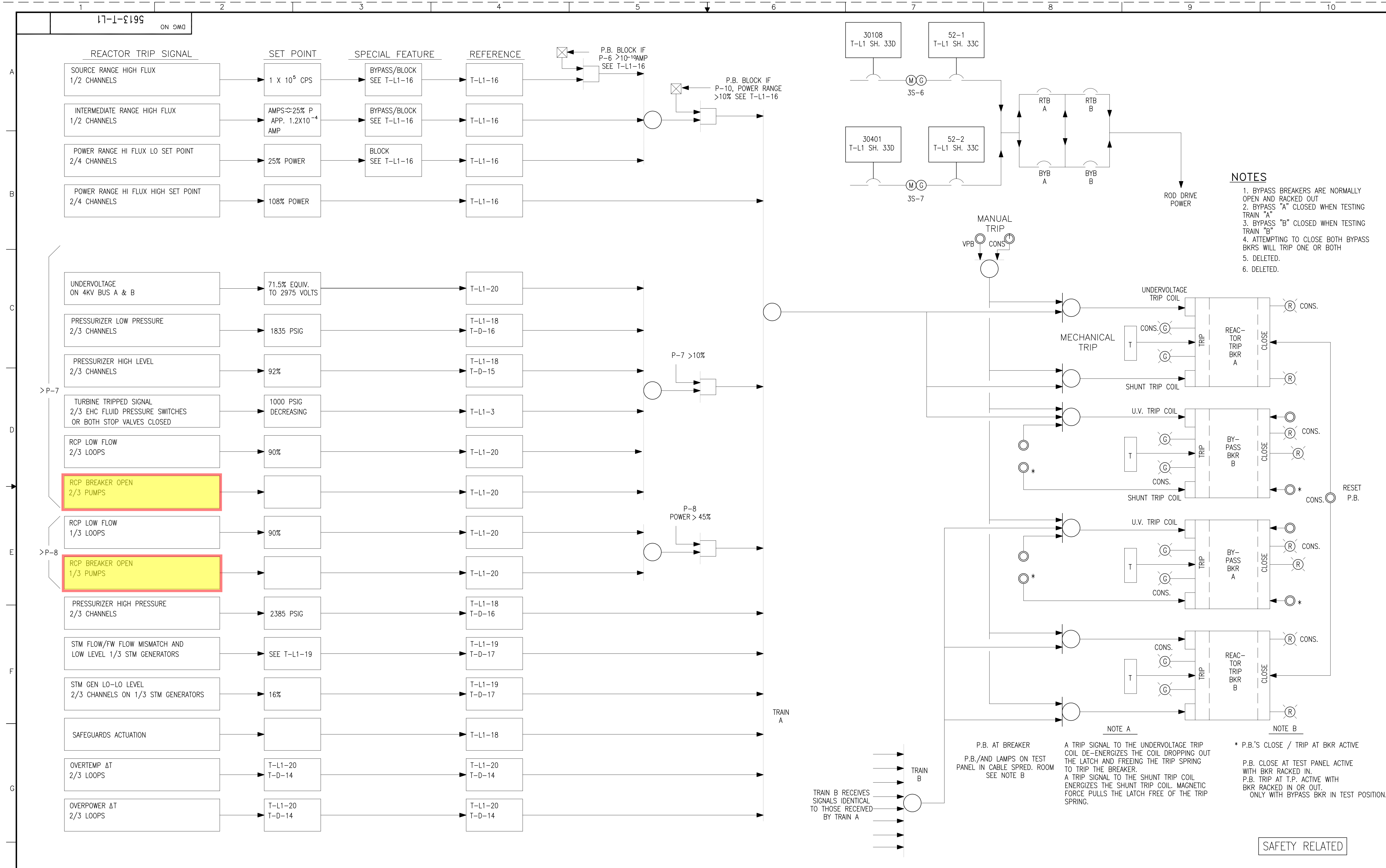
DRAWING NUMBER 5610-T-L1

SYS

SHEET 20

REV 25

5610-T-L1_SH 20.DWG



REV	DATE	REVISION	BY	CH	APP	APP
0	DDM	ISSUED AS-BUILT PER EC 246849 (PC/M 08-077) AND ENTERED INTO THE FPL SYSTEM.	BHS	BB	DDM	PMB

FPL

TURKEY POINT NUCLEAR UNIT 3

LOGIC DIAGRAM

REACTOR TRIP SIGNALS

POD

DRAWING NUMBER 5613-T-L1

SYS —

REV 0

SHEET 2

5610-T-L1_SH 2.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	004		K6.09
	Importance Rating	2.8		

Knowledge of the effect of a loss or malfunction on the following CVCS components: Purpose of VCT divert valve

Proposed Question: RO Question # 29

Given the following conditions:

- Unit 3 is at 100% power.
- NRHX outlet temperature on TI-3-144 is 151°F and rising.
- Letdown pressure on PI-3-145 is 295 psig.
- NRHX (Non Regenerative Heat Exchanger) CCW flow is 105 gpm.
- TCV-3-143, Letdown Demin Divert Valve, remains aligned to the demineralizers.

Which one of the following identifies the effect if no action is taken?

A.	Demineralizer vessel over pressure condition
B.	Demineralizer bed high temperature degradation
C.	Flashing of the letdown line upstream of the NRHX
D.	Thermal stress on NRHX tubes due to insufficient cooling

Proposed Answer: B

A.	Incorrect. Plausible because the candidate may think 295 psig would over pressurize the vessel. Candidate may confuse with VCT pressure which is normally 30 psig. Candidate may also think 295 psig is way to high a pressure relative to saturation conditions for 151°F.
B.	Correct. Design restrictions on demineralizer operation require the temperature of the water entering the inlet header to be less than 140°F.

C.	Incorrect. Letdown flow is aligned with 2 orifices in service and flow is relatively high but the divert valve is for temperature and should divert at 135°F to prevent degradation due to temperature. Plausible because this could be a concern if the Letdown Pressure Controller failed and hot letdown water was flowing to the orifices.		
D.	Incorrect. Plausible because CCW flow is insufficient but divert is strictly for protection of demins		
Technical Reference(s)	LP 6902113 3-ARP-097-CR.A 3/5 3-OP-047 P&Ls		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902113 obj 5		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

CAUSES:
1. High letdown flow
2. Low CCW flow to Non Regenerative Heat Exchanger
3. Low charging flow to Regenerative Heat Exchanger

A3/5

**LTDN DEMIN
HI TEMP/
FLOW DIVERTED**

DEVICE:	**SETPOINT:**	**LOCATION:**
TI-3-143	135°F	N/A

ALARM CONFIRMATION

1. **CHECK** TCV-3-143 diverting letdown to VCT.
2. **CHECK** TI-3-144, NONREGEN HX OUTLET TEMPERATURE is greater than 135°F, but less than TI-3-140, REGEN HX LTDN OUTLET TEMPERATURE on VPA.

Divert valve failed to re-position and anion resin is in jeopardy, due to high temperature

OPERATOR ACTIONS

1. **ENSURE** TCV-3-143 diverting letdown to VCT.
2. IF TI-3-143, NONREGEN HX LTDN TEMPERATURE is approximately equal to TI-3-140, THEN:
   - **CHECK** FI-3-620, NRHX CCW flow local indicator between 100 and 800 gpm.
   - **CHECK** FI-3-620A, Flow Indicator for Non-Regen HX CCW Outlet between 100 and 800 gpm.
3. IF alarm is due to low CCW flow, THEN:

   A. **PLACE** TC-3-144A, L/D TEMP CONTROLLER in MANUAL.

   B. **REDUCE** Letdown Temperature manually with TC-3-144A.

   C. IF Letdown Temperature can **NOT** be reduced with TC-3-144A in MANUAL, THEN:

   (1) **THROTTLE** open 3-834, NON-REGEN HX TEMP CONTROL VLV, TCV-3-144 BYPASS.

   (2) **CLOSE** TCV-3-144A, NON-REGEN HX TEMP CONTROL VLV using TC-3-144A.

   (3) **CLOSE** 3-833, NON-REGEN HX TEMP CONTROL VLV TCV-3-144 INLET.

   (4) **CONTROL** Letdown Temperature using 3-834, NON-REGEN HX TEMP CONTROL VLV, TCV-3-144 BYPASS, while maintaining either of the following below 800 gpm:

   - FI-3-620, NRHX CCW flow local indicator
   - FI-3-620A, Flow Indicator for Non-Regen HX CCW Outlet.

4.0 PRECAUTIONS/LIMITATIONS

4.1 Before changing system status, Technical Specifications should be consulted for system requirements for that plant mode.

4.2 Design restrictions on demineralizer operation require the letdown flow rate to be maintained below 120 gpm and the temperature of the water entering the inlet header to be less than 140 °F.

4.3 Explosive mixtures of hydrogen and oxygen concentration shall be avoided at all times. The oxygen concentration in the VCT shall be maintained less than or equal to 2 percent by volume when hydrogen is greater than 4 percent.

4.4 The CVCS Demineralizers are required to be bypassed prior to adding hydrazine to the CVCS **EXCEPT** a demineralizer with PRC-01.

4.5 All work performed in the Radiation Controlled Area shall be performed in accordance with the requirements of the Radiation Work Permit and ALARA program.

4.6 When aligning remotely operated valves (i.e., chain operated, reach rods, etc.), the position shall be verified by local valve position. This requirement may be waived by the Shift Manager in cases of significant radiation exposure, which occur in areas designated as high radiation areas or in areas deemed inaccessible by the Shift Manager.

4.7 Letdown flow should be maintained through the CVCS Demineralizers to maintain system cleanliness. Securing letdown during plant cooldown may result in high dose rates in the RHR System. The RP Supervisor and the Radiochemist shall be notified if letdown is to be secured.

4.7.1 Letdown orifices should not be changed during delithiation operations. If letdown flow has to be changed, then Chemistry should be notified so that the delithiation bed run time can be recalculated.

4.8 If a charging pump exhibits primary packing leakage symptoms as described below, then issue a PWO to Mechanical Maintenance Department to repack the pump.

4.8.1 Primary packing leakage of greater than 0.05 gpm: place on Plant Status Sheet and repack within 4 weeks.

4.8.2 Primary packing leakage of greater than 0.08 gpm: place on Plant Status Sheet and repack within 2 weeks.

4.8.3 Abnormally high airborne gas concentration in the Charging Pump Room.

4.9 If a charging pump exhibits secondary packing leakage symptoms as described below, then issue a PWO to Mechanical Maintenance Department to repack the pump.

4.9.1 Decreasing seal pot level that requires shiftly seal pot fills.

4.9.2 A steady stream of water leaking out any one of the plungers in the charging pump plunger well.

4.10 Temperature changes of the CVCS letdown will affect the ability of the in-service resin bed to retain boric acid. A temperature increase will cause a minor boration and a temperature decrease will cause a minor dilution. Reactor power should be closely monitored when changing letdown temperatures or changing resin beds.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	004		K6.22
	Importance Rating	2.6		

Knowledge of the effect of a loss or malfunction on the following CVCS components: Design minimum and maximum flow rates for letdown system.

Proposed Question: RO Question # 30

Given the following conditions:

- Unit 3 is at 100% power.
- An inadvertent letdown isolation occurred.
- The cause has been identified and corrected.
- PRZ level is 68% and rising.

Which one of the following describes (1) a requirement for re-establishing letdown, and (2) the maximum allowable letdown flow when letdown is in service?

A.	(1) Orifice Isolation Valves CV-3-200A/B/C must be open prior to opening Letdown Isolation valve LCV-3-460. (2) 120 gpm
B.	(1) Orifice Isolation Valves CV-3-200A/B/C must be open prior to opening Letdown Isolation valve LCV-3-460. (2) 165 gpm
C.	(1) Letdown Isolation Valve LCV-3-460 must be open prior to opening Orifice Isolation Valves CV-3-200A/B/C (2) 120 gpm
D.	(1) Letdown Isolation Valve LCV-3-460 must be open prior to opening Orifice Isolation Valves CV-3-200A/B/C (2) 165 gpm

Proposed Answer: C

A.	Incorrect. Plausible because part 2 is correct and because there is a sequence for opening the valves, but the interlock is the other way around
B.	Incorrect. Plausible for same reason as in Option A and part 2 is plausible because this would be the flow if all 3 orifices were placed in service, but administrative limit is 120 gpm
C.	Correct. Orifice valves are interlocked with LCV-3-460 in this manner. Design restrictions on demineralizer operation require the letdown flow rate to be maintained below 120 gpm.
D.	Incorrect. First part is correct and 2nd part is plausible for same reason as in option B

Technical Reference(s)	6902113 3-OP-047 P&Ls		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902113 obj 5		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 1
Page 1 of 1

ESTABLISH LETDOWN FLOW

1. **Verify B CCW Header Flow - NORMAL**

2. **Verify Letdown Orifice Isolation Valves - CLOSED**
   - **CV-3-200A**
   - **CV-3-200B**
   - **CV-3-200C**

3. **OPEN Letdown From Regen Heat Exchanger Isolation, CV-3-204**

4. **OPEN High Pressure Letdown Isolation From Loop B Cold Leg, LCV-3-460**

5. **Manually Control Low Pressure Letdown Controller, PCV-3-145, To Limit Pressure Spike When Opening Letdown Orifice Isolation Valves**

   Interlock <u>prevents</u> opening of orifice isolation valves before LCV-3-460 is opened

6. **Verify Adequate Charging Flow For Desired Letdown Flow**

7. **Verify Adequate CCW Flow For The Desired Letdown Flow**

8. **Open Letdown Orifice Isolation Valves To Establish Desired Letdown Flow**

9. **Manually Control Low Pressure Letdown Controller PCV-3-145, To Establish Pressure Of 250 To 300 Psig As Indicated On PI-3-145**

10. **Place Low Pressure Letdown Controller, PCV-3-145, In AUTOMATIC**

11. **Open Excess L/D and RCP Seal Return Isolation Valve, MOV-3-6386**

4.0 PRECAUTIONS/LIMITATIONS

<120 gpm to prevent resin channeling

4.1 Before changing system status, Technical Specifications should be consulted for system requirements for that plant mode.

4.2 Design restrictions on demineralizer operation require the letdown flow rate to be maintained below 120 gpm and the temperature of the water entering the inlet header to be less than 140 °F.

4.3 Explosive mixtures of hydrogen and oxygen concentration shall be avoided at all times. The oxygen concentration in the VCT shall be maintained less than or equal to 2 percent by volume when hydrogen is greater than 4 percent.

4.4 The CVCS Demineralizers are required to be bypassed prior to adding hydrazine to the CVCS **EXCEPT** a demineralizer with PRC-01.

4.5 All work performed in the Radiation Controlled Area shall be performed in accordance with the requirements of the Radiation Work Permit and ALARA program.

4.6 When aligning remotely operated valves (i.e., chain operated, reach rods, etc.), the position shall be verified by local valve position. This requirement may be waived by the Shift Manager in cases of significant radiation exposure, which occur in areas designated as high radiation areas or in areas deemed inaccessible by the Shift Manager.

4.7 Letdown flow should be maintained through the CVCS Demineralizers to maintain system cleanliness. Securing letdown during plant cooldown may result in high dose rates in the RHR System. The RP Supervisor and the Radiochemist shall be notified if letdown is to be secured.

4.7.1 Letdown orifices should not be changed during delithiation operations. If letdown flow has to be changed, then Chemistry should be notified so that the delithiation bed run time can be recalculated.

4.8 If a charging pump exhibits primary packing leakage symptoms as described below, then issue a PWO to Mechanical Maintenance Department to repack the pump.

4.8.1 Primary packing leakage of greater than 0.05 gpm: place on Plant Status Sheet and repack within 4 weeks.

4.8.2 Primary packing leakage of greater than 0.08 gpm: place on Plant Status Sheet and repack within 2 weeks.

4.8.3 Abnormally high airborne gas concentration in the Charging Pump Room.

4.9 If a charging pump exhibits secondary packing leakage symptoms as described below, then issue a PWO to Mechanical Maintenance Department to repack the pump.

4.9.1 Decreasing seal pot level that requires shiftly seal pot fills.

4.9.2 A steady stream of water leaking out any one of the plungers in the charging pump plunger well.

4.10 Temperature changes of the CVCS letdown will affect the ability of the in-service resin bed to retain boric acid. A temperature increase will cause a minor boration and a temperature decrease will cause a minor dilution. Reactor power should be closely monitored when changing letdown temperatures or changing resin beds.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	005		K2.03
	Importance Rating	2.7		

Knowledge of bus power supplies to the following: RCS pressure boundary motor-operated valves

Proposed Question: RO Question # 31

Given the following condition:

- Unit 4 is in Mode 3.

Which one of the following identifies (1) the power supply to RHR Isolation Valve MOV-4-750 and (2) the current status of power to MOV-4-750 in accordance with 4-GOP-503, Cold Shutdown to Hot Standby?

A.	(1) 4D MCC (2) energized
B.	(1) 4D MCC (2) de-energized
C.	(1) 4B MCC (2) energized
D.	(1) 4B MCC (2) de-energized

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible there are other safety related MOVs powered by 4D MCC. Part 2 is incorrect, but plausible if candidate confuses mode at which MOV is de-energized. The MOV is energized in mode 4 through mode 6.
B.	Incorrect. Part 1 is incorrect. Part 2 is correct.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect.
D.	Correct. MOV-4-750 is supplied by MCC 4B and is de-energized at power to prevent overpressurizing the RHR system when RCS temperature is >350°F

Technical Reference(s)	LP 6902107 LP 6902121A		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			NO
Learning Objective:	6902121A obj 8b		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

INITIALS
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5.17 Verify the SI Pump Recirc Phase Suct Stops are aligned as follows:

____ ____ 5.17.1 MOV-4-863A, Closed.

____ ____ 5.17.2 MOV-4-863B, Closed.

____ ____ 5.17.3 Open and lock breaker 40726, MOV-4-863A.

____ ____ 5.17.4 Open and lock breaker 40626, MOV-4-863B.

____ 5.18 **IF** 0-ADM-523, ASME Section XI Pressure Tests for Quality Group A, B, and C Systems/Components, is **NOT** required to be performed, **THEN** establish a hydrogen blanket in the VCT using 4-OP-047.1, VCT Gas Space Concentration Control, while continuing heat up (35-40 psig) **OR** as directed by Chemistry.

____ 5.19 **WHEN** steam generator steam production is increasing, **THEN** ensure Closed the steam generator atmospheric dumps, and coordinate with the Turbine Operator to throttle steam traps ST-1, 2, and 3 drains to atmosphere, as necessary, to heatup and pressurize.

CAUTION

The RHR System is required to be isolated from the RCS prior to reaching 350°F and 450 psig after a Pressurizer bubble has been formed.

NOTE

RHR loops shall be aligned for low head SI prior to reaching 350°F on the highest reading T_{HOT} OR T_{COLD} loop indicator.

____ 5.20 Initiate realignment of the Residual Heat Removal Loops for Low Head SI using 4-OP-050, Residual Heat Removal System, Section 6.0, Shutdown.

5.21 **IF** required, **THEN** perform the following Post-RHR Cooldown Operation Testing. (N/A if testing not performed.)

____ 5.21.1 Applicable sections of 4-OSP-041.17, RCS/RHR Loop Pressure Boundary Leak Test, as determined by the Engineering Department.

____ 5.21.2 4-OSP-050.7, RHR MOVs/System Pressure Interlock Test.

INIT Date/Time Started: ________________/______________

6.0 SHUTDOWN

6.1 Removing RHR from Cooldown Operation

CAUTION

If MOV-4-863A and/or MOV-4-863B and MOV-4-872 have been opened or used for Alternate RHR Cooldown Lineup, RHR temperature is required to be verified less than 200°F on TR-4-604 prior to closing MOV-4-863A, MOV-4-863B and/or MOV-4-872. [Commitment - Step 2.3.9]

6.1.1 Initial Conditions

____ 1. All applicable prerequisites listed in Section 3.0 are satisfied.

____ 2. The RHR Loop is ready to be isolated from the RCS and be removed from service as directed by 4-GOP-503, Cold Shutdown to Hot Standby.

____ 3. **IF** shutdown for a Refueling Outage and **NOT** already performed, **THEN** perform 4-OSP-050.8, RHR MOVs 750, 751, 862 and 863 Interlock Test. (Mark N/A if already performed.)

____ 4. **IF** MOV-4-863A and/or MOV-4-863B **AND** MOV-4-872 have been opened **OR** used for Alternate RHR Cooldown Lineup, **THEN** verify RHR temperature less than 200°F on TR-4-604 prior to closing MOV-4-863A, MOV-4-863B, and/or MOV-4-872. [Commitment - Step 2.3.9]

____ 5. A bubble has been established in the pressurizer using 4-OP-041.2, Pressurizer Operation.

6.1.2 Procedure Steps

1. Verify RHR Ltdn to CVCS, HCV-4-142, open.

____ 2. Verify open Letdown from Regen Hx Iso Vlv, CV-4-204.

3. Perform the following to prepare for expected reduction in letdown flow:

____ a. Place excess letdown in service per 4-OP-047, CVCS Charging and Letdown.

____ b. Open Low Pressure Letdown Controller, PCV-4-145, as necessary to establish desired letdown flow.

____ c. **IF** desired, **THEN** take manual control of TCV-4-144 by placing TC-4-144A in MAN. Throttle TCV-4-144 as needed to minimize the temperature effects on the demineralizers due to the rise in the letdown temperature.

INITIALS
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6.1.2 (Cont'd)

NOTE

When MOV-4-750 and/or MOV-4-751 is closed in the following step, a significant decrease in letdown flow will be observed. Manual actions may be necessary to control pressurizer level, or RCS pressure.

4. Close the following valves:

____ a. Verify RHR Pumps none running.

____ b. Loop 4A RHR Pump Suction Stop Vlv, MOV-4-750

____ c. Loop 4A RHR Pump Suction Stop Vlv, MOV-4-751

____ 5. Open **AND** lock Breaker 40731 to de-energize Loop 4A RHR Pump Suction Stop Valve, MOV-4-751. (N/A if 4-OSP-041.17, RCS/RHR Loop Pressure Boundary Leak Test, Subsection 7.1 is to be performed.)

____ 6. Open **AND** lock Breaker 40615 to de-energize Loop 4A RHR Pump Suction Stop Valve, MOV-4-750. (N/A if 4-OSP-041.17, RCS/RHR Loop Pressure Boundary Leak Test, Subsection 7.1 is to be performed.)

____ 7. Unlock **AND** open RHR Recirc Line Isolation Valve, 4-741A.

8. Ensure that the following valves are closed:

____ a. RHR Hx Outlet Flow, HCV-4-758

____ b. RHR Hx Bypass Flow, FCV-4-605

INPUT DATA FILE PTP.BL.134   MCC 4B (REACTOR AREA) (4B06)

CODE	HEATER BREAKER	SCHEMATIC LOCATION PNL SCHED.	EQUIP. NO. CKT DESC	EQUIPMENT AND OR LOCATION		BREAKER NUMBER
78		E-10SH1	PWR VITAL	MOV-4-750 (RHR INLET ISOL FROM RCS)		40615
78		E-10SH1	PWR VITAL	MOV-4-862B (RWST TO RHR HEADER STOP)		40616
78		E-10SH1	PWR VITAL	MOV-4-749B (RHR HX CC WTR ISOL)		40617
78		E-10SH1	PWR VITAL	MOV-4-1402 (MAIN STM BYPASS 4C)		40618
129		E-10SH1	4H6 NON-VITAL	CONTAINMENT ELEVATOR #4 (PC/M 86-101)		40619
347		E-28 SH 63A E-10SH1	4Y05A VITAL	CVT FOR INVERTER 4Y05	(DCR-TPE-93-517)	40620
597		E-10SH1	PWR VITAL	MOV-4-866B (SAFETY INJ LOOP B HOT LEG)	EC-DCR 273080	40621
78		E-10SH1	PWR VITAL	MOV-4-843B (BORON INJ TK SI LOOP B COLD LEG)		40622
78		E-10SH1	PWR VITAL	MOV-4-1421 (STM GEN FEED PP 4B DISCH)		40623
78		E-10SH1	PWR VITAL	MOV-4-1418 (NORMAL CONTAIN. COOLER CCW - OUTLET)		40624
347		E-10SH1	E16F VITAL	BATTERY ROOM A/C	(DCR-TPE-93-517)	40625
78		E-10SH1	PWR VITAL	MOV-4-863B (RECIRC SUPPLY TO HH SI PP & CONTAIN SPRAY PP'S RHR HEAT EXCHANGER OUTLET TO RWST)		40626

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	006		A4.08
	Importance Rating	4.2		

Ability to manually operate and/or monitor in the control room: ESF system, including reset

Proposed Question: RO Question # 32

Given the following conditions:

- A plant heatup is in progress in accordance with 3-GOP-503, Cold Shutdown to Hot Standby.
- RCS temperature is 520°F and rising.
- Pressurizer pressure is 2075 psig and rising.

Which one of the following completes the statement below?

The BLOCK LOW PRZ PRESS SI status light on VPA is __(1)__ .

The LOW PRZ PRESS SI BLOCKED status light on VPA is __(2)__ .

A.	(1) lit (2) lit
B.	(1) lit (2) NOT lit
C.	(1) NOT lit (2) lit
D.	(1) NOT lit (2) NOT lit

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible if candidate believes that the “pressure” Si signals are blocked on Tavg <543°F or if candidate believes that the signals must be RESET manually or if candidate believes signals RESET at 2100 psig vs 2000 psig. Part 2 is incorrect, but plausible for the same reasons above.
B.	Incorrect. Part 1 is incorrect. Part 2 is correct. This combination is plausible if candidate believes the permissive to block must be RESET manually, however the block permissive automatically RESETS.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect. This combination is plausible if candidate believes the blocked status RESETS automatically at 2000 psig, however the block permissive must be manually RESET.
D.	Correct. Both the permissive to block and the blocked status automatically RESET above 2000 psig.

Technical Reference(s)	3-GOP-305 3-GOP-503		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902163 obj 8		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

QA RECORD PAGE

INITIALS
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NOTE

The pressurizer low pressure and steam line high differential pressure safety injection signals will be unblocked automatically if RCS pressure increases to 2000 psig.

CAUTION

Technical Specifications require pressurizer pressure to be less than 2000 psig prior to blocking the pressurizer low pressure and the steamline high differential pressure safety injection signals.

5.5 **WHEN** pressurizer pressure decreases to less than 2000 psig, **THEN** block the pressurizer low pressure and steamline high differential pressure safety injection signals as follows:

____ 5.5.1 Verify the BLOCK LO PRZ PRESS SI plant status light is ON.

____ 5.5.2 Momentarily place both Safety Injection Block switches to the BLOCK position.

____ 5.5.3 Verify the LOW PRZ PRESS SI BLOCKED plant status light is ON.

CAUTION

RCS pressure shall be reduced to less than 2000 psig and safety injection steam line high differential pressure signal blocked before steam generator pressures are reduced to less than 500 psig. This will prevent an inadvertent safety injection that will occur when steam generator pressure decreases to 485 psig if the safety injection signal is unblocked.

____ 5.6 Verify the LOW PRZ PRESS SI BLOCKED plant status light is ON prior to steam generator pressures decreasing to less than 500 psig.

As RCS pressure rises above 2000 psig, the low-pressurizer-pressure SI signal is automatically unblocked and the BLOCK LO PRZ PRESS SI and LOW PRZ PRESS SI BLOCKED status lights are extinguished

INITIALS
CK'D VERIF

5.39.3 **IF** level is less than 50%, **THEN** add feedwater to affected steam generator(s) as follows:

1. Ensure applicable steam generator feedwater bypass isolation valve is Open:

____ • POV-3-477, 3A F/W Bypass Isolation

____ • POV-3-487, 3B F/W Bypass Isolation

____ • POV-3-497, 3C F/W Bypass Isolation

2. Manually control feedwater by using any of the following Feedwater Bypass Flow Control Valves: (N/A substeps not used)

____ a. FCV-3-479 for S/G A

____ b. FCV-3-489 for S/G B

____ c. FCV-3-499 for S/G C

CAUTION

Do NOT exceed RCS pressure of 2000 psig until Steam Generator pressures are greater than 585 psig to prevent an inadvertent SI signal.

____ ____ 5.40 **WHEN** RCS pressure reaches 2000 psig, **THEN** verify that the Low Prz Press SI Blocked bistable light on VPA is Out.

NOTE

If returning from a Refueling Outage, cold shutdown boron concentration shall be maintained as required by the Reactor Engineer.

____ 5.41 **WHEN** RCS temperature reaches 541°F, **THEN** the RCS may be diluted to the critical boron concentration with Shift Manager permission. (N/A if cold concentration is to be maintained.)

____ 5.42 **WHEN** RCS temperature (Tavg) reaches 543°F, **THEN** verify that the LO Tave SI Blocked status light on VPA is Out.

____ 5.43 **WHEN** RCS pressure is between 2225 and 2235 psig **AND** Attachment 15 (if applicable) is complete, **THEN** establish Auto pressure control using 3-NOP-041.02, Pressurizer Operation.

As RCS pressure rises above 2000 psig, the low-pressurizer-pressure SI signal is automatically unblocked and the BLOCK LO PRZ PRESS SI and LOW PRZ PRESS SI BLOCKED status lights are extinguished

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	007		A4.10
	Importance Rating	3.6		

Ability to manually operate and/or monitor in the control room: Recognition of leaking PORV/code safety

Proposed Question: RO Question # 33

Given the following conditions:

- Unit 3 is at 100% power.
- RCS pressure is 2235 psig and stable.
- RCS leakage has risen by 0.5 gpm.

Which one of the following indications can SOLELY be used to distinguish a leaking PRZ PORV from a leaking PRZ Safety?

A.	PRZ relief tank level, LI-3-470
B.	PRZ relief tank pressure, PI-3-472
C.	PRZ relief line temperature, TI-3-463
D.	PRZ PORV/safety acoustic monitor, S-3-6303

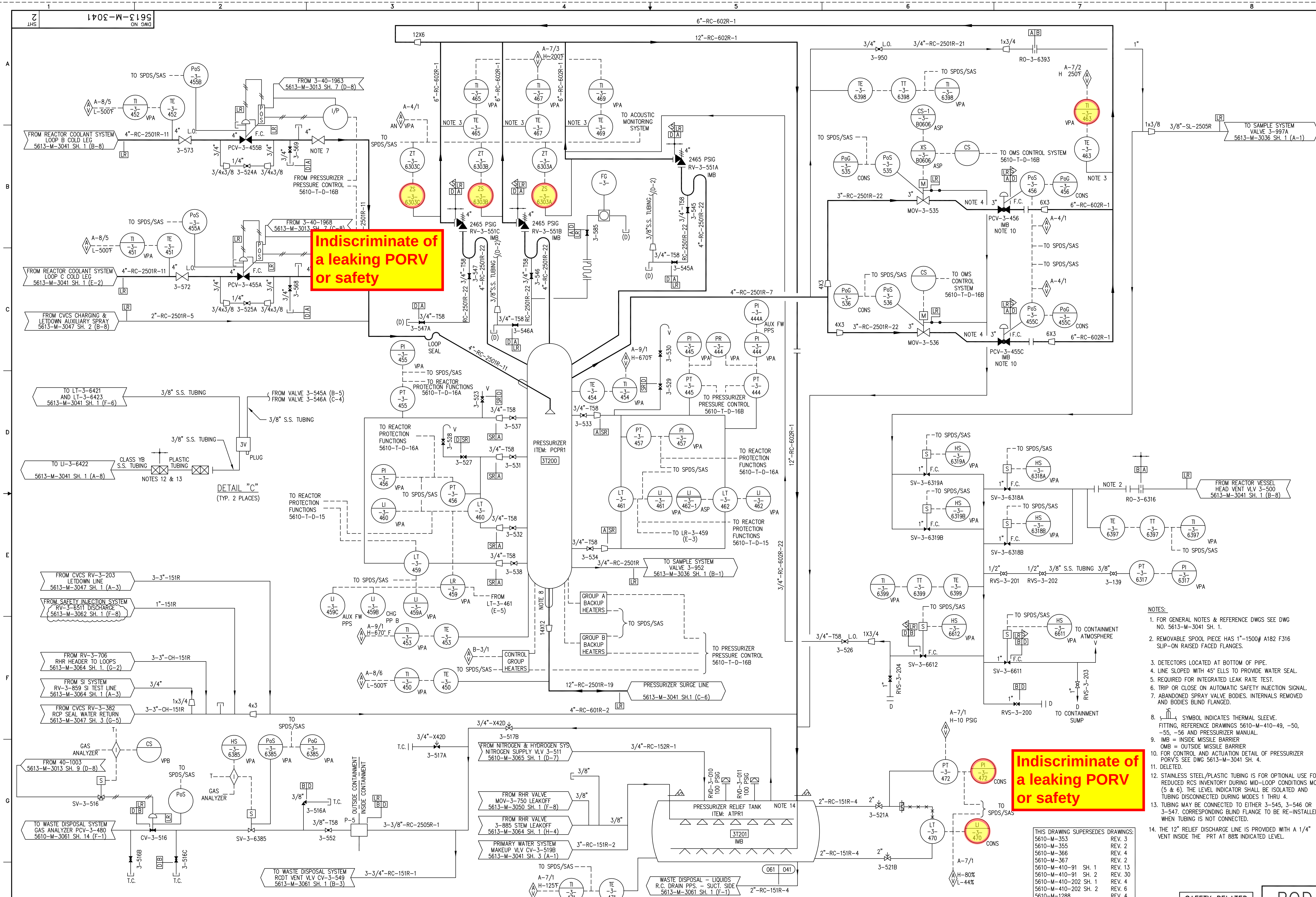
Proposed Answer: C

A.	Incorrect. Plausible because PRT level may rise for a leaking PORV / safety. This may be chosen over answer B when candidate believes that pressure doesn’t rise as much as level due to the PRT sparger.
B.	Incorrect. Plausible because PRT pressure may rise for a leaking PORV / safety; also candidate believes the PORV relieves to the PRT and the safety valve to the RCDT or the containment floor.
C.	Correct. All other options are indicative of a PRZ safety valve / PORV leak IAW 3-ONOP-041.5, however only the PRZ relief line temperature can be used to identify a leaking PORV based on the sensor location on the piping when a bubble is present in the PRZ.

D.	Incorrect. Plausible if candidate believes the acoustic monitor is selectable (PORV /safety) or believes there is indication for both PORVs and safety valves either on the rack or on DCS.		
Technical Reference(s)	3-ONOP-041.5		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902204 obj 1		(As available)
Question Source:	Bank	X	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			
PTN Bank question 69022040101			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12	**Check PZR Heaters Operable**	Perform the following:
	a. Check PZR Heaters ON	a. Perform the following: * Verify PZR Control Group Heater Distribution Panel B-11 Breakers, West Electrical Penetration Room, CLOSED. * Verify PZR Backup Group A Heater Distribution Panel B-12 Breakers, West Electrical Penetration Room, CLOSED. * Verify PZR Backup Group B Heater Distribution Panel B-13 Breakers, West Electrical Penetration Room, CLOSED.
	b. Check PZR Htrs capable of maintaining pressure	b. Dispatch an operator to perform Attachment 1, Pressurizer Heater Output Worksheet, to determine heater output. c. Notify the Electrical Department.
13	**Check If A PORV Is Leaking** * PZR relief line temperature, TI-3-463 - INCREASING or at the saturation temperature associated with the PZR relief tank pressure according to Attachment 2 * PZR relief tank level, LI-3-470 - INCREASING * PZR relief tank temperature, TI-3-471 - INCREASING **OR** above ambient temperature for containment conditions * PZR relief tank pressure PI-3-472 - INCREASING * PZR PORV/Safety Acoustic Monitor - LEDs LIT	Go to Step 15.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15	**Determine If A Leaking PZR Safety Is Causing Pressure To Decrease**	
	a. Check if a PZR Safety is leaking	a. Go to Step 16.
	* PZR safety line temperature, TI-3-465 - INCREASING or at saturation temperature associated with the PZR relief tank pressure per Attachment 2	
	* PZR safety line temperature, TI-3-467 - INCREASING or at saturation temperature associated with the PZR relief tank pressure per Attachment 2	
	* PZR safety line temperature, TI-3-469 - INCREASING or at saturation temperature associated with the PZR relief tank pressure per Attachment 2	
	* PZR relief tank level, LI-3-470 - INCREASING	
	* PZR relief tank temperature, TI-3-471 - INCREASING	
	* PZR relief tank pressure, PI-3-472 - INCREASING	
	* PZR PORV/Safety Acoustic Monitor - LEDs LIT	
	b. Refer to Technical Specifications for a leaking PZR SAFETY	
16	**Determine If RCS Leakage Is Causing Pressure To Decrease**	
	• Monitor RCS Leakage using 3-OSP-041.1, RCS LEAK RATE CALCULATION	
17	**Check Pressurizer Pressure Decreasing**	Go to Step 20.



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5613-M-3041 SH. 1.
2. REMOVABLE SPOOL PIECE HAS 1"-1500# A182 F316 SLIP-ON RAISED FACED FLANGES.
3. DETECTORS LOCATED AT BOTTOM OF PIPE.
4. LINE SLOPED WITH 45° ELLS TO PROVIDE WATER SEAL.
5. REQUIRED FOR INTEGRATED LEAK RATE TEST.
6. TRIP OR CLOSE ON AUTOMATIC SAFETY INJECTION SIGNAL.
7. ABANDONED SPRAY VALVE BODIES. INTERNALS REMOVED AND BODIES BLIND FLANGED.
8. SYMBOL INDICATES THERMAL SLEEVE. FITTING, REFERENCE DRAWINGS 5610-M-410-49, -50, -55, -56 AND PRESSURIZER MANUAL.
9. IMB = INSIDE MISSILE BARRIER
   OMB = OUTSIDE MISSILE BARRIER
10. FOR CONTROL AND ACTUATION DETAIL OF PRESSURIZER PORV'S SEE DWG 5613-M-3041 SH. 4.
11. DELETED.
12. STAINLESS STEEL/PLASTIC TUBING IS FOR OPTIONAL USE FOR REDUCED RCS INVENTORY DURING MID-LOOP CONDITIONS MODES (5 & 6). THE LEVEL INDICATOR SHALL BE ISOLATED AND TUBING DISCONNECTED DURING MODES 1 THRU 4.
13. TUBING MAY BE CONNECTED TO EITHER 3-545, 3-546 OR 3-547. CORRESPONDING BLIND FLANGE TO BE RE-INSTALLED WHEN TUBING IS NOT CONNECTED.
14. THE 12" RELIEF DISCHARGE LINE IS PROVIDED WITH A 1/4" VENT INSIDE THE PRT AT 88% INDICATED LEVEL.

THIS DRAWING SUPERSEDES DRAWINGS:		
5610-M-353		REV. 3
5610-M-355		REV. 2
5610-M-366		REV. 4
5610-M-367		REV. 2
5610-M-410-91	SH. 1	REV. 13
5610-M-410-91	SH. 2	REV. 30
5610-M-410-202	SH. 1	REV. 4
5610-M-410-202	SH. 2	REV. 6
5610-M-1288		REV. 4
5610-M-1310		REV. 2

SAFETY RELATED

POD

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4501 SH. 1 REV. 90
FPL POD 5610-T-E-4517 SH. 2 REV. 8

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
43	7/13/12	ISSUED AS-BUILT PER EC 246977 (PC/M 09-083).	RV	BB	JN	JD	37	3-26-06	ISSUED AS-BUILT PER PC/M 04-012 AND INCORP. CRN M-11529.	RH	SB	BSG	JTL
42	7/12/12	ISSUED AS-BUILT PER EC 246924.	RH	RV	PSB	JD	36	7-11-05	ISSUED AS-BUILT PER CRN-M-11395 (PC/M 03-106).	RH	RRW	AAP	PRB
41	7/1/12	ISSUED AS-BUILT PER EC 246924 (PARTIAL).	AFG	SB	–	BP	35	12-2-04	ISSUED AS-BUILT PER PC/M 04-157.	RH	RV	AAP	JMM
40	5/24/12	ISSUED AS-BUILT PER EC 242470.	AFG	BB	–	AC	44	8-02-12	ISSUED AS-BUILT PER EC-DCR 277023.	RH	RV	–	JG
39	9/28/09	ISSUED AS-BUILT PER CRN-I-5092 (PC/M 09-070).	RV	RH	–	LG	0	2-15-91	THIS DRAWING CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	TAM	HLG	AM/RAS	LRB
38	9/25/06	ISSUED AS-BUILT PER CRN-M-11759 (PC/M 03-106)	RH	RV	BSG	JTL	—	—	AND ISSUED INTO THE FPL SYSTEM PER DCR-TPM-91-043.	—	—	—	—

FPL

TURKEY POINT NUCLEAR UNIT 3

P & I D

REACTOR COOLANT SYSTEM

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3041
SHEET 2
SYS 041
REV 44

30x42 E SIZE 05/04/93

5613-M-3041_SH 2.DWG

Item: 1.1.25.4.1.1

Question 33 original

69022040101;

Given the following conditions:

- Unit 3 is at 100% steady-state conditions
- A-7/1, PRT HI/LO LEVEL HI PRESS/TEMP, is in alarm
- A-7/2, PZR PORV HI TEMP, is in alarm
- A-7/3, PRZ SAFETY VALVE A/B/C HI TEMP, is in alarm
- Pressurizer level is 54%
- Pressurizer pressure is 2260 psig and lowering, as indicated on pressure transmitters PT-3-444 and PT-3-445

Based on these conditions, which one of the following identifies the appropriate action(s) to be taken?

A) Check the pressurizer PORVs closed.

B) Trip the reactor and enter 3-EOP-E-0 (Reactor Trip or Safety Injection).

C) Place the pressurizer spray valves (PCV-3-455A, 455B) in manual.

D) Take manual control of pressurizer pressure, using PC-3-444J (pressurizer pressure controller).

Item Classification: Knowledge
Item difficulty: 0.50
Keywords: 2.4.4, RCO, LOCT, 2.4.11, 2.4.31
Item weight: 10
Points required for mastery: 1
Correct alternative(s): A
Judging values of alternatives:
A=1 B=-1 C=-1 D=-1
Memo Field: REFERENCES: 3/4-ONOP-041.5 step 2
HBR RCO 7/94 Q 22

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	008		K4.01
	Importance Rating	3.1		

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump

Proposed Question: RO Question # 34

Given the following conditions:

- Unit 3 is operating at 100% power.
- The 3B CCW Pump is in operation.
- The following alarms are received:
- H 8/2, CCW PP A/B/C/ MOTOR OVERLOAD
- H 8/1, CCW PP A/B/C TRIP
- H 8/3, CCW HEADER LO PRESS

Which one of the following identifies the subsequent automatic action?

A.	3A CCW pump will start immediately after initiating signal
B.	3A CCW pump will start after at least 10 seconds
C.	3C CCW pump will start immediately after initiating signal
D.	3C CCW pump will start after at least 10 seconds

Proposed Answer: B

A.	Incorrect. 3A CCW pump will be the first to start, but there will be a time delay of 10 seconds prior to starting. Some PTN pumps start immediately on low pressure, or trip of another pump.
B.	Correct. 3A CCW pump will be the first to start, but there will be a time delay of 10 seconds prior to starting. Some PTN pumps start immediately on low pressure, or trip of another pump.

C.	Incorrect. Plausible for same reasons as A. 3C CCW pump has a longer time delay so that it won't start unless pressure is not restored within 30 seconds. Normally the 3D 4kV Bus is aligned to 3B, the candidate assumes the backup pump 3C (power supply) would start.		
D.	Incorrect. Plausible for same reasons as A. 3C CCW pump has a longer time delay so that it won't start unless pressure is not restored within 30 seconds. Normally the 3D 4kV Bus is aligned to 3B, the candidate assumes the backup pump 3C (power supply) would start.		
Technical Reference(s)	LP 6902140		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902140 obj 5		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

CAUSES:
1. Trip of CCW pump
2. System rupture

H8/3

CCW HEADER LO PRESS

DEVICE: PC-3-611

SETPOINT: 73 psig at CCW pump discharge header.

LOCATION: Common CCW heat exchanger inlet header near 3B CCW Heat Exchanger

ALARM CONFIRMATION

1. **CHECK** the following:
   - CCW pump breaker indications on VPB
   - CCW pump motor ammeters on VPB
   - CCW header flow indications on VPB
   - CCW Surge Tank level on VPB

OPERATOR ACTIONS

NOTE

This corresponds to 77 psig at PC-3-611 and PI-3-612 due to 4 psi static head from header elevation.

1. **ENSURE** Standby CCW pump starts at 73 psig CCW pump discharge header pressure with a time delay of 10/20/30 seconds for A/B/C pumps, respectively.)
2. **REFER TO** 3-ONOP-030, Component Cooling Water Malfunction.
3. **REFER TO** TS 3.7.2 for any additional required actions.

REFERENCES:
1. 5613-E-25, Sh 2A, 2B, 2C
2. 5613-M-3030, Sh 1
3. 5610-T-L1, Sh 24D, 144A, 144B
4. Tech Spec 3.7.2
5. PC/M 96-092, Addition of U-3 CCW Head Tank

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	010		K2.02
	Importance Rating	2.5		
Knowledge of bus power supplies to the following: Controller for PRZ spray valve				
Proposed Question: RO Question # 35				

Given the following conditions:

- All systems are in automatic when a vital instrument AC bus is lost.
- PRZ spray control is as follows :

Which one of the following completes the statement below?

120V Vital Instrument Bus power __(1)__ is lost to the PCV-3-455B controller.

Due to the loss of power, PCV-3-455B LOOP B spray valve will __(2)__ .

A.	(1) 3P06 (2) fail CLOSED

B.	(1) 3P06 (2) remain AS IS
C.	(1) 3P09 (2) fail CLOSED
D.	(1) 3P09 (2) remain AS IS

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible if candidate believes RED (3P06) on placard is the fail state for going to manual based on the location of the placard (placard is on the right side therefore failure is to the right side). Candidate may also confuse loss of 3P06 with 3P09 given that a loss of 3P06 causes the spray valve to lock-up in automatic. Part 2 is incorrect, but plausible if the candidate believes the solenoid powered instrument air valves fails open and vents air slowly driving the valve to its fail closed position (fails closed on loss of air).
B.	Incorrect. Part 1 is incorrect. Part 2 is correct. This combination is plausible if candidate believes 3P06 has failed and the valve fails as is.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect. This combination is plausible if candidate believes 3P09 failed and the valves fails closed on loss of air.
D.	Correct. Part 1 is correct. Part 2 is correct. The PRZ spray controller fails to manual. IAW 3-ONOP-003.9, a loss of vital instrument panel 3P09 would cause this to occur.

Technical Reference(s)	3-ONOP-003.6, 3-ONOP-003.9		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	LP 6902260 obj 2		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	7
	55.43	
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		
Comments:		

1.0 **PURPOSE**

This procedure provides instructions to be followed upon receipt of Loss of 120V Vital Instrument Panel 3P06.

2.0 **SYMPTOMS OR ENTRY CONDITIONS**

2.1 Indications

2.1.1 Power Range N-41 Failure (NIS Racks Channel I Lights Out)

2.1.2 Loss of Channel I Vital Instrumentation/Indications

2.1.3 Transfer of Feedwater Control to Backup Controller for Steam Generator A

2.1.4 Loss of Power to Pressurizer pressure control Auto/Manual Station (auto lockup)

2.1.5 Loss of Power to the Pressurizer Spray Valve Auto/Manual Station (auto lockup)

2.1.6 Loss of Pressurizer Heaters (Control and Backup)

2.1.7 Isolation of CVCS Letdown Flow

2.1.8 Loss of Power to Pressurizer Level Auto/Manual Station (auto lockup)

2.1.9 Loss of Power to 3A Charging Pump Auto/Manual Station (auto lockup)

2.1.10 PCV-3-456 Auto Open Disabled (if in OMS LOW PRESSURE OPS)

2.2 Alarms

2.2.1 F 1/2, VITAL AC BUS INVERTER TROUBLE

2.2.2 B 6/5, POWER RANGE LOSS OF DETECTOR VOLTAGE

2.2.3 B 7/1, NIS/RPI ROD DROP ROD STOP

2.2.4 C 6/1, SG A LEVEL DEVIATION CNTRL TROUBLE

2.2.5 A 1/5 RCP CBO HI FLOW

2.2.6 A 6/4, RCP P2 SEAL HI PRESSURE

ENCLOSURE 1
(Page 1 of 5)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON LOSS OF 3P06

FUNCTIONS, Operating

Lock up of Pressurizer Pressure Controllers causing spray valves to stay as is
FCV-3-478, A Feedwater Control Valve - On Backup Controller
Lose Auto and Manual 3A Charging Pump Control causing Auto Lock-up
Lose Auto Speed Control of 3B and 3C Charging Pumps
Lose the Auto Makeup Control to the Volume Control Tank
Lose power to Control Relay from MOV-3-115C which opens LCV-3-115B
Letdown Isolation
Pressurizer heaters de-energize
Lose Auto and Manual control of PCV-3-145, Letdown Pressure Controller
Loss of 3B Diesel Load Sequencer, 3C23B-1 deenergized
Lose AMSAC A Processor
Lose the Ability to Block the Source Range Trip
Lose Feedwater Isolation signal (Reactor Trip with Tavg ≤554°F)
Loss of power to hand/auto station for CV-3-1607 which fails closed

NOTES

- *The following conditions exist which affect Pressurizer Pressure control:*
  - *Pressurizer Pressure Controller PC-444J - AUTO LOCKUP*
  - *PZR Spray Valve Controllers - AUTO LOCKUP*
  - *PZR heaters deenergized*
  - *Letdown isolation*
  - *3A charging pump - AUTO LOCKUP*
  - *3B AND 3C Charging pump loss of auto speed control*

1.0 **PURPOSE**

1.1 This procedure provides instructions to be followed upon Loss of 120V Vital Instrument Panel 3P09

2.0 **SYMPTOMS OR ENTRY CONDITIONS**

2.1 Indications

2.1.1 Power Range N-44 Failure (NIS Racks Channel IV Lights Out)

2.1.2 Loss of Channel IV Vital Instrumentation/Indications

2.1.3 Loss of Power to Backup Controllers for Feedwater Control to Steam Generator A, B, C.

2.1.4 Loss of Pressurizer Heaters (Control and Backup)

2.1.5 Isolation of CVCS Letdown Flow

2.1.6 Transfer of all Charging Pump Controllers from Automatic to Manual

2.1.7 Transfer of Pressurizer Pressure Controller from Automatic to Manual

2.1.8 Transfer of Pressurizer Spray Valve Controllers from Automatic to Manual

2.1.9 PORV-455C Auto Open Disabled (if in Normal Ops)

2.2 Alarms

2.2.1 F 1/2, VITAL AC BUS INVERTER TROUBLE

2.2.2 B 6/5, POWER RANGE LOSS OF DETECTOR VOLTAGE

2.2.3 B 7/1, NIS/RPI ROD DROP ROD STOP

2.2.4 G 9/1, SI PUMP 3A LO SUCTION PRESSURE

2.2.5 UNIT 4: G 9/1, SI PUMP 3A LO SUCTION PRESSURE

2.2.6 C 6/1, SG A LEVEL DEVIATION/TROUBLE

2.2.7 C 6/2, SG B LEVEL DEVIATION/TROUBLE

2.2.8 C 6/3, SG C LEVEL DEVIATION/TROUBLE

ENCLOSURE 1
(Page 1 of 3)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON FAILURE OF VITAL INSTRUMENT PANEL 3P09

FUNCTIONS, OPERATING

Lose Auto Rod Control, due to loss of TM-408
Lose Backup Controllers for Stm Gen A,B, and C
LT-3-112 VCT Level Transmitter
Letdown Isolation Occurs
Lose Pressurizer Auto Control of spray valves
Lose Pressurizer Heaters
Lose Steam Dumps to the Condenser
Lose Auto speed control of 3A, 3B, 3C charging pumps
Lose automatic operation of PORV PCV-3-455C
Disarms AMSAC due to loss of PT-447 (after six minute time delay)
LP Heater 2B Dump Valve Fails Open
LP Heater 2A Dump Valve Fails Open
LP Heater 1B Dump Valve Fails Open
TC-3-607A CCW Ht Exchanger Outlet Temp.
Lose R-11/12 Due to loss of power to SV-3-2911/2912
Loss of power to the Rod Insertion Limit Monitors (TR-309A - TR-309D)
Lose sample flow to Cold Chem Lab
Loss of Containment Evacuation Alarm
Loss of Feedwater Isolation Signal after reactor trip at 554°F
Loss of Megawatt Recorder Display
Possible loss of power to the hand/auto station for CV-3-1608, if aligned to 3P09 and would fail closed.

NOTE

The following conditions exist which affect Pressurizer Pressure control:

- *PZR Spray Valves -Lose auto control*
- *PZR heaters - Lose auto control*
- *Letdown isolation*
- *3B and 3C Charging pump loss of auto speed control*
- *PORV PCV-3-455C, lose automatic operation (if OMS in NORMAL OPS)*
- *PCV-3-145, MANUAL control ONLY*

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	012		K1.04
	Importance Rating	3.2		

Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems: RPIS

Proposed Question: RO Question # 36

Given the following conditions:

- Unit 3 is stable at 75% power.
- Control bank D is at 210 steps.

Which one of the following completes the statements below?

The Tech Spec maximum allowed rod position misalignment from group step counter demand is __(1)__ steps.

If two C bank control rods drop into the core (one dropped rod RPI is at 2 steps and the other dropped rod RPI indicates 4 steps), the crew will __(2)__ .

A.	(1) 18 (2) trip the reactor
B.	(1) 12 (2) trip the reactor
C.	(1) 18 (2) stabilize the plant at power
D.	(1) 12 (2) stabilize the plant at power

Proposed Answer: A

A.	Correct. Part 1 is correct. IAW tech specs, <90% maximum mismatch is 18 steps. Part 2 is correct. IAW ARP, IF 2 or more control rods have dropped (rod bottom lights turn on at 20 steps from bottom), THEN: TRIP the reactor.
B.	Incorrect. Part 1 is incorrect, but plausible if candidate confuses with >90% power 12 step mismatch criteria. Part 2 is correct.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible because if only one rod has dropped the crew will stabilize the plant and withdrawal the rod IAW 3-ONOP-028.3, Dropped RCC. Also plausible if candidate dismisses rod bottom light criteria and considers rods not fully dropped (RPIs not at zero) therefore trip is not required.
D.	Incorrect. Same reasons as B and C.

Technical Reference(s)	LP 6902106		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902106 no specific objective		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

CAUSES:
1. One or more control rod below rod bottom trip setpoint
2. NI dropped rod signal from any PR channel
3. PR NI malfunction

B7/1

NIS/RPI
ROD DROP
ROD STOP

DEVICE:	SETPOINT:	LOCATION:
RPI: Software	Any rod between 0 and 20 steps	DCS
NIS: • Dropped rod circuit NM-311 in each A PR drawer	Rate of flux decrease of 5% in 5 seconds by any PR channel	NI Racks 59, 60, 61, and 62

NOTE

Bypass logic will block alarm and function for banks B, C, and D when respective bank is less than 35 steps from bottom.

ALARM CONFIRMATION

1. **CHECK** for any RPI or rod bottom lights on console indicating one or more dropped control rods.
2. **CHECK** for any PR channel indicating instrument malfunction, loss of power supply, or blown fuses.

OPERATOR ACTIONS

1. **ENSURE** auto rod withdrawal block has occurred (Auto rod withdrawal capability currently disabled).

CAUTION

If this procedure was entered as a result of performing either 3-PTP-028.2, Rod Position Indication System Replacement Testing, Phase 2, Mode 3 Tests, or 3-SMI-028.03, RPI Hot Calibration, CRDM Stepping Test, and Rod Drop Test, and the RCS boron concentration is greater than or equal to test requirements, the following two operator actions shall **NOT** be performed.

2. IF 2 or more control rods have dropped, THEN:
   A. **TRIP** the reactor.
   B. **PERFORM** 3-EOP-E-0, Reactor Trip or Safety Injection.
3. IF one control rod has dropped, THEN **PERFORM** 3-ONOP-028.3, Dropped RCC.
4. IF caused by loss of Vital AC, THEN **RESET** Dropped Rod/Rod Stop Bistables on Power Range Drawers.

REFERENCES:
1. FPL Logic Diagram, 5610-T-L1, Sheet 21
2. Tech Spec Section 3/4.2, 3/4.1.3
3. PC/M 09-006

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within the Allowed Rod Misalignment between the Analog Rod Position Indication and the group step counter demand position within one hour after rod motion. The Allowed Rod Misalignment shall be defined as:

a. for THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 18 steps, and

b. for THERMAL POWER greater than 90% of RATED THERMAL POWER, the Allowed Rod Misalignment is ± 12 steps.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps and THERMAL POWER greater than 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER and confirm that all indicated rod positions are within the Allowed Rod Misalignment, or

   3. Be in HOT STANDBY within the following 6 hours.

c. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 18 steps and THERMAL POWER less than or equal to 90% of RATED THERMAL POWER, within 1 hour either:

   1. Restore all indicated rod positions to within the Allowed Rod Misalignment, or

   2. Be in HOT STANDBY within the following 6 hours.

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* See Special Test Exceptions 3.10.2 and 3.10.3.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	013		K5.01
	Importance Rating	2.8		

Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Definitions of safety train and ESF channel

Proposed Question: RO Question # 37

Given the following conditions:

- Unit 3 is at 100% power.
- 3-SMI-063.01A, Train A Safeguards Matrix Logic Test, is in progress.
- Blue channel STM HEADER PRESSURE, PT-3-466, fails HIGH.
- The test is stopped and all test switches are returned to normal.
- I&C wants to open the associated channel rack.

Which one of the following completes the statements below?

The ___(1)___ PROTECTION RACK OPEN annunciator is expected to alarm when I&C performs the investigation.

If one of the remaining steam header pressure protection transmitters subsequently fails LOW, the 3A train sequencer ___(2)___ start safeguards equipment.

A.	(1) CHANNEL II (2) will
B.	(1) CHANNEL III (2) will
C.	(1) CHANNEL II (2) will NOT
D.	(1) CHANNEL III (2) will NOT

Proposed Answer: D

A.	Incorrect. Both parts incorrect. See reasoning in B and C.
B.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate confuses sequencer trains with ESF channels. Candidate may also confuse the header higher than S/G pressure DP signal by thinking the SG pressures should be higher.
C.	Incorrect. Part 1 is incorrect, but plausible if student confuses hi dp with high steam flow which only has 2 inputs, one from channel III and one from channel II. Part 2 is correct.
D.	Correct. Part 1 is correct. STM HEADER PRESSURE, PT-3-466 provides RPS protection on channel 3. Part 2 is correct. The reactor will not trip. Only 1 of 3 steam header higher than steam generator channels are made up and Train A sequencer is not instrument channel dependent.

Technical Reference(s)	3-ARP-097-CR.C 1/4 5613-T-L-1 5610-T-L-1 sh 19	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:	N

Learning Objective:	6902163 obj 7	(As available)

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	

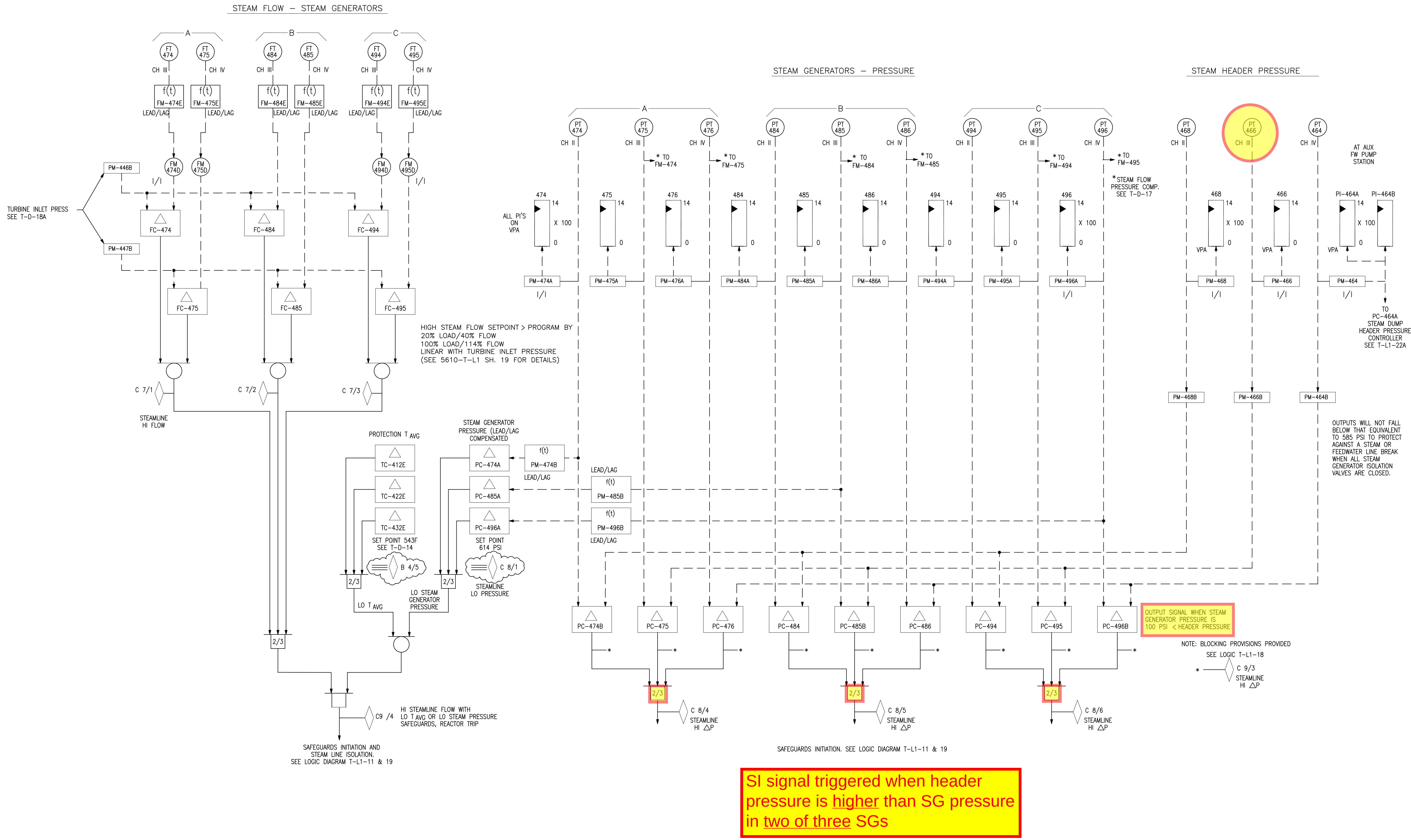
Question History:	Last NRC Exam:	2009	

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:



REV	DATE	REVISION	BY	CH	APP	APP
27	7/15/14	ISSUED AS-BUILT PER EC 281974.	RH	BB	–	AFG
26	06-01-13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-06-001).	RV	JM	–	TK
25	05-30-13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-05-129) PARTIAL.	AFG	RV	–	JLR
24	05-20-13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-05-102) PARTIAL.	RV	RH	–	PMB
23	05-18-13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-05-094) PARTIAL.	RV	RS	–	GD
22	05-14-13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-05-065) PARTIAL.	RV	RS	–	TK
21	05-09-13	ISSUED AS-BUILT PER EC'S 246974 (ITOP 13-05-015), 247049 (ITOP'S 13-04-155 & 13-05-047) PARTIAL.	RV	BB	–	TK
20	04-18-13	ISSUED AS-BUILT PER EC 247048 (ITOP 13-04-136).	RV	RH	–	PMB
19	04-12-13	ISSUED AS-BUILT PER EC 247048 (PARTIAL) (ITOP 13-04-109).	RV	BB	–	PMB
18	03-22-13	ISSUED AS-BUILT PER EC 247048 (PARTIAL) (ITOP 13-03-223).	RV	BB	–	PMB
17	2-10-13	ISSUED AS-BUILT PER EC 247048 (PARTIAL) (ITOP 13-02-049).	RH	RL	–	BEB
16	2-7-13	ISSUED AS-BUILT PER EC 247048 (PARTIAL) (ITOP 13-02-028).	RH	AFG	–	SB

SAFETY RELATED | POD

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

CONTROL SYSTEM DIAGRAM

STEAM BREAK PROTECTION

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-D-18B | SYS – | SHEET: 1 | REV 27

5610-T-D-18B_SH 1.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	022		2.2.12
	Importance Rating	3.7		

Equipment Control: Knowledge of surveillance procedures.

Proposed Question: RO Question # 38

Given the following conditions:

- Unit 3 is at 100% power.
- 3-OSP-055.1, Emergency Containment Cooler Operability Test, is in progress.

Which one of the following completes the statement below?

In accordance with 3-OSP-055.1, a minimum of __(1)__ CCW Heat Exchangers must be in service to prevent exceeding the __(2)__ during an ECC test.

A.	(1) two (2) 5500 gpm individual ECC coil flow rate limit
B.	(1) two (2) 6840 gpm individual CCW heat exchanger flow rate limit
C.	(1) three (2) 5500 gpm individual ECC coil flow rate limit
D.	(1) three (2) 6840 gpm individual CCW heat exchanger flow rate limit

Proposed Answer: D

A.	Incorrect. Both incorrect. See B and C reasons.
B.	Incorrect. Part 1 is incorrect, but plausible if candidate believes 2 HXs must be in service to run the test as when we split CCW headers for other tests. Part 2 is correct.

C.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible since it is a caution / note in 3-OSP-055.1 but for different reasons.		
D.	Correct. Three CCW HXs must be in service IAW 3-OSP-055.1 this prevents exceeding individual CCW heat exchanger flow rate limits.		
Technical Reference(s)	3-OSP-055.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

4.2 3A Emergency Containment Cooler Test

4.2.1 3A ECC Test Performance

NOTE

- Inservice Testing (IST) of CCW valves is performed during quarterly ECC Fan testing.
- H 9/5 – RCP MOTOR BEARING COOLING WATER LOW FLOW and other Component Cooling Water annunciators may alarm while performing this procedure.

CAUTION

Three CCW Heat Exchangers shall be in service to prevent exceeding 6840 gpm individual CCW Heat Exchanger flow rate, above which could cause heat exchanger damage. During performance of this test, CCW flow rates will change.

1. **ENSURE** Component Cooling Water System operating with all three CCW Heat Exchangers in service. _______

2. **MONITOR** CCW Heat Exchanger flow rates to ensure limits are **NOT** exceeded. _______

3. **INDICATE** the reason(s) for performing this test. _______

   ☐ Monthly Fan Start ☐ Quarterly Valve IST

   ☐ 18 Month Valve Remote Position indication

   ☐ Increased Surveillance frequency for ______________________

   ☐ Other (Specify) ____

4. **OBTAIN** a portable ammeter. _______

5. **RECORD** portable ammeter M&TE number and calibration due date. _______

   Ammeter M&TE #:_____________Calibration Due Date: ___________

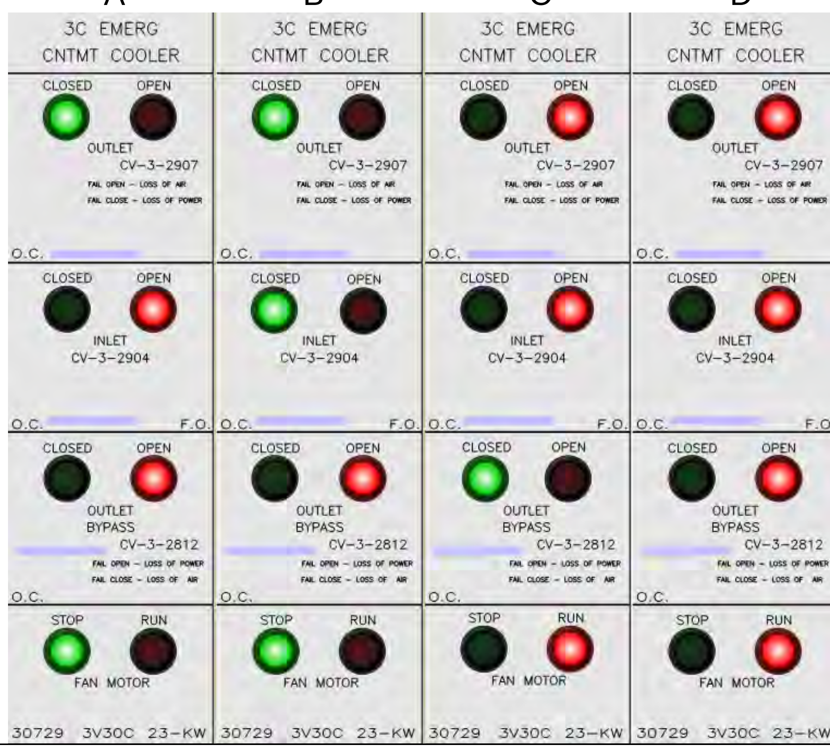
Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	022		K1.01
	Importance Rating	3.5		

Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS/cooling system

Proposed Question: RO Question # 39

Which one of the following matches the letters identifying the alignments of the 3C ECC CCW valves in the (1) normal standby and (2) following a safety injection signal?

A B C D



A.	(1) normal standby alignment- A (2) emergency alignment- D
B.	(1) normal standby alignment- B (2) emergency alignment- D
C.	(1) normal standby alignment- A (2) emergency alignment- C

D.	(1) normal standby alignment- B (2) emergency alignment- C		
Proposed Answer:	A		
A.	Correct. First part is correct because the normal standby alignment is pictured. The CCW bypass valve around the CCW outlet valve for the ECC is open with the inlet open to prevent water hammer and maintains a minimum flow through the ECC. Second part is correct because the emergency alignment is pictured. All CCW valves should be open.		
B.	Incorrect. First part is incorrect, but plausible if candidate believes only the bypass valve is open- stands to reason for a standby alignment especially if candidate believes the bypass valve pressurizes the HX and that no flow should be going through the HX when in standby.		
C.	Incorrect. First part is correct. Second part is incorrect, but plausible if student believes that when on an ECC fan start, the inlet and outlet CCW valves go open and the bypass closes for maximum flow through the ECC (student believes the bypass valve bypasses the ECC).		
D.	Incorrect. Both incorrect. See B and C reasons.		
Technical Reference(s)	LP 6902129, Containment Ventilation and Heat Removal Systems 4-NOP-055, Emergency Containment Cooling System 3-EOP-E-0, Reactor Trip Or Safety Injection		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	LP 6902129, Obj. 9		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X

	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	7
	55.43	
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.		
Comments:		

4.1.1 Placing Emergency Containment Coolers in Standby Condition (continued) **INITIAL**

3. **CHECK** Emergency Containment Cooler valve positions as follows:

- CV-3-2908, 3A EMERG CNTMT COOLER OUTLET is CLOSED ______ ______ IV
- CV-3-2905, 3A EMERG CNTMT COOLER INLET is OPEN ______ ______ IV
- CV-3-2814, 3A EMERG CNTMT COOLER OUTLET BYPASS is OPEN ______ ______ IV
- CV-3-2906, 3B EMERG CNTMT COOLER OUTLET is CLOSED ______ ______ IV
- CV-3-2903, 3B EMERG CNTMT COOLER INLET is OPEN ______ ______ IV
- CV-3-2810, 3B EMERG CNTMT COOLER OUTLET BYPASS is OPEN ______ ______ IV
- CV-3-2907, 3C EMERG CNTMT COOLER OUTLET is CLOSED ______ ______ IV

4.1.1 Placing Emergency Containment Coolers in Standby Condition (continued) **INITIAL**

3. (continued)

- CV-3-2904, 3C EMERG CNTMT COOLER INLET is OPEN ______

  ______ IV

- CV-3-2812, 3C EMERG CNTMT COOLER OUTLET BYPASS is OPEN ______

  ______ IV

4. **ENSURE** Component Cooling Water flow indications to the Emergency Containment Coolers are as follows:

- 3A ECC flow is greater than 0 and less than or equal to 1000 gpm as indicated on FI-3-1470, A ECC CCW FLOW.
- 3B and 3C ECC flow is greater than 0 and less than or equal to 2000 gpm as indicated on FI-3-1471, B&C ECC CCW FLOW.

5. IF CCW flow through the ECCs does **NOT** meet the criteria specified in Section 4.1.1 Step 4, THEN **NOTIFY** the Shift Manager and the System Engineer.

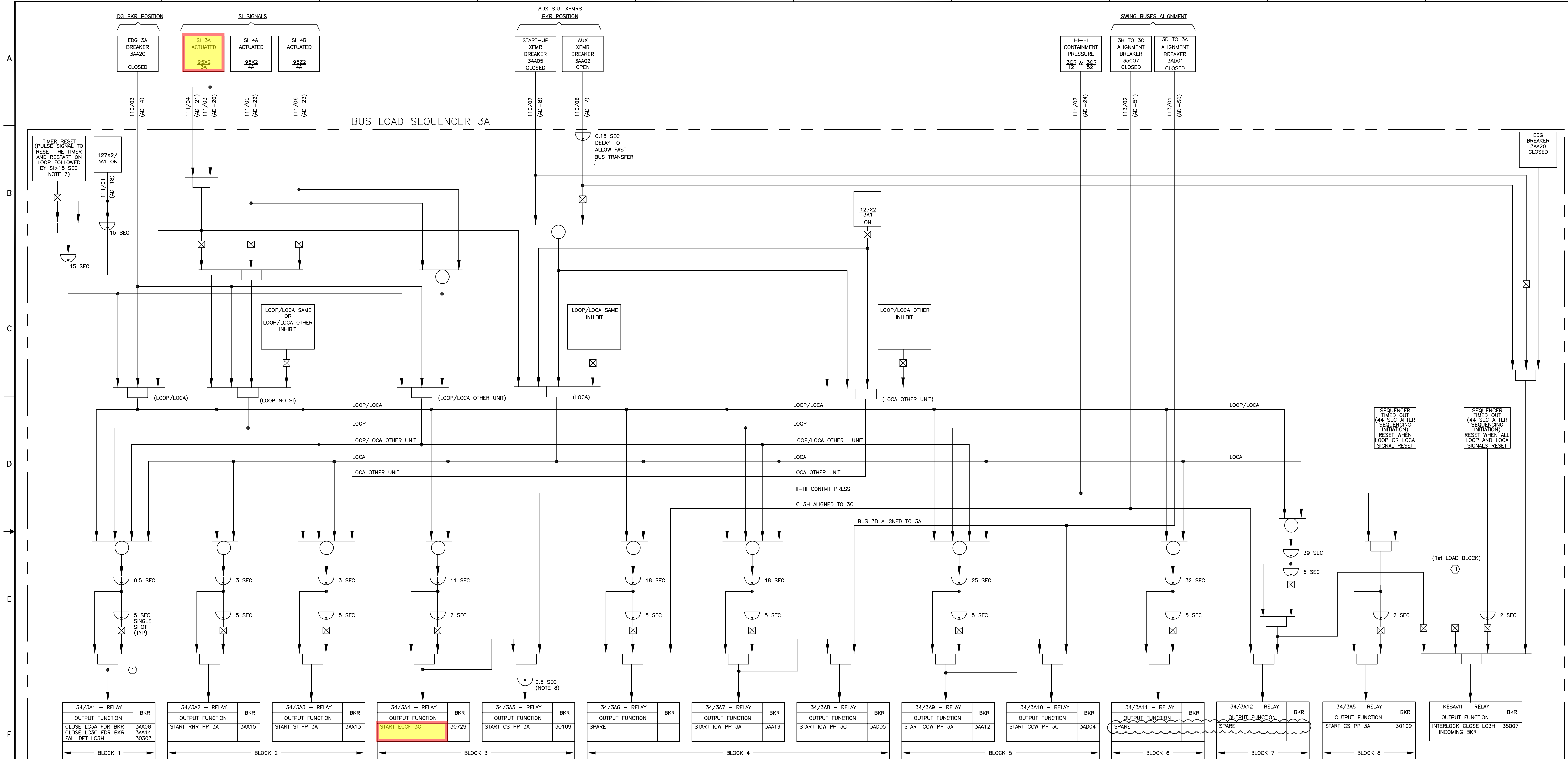
6. **CHECK** Annunciator I 9/4, ECC A/B/C TRIP, is CLEAR.

ATTACHMENT 3

Prompt Action Verifications

(Page 5 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**Verify Proper ICW System Operation**	
	a. Verify ICW Pumps – AT LEAST TWO RUNNING	**a.** Start ICW Pump(s) to establish at least two running.
	b. Verify ICW To TPCW Heat Exchanger – ISOLATED: • POV-3-4882 – CLOSED • POV-3-4883 – CLOSED	**b.** Manually close valve(s). IF valve(s) can **NOT** be closed, THEN locally close the following valves: * 3-50-319 for POV-3-4882 * 3-50-339 for POV-3-4883
	c. Check ICW Headers – TIED TOGETHER	**c.** IF both ICW headers are intact, THEN direct operator to tie headers together.
8.	**Verify Containment Cooling**	
	a. Check Emergency Containment Coolers – ONLY TWO RUNNING	**a.** Manually start or stop Emergency Containment Coolers to establish only two running.
9.	**Verify Containment Ventilation Isolation**	
	a. Unit 3 Containment Purge Exhaust And Supply Fans – OFF	**a.** Manually stop fans.



BUS LOAD SEQUENCER 3A

***	LOAD CONDITIONS (EVENT)	1st LOAD BLOCK	2nd LOAD BLOCK		3rd LOAD BLOCK		4th LOAD BLOCK			5th LOAD BLOCK		6th LOAD BLOCK	7th LOAD BLOCK	8th LOAD BLOCK
1	LOOP	X						X	X	X	X			
2	LOOP WITH A SIMULTANEOUS LOCA SAME TRAIN (3A)	X	X	X	X		X	X	X	X	X	X	X	
3	LOOP FOLLOWED SOMETIME LATER BY A LOCA SAME TRAIN (3A)	X	X	X	X		X	X	X	X	X	X	X	
4	LOOP WITH A SIMULTANEOUS LOCA OTHER UNIT	X		X				X	X	X	X			
5	LOOP FOLLOWED SOMETIME LATER BY A LOCA OTHER UNIT	X		X				X	X	X	X			
6	LOCA SAME TRAIN (3A)	X	X	X	X		X	X	X	X	X	X	X	
7	LOCA OTHER UNIT			X										
8	HI-HI CONTAINMENT PRESSURE CONCURRENT OR LESS THAN 13 SECONDS AFTER A LOOP/LOCA	X	X	X	X	X	X	X	X	X	X	X	X	
9	HI-HI CONTAINMENT PRESSURE CONCURRENT OR LESS THAN 13 SECONDS AFTER A LOCA	X	X	X	X	X	X	X	X	X	X	X	X	
10	HI-HI CONTAINMENT PRESSURE OCCURING MORE THAN 13 SECONDS AFTER A LOCA	X	X	X	X		X	X	X	X	X	X	X	X
11	HI-HI CONTAINMENT PRESSURE OCCURING MORE THAN 13 SECONDS AFTER A LOOP/LOCA	X	X	X	X		X	X	X	X	X	X	X	X
12	LOCA SAME TRAIN FOLLOWED SOMETIME LATER BY LOOP	X	X	X	X		X	X	X	X	X	X	X	
13	LOCA OTHER UNIT FOLLOWED SOMETIME LATER BY LOOP	X		X				X	X	X	X			

LC 3H TRANSFER PERMISSIVE

THE KESAVI1 RELAY IS NORMALLY DE-ENERGIZED ALLOWING FOR LC-3H TRANSFER. THE RELAY ENERGIZES AS SOON AS THE AUX AND S/U XFMR BKRS ARE OPEN AND THE EDG 3A BREAKER CLOSES. IT THEN DE-ENERGIZES DURING THE FIRST LOAD BLOCK, ENERGIZES AFTER THE FIRST LOAD BLOCK, DE-ENERGIZES DURING THE SEVENTH LOAD BLOCK, ENERGIZES AFTER THE SEVENTH LOAD BLOCK, AND DE-ENERGIZES TWO SECONDS AFTER THE SEQUENCER HAS TIMED OUT.

NOTES:

1. THIS DRAWING REPRESENTS A CONCEPTUAL LOGIC DIAGRAM OF THE SEQUENCING LOADING FUNCTION OF THE EMERGENCY BUS LOAD SEQUENCER 3A.
2. DELETED
3. DELETED
4. THE EIGHTH LOAD BLOCK REPRESENTS A LATE START OF THE CONTAINMENT SPRAY PUMP DUE TO A LATE RECEIPT (TIME LATER THAN 13 SECONDS AFTER LOOP OR LOOP/LOCA) OF A HIGH-HIGH CONTAINMENT PRESSURE SIGNAL.
5. DELETED.
6. THE BLOCKS SHOWN INSIDE THE SEQUENCER BOX ARE GENERATED INTERNALLY BY THE SEQUENCER LOGIC.
7. ON LOOP FOLLOWED BY LOCA SCENARIO, THE TIMER IS RESET AND RESTARTS, IF SI OCCURS IN MORE THAN 15 SEC'S AFTER LOOP.
8. TIME DELAY PROVIDED TO ENSURE CONTAINMENT SPRAY PUMP START IF HI-HI CONTAINMENT PRESSURE IS RECEIVED NEAR END OF TWO SECOND LOAD BLOCK 3 PULSE.

NUCLEAR SAFETY RELATED

JPN CAD ESS

NOTE: THIS DWG IS MADE FROM:
_____ (VENDOR) DWG. NO. _____
_____ (A/E) DWG. NO. _____ REV. ___
FPL DWG NO. 5610-T-L1, SH.12B REV. 15

REV	DATE	REVISION	BY	CH	APP	APP
2	06-11-12	ISSUED AS-BUILT PER EC 246930 (PC/M 08-181).	RV	BB	VPL	MFM
1	10/3/96	ISSUED AS-BUILT PER PC/M 95-147.	RV	RH	BP	PMB
0	9/17/95	ISSUED AS-BUILT PER PC/M 95-032 AND ISSUED INTO THE FPL SYSTEM.	RV	RH	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNIT 3

LOGIC DIAGRAM

EMERGENCY BUS LOAD SEQUENCER LOADING LOGIC DIAGRAM

POD

DRAWING NUMBER 5613-T-L1

SHEET 12A

SYS

REV 2

5613-T-L1_SH 12A.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	026		2.4.4
	Importance Rating	4.5		

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: RO Question # 40

Given the following conditions:

- Unit 3 trips from full power due to a steam line break inside containment.
- Containment pressure on DCS is 22 psig.
- The following is observed on VPB:

★ identifies a lit bistable

Which one of the following completes the statements below?

(Assuming no operator action)

Containment spray pumps ___(1)___ automatically started.

An ORANGE path entry into 3-EOP-FR-Z.1, Response to High Containment Pressure, ___(2)___ required.

A.	(1) have (2) is
B.	(1) have NOT (2) is
C.	(1) have (2) is NOT
D.	(1) have NOT (2) is NOT

Proposed Answer: B

A.	Incorrect. Part 1 is incorrect but plausible if candidate believes only HI HI containment pressure is required to actuate containment spray. Part 2 is correct.
B.	Correct. 2/3 HI and HI HI containment pressure signals are required to start containment spray pumps. In this case only 2/3 HI HI logic is made up therefore pumps will NOT have started. FRP entry is required with containment pressure > 20 psig and no containment spray pumps running.
C.	Incorrect. Part 1 is incorrect. Part 2 is incorrect but plausible if candidate believes containment spray pumps have started therefore FRP entry is NOT required.
D.	Incorrect. Part 1 is correct. Part 2 is incorrect. This combination is plausible if the candidate believes pumps have not started and confuses ORANGE path FR-Z.1 entry with RED path entry conditions. Candidate thinks the RED path condition applies.

Technical Reference(s)	3-EOP-FR-Z.1 step 3 caution 5610-T-L1	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:	N

Learning Objective:		(As available)

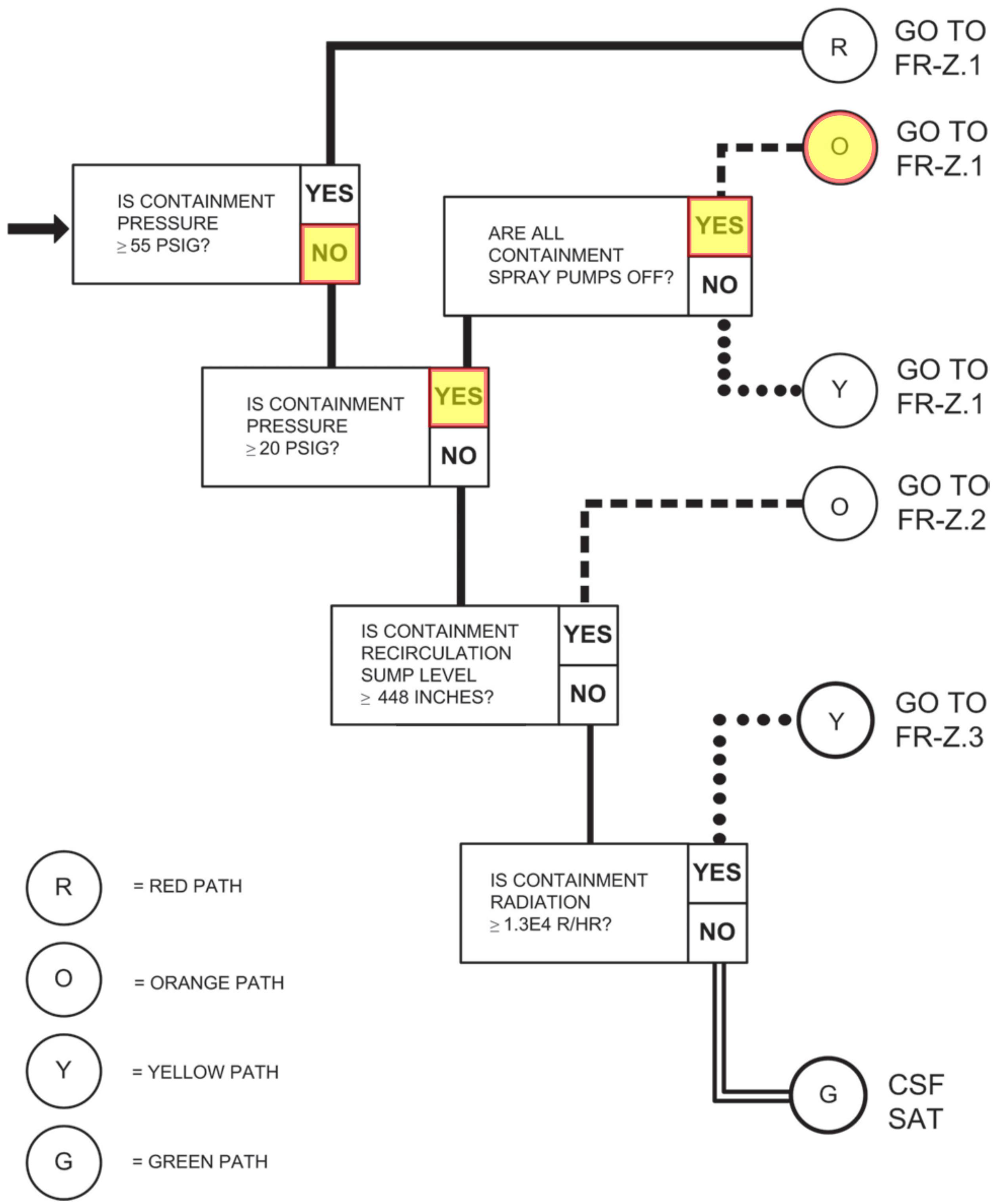
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	

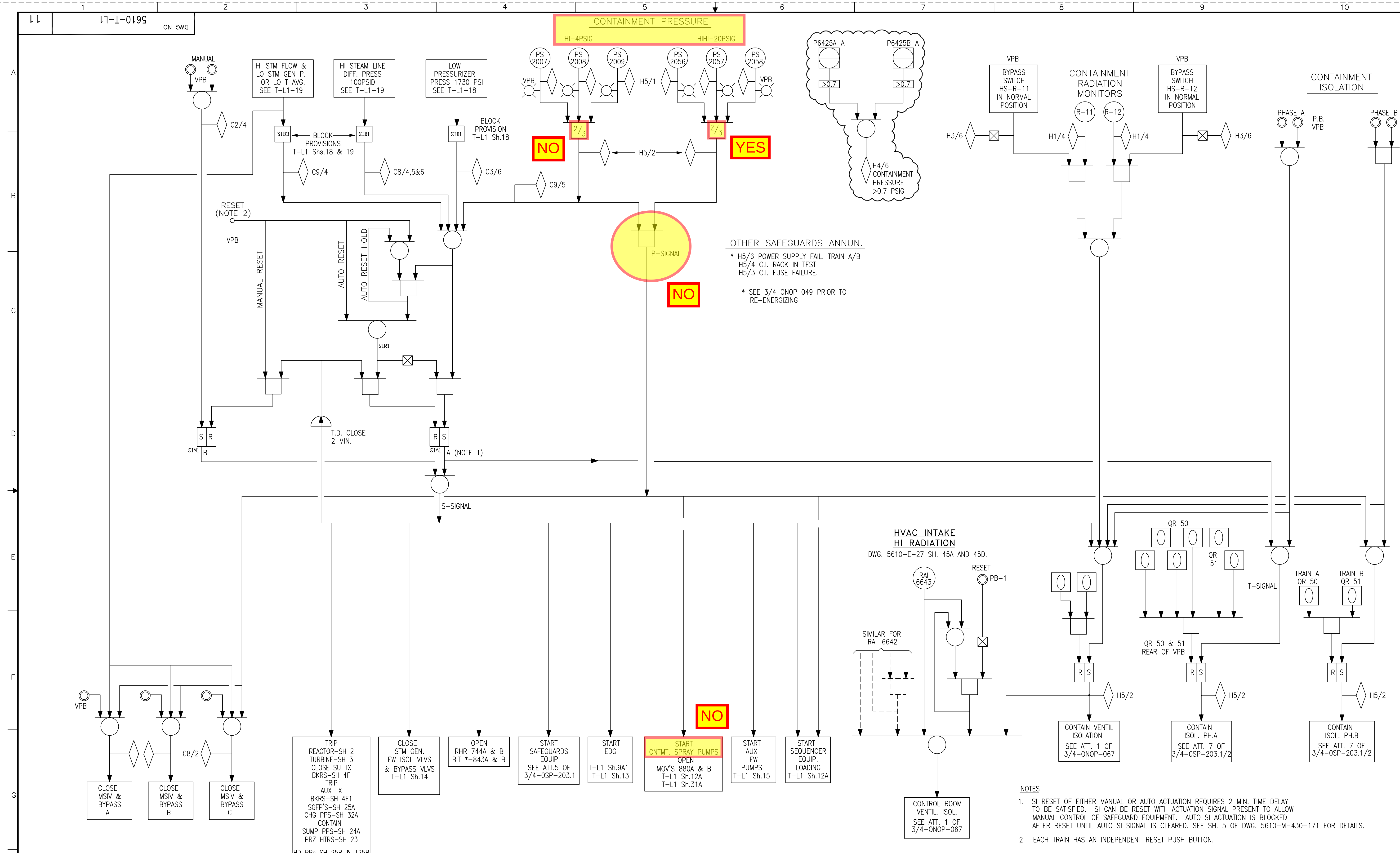
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

ENCLOSURE 5

CSF F-0.5 Containment

(Page 1 of 1)





NOTES

1. SI RESET OF EITHER MANUAL OR AUTO ACTUATION REQUIRES 2 MIN. TIME DELAY TO BE SATISFIED. SI CAN BE RESET WITH ACTUATION SIGNAL PRESENT TO ALLOW MANUAL CONTROL OF SAFEGUARD EQUIPMENT. AUTO SI ACTUATION IS BLOCKED AFTER RESET UNTIL AUTO SI SIGNAL IS CLEARED. SEE SH. 5 OF DWG. 5610-M-430-171 FOR DETAILS.
2. EACH TRAIN HAS AN INDEPENDENT RESET PUSH BUTTON.

SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP
31	08-28-12	ISSUED AS-BUILT PER EC 246935 (PC/M 08-187).	RV	RH	–	PMB
30	12-28-09	ISSUED AS-BUILT PER CRN-E-18083 (PC/M 09-106).	RV	BB	–	LG
29	11-20-09	ISSUED AS-BUILT PER PC/M 08-037.	RV	RH	–	AD
28	4-9-09	ISSUED AS-BUILT PER PC/M 08-006.	RH	KCP	–	RCS

REV	DATE	REVISION	BY	CH	APP	APP
27	07-15-03	ISSUED AS-BUILT PER CRN-E-16419 (PC/M 02-065).	RV	RH	BP	TS
26	1-20-98	ISSUED AS-BUILT PER CRN'S I-3659 & I-3663(PC/M 97-042).	RV	RH	MA	TS
25	11-7-97	ISSUED AS-BUILT PER CRN-I-3640(PC/M 97-036)	RV	RH	MA	TS/JG
24	1-23-97	ISSUED AS-BUILT PER CRN-I-3482(PC/M 96-080)	RV	RH	BP	TS
23	9-30-96	INCORPORATED CRN-I-3418 FOR PC/M 96-058.	RV	BP	–	TS
22	4-7-93	AS-BUILT PER DCR-TPI-92-172	RH	RV	ET	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

POD

FLORIDA POWER & LIGHT

SAFEGUARDS ACTUATION AND STEAM LINE ISOLATION

DRAWING NUMBER 5610-T-L1

SYS 063

REV 31

SHEET 11

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	039		A3.02
	Importance Rating	3.1		

Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

Proposed Question: RO Question # 41

Given the following conditions:

- Unit 3 experiences a steam header break from 50% power.
- RCS pressure is 1800 psig.
- RCS Tavg is 525°F.
- Containment pressure is 0.1 psig.
- All SG pressures are 720 psig.
- All SG steam flows are 40% of full-scale.

Which one of the following completes the statement below?

(Assuming no operator action.)

_______ will receive an automatic closure signal.

A.	Only the Turbine Stop valves
B.	Only the Turbine Stop valves and MSR stop valves
C.	Only the Turbine Stop valves, MSR stop valves and MSIVs
D.	The Turbine Stop valves, MSR stop valves, MSIVs and MSIV bypasses

Proposed Answer: D

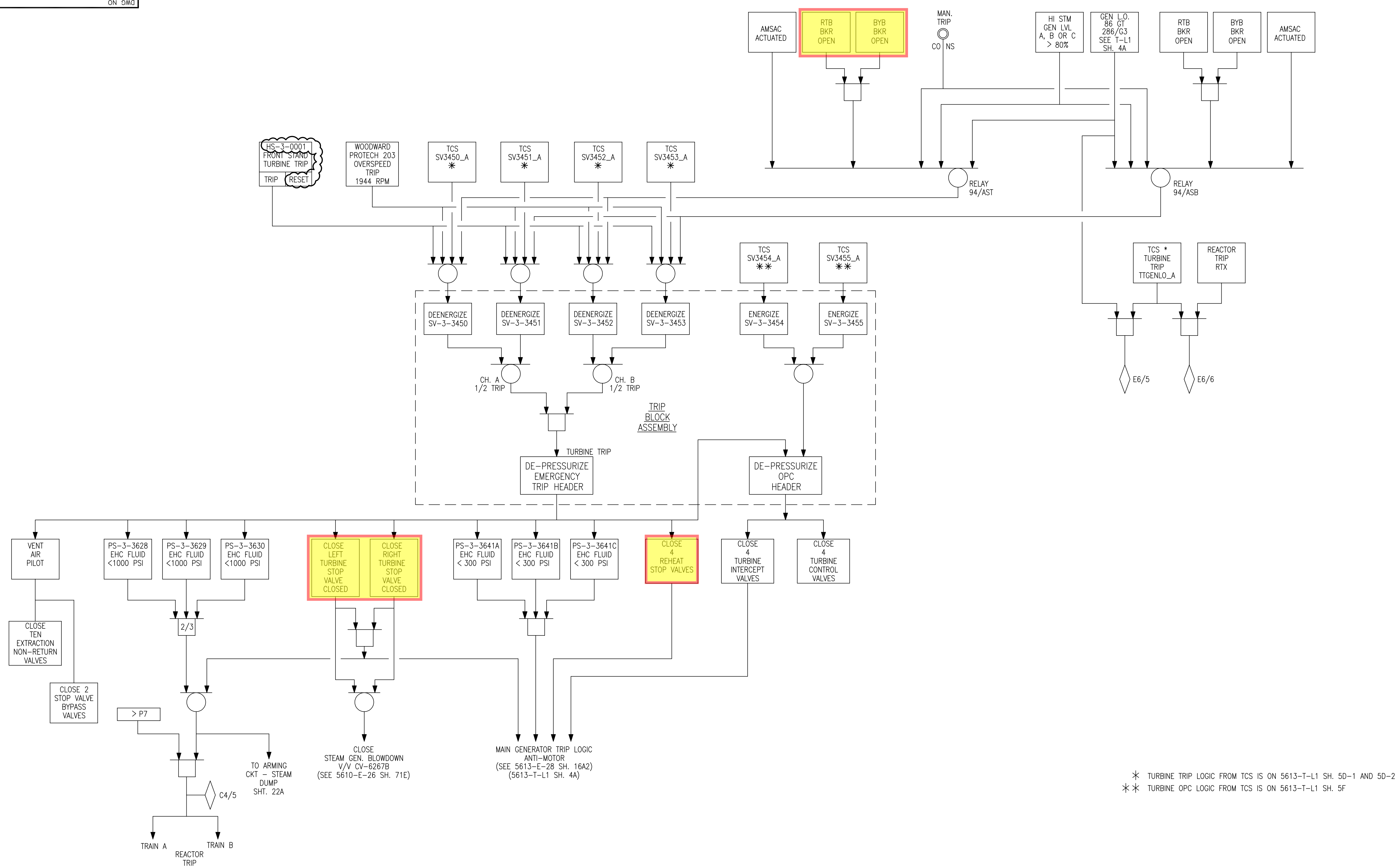
A.	Incorrect. Plausible because the candidate assumes only the Turbine Stop Valves close on a turbine trip actuated by a safety injection signal. Candidate also forgets MSR stop valves automatically close post EPU. Candidate also does not recognize main steam line isolation has actuated when candidate confuses SG steam flow with post trip SG level and/or candidate assumes since SG pressures are higher than Hi stm flow with lo SG pressure (614#) SI signal that main steam line actuation / safety injection has not occurred.
B.	Incorrect. Plausible when candidate recalls the post EPU mod for MSR stop valves and fails to recognize main steam line isolation conditions per discussion above.
C.	Incorrect. Plausible when candidate fails to recall that MSIV bypasses automatically close given these valves are normally closed and de-energized at power. Candidate must recall logic to answer correctly.
D.	Correct. All will receive an auto closure signal. Turbine Stop valves & MSR stop valves close on a turbine trip signal and MSIVs and MSIV bypasses close on Main Steam Isolation

Technical Reference(s)	LP 6902163 5610-T-L1 (sh. 11, 19)		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902163 obj 8		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

DWG NO 5613-T-L1 SHT 3



* TURBINE TRIP LOGIC FROM TCS IS ON 5613-T-L1 SH. 5D-1 AND 5D-2

** TURBINE OPC LOGIC FROM TCS IS ON 5613-T-L1 SH. 5F

SAFETY RELATED

REV	DATE	REVISION	BY	CH	APP	APP
1	11-22-13	ISSUED AS-BUILT PER CRN-337 (EC 246849). (ITOP 13-11-003)	RV	BB	–	PE
0	08-24-12	ISSUED AS-BUILT PER EC 246849 (PC/M 08-077); INCORP. CRN-222 & ENTERED INTO FPL SYSTEM.	JPH	BB	–	PMB

FPL

TURKEY POINT NUCLEAR UNIT 3

LOGIC DIAGRAM

TURBINE TRIPS

POD

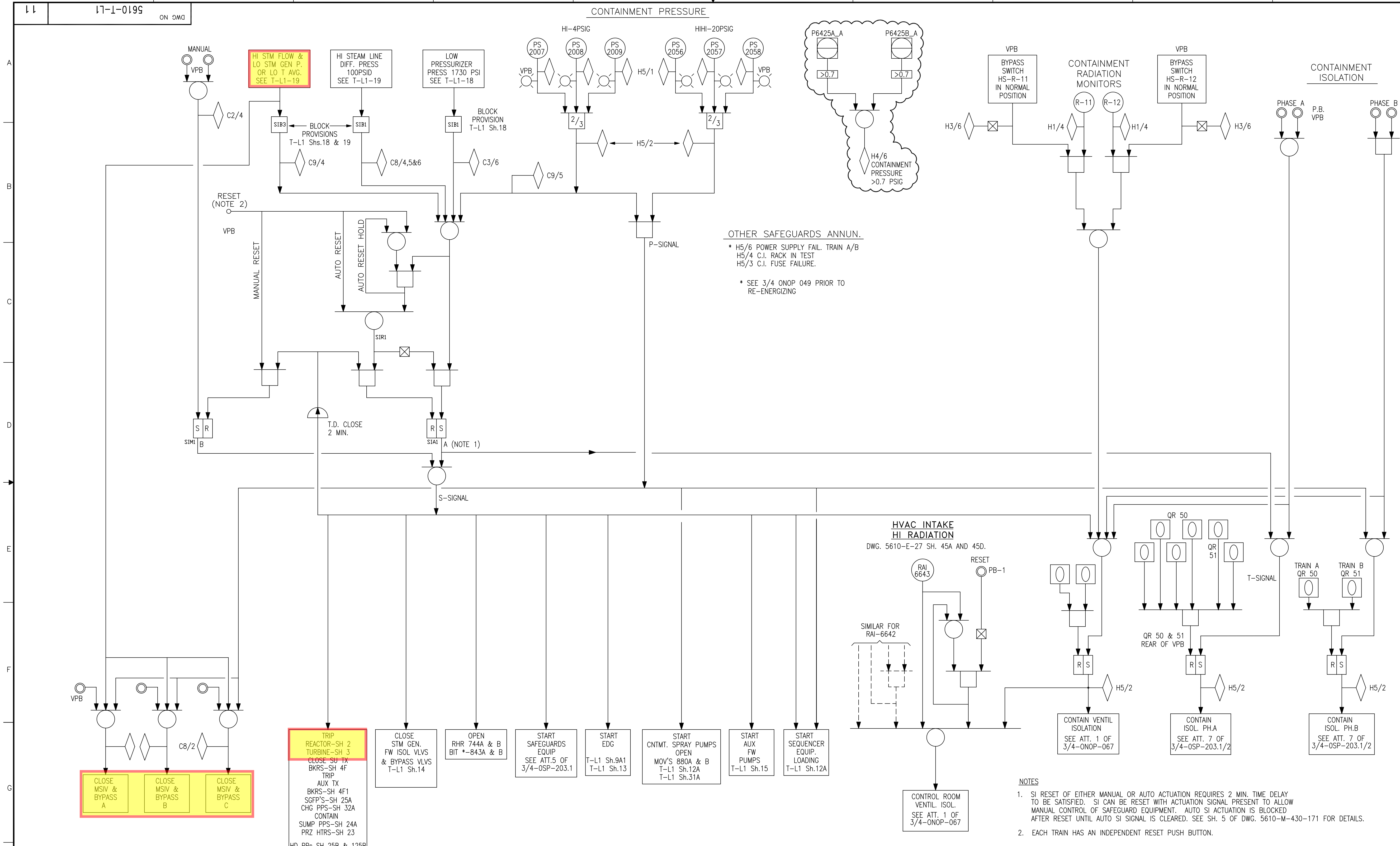
DRAWING NUMBER 5613-T-L1

SYS –

SHEET 3

REV 1

5613-T-L1_SH_3.DWG



NOTES

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2. EACH TRAIN HAS AN INDEPENDENT RESET PUSH BUTTON.

SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP
31	08-28-12	ISSUED AS-BUILT PER EC 246935 (PC/M 08-187).	RV	RH	–	PMB
30	12-28-09	ISSUED AS-BUILT PER CRN-E-18083 (PC/M 09-106).	RV	BB	–	LG
29	11-20-09	ISSUED AS-BUILT PER PC/M 08-037.	RV	RH	–	AD
28	4-9-09	ISSUED AS-BUILT PER PC/M 08-006.	RH	KCP	–	RCS

REV	DATE	REVISION	BY	CH	APP	APP
27	07-15-03	ISSUED AS-BUILT PER CRN-E-16419 (PC/M 02-065).	RV	RH	BP	TS
26	1-20-98	ISSUED AS-BUILT PER CRN'S I-3659 & I-3663(PC/M 97-042).	RV	RH	MA	TS
25	11-7-97	ISSUED AS-BUILT PER CRN-I-3640(PC/M 97-036)	RV	RH	MA	TS/JG
24	1-23-97	ISSUED AS-BUILT PER CRN-I-3482(PC/M 96-080)	RV	RH	BP	TS
23	9-30-96	INCORPORATED CRN-I-3418 FOR PC/M 96-058.	RV	BP	–	TS
22	4-7-93	AS-BUILT PER DCR-TPI-92-172	RH	RV	ET	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

FLORIDA POWER & LIGHT

SAFEGUARDS ACTUATION AND STEAM LINE ISOLATION

POD

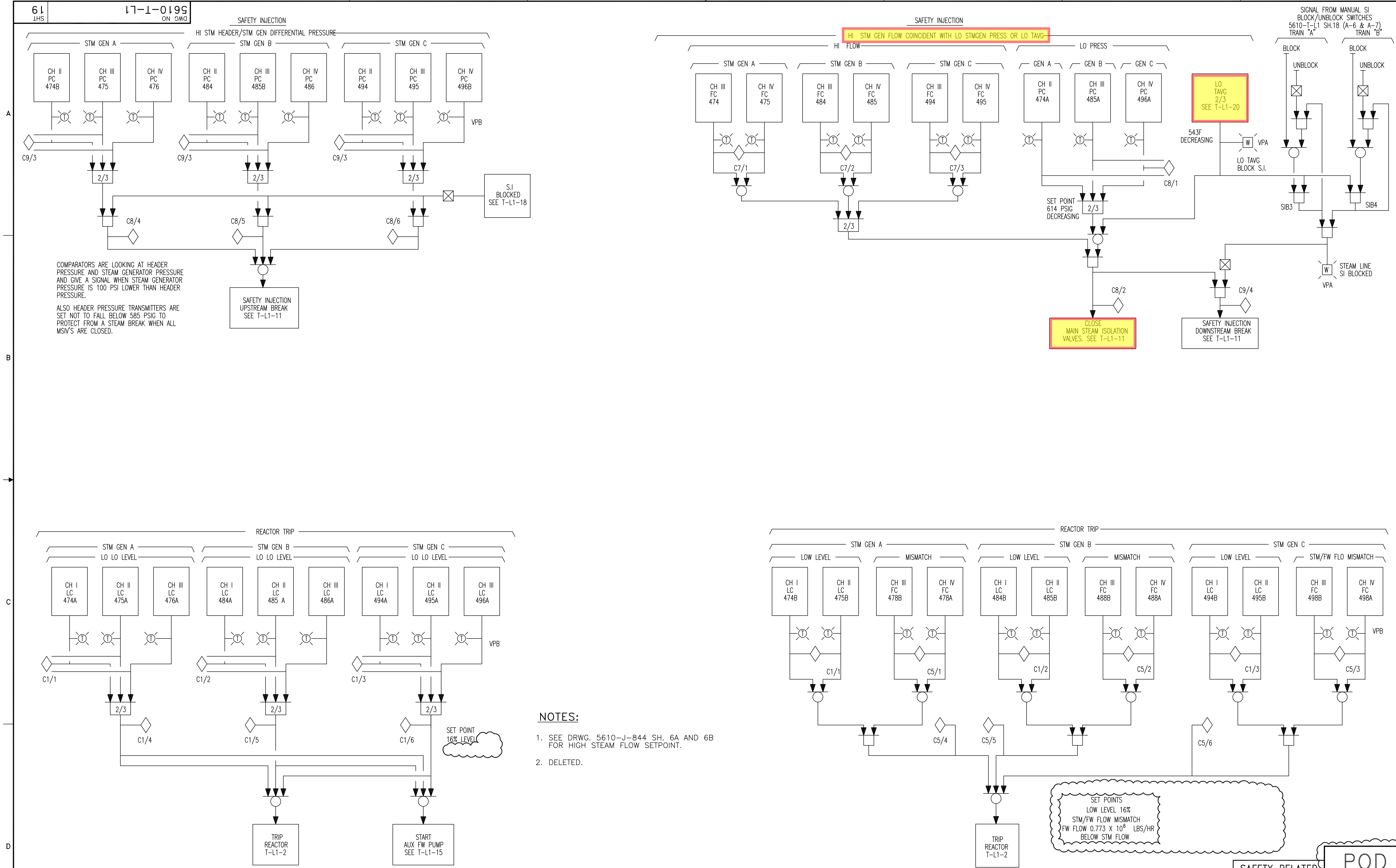
DRAWING NUMBER 5610-T-L1

SYS 063

REV 31

SHEET 11

5610-T-L1_SH 11.dwg



NOTES:

1. SEE DRWG. 5610-J-844 SH. 6A AND 6B FOR HIGH STEAM FLOW SETPOINT.
2. DELETED.

SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
19	10-28-96	ISSUED AS-BUILT FOR PC/M 95-171. (PARTIAL)	RV	RH	JG	JJN	24	7-17-12	ISSUED AS-BUILT PER EC 247048 (PC/M 10-039).	RV	BB	–	PMB
18	10-17-96	ISSUED AS-BUILT FOR PC/M 95-171 (PARTIAL) & INCORP. CRN-I-3465.	RV	RH	JG	TS	23	07-31-01	ISSUED AS-BUILT PER CRN-I-3967 (PC/M 00-016).	RV	RH	BP	BAP
17	10-9-96	ISSUED AS-BUILT PER PC/M 95-170. (PARTIAL)	RV	RH	–	TS	22	1-20-98	ISSUED AS-BUILT PER CRN-I-3663 (PC/M 97-042).	RV	RH	MA	TS
26	5/1/13	ISSUED AS-BUILT PER EC 247049 (ITOP 13-04-155).	RV	BB	–	PMB	21	1-21-97	ISSUED AS-BUILT FOR PC/M 95-171.	RV	RH	JG	TS
25	8-9-12	ISSUED AS-BUILT PER CRN-006 (EC 247048).	RV	RH	–	PMB	20	1-13-97	ISSUED AS-BUILT FOR PC/M 95-170.	RV	RH	JG	TS

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

LOGIC DIAGRAM

STEAM GENERATOR CAUSED REACTOR TRIP AND SAFETY INJECTION SIGNALS

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-L1

SYS –

SHEET: 19

REV 26

5610-T-L1_SH 19.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	059		A1.03
	Importance Rating	2.7		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves.

Proposed Question: RO Question # 42

Given the following conditions:

- Unit 3 is performing a power ascension.
- The 3B Steam Generator Feed Pump is OOS.

Which one of the following identifies the MAXIMUM power level allowed in accordance with 3-GOP-301, Hot Standby to Power Operation?

A.	88%
B.	85%
C.	58%
D.	50%

Proposed Answer: C

A.	Incorrect. 88% is plausible because on loss of condensate or heater drain pumps, if power is greater than 88% the unit will run back to 85%
B.	Incorrect. 85% is plausible because of same reason as option A. If heater drain pump, condensate pump trip, or LP heater bypass is open, the turbine runback will occur to 85% if reactor power is greater than 88%.

C.	Correct. If a SGFP pump motor overloads or otherwise fails, power must be reduced to less than 58%. If power is 60% - 85%, a runback will occur to 50%. Although the runback is to 50%, procedure requirements maintain power less than 58% with only one SGFP in service.		
D.	Incorrect. Plausible because if a SGFP trips with power 60% or greater, the turbine will run back to 50% power. Although the runback is to 50%, procedure requirements are less than 58%		
Technical Reference(s)	3-ARP-097.CR.D 5/1 and others 3-GOP-301 step 5.83.4 table		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		4
	55.43		
Secondary coolant and auxiliary systems that affect the facility.			
Comments:			

INIT

NOTES

Power increase may continue at this time.

5.83 **IF** desired to start a second SGFP at less than 425 MWe, **THEN** perform the following:

____ 5.83.1 Increase monitoring frequency of condenser vacuum.

____ 5.83.2 Establish a compensatory action to monitor pump vibrations while less than 425 MWe.

____ 5.83.3 Verify proper operation of SGFP recirculation flow control valves.

____ 5.83.4 **IF** the second S/G Feedwater Pump is **NOT** available, **THEN** power escalation may proceed, but **NOT** to exceed 58% or other power level as determined by the Shift Manager based on plant operating parameters.

____ 1. Ensure the following parameters are monitored and the given limits **NOT** exceeded:

Parameter	Limit
Max Reactor Power, percent	60
Max SGFP A (3P1A) Motor Current, amps	950
Max SGFP B (3P1B) Motor Current, amps	950
Max Feedwater Regulating Valve Controller Demand, percent	95
Steam Generator Feedwater Pump Suction Pressure, psig	In Green Band
Steam Generator Feedwater Pump Discharge Pressure, psig	

5.83.5 **WHEN** the second S/G Feedwater Pump is available to be placed in service, **THEN** perform the following:

____ 1. Lower reactor power, as required, per the direction of the Shift Manager.

____ 2. Place second S/G Feedwater Pump in service using 3-NOP-074, Steam Generator Feedwater System.

____ 3. Record MWe SGFP place in Service: ____________________________

____ 4. Check running amps for both pumps approximately equal.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	059		K4.19
	Importance Rating	3.2		

Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: Automatic feedwater isolation of MFW

Proposed Question: RO Question # 43

Given Unit 3 is at 15% power.

Which one of the following completes the statements below?

A(n) ___(1)___ signal will cause an automatic feedwater isolation and an AFW actuation.

A feedwater isolation signal ___(2)___ close Main Feedwater Control Valves FCV-3-478, FCV-3-488 and FCV-3-498.

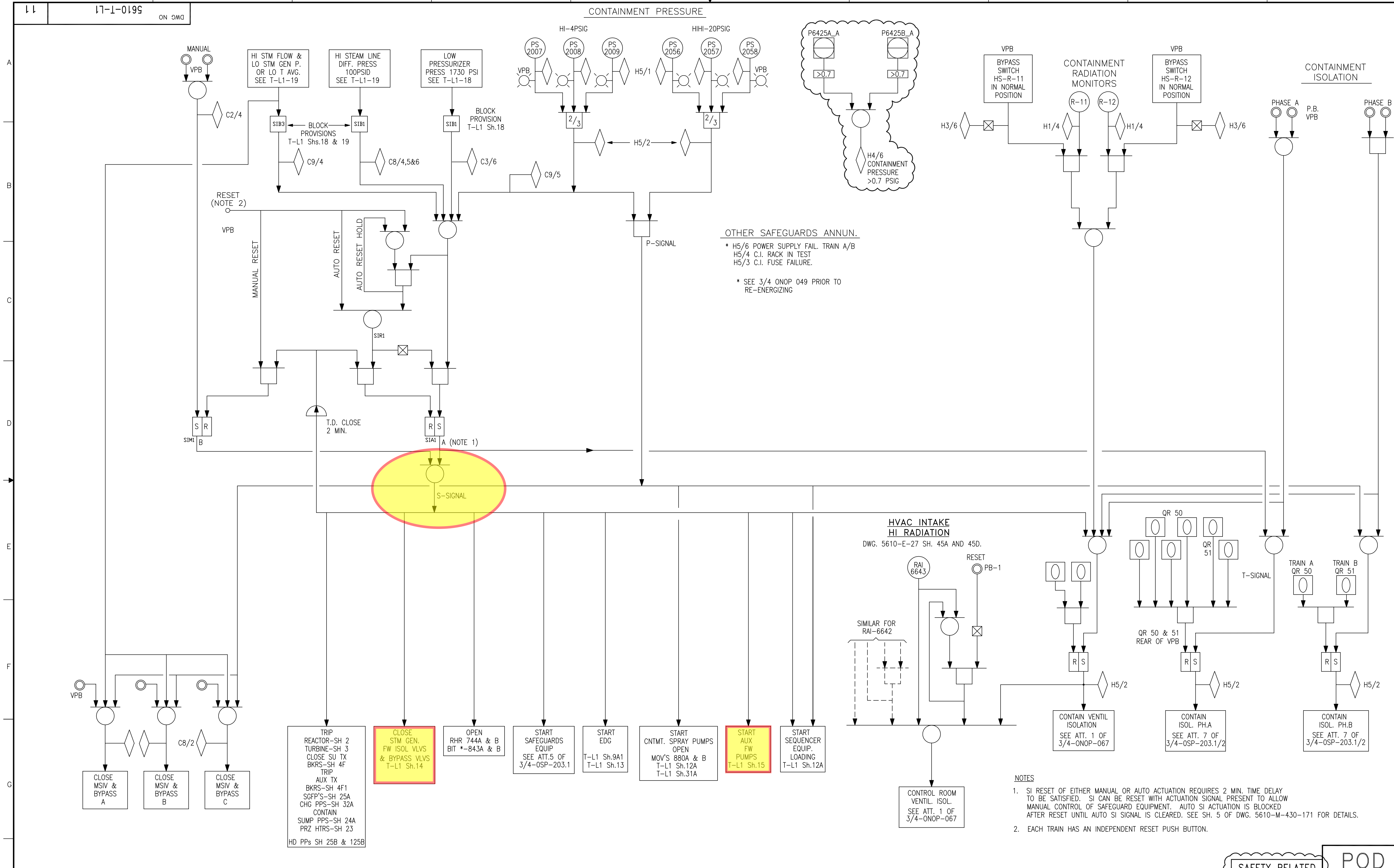
A.	(1) Reactor Trip with Lo Tavg (2) will
B.	(1) Safety Injection actuation (2) will
C.	(1) Reactor Trip with Lo Tavg (2) will NOT
D.	(1) Safety Injection actuation (2) will NOT

Proposed Answer: B

A.	Incorrect. Part 1 is incorrect. Plausible because reactor trip with lo tavg (554°F) will close main feed reg valves but will not trip the SGFPs which would actuate AFW. Part 2 is correct.

B.	Correct. Part 1 is correct. SI-actuation will cause Feedwater isolation. A feedwater isolation will trip main feed pumps causing AFW actuation. Part 2 is correct. A Feedwater isolation signal will close both the FRV and the associated Feedwater Isolation MOVs-3-1407/1408/1409.
C.	Incorrect. Part 1 is incorrect. Part 2 is incorrect, but plausible if candidate believes only Feedwater isolation valves and Feedwater bypass isolation valves go closed. It is logical to think that since the words contain isolation that only those valve isolate. Also, since main feed reg valve program and setpoint values remain at 50% following a trip candidate may believe that they remain in a throttle state ready to receive flow once the Feedwater isolation signal is RESET. Also plausible when candidate confuses feed water system with blowdown system, where blowdown isolation (MOVs) valves receive a phase A signal and blowdown control valves do NOT.
D.	Incorrect. Part 1 is correct. Part 2 is incorrect.

Technical Reference(s)	3-ARP-097.CR.E LP 6902163 figure 14, 5610-T-L1 sheets 11, 14A		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902163 Objectives 6.f, 7.e, and 8.e		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			



NOTES

1. SI RESET OF EITHER MANUAL OR AUTO ACTUATION REQUIRES 2 MIN. TIME DELAY TO BE SATISFIED. SI CAN BE RESET WITH ACTUATION SIGNAL PRESENT TO ALLOW MANUAL CONTROL OF SAFEGUARD EQUIPMENT. AUTO SI ACTUATION IS BLOCKED AFTER RESET UNTIL AUTO SI SIGNAL IS CLEARED. SEE SH. 5 OF DWG. 5610-M-430-171 FOR DETAILS.
2. EACH TRAIN HAS AN INDEPENDENT RESET PUSH BUTTON.

SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP
31	08-28-12	ISSUED AS-BUILT PER EC 246935 (PC/M 08-187).	RV	RH	–	PMB
30	12-28-09	ISSUED AS-BUILT PER CRN-E-18083 (PC/M 09-106).	RV	BB	–	LG
29	11-20-09	ISSUED AS-BUILT PER PC/M 08-037.	RV	RH	–	AD
28	4-9-09	ISSUED AS-BUILT PER PC/M 08-006.	RH	KCP	–	RCS
27	07-15-03	ISSUED AS-BUILT PER CRN-E-16419 (PC/M 02-065).	RV	RH	BP	TS
26	1-20-98	ISSUED AS-BUILT PER CRN'S I-3659 & I-3663(PC/M 97-042).	RV	RH	MA	TS
25	11-7-97	ISSUED AS-BUILT PER CRN-I-3640(PC/M 97-036)	RV	RH	MA	TS/JG
24	1-23-97	ISSUED AS-BUILT PER CRN-I-3482(PC/M 96-080)	RV	RH	BP	TS
23	9-30-96	INCORPORATED CRN-I-3418 FOR PC/M 96-058.	RV	BP	–	TS
22	4-7-93	AS-BUILT PER DCR-TPI-92-172	RH	RV	ET	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

FLORIDA POWER & LIGHT

SAFEGUARDS ACTUATION AND STEAM LINE ISOLATION

POD

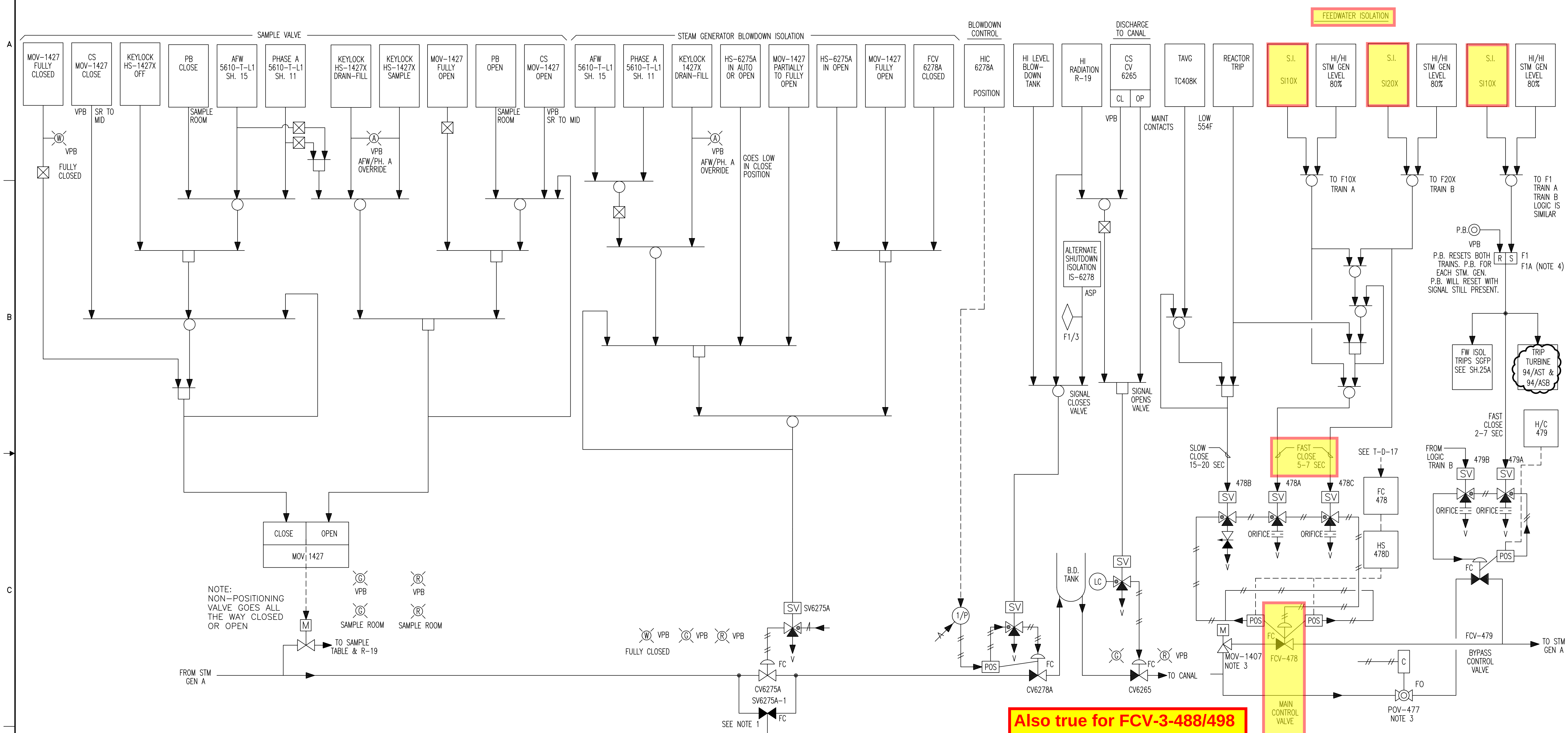
DRAWING NUMBER 5610-T-L1

SYS 063

REV 31

SHEET 11

5610-T-L1_SH 11.dwg



REFERENCES:

EWD 5610 – E26 SHEETS 27–28
5610 – M – 430 – 171, SHEET 7
5610 – E – 25, SH. 32A
5610 – E – 26, SH. 71A THRU J

NOTE:

1. SV-✳-6275A/B/C-1 COILS DISCONNECTED BY PC/Ms 95-168 (U3)
2. MAIN FEED REG SV'S ARE SHOWN IN ENERGIZED STATE.
3. FOR CONTROL AND ACTUATION OF THE BACKUP FEEDWATER ISOLATION VALVES, SEE DWG. 5610-T-L1, SH. 14A1.
4. RELAYS FIA THROUGH F6A ARE MULTIPLYING RELAYS WHICH MIRROR THE OPERATION OF THE ASSOCIATED F1 THROUGH F6 MG6 RELAY.

NOTE: THIS DWG IS MADE FROM:
(VENDOR) DWG. NO.
(A/E) DWG. NO. REV.
FPL DWG. NO. 5610-T-L1 SH 14A/05-073 REV. 0

SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP
6	11-13-13	ISSUED AS-BUILT PER EC 246849. (ITOP 13-11-003)	RV	BB	–	JN
5	08-28-12	ISSUED AS-BUILT PER EC 246870 (PC/M 08-107); INCORP. CRN-036.	RV	BB	–	PMB
4	8-9-12	ISSUED AS-BUILT PER EC 246870 (PC/M 08-107). (PARTIAL)	RV	RH	–	VPC

REV	DATE	REVISION	BY	CH	APP	APP
3	7/27/12	ISSUED AS-BUILT PER EC 242442 (PC/M 08-106).	RV	BB	–	CW
2	5/24/11	ISSUED AS-BUILT PER EC-DCR 249689.	RV	RH	–	MC
1	7/16/07	ISSUED AS-BUILT PER CRN-I-4604 (PC/M 05-073).	RV	RH	BSG	PRB
0	4/3/06	THIS DRAWING SUPERSEDES DWG. 5610-T-L1 SH 14. ISSUED AS-BUILT PER PC/M 05-073 AND ENTERED INTO THE FPL SYSTEM.	RV	BB	ASD	JMM

FPL

TURKEY POINT NUCLEAR UNIT 3

LOGIC DIAGRAM

FEEDWATER ISOLATION AND STEAM GENERATOR BLOWDOWN ISOLATION

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-L1

SYS –

SHEET 14A

REV 6

5610-T-L1_SH 14A.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	061		K1.10
	Importance Rating	2.6		

Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: Diesel fuel oil

Proposed Question: RO Question # 44

Given the following conditions:

- Unit 4 trips due to a LOOP.
- The crew is securing AFW to Unit 4.
- SG pressures are 200 psig.

Which one of the following identifies (1) which valve(s) is / are required to be throttled open PRIOR to starting the B Standby Feed Pump (SSGFP) and (2) the amount of pump run time that the B SSGFP fuel tank capacity will provide at full flow?

A.	(1) FCV-3-479, FCV-3-489 and FCV-3-499, FW Bypass Valves (2) 11 hours
B.	(1) DWDS-4-012, STBY SGFP Discharge Header to Unit 4 Isolation Valve (2) 11 hours
C.	(1) FCV-3-479, FCV-3-489 and FCV-3-499, FW BYPASS Valves (2) 72 hours
D.	(1) DWDS-4-012, STBY SGFP Discharge Header to Unit 4 Isolation Valve (2) 72 hours

Proposed Answer: B

A.	Incorrect. Part 1 is incorrect. Part 2 is correct.
B.	Correct. Both parts correct IAW 0-NOP-074.01.

C.	Incorrect. Part 1 is incorrect, but plausible given Feedwater bypass valves will be throttled to control flow after the pump is started. Part 2 is incorrect, but plausible given 72 hours is a common action in technical specifications, is also consistent with tech spec basis for AFW maintaining the unit at hot standby for 72 hours on a loss of all AC power		
D.	Incorrect. Part 1 is correct. Part 2 is incorrect.		

Technical Reference(s)	0-NOP-074.01		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		8
	55.43		
Components, capacity, and functions of emergency systems.			
Comments:			

2.2 Limitations (continued)

9. The following parameters alarm on Unit 4 Annunciator J-9/1, SSGFP B TROUBLE:

- Low gear oil press at 11 psig
- High gear oil temperature at 225°F
- Loss of charger output
- Loss of ac power
- Battery #1 OR #2 failure
- Over cranking at 6 attempts

10. J-9/1 SSGFP B TROUBLE, IN ALARM conditions which are only alarms, will reset when the condition is clear.

11. J-9/1 SSGFP B TROUBLE alarm conditions which are trip functions require the Master Control Switch to be placed in OFF then LOCAL/REMOTE to CLEAR.

12. The SSGFP B engine is equipped with an automatic cranking circuit. This circuit will provide six crank periods separated by five rest periods each of approximately 15 seconds duration. The engine may require up to four minutes to start and come up to full speed.

13. Standby Feedwater had proven ineffective at maintaining Steam Generator Levels during Turbine rollup to speed. Ref. CR-1865591

14. The B SSGFP fuel tank capacity allows the engine to run for ~11 hours (AR 1809492).

4.1 Startup (continued)

7. **ENSURE** DWDS-188, DWST HOSE STATION ISOLATION, CLOSED.

8. **ENSURE** the following valves are CLOSED to prevent filling the condenser hotwell:

   - 3(4)-20-120, A SGFP DISCH MOV DOWNSTREAM BYPASS.
   - 3(4)-20-220, B SGFP DISCH MOV DOWNSTREAM BYPASS.

NOTE

Throttling DWDS-3(4)-012, SSGFP DISCH TO UNIT 3(4) ISOL prevents excessive differential pressure and allows manual valve operation.

9. IF the Feedwater System pressure is less than 500 psig as indicated on PI-3(4)-1616, PRESSURE INDICATOR FOR FEEDWATER HEADER REMOTE, THEN **THROTTLE** DWDS-3(4)-012, SSGFP DISCH TO UNIT 3(4) ISOL, three turns OPEN on the Unit to be supplied with feedwater.

10. IF the Feedwater System is pressurized greater than 500 psig as indicated on PI-3(4)-1616, PRESSURE INDICATOR FOR FEEDWATER HEADER REMOTE, THEN **ENSURE** DWDS-3(4)-012, SSGFP DISCH TO UNIT 3(4) ISOL is CLOSED.

11. IF starting Standby Steam Generator Feedwater Pump A, THEN:

    A. **ENSURE** oil level is between the upper and lower line marks on the inboard and outboard motor bearing bulls-eye.

    B. **ENSURE** motor bearing temperature pyrometers are set at 180°F.

    C. **START** SSGFP A.

    D. **ENSURE** motor operating amperage lower than 115 amps.

    E. **OPEN** DWDS-3(4)-012, SSGFP DISCH TO UNIT 3(4) ISOL.

    F. **ENSURE** motor operating amperage lower than 115 amps.

    G. **NOTIFY** Reactor Control Operator that feedwater is available up to the S/G Feedwater Bypass Valves.

4.1 Startup (continued)

12. IF starting Standby Steam Generator Feedwater Pump B, THEN,

A. **CHECK** Unit 4 Annunciator Window J-9/1, SSGFP B TROUBLE alarm CLEAR.

B. **ENSURE** Master Control Switch in LOCAL/REMOTE.

C. **CHECK** CB-1 SWITCH Battery/Battery Charger Disconnect is ON.

D. **CHECK** CB-2 SWITCH Battery/Battery Charger Disconnect is ON.

E. **CHECK** Local/Remote light is ON.

F. **CHECK** BATT 1 light is ON.

G. **CHECK** BATT 2 light is ON.

H. **ENSURE** Idle Switch in CLOSE (RATED SPEED).

NOTE

SSGFP B operability requires a fuel volume of 240 gallons which is a level of greater than 7.5 inches above the bottom of the fuel tank at the filler location or greater than 5/8 full.

I. **ENSURE** SSGFP B fuel tank to be greater than 3/4 full by:

- Fuel level gauge on top of tank

  OR

- Using clean metal or wood ruler, with no loose parts, measure equal to or greater than 9 inches above bottom of tank at the filler location

J. **CHECK** jacket water temperature on TI-7033, SSGFP B ENGINE COOLANT TEMP, is approximately 150°F.

K. **ENSURE** oil and coolant fluid levels are NORMAL.

L. **INSPECT** the engine for evidence of fuel, coolant, or oil leakage.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	061		K5.01
	Importance Rating	3.6		

Knowledge of the operational implications of the following concepts as the apply to the AFW: Relationship between AFW flow and RCS heat transfer

Proposed Question: RO Question # 45

Given the following conditions:

- Unit 4 trips due to a loss of main feedwater.
- 4B 4kV Bus is locked out.
- Only the A AFW Pump is in service.
- A AFW Pump speed begins to lower due to a malfunctioning governor.

Which one of the following describes (1) how the change in AFW flow will initially affect pressurizer level and (2) the reason why?

A.	(1) pressurizer level will rise (2) changing saturation conditions in the vessel head
B.	(1) pressurizer level will rise (2) changing primary to secondary heat transfer rate
C.	(1) pressurizer level will lower (2) changing saturation conditions in the vessel head
D.	(1) pressurizer level will lower (2) changing primary to secondary heat transfer rate

Proposed Answer: B

A.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate believes that saturation conditions may develop quickly without a secondary heat sink. The candidate assumes that given only the 3A RCP is running (no spray capability) and only one SG can be steamed and only for a limited time, that RCS conditions deteriorate.

B.	Correct. As AFW Pump speed decreases due to the governor valve closing, less heat is removed from the RCS via less steam to the pump turbine and less feedwater flow generated. PRZ will rise as the RCS fluid expands.
C.	Incorrect. Part 1 is incorrect. Part 2 is incorrect. Combination is plausible, if candidate believes overall vessel temperature rises causing the steam bubble in the PRZ to also heatup along with pressure to rise. This pressure rise compresses the PRZ fluid. The candidate assumes that given only the 3A RCP is running (no spray capability) and only one SG can be steamed and only for a limited time, that RCS conditions deteriorate.
D.	Incorrect. Part 1 is incorrect. Part 2 is correct. Combination is plausible if the candidate believes reactor power rises due to cold water addition from AFW and therefore opposite holds true- power lowers due to less cold water addition. This causes the RCS to cool and shrink.

Technical Reference(s)	3-OSP-075.1, Auxiliary Feedwater Train 1 Operability Verification		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	12911	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2011	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	14	
	55.43		
Principles of heat transfer, thermodynamics and fluid mechanics.			
Comments: 12911. Made 2x2.			

Facility: WTSI Corporate

Question 45 original

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following:

- Unit 4 was operating at 100% power.
- A Reactor Trip due to a Loss of Main Feedwater.
- 4B 4KV Bus is locked out.
- Due to equipment malfunctions, ONLY A AFW Pump is in service.
- The A AFW Pump speed has begun to slowly LOWER due to a malfunctioning governor.

Which ONE of the following describes how the change in AFW flow will affect Pressurizer Level, including the reason?
Indicated Pressurizer Level will initially 5

A. rise due to a bubble formation in the Rx Vessel Head

B. rise due to decreased primary to secondary heat transfer

C. lower due to the density change in the RCS

D. lower due to decreasing Charging flow

Proposed Answer: B

Explanation (Optional):

A. Incorrect since a bubble should not form in the Reactor Vessel Head with an RCP running. Plausible because rises could be correct if such a bubble was to form.

B. CORRECT. As AFW Pump speed decreases due to the governor valve closing, less heat is removed from the RCS via less steam to the pump turbine and less feedwater

flow generated. PZR will rise as the RCS fluid expands.

C. Incorrect since PZR level will rise, not lower. Plausible because density decrease in Pressurizer level is possible with an insurge of cooler water lowering the saturation temperature of the fluid. This effect will cause the water volume to contract.

D. Incorrect since PZR level will rise, not lower. Plausible because PZR level will lower after the initial increase as charging pump speed adjusts to the rising level. This is a subsequent effect

Techical Reference(s): - (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: - (As available)

Question Source: Bank 12911

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 14

55.43

Principles of heat transfer, thermodynamics and fluid mechanics.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	062		A1.01
	Importance Rating	3.4		

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

Proposed Question: RO Question # 46

Given the following conditions:

- A loss of off-site power and SI has occurred on Unit 4.
- 4A EDG is powering the 4A 4KV Bus.
- 4A EDG is loaded to 2500 KW with essential loads.
- The RO is directed to load additional equipment in the following order:

A containment spray pump	212 KW
A battery charger	56 KW
A CRDM fan	48 KW
A computer room chiller	43 KW
A battery room ac	30 KW
An electrical equipment room A/C	25 KW

Which one of the following indicates the maximum number of components (if any) that may be started, in the order listed, before steady state loading limits on the 4A EDG are exceeded?

A.	0
B.	1
C.	4
D.	6

Proposed Answer: C

A.	Incorrect, plausible 2500 = steady state limit 2500KW unit 3
B.	Incorrect, plausible 2500+212= 2712, 2750 transient limit 2750KW
C.	Correct. 2500+212+56+48+43 =2859 2874KW is the limit
D.	Incorrect, plausible 2500+212+56+48+30+25, confuses with U4 transient limit 3162KW

Technical Reference(s)	4-ONOP-023.2 4-EOP-ES-0.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			
Learning Objective:	6902136 obj 4		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	8	
	55.43		
Components, capacity, and functions of emergency systems.			
Comments:			

ATTACHMENT 3
Unit 4 Component KW Load Rating Chart
(Page 1 of 2)

CAUTION

- When using this chart for Attachment 2 with one Unit 4 EDG supplying power to 4A or 4B 4KV bus and to Unit 3 via the SBO Tie, the 2000-hour rating in brackets may be used.
- Steady state loading on each Unit 4 EDG shall **NOT** exceed 2874[3095] KW When starting additional equipment, diesel load is required to be monitored to ensure the transient limit of 3162[3195] KW is **NOT** exceeded.

NOTE

- One Computer Room Chiller is required to be restarted within 60 minutes of Loss of Offsite Power to maintain operability of DCS and QSPDS.
- Battery Charger load is dependent on the status of its parallel charger (i.e., in service or de-energized).

ESSENTIAL LOADS

COMPONENT	KW	COMPONENT	KW
CCW PUMP	380	LIGHTING PANEL 4DP87	22
HIGH-HEAD SI PUMP	302	BATTERY CHARGER 3B2	20/39
INTAKE COOLING WATER PUMP	265	BATTERY CHARGER 4A1	20/39
RHR PUMP	222	EMERGENCY LIGHTING	18
CONTAINMENT SPRAY PUMP	212	INSTRUMENT AIR DRYER	18
NORMAL CONTAINMENT COOLER	77	SWITCHGEAR/LC 4A A/C AHU	17
PRIMARY WATER PUMP	49	SWITCHGEAR/LC 4B A/C AHU	17
CRDM COOLER FAN	48	DG AIR COMPRESSOR	13
COMPUTER ROOM CHILLER	43	EDG RM LIGHTING PANEL 4PD88	11
AUXILIARY BLDG EXHAUST FAN	33	AUXILIARY BLDG SUPPLY FAN	9
BATTERY ROOM A/C	30	H2 ANALYZER HEAT TRACE	6
BATTERY CHARGER 3A2	29/56	CABLE SPREADING ROOM A/C	5
BATTERY CHARGER 4B1	29/56	CONTROL ROOM FILTER FAN	3
CONTROL ROOM A/C COMPR	27	COMPUTER ROOM AIR UNIT	3
SWITCHGEAR/LC 4A A/C CHILLER	26	SWITCHGEAR 4D SUPPLY FAN	2
SWITCHGEAR/LC 4B A/C CHILLER	26	DG CONTROL ROOM SUPPLY FAN	2
ELECTRICAL EQUIP RM A/C	25	DG CIRC OIL PUMP	1
EMERGENCY CNTMT COOLER	23	DG FUEL OIL TRANSFER PUMP	1
LIGHTING PANEL 4DP86	22	DG TURBO OIL PUMP	1
		H2 ANALYZER PUMP	1

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***Operating the isolation switches with the EDG loaded, will trip the EDG.***
- ***Steady state loading on each Unit 4 EDG shall not exceed 2874 KW. Load transients up to 3162 KW are acceptable when starting additional equipment.***

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16	**Maintain Running Diesel Generator(s)**	
	a. Verify voltage - BETWEEN 3950 AND 4350 – VOLTS	a. Adjust Voltage Adjust Control Switch.
	b. Verify frequency - BETWEEN 59.4 AND 60.6HZ	b. Adjust Governor Control Switch.
	c. Verify load - LESS THAN 2874 KW	c. Notify RO to shed non-essential loads.
	d. Operate diesel generator controls as directed by the RO	
17	**Obtain Permission To Transfer Affected Emergency Diesel Generator Operation To Control Room**	Perform the following: a. **IF** Control Room has been evacuated, **THEN** return to 0-ONOP-105, CONTROL ROOM EVACUATION. b. Return to Step 16.
18	**Place Affected Emergency Diesel Generator Master Control Switch In NORMAL**	
19	**Check Unit 4 Emergency Diesel Generators - BOTH RUNNING**	**IF** local start of second emergency diesel generator is required, **THEN** return to Step 1.
20	**Return To Procedure And Step In Effect**	

END OF TEXT

FINAL PAGE

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	063		A1.01
	Importance Rating	2.5		

Ability to predict and/or monitor changes in parameters associated with operating the dc electrical system controls including: Battery capacity as it is affected by discharge rate

Proposed Question: RO Question # 47

Given the following conditions:

- A loss of all AC power has occurred on Unit 3.
- 3-EOP-ECA-0.0, Loss of All AC Power, is entered.
- 4kV bus power can NOT be restored.

Which one of the following completes the statement below?

Operators are directed to shed non-essential loads in order to ___(1)___ .

Vital DC battery discharge rate will be monitored by reading Vital DC bus voltages ___(2)___ .

A.	(1) lower battery discharge rates to lengthen availability of vital equipment (2) on DCS
B.	(1) lower battery hydrogen generation rates while no ventilation is available (2) on DCS
C.	(1) lower battery discharge rates to lengthen availability of vital equipment (2) at VPA
D.	(1) lower battery hydrogen generation rates while no ventilation is available (2) at VPA

Proposed Answer: A

A.	Correct. Both correct IAW 3-EOP-ECA-0.0.

B.	Incorrect. Part 1 is incorrect but plausible since 90 min is time DC load shedding is required to be completed. Part 2 is correct.		
C.	Incorrect. Part 1 is correct. Part 2 is incorrect but plausible since 480V LCs, bus breakers and battery charger status is on VPA.		
D.	Incorrect. Both parts incorrect. Plausible for same reasons as options B and C		
Technical Reference(s)	3-EOP-ECA-0.0		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902348 obj 4		(As available)
Question Source:	Bank	9634	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Ginna
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments: 9634. Modified for flex and made 2x2 to monitor changes.			

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

CAUTION

If both Units have experienced a Loss Of All AC Power, AND **neither** Unit's A **NOR** B 4KV Buses can be promptly re-energized, then Vital DC Load Shedding per 0-FSG-04 is required to be completed within 90 minutes of event initiation.

NOTE

An Extended Loss of AC Power (ELAP) exists if either of the following occurred:

* Dual unit Loss Of All AC Power

OR

* Single unit Loss Of All AC Power with inability to meet SBO power restoration time requirements

9. Check If ELAP In Progress

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check opposite unit 4KV buses (A AND B) – AT LEAST ONE ENERGIZED	**a.** Go to Step 9.d.
b. Check either of the following – * Elapsed time since reaching Attachment 6, Step 5 NOTE – GREATER THAN 10 MINUTES OR * Elapsed time since reaching Attachment 7, Step 4 NOTE – GREATER THAN 10 MINUTES	**b.** WHEN greater than 10 minutes has elapsed, THEN continue with Step 9.v.
c. Go to Step 9.v	
d. Initiate Containment Isolation Phase A	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	**(continued)**	
	t. Direct personnel to periodically, locally or on DCS, monitor Vital DC Bus voltages: • 3D01 • 3D23 • 4D01 • 4D23	
	u. Observe NOTE prior to Step 10, and go to Step 10	
	v. Direct <u>opposite</u> Unit RO to verify at least <u>one</u> battery charger supplying each Vital DC Bus using 0-FSG-99, Attachment 23, Battery Charger Alignment from Unit 4 Verification	
	w. Dispatch operator(s) to perform 0-FSG-05, INITIAL ASSESSMENT AND FLEX EQUIPMENT STAGING	
	x. Check AFW pumps – MORE THAN ONE PUMP RUNNING ON A SINGLE TRAIN	**x.** Go to Step 9.z.
	y. Direct Personnel to establish <u>only one</u> AFW pump running per train by coordinating with opposite unit, and shutting down the appropriate AFW pump using 3-NOP-075, AUXILIARY FEEDWATER SYSTEM	
	z. Check AFW trains – TWO OPERATING	**z.** Go to Step 9.cc.

Facility: WTSI Corporate

Question 47 original

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		
Group #		
Topic & KA #		
Importance Rating:		

KA Statement

Proposed Question:

Plant conditions:

- A loss of all AC power has occurred.
- The crew is performing actions of ECA-0.0, Loss of All AC Power.
- The crew is evaluating load shed of the Batteries

Which ONE of the following describes the reason for requirement to shed non-essential DC loads in accordance with ECA-0.0, Loss of All AC Power?

A. Battery discharge rate is reduced to ensure the station meets the 2 hour technical specification design basis requirement for battery capacity following a loss of AC power

B. Battery discharge rate is reduced to ensure the station meets the 4 hour technical specification design basis requirement for battery capacity following a loss of AC power

C. Battery discharge rate is reduced to ensure the station meets the 2 hour coping requirement for loss of all AC power

D. Battery discharge rate is reduced to ensure the station meets the 4 hour coping requirement for loss of all AC power

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible since there is a design basis assumption contained in TS, but it is 4 hours, not 2 hours. 2 hours is plausible because it is the allowed TS action time for loss of DC.

B. Incorrect. Plausible since the TS design basis is 4 hours, but load shedding is not required to achieve design basis operation of the battery

C. Incorrect. In accordance with the Station Blackout Program Plan, the coping requirement is 4 hours. 2 Hours is plausible because of the TS action time in section 3.8 for battery or DC bus inoperability

D. Correct. Load is shed to ensure that a loss of AC power lasting up to 4 hours will not fully discharge the batteries

Techical Reference(s): ECA-0.0 step 17 and background document
Station Blackout Program Plan (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: REC00C, Obj 1.03 (As available)

Question Source: Bank 9634
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2010 Ginna

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	063		A4.01
	Importance Rating	2.8		

Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses

Proposed Question: RO Question # 48

Given the following conditions:

Unit 3 is at 10% power.
A loss of Vital DC bus 3D23 occurs.

Which one of the following describes the direct effect on the RTBs?

A.	3A RTB opens due to loss of power to the undervoltage trip coil
B.	3B RTB opens due to loss of power to the undervoltage trip coil
C.	3A RTB opens due to loss of power to the shunt trip coil
D.	3B RTB opens due to loss of power to the shunt trip coil

Proposed Answer: B

A.	Incorrect. Plausible when the candidate confuses A and B trains and confuses the shunt trip with the undervoltage trip coil. The 3A RTB does not trip on loss of B train control power.
B.	Correct. Correct Train. Indication, UV coil, and shunt trip coil receive power from DC bus. Loss of DC results in a loss of power to the UV coil, causing it to drop out and causing the breaker to open
C.	Incorrect. Plausible when the candidate confuses A and B trains and confuses the shunt trip with the undervoltage trip coil. The 3A RTB does not trip on loss of power to shunt coil because the shunt coil requires control power to operate.

D.	Incorrect. Correct train. Plausible since indication is lost and power is lost to UV coil, however the breaker shunt coil uses control power and it will not be capable of tripping on a shunt trip		
Technical Reference(s)	6902163		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902163 Obj 5		(As available)
Question Source:	Bank	9649	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Ginna
Question Cognitive Level:		Memory or Fundamental Knowledge	
		Comprehension or Analysis	X
10 CFR Part 55 Content:		55.41	7
		55.43	
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments: 9649. Added train specific content on loss of vital dc bus 3D23.			

1.0 PURPOSE

Provides instructions to stabilize the plant and recover 3D23 and 3D23A, 3B DC BUS, in the event they are de-energized with the unit initially in MODE 1, Power Operation, and the Auxiliary Transformer supplying Plant loads. This procedure is performed after the unit has been stabilized per 3-EOP-ES-0.1, Reactor Trip Response.

2.0 ENTRY CONDITIONS

2.1 Indications

- Reactor and Turbine Trip due to loss of DC Power (3D23-8) to Reactor Trip Breaker B undervoltage coil.
- DC Load Center 3D23 and 3D23A, 3B DC BUS, voltmeter indicates voltage is zero.
- IF initially aligned to 3D23 and 3D23A, THEN 480V LC 4H and 4160V Swgr 4D control power transferred to 3D01 and 3D01A.
- Any of the following inverters that are in service transfer to CVT
  - 3Y05, INVERTER
  - 4Y05, INVERTER
  - 3Y06, SPARE INVERTER
- IF LC 3C was available and LC 3H was aligned to LC 3D, THEN LC 3H will transfer to LC 3C.
- IF LC 3C was **NOT** available, THEN LC 3H and MCC 3D are deenergized.
- Various valves fail as indicated on Attachment 1, Valve Failure Positions For Loss of DC Bus 3D23 and 3D23A (3B).
- Loss of power to Backup Generator Lockout Relay
- Loss of 3B Bus Load Sequencer

NOTE
The 3B EDG does **NOT** have black start capabilities.

- Loss of 3B EDG
- Loss of Train 2 feedwater isolation capability of S/G Feedwater Bypass valves.

Facility:	WTSI Corporate	Question 48 original	
Vendor	WEC		
Exam Date:			
Exam Type:			

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

During operation at power with the Reactor Trip Breakers (RTBs) closed, a loss of 125 VDC control power to one of the RTBs occurs.

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

A. RTB opens due to loss of power to the undervoltage trip coil.

B. RTB opens due to loss of power to the shunt trip coil

C. RTB remains closed, and the undervoltage trip coil will not function on a reactor trip signal from the Reactor Protection System

D. RTB remains closed, and the shunt trip coil will not open on a reactor trip signal from the Reactor Protection System

Proposed Answer: A

Explanation (Optional):

A. Correct. Indication, UV coil, and shunt trip coil receive power from DC bus. Loss of DC results in a loss of power to the UV coil, causing it to drop out and causing the breaker to open

B. Incorrect. The breaker does not trip on loss of power to shunt coil because the shunt coil requires control power to operate. The undervoltage coil losing power

would cause a reactor trip breaker to open

C. Incorrect. Indication is lost and power is lost to UV coil, but plausible because the breaker shunt coil uses control power and it will not be capable of tripping on a shunt trip

D. Incorrect. Indication and shunt trip capability lost. Plausibility is as described in options above

Techical Reference(s): R3501C Rev 28 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: R3501C, Obj 1.10 (As available)

Question Source: Bank 9649
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2010 Ginna

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	064		K6.07
	Importance Rating	2.7		

Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers

Proposed Question: RO Question # 49

Given the following conditions:

- With Unit 3 at 100% power, the following alarm is received:
- F 8/2, EDG A TROUBLE
- ANN 1/10, LOW AIR PRESSURE, is locked in at the local panel.
- The left-side Air Receivers for the 3A EDG are at 195 psig.
- The right-side Air Receivers for the 3A EDG are at 205 psig.

Which one of the following completes the statement below?

In accordance with 3-OP-023, Emergency Diesel Generator, the 3A EDG ___(1)___ available to start and the right-side air receivers ___(2)___ be aligned to supply both the left-side and right-side set of Air Start Motors.

A.	(1) is (2) will
B.	(1) is NOT (2) will
C.	(1) is (2) will NOT
D.	(1) is NOT (2) will NOT

Proposed Answer: C

A.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate believes flasks can be cross-tied just like 3A and 3B EDG starting air compressor can be cross-tied IAW 3-OP-023 as directed by the ARP.
B.	Incorrect. Part 1 is incorrect, but plausible because 195 psig is below the alarm setpoint but the EDG is operable above 160 psig, and the 4A EDG has a diesel air compressor, but not the 3A EDG 195 psig is incorrect but plausible because it is below both the normal operating pressure of 225 psig and the alarm setpoint of 200 psig. Also because the unit 4 EDGs have air compressors have diesel air compressors but the unit 3 air start system does not. Part 2 is incorrect.
C.	Correct. EDG is operable above 160 psig. The air flasks can NOT be cross-tied. 3-OP-023. Violates basis for crank attempts on one train of air.
D.	Incorrect. Part 1 is incorrect. Part 2 is correct.

Technical Reference(s)	3-OSP-023.1, Att 3 3-OP-023 3-ARP-097.DG.A 1/10		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902136 obj 11a		(As available)
Question Source:	Bank	8605	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		8
	55.43		
Components, capacity, and functions of emergency systems.			
Comments:			
Question Bank 8605 2010 PTN modified.			

CAUSES:
1. Starting Air Compressor Failure
2. Failure of PS-3-6696A1 or A2
3. Loss of power to Starting Air Compressors
4. Leak in Starting Air System

1/10

LOW AIR PRESSURE

DEVICE:
PS-3-6696A1
PS-3-6696A2

SETPOINT:
200 psig

LOCATION:
At Starting Air Compressor

ALARM CONFIRMATION

1. **CHECK** PI-3-3669A, STARTING AIR PRESSURE, indicates less than 200 psig.

OPERATOR ACTIONS

1. **NOTIFY** Control Room.
2. IF EDG A STARTING AIR COMPRESSOR is **NOT** operating, THEN:
   A. **PLACE** EDG A STARTING AIR COMPRESSOR is in RUN.
   B. **ENSURE** breaker 30504, 3A EDG STARTING AIR COMPRESSOR BKR, is ON.
3. IF EDG A STARTING AIR COMPRESSOR can **NOT** be started, THEN with permission from Control Room, **PERFORM** Cross-Tying Air Start Systems per 3-OP-023, Emergency Diesel Generator.
4. IF EDG A STARTING AIR COMPRESSOR is operating, THEN **CHECK** for leaks in Air Start System.

NOTE

An EDG may be considered operable with receiver tank air pressure below 200 psig, as long as tank pressure is above 160 psig.

5. **RESTORE** Air Pressure to at least 200 psig expeditiously.
6. **NOTIFY** Control Room of EDG 3A status.
7. IF EDG 3A is inoperable, THEN **REFER TO**:
   - TS 3.8.1.1, AC Sources Operating
   - TS 3.8.1.2, AC Sources Shutdown

REFERENCES:
1. TS 3.8.1.1, AC Sources Operating
2. TS 3.8.1.2, AC Sources Shutdown
3. 5613-M-3022, Sh 1
4. 5613-M-16-69, Sh 1A2
5. CR 02-0153

4.20 Automatic or manual start of any of the following pumps on the same bus as a loaded diesel could cause the loaded EDG to Trip or sustain damage and shall be avoided:

4.20.1 Steam Generator Feed Pump (3A EDG only)

4.20.2 Reactor Coolant Pump

4.20.3 Condensate Pump

4.20.4 Circulating Water Pump

4.20.5 Heater Drain Pump

4.21 Due to the possibility of inadvertently removing an Emergency Diesel Generator from service while performing maintenance on Starting Air Systems, the Shift Manager should perform an operability assessment on Starting Air. This evaluation should take into account the last time the train that is to be left in service was tested.

4.22 When a set of Air Start Motors is declared inoperable, the EDG can be considered operable if the remaining set of Air Start Motors started the EDG during its last surveillance (60 days ago). One of the following actions should be performed expeditiously:

4.22.1 Repair the Air Start Motors and start the EDG with the repaired set of Air Start Motors,

OR

4.22.2 Start the EDG with the remaining operable set of Air Start Motors,

OR

4.22.3 Declare the EDG inoperable and test the remaining trains as directed by TS 3.8.1 for an inoperable EDG.

The EDG failure to start may still require a special report in accordance with TS 3.8.1.

4.23 An EDG may be considered operable if the Air Start Receivers have a pressure of less than 200 psig, but greater than 160 psig. Air pressure should be restored to the Air Start Receivers expeditiously to greater than 200 psig or declare the affected EDG inoperable in accordance with TS 3.8.1. (CR 02-0153)

4.24 A minimum of two Air Reservoirs are required to be available for each set of Air Start Motors as per design basis. One pair of Air Reservoirs can not be used to provide air to both sets of Air Start Motors.

4.25 Ensure any make up water added to the EDG Cylinder Jacket Cooling Water System contains less than 200 ppb ammonia unless water is added to maintain EDG operable.

4.26 The EDG Cooling System contains chromium and its compounds which are known carcinogens. Avoid inhalation or contact with skin and eyes.

4.27 Any waste generated containing chromium is a hazardous waste and must be placed in the designated satellite accumulation drum.

Facility:	WTSI Corporate		Question 49 original	
Vendor	WEC			
Exam Date:				
Exam Type:				
Examination Outline Cross-reference:		Level	RO	SRO
		Tier #		
		Group #		
		Topic & KA #		
		Importance Rating:		

KA Statement

Proposed Question:

The ANPO reports A and B Air Receivers for the 4A EDG are at 155 psig and the associated Air Compressor will not load.

In accordance with 4-ARP-097.DG, Diesel Generator Panel Annunciator Response, which ONE of the following identifies if the 4A EDG is OPERABLE and the required response to the above event?

A.	OPERABLE	start the 4A EDG Diesel Air Compressor
B.	NOT OPERABLE	start the 4A EDG Diesel Air Compressor
C.	OPERABLE	Cross-tie with the 4B EDG starting air7
D.	NOT OPERABLE	Cross-tie with the 4B EDG starting air

Proposed Answer: B

Explanation (Optional):

A. Incorrect IAW above discussion. Plausible - the value is stem is near the value at which must be declared inoperable

B. Correct IAW above discussion

C. Incorrect IAW above discussion. Plausible - this is the required action for Unit 3

D. Incorrect IAW above discussion. Plausible - this is the required action for Unit 3 and the value is stem is near the value at which must be declared inoperable

Techical Reference(s):
1. 5614-M-3022 sheet 1 rev. 10
2. 5613-M-3022 sheet 1 rev. rev. 16
3. 4-ARP-097.DG p. 18 rev. 6/15/02

IAW the ARP, may be considered operable as long as tank pressure is above 160 psig. Required to start the diesel air compressor

(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: 6900136 EO 11a (As available)

Question Source: Bank 8605

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010 Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	073		K3.01
	Importance Rating	3.6		

Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases

Proposed Question: RO Question # 50

Given the following conditions:

- Blowdown is 60k lbm/hr per SG on Unit 3.

Subsequently:

- Annunciator H-1/6, PRMS CHANNEL FAILURE actuates.
- R-19, SG Blowdown Radiation Monitor, high alarm is present.

Which one of the following completes the statements below?

Blowdown Isolation Valves, CV-3-6275A, B and C ___(1)___ auto close.

Blowdown Flow Control Valves, FCV-3-6278 A, B and C ___(2)___ auto close.

A.	(1) will (2) will
B.	(1) will (2) will NOT
C.	(1) will NOT (2) will
D.	(1) will NOT (2) will NOT

Proposed Answer: C

A.	Incorrect. Part 1 is incorrect. Plausible since blowdown isolation valves may logical thought to isolate on R-19 failing high. Also, these CVs automatically close on a phase A or AFW actuation. Part 2 is correct.
B.	Incorrect. Plausible for same reason as A. Part 2 is incorrect. Plausible since blowdown isolation valves are thought to isolate, FCVs which are downstream are not needed.
C.	Correct. CVs close on phase A or AFW actuation not R-19 failing high. All blowdown flow control valves close on the failure.
D.	Incorrect. Part 1 is correct. Part 2 is incorrect. This combination is plausible when the candidate believes that manual isolation must be required given that the channel has failed and will not cause an automatic isolation.

Technical Reference(s)	3-ARP-097.CR.H 1/6		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		11
	55.43		
Purpose and operation of radiation monitoring systems, including alarms and survey equipment.			
Comments:			

CAUSES:
1. Loss of detector counts for three minutes (30 seconds for R20)
2. Loss of power to PRMS drawer
3. RANGE switch **NOT** in normal position(except R15 and R20)
4. Loss of power to R20 Local Ratemeter in Pipe and Valve Room.

H1/6

PRMS CHANNEL FAILURE

DEVICE:	SETPOINT:	LOCATION:
• RANGE switch (except R15 and R20)	In other than normal position	PRMS drawer
OR • FAIL relay K-3 (except R15 and R20)	N/A	PRMS drawer

ALARM CONFIRMATION

1. **CHECK** the following:
   - Fail lamp on PRMS drawer or on R20 Local Ratemeter in Pipe and Valve Room
   - Loss of power to PRMS channel or to R20 Local Ratemeter in Pipe and Valve Room
   - Except for R15, loss of detector counts for three minutes (30 seconds for R20)
   - For R15, green OPERATE LED is OFF

OPERATOR ACTIONS

1. IF **NOT** under test, THEN **DETERMINE** which channel is alarming AND **RETURN** switches or power alignment to normal.
2. **CHECK** for channel failure.
3. IF R-14 fails, THEN **STOP** gas decay release.
4. IF R-18 fails, THEN **STOP** liquid release.
5. IF R-19 fails, THEN **SECURE** S/G blowdown.
6. **REFER TO** Tech Spec 3/4.3.3, 3/4.4.6, and 3/4.9.13.

REFERENCES: Tech Spec Sections 3/4.3.3, 3/4.4.6 and 3/4.9.13

FOLDOUT PAGE

1. 3-EOP-E-0 TRANSITION CRITERIA

a) **IF** RCS Tavg - GREATER THAN Tref by 6 °F, **THEN** trip the Reactor and Turbine **AND** go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

b) **IF** any of the following limits are reached, **THEN** trip the Reactor and Turbine, initiate Safety Injection and Phase A, **AND** go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.
   1) RCS Leakage greater than Charging Pump capacity **AND** letdown isolated
   2) PZR Level can **NOT** be maintained within 10% of program

2. CONTROL ROOM VENTILATION MANUAL ISOLATION CRITERIA

IF a reactor trip occurs and any PRMS channels listed below is in the alarm state, **THEN** manually align Control Room ventilation for emergency recirculation mode with 30 minutes of the alarm:
* R-15 Condenser Air Ejector Monitor
* R-19 Steam Generator Blowdown Monitor
* R-20 CVCS Letdown Line Radioactivity Monitor

3. TURBINE LOAD WITHIN 10% OF TARGET POWER LEVEL

WHEN turbine load is within 10% of end target load, **THEN** stop boration by performing the following:
   1) Place the Reactor Makeup Selector Switch to Auto.
   2) Set FC-3-113A, Boric Acid Flow Controller pot setting as desired.
   3) Place the RCS Makeup Control Switch to Start.

4. BLOWDOWN RELEASE PATH ISOLATION

IF PRMS R-19 Count Rate is increasing **OR** High Alarm is present, **THEN** verify the following:

a) Steam Generator Blowdown Flow Control Valves are Closed.
   - FCV-3-6278A
   - FCV-3-6278B
   - FCV-3-6278C

b) Blowdown Tank to Canal Level Control Valve, LCV-3-6265B is Closed.

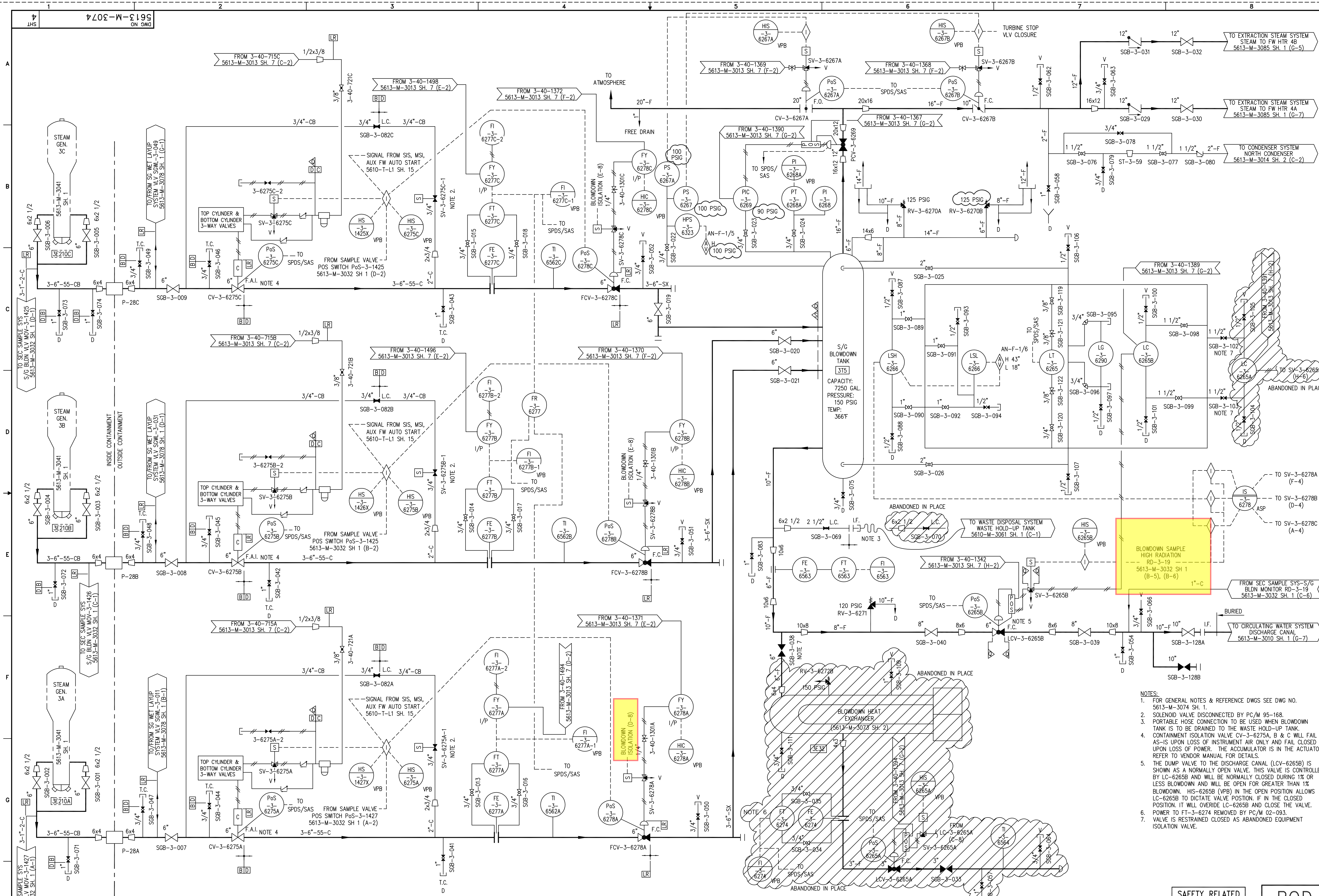
c) **WHEN** R-19 High Alarm is present, **THEN** verify NO FLOW on S/G Sample Flow Indicators at the Cold Chem Lab. (Ensures Sample Valves SV-3-2800, SV-3-2801, SV-3-2802 are Closed.)

5. AFW STEAM SUPPLY RELEASE PATH ISOLATION

WHEN the affected Steam Generator is identified, **THEN** perform the following**:**

a) Verify Steam Supply aligned to both trains of AFW from the Intact Steam Generators

b) Verify Closed **AND** De-Energize the affected Steam Generator AFW Steam Supply MOV using ATTACHMENT 4.



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5613-M-3074 SH. 1.
2. SOLENOID VALVE DISCONNECTED BY PC/M 95-168.
3. PORTABLE HOSE CONNECTION TO BE USED WHEN BLOWDOWN TANK IS TO BE DRAINED TO THE WASTE HOLD-UP TANK.
4. CONTAINMENT ISOLATION VALVE CV-3-6275A, B & C WILL FAIL AS-IS UPON LOSS OF INSTRUMENT AIR ONLY AND FAIL CLOSED UPON LOSS OF POWER. THE ACCUMULATOR IS IN THE ACTUATOR, REFER TO VENDOR MANUAL FOR DETAILS.
5. THE DUMP VALVE TO THE DISCHARGE CANAL (LCV-6265B) IS SHOWN AS A NORMALLY OPEN VALVE. THIS VALVE IS CONTROLLED BY LC-6265B AND WILL BE NORMALLY CLOSED DURING 1% OR LESS BLOWDOWN AND WILL BE OPEN FOR GREATER THAN 1% BLOWDOWN. HIS-6265B (VPB) IN THE OPEN POSITION ALLOWS LC-6265B TO DICTATE VALVE POSTION. IF IN THE CLOSED POSITION. IT WILL OVERIDE LC-6265B AND CLOSE THE VALVE.
6. POWER TO FT-3-6274 REMOVED BY PC/M 02-093.
7. VALVE IS RESTRAINED CLOSED AS ABANDONED EQUIPMENT ISOLATION VALVE.

SAFETY RELATED

POD

THIS DRAWING SUPERSEDES DRAWINGS:
5610-M-1316 REV. 15
5610-M-341 REV. 11

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4062 SH. 4 REV. 53

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
26	11-02-10	ISSUED AS-BUILT PER EC 242489.	RV	CBW	PJV	MG	30	06-05-12	ISSUED AS-BUILT PER EC 247006. (PARTIAL)	RV	SB	–	CW
25	10-29-10	ISSUED AS-BUILT PER EC 247066 (PC/M 10-061) (PARTIAL).	RH	BB	DW	PRB	29	12-27-11	ISSUED AS-BUILT PER EC 247066. (PARTIAL)	RV	BB	–	PRB
24	5-12-09	ISSUED AS-BUILT PER CRN-I-4315 (PC/M 05-026).	RV	RH	JPB	SRR	28	12-17-11	ISSUED AS-BUILT PER EC-DCR 274982.	RH	RV	–	BB
23	4-10-09	ISSUED AS-BUILT PER PC/M 09-043. (PARTIAL)	RV	BB	DW	JEA	27	1-10-11	ISSUED AS-BUILT PER EC-DCR 270586.	RH	RV	BSG	PRB
32	7-28-12	ISSUED AS-BUILT PER EC 247006 (PC/M 09-137).	RV	AFG	–	PSB	0	5-14-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	MAB	MD	AM/RS	LRB
31	6-21-12	ISSUED AS-BUILT PER EC 247006 (PC/M 09-137) (PARTIAL).	RH	RV	–	CW	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-137.	—	—	—	—

FPL

TURKEY POINT NUCLEAR UNIT 3

P & ID

FEEDWATER SYSTEM
STEAM GENERATOR
BLOWDOWN RECOVERY

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3074

SYS 074

SHEET 4

REV 32

5613-M-3074_SH 4.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	039		K5.08
	Importance Rating	3.6		

Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity

Proposed Question: RO Question # 51

Given the following conditions:

- Unit 3 is at 8% power, MOL.
- 3-GOP-301, Hot Standby to Power Operation, startup is in progress.
- The main generator is rolling unloaded at 1800 rpm.
- Steam Dump to Atmosphere (SDTA) valves are controlled as follows:
- 3A is throttled in manual.
- 3B is closed in manual.
- 3C is throttled in AUTO.

Subsequently:

- 3C SDTA setpoint is lowered by 20 psig.

Which one of the following completes the statements below?

With no additional operator action, reactor power will __(1)__ .

Control rods __(2)__ automatically compensate for the SDTA adjustment.

A.	(1) lower (2) will
B.	(1) rise (2) will
C.	(1) lower (2) will NOT
D.	(1) rise (2) will NOT

Proposed Answer:	D
A.	Incorrect. Power trend is incorrect and rod movement is incorrect but plausible because the candidate may misunderstand the effect of lowering the steam dump setpoint in pressure mode, and may not recall that rod control is in manual below 15% power.
B.	Incorrect. Plausible same as option A and second part is correct.
C.	Incorrect. Plausible same as Option A and first part is also correct.
D.	Correct. With steam dumps open in MS pressure control auto, lowering the setpoint will cause steam dumps to open to reduce steam header pressure to setpoint pressure This will cause a higher steam flow and lower temperature. which causes higher Rx power. Control rods are not placed in auto until >15% Turbine power, which is not online yet, so rods will be in manual and no operator action stated in stern.

Technical Reference(s)	3-GOP-301		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			NO
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		6
	55.43		
Design, components, and function of reactivity control mechanisms and instrumentation.			
Comments:			

INIT

5.92 Prior to exceeding 90% power, perform the following:

5.92.1 Perform one of the following:

____ 1. Verify all Control and Shutdown Rods are aligned within 12 steps from the Group Step Demand position,

OR

____ 2. Verify less than 1 hour since last rod motion and that continuing to maneuver the plant will require further rod motion,

OR

____ 3. Hold reactor power less than 90 percent until all Control and Shutdown Rods are aligned within 12 steps from the Group Step Demand position.

CAUTION

When Venturi Power is selected for calorimetric reactor power indication, reactor power must be limited to 97% Venturi Percent Power indicated value. This can be maintained indefinitely.

NOTE

With Annunciator D-5/6, LEFM TROUBLE CLEAR, LEFM % power is accurate and can be used for 100% power determination. Venturi Corrected Power can NOT be used until after 60 minutes once D-5/6 is clear to allow time to restore the Correction Factor . Refer to 3-ONOP-074.1, Leading Edge Flow Meter (LEFM) Trouble, if alarm D-5/6 is NOT clear.

5.93 Perform the following per direction of the Shift Manager:

5.93.1 Maintain power less than or equal to 97% if using Venturi Power Indication.

5.93.2 Prior to exceeding 97% power, ensure LEFM power indication selected as follows:

1. Verify Annunciator D-5/6, LEFM TROUBLE, is Clear.

2. Select LEFM in DCS by selecting the following:

____ a. Power Menu button.

____ b. Calorimetric Input Overlay button.

____ c. LEFM button.

____ d. Verify LEFM shows as selected.

____ e. Close overlay.

INIT 5.93.2 (Cont'd)

3. Ensure Correction Factor Reset and Good quality.

____ a. From Power Menu, select Calorimetric Correction Factor.

____ b. Verify Correction Factor 60 Minute Average for SG A, B, and C are Good quality (green).

____ c. **IF** Correction Factor 60 Minute Averages for SG A, B, and C are **NOT** Good quality, **THEN** select Reset Correction Factor.

____ d. **WHEN** 60 minutes has elapsed after resetting correction faction, **THEN** verify Correction Factor 60 Minute Averages for SG A, B, and C are Good quality (green).

____ 5.93.3 **WHEN** permission is obtained from the Shift Manager **THEN** continue reactor power increase to 99.99% LEFM power.

____ 5.94 **IF** the reactor has **NOT** operated at full power since the last outage **AND** plant changes / modifications were made during the outage that could affect previously understood indication of reactor power (from Engineering input in accordance with 0-ADM-542, Plant Start-up Equipment Monitoring Plan), **THEN** prior to exceeding 98% power, ensure the section titled Start-up Monitoring at 98 % Power within 0-ADM-542 is completed satisfactorily. [Commitment - Step 2.3.10 - CAPR]

NOTES

- *Enclosure 3 provides instructions for reactivity manipulation using control rods or turbine control valves when at or near full power.*
- *Enclosure 4 provides instructions for maintaining reactor power below 100 percent to prevent exceeding the Tech Spec power limit.*

____ 5.95 **WHEN** steady state power conditions have been established **AND** Tavg - Tref deviation is within 1°F, **THEN** the Rod Control Selector switch should be placed in Auto. (N/A if rods are to be left in Manual)

5.96 **IF** the reactor has **NOT** operated at full power since the last refueling outage, **THEN** perform the following:

____ 5.96.1 Initiate two thermal calorimetrics.

____ 5.96.2 Adjust the Power Range NIS to be within plus or minus 1 percent of the calorimetric power using 3-OSP-059.5, Power Range Nuclear Instrumentation Shift Checks and Daily Calibrations.

____ 5.96.3 Record NIS Intermediate Range currents in the Remarks Section of 3-OSP-059.5, Power Range Nuclear Instrumentation Shift Checks and Daily Calibrations.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	003		A2.02
	Importance Rating	3.7		

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: Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP

Proposed Question: RO Question # 52

Given the following conditions:

- Unit 3 is at 75% power.
- Annunciator F 1/1, RCP MOTOR/SHAFT HI VIB, is received.
- R-3-369 RCP Vibration Recorder indicates the following for 3B RCP:
- Shaft vibration is 20 mils and rising slowly.
- Motor frame vibration is 4 mils and stable.

Which one of the following completes the statements below?

The crew will ___(1)___ in accordance with ___(2)___ .

A.	(1) trip the reactor then trip the 3B RCP (2) 3-ARP-097.CR.F, Control Room Annunciator Response Panel F
B.	(1) trip the reactor then trip the 3B RCP (2) 3-ONOP-041.1, Reactor Coolant Pump Off-Normal
C.	(1) reduce reactor power and monitor the RCP (2) 3-ARP-097.CR.F, Control Room Annunciator Response Panel F
D.	(1) reduce reactor power and monitor the RCP (2) 3-ONOP-041.1, Reactor Coolant Pump Off-Normal

Proposed Answer: B

A.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate believes ARP contains RCP trip criteria and since ARPs are entered 1st, it takes precedence.
B.	Correct. ONOP foldout page contains RCP trip criteria.
C.	Incorrect. Part 1 is incorrect, but plausible if candidate believes reactor power must be reduced to satisfy an RPS permissive (e.g. <P8) prior to tripping an RCP to prevent a unit trip. Also plausible because this action is correct for other RCP malfunctions such as loss of seal injection and high CBO flow. Part 2 is incorrect.
D.	Incorrect. Part 1 is incorrect, but plausible per discussion above. Part 2 is correct.

Technical Reference(s)	3-ONOP-041.1 Foldout Page		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902205 obj 3		(As available)
Question Source:	Bank	69022050302	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			
Modified to change conditions and change correect answer from PTN Bank item 69022050302.			

FOLDOUT PAGE FOR PROCEDURE 3-ONOP-041.1

1. **RCP STOPPING CRITERIA**

IF any of the following RCP limits are reached, **THEN** manually Trip the Reactor, and verify Reactor Trip using the EOP network, and then stop the affected RCP, and close PCV-3-455A, PZR Spray Valve Loop C, or PCV-3-455B, PZR Spray Valve Loop B, if applicable.

* RCP pump bearing temperature on DCS - GREATER THAN OR EQUAL TO **225°F.**
* RCP motor bearing temperature on DCS - GREATER THAN OR EQUAL TO **195°F.**
* RCP stator winding temperature on DCS - GREATER THAN OR EQUAL TO **248°F.** Note exception in Foldout Page Item 4.
* Motor frame vibration, R-3-369 (Points 1, 2, 5, 6, 9, 10) - GREATER THAN OR EQUAL TO **5 MILS**. Note exception in Foldout Page Item 4.
* RCP shaft vibration, R-3-369 (Points 3, 4, 7, 8, 11, 12) - GREATER THAN OR EQUAL TO **20 MILS**. Note exception in Foldout Page Item 4.

2. **RCP SEAL CRITERIA FOR STOPPING RCP**

IF any of the following RCP limits are reached, **THEN** manually Trip the Reactor, and verify the Reactor Tripped using the EOP network, and stop the affected RCP, Close the applicable RCP CBO Isolation Valve 303A, 303B, or 303C, and Close PCV-3-455A, PZR Spray Valve Loop C, or PCV-3-455B, PZR Spray Valve Loop B, if applicable.

* RCP CBO temperatures on DCS - GREATER THAN OR EQUAL TO **260°F**.
* RCP CBO flow exceeds **4.1 gpm**
* Any Seal Stage differential pressure exceeds **2000 psid AND** respective CBO Isolation valve (CV-3-303A, 303B or 303C) is Open

3. **FAST LOAD REDUCTION CRITERIA**

IF any of the following RCP limits are reached, **THEN** perform 3-GOP-100, Fast Load Reduction.

* RCP CBO Flow - GREATER THAN **3.7 gpm AND increasing**
* DP across any Seal Stage - GREATER THAN **1700 psid AND** respective CBO Isolation valve (CV-3-303A, 303B or 303C) is Open
* ALL of the following indications exist on the same RCP indicating a failed #3 Seal
  - RCP CBO Flow - LESS THAN **0.5 gpm**
  - RCP CBO isolation vale - OPEN
  - P3 pressure - LESS THAN **100 psig**
  - P2 pressure - GREATER THAN **1000 psig**

4. **EXCEEDING VIBRATION OR STATOR TEMPERATURE LIMITS**

* For the basis of obtaining data for startup, for balancing an RCP, or for shutdown operations; the Electrical Maintenance Supervisor or Component Engineering Supervisor may authorize continued RCP operations with vibration level or stator winding temperature above stopping criteria noted in Foldout Page Item 2. This authorization is required to be obtained prior to starting the RCP.
* When in EOP network, RCP stator winding temperature on DCS - GREATER THAN OR EQUAL TO 300°F.

5. **RCP VIBRATION ASSESSMENT CRITERIA**

IF motor frame vibration, R-3-369 (Points 1, 2, 5, 6, 9, 10), is greater than or equal to 3 mils, but less than 5 mils, **THEN** contact Engineering to evaluate the condition.

W2010:/njw/rr/cls

Item: 1.1.25.5.3.2

Question 52 original

69022050302;

Given the following conditions:

- Unit 3 is at 100% power
- Annunciator F-1/1 (RCP MOTOR/SHAFT HI VIB) actuates
- Recorder R-3-369 (RCP shaft vibration) indicates 15 mils on the 3B RCP and increasing slowly
- Recorder R-3-369 (motor frame vibration) indicates 2 mils on the 3B RCP and stable

Which ONE of the following describes the correct operator response?

A) Trip the B RCP and then verify the reactor is tripped in accordance with the EOP network.

B) Trip the reactor, verify the reactor is tripped in accordance with the EOP network, and then trip the B RCP.

C) Reduce reactor power in accordance with ONOP-100, "Fast Load Reduction," to below P-10. Then trip the B RCP.

D) Cross check the B RCP parameters. If other RCP parameters are within limits, continue B RCP operation.

CORRECT or INCORRECT feedback for item: 1.1.25.5.3.2

RCO Group 19 Audit Exam
3-ARP-097.CR, F1/1
ONOP-041.1 FO Page Items 1 & 4

Item Classification: Knowledge
Item difficulty: 0.50
Keywords: 015 AA1.23
Item Nonselectable
Item weight: 10
Points required for mastery: 1
Correct alternative(s): D
Judging values of alternatives:
A=-1 B=-1 C=-1 D=1

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	076		A2.02
	Importance Rating	2.7		

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: Service water header pressure

Proposed Question: RO Question # 53

Given the following conditions:

- Unit 3 is at 100% power.
- The 3A and 3B ICW pumps are running.
- The ICW header piping ruptures.

Which one of the following completes the statements below?

An ICW header __(1)__ alarm will come in.

3-ONOP-019, Intake Cooling Water Malfunction, will direct __(2)__ start of the 3C ICW pump.

A.	(1) low pressure (2) a manual
B.	(1) low pressure (2) verifying an automatic
C.	(1) high flow (2) a manual
D.	(1) high flow (2) verifying an automatic

Proposed Answer: A

A.	Correct. ICW low header pressure alarms at 10 psig for a header rupture. 3C ICW pump will be manually started IAW ARP.			
B.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible because automatic pump starts occur for other systems (e.g. CCW).			
C.	Incorrect. Part 1 is incorrect, but plausible because this is an indication of a leak and there are high flow alarms for other systems. Part 2 is correct.			
D.	Incorrect. Both parts incorrect, but plausible per discussion above.			
Technical Reference(s)	3-ARP-097-CR.I.4/4 3-ONOP-019		(Attach if not previously provided)	
Proposed Reference to be provided to applicants during examination:			N	
Learning Objective:			(As available)	
Question Source:	Bank			
	Modified Bank		(Note changes or attach parent)	
	New	X		
Question History:	Last NRC Exam:			
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	10		
	55.43			
Administrative, normal, abnormal, and emergency operating procedures for the facility.				
Comments:				

CAUSES:
1. Leak in ICW System
2. Trip of a running ICW pump

I4/4

ICW HEADER A/B LO PRESS

DEVICE:
- PS-3-1619 (A HDR)
- PS-3-1620 (B HDR)

SETPOINT: 10 psig

LOCATION: N/A

ALARM CONFIRMATION

1. **CHECK** ICW header pressure indicators, PI-3-1619 or 3-1620 less than or equal to 10 psig on VPA.
2. IF operating a single ICW Pump, THEN **CHECK** total ICW flow is less than 18,500 gpm.

OPERATOR ACTIONS

1. **START** standby ICW pump using 3-NOP-019, Intake Cooling Water System.
2. Locally **CHECK** ICW piping and heat exchangers for leaks.
3. **REFER TO** 3-ONOP-019, Intake Cooling Water Malfunction.
4. IF operating a single ICW Pump AND total ICW flow is greater than 18,500 gpm, THEN immediately **REDUCE** total ICW flow by performing the following:
   A. **THROTTLE** TPCW Combined Outlet Valve, 3-50-401, while maintaining TPCW Hx outlet temperature less than 105°F.
   B. **THROTTLE** 3-50-406, CCW HX ICW OUTLET SPOOL PIECE BYPASS and 3-50-407, CCW HX ICW OUTLET SPOOL PIECE ISOL while maintaining minimum ICW flows through CCW Hxs as determined by 3-NOP-019, Intake Cooling Water System.
5. IF unable to reduce total ICW flow through a single ICW Pump to less than 18,500 gpm, THEN **REDUCE** unit load using 3-GOP-103, Power Operation to Hot Standby, to limit heat input into TPCW AND **THROTTLE** TPCW Hx ICW flows using TPCW COMBINED OUTLET VALVE, 3-50-401, until total ICW flow is below 18,500 gpm.
6. IF a single ICW Pump has operated at flows greater than 18,500 gpm, THEN **REFER TO** 3-NOP-019, Intake Cooling Water System.

REFERENCES:
1. FPL Dwg 5613-M-3019, Sh 1
2. FPL EWD 5610-E-27, Sh 25, Misc. Alarms
3. PTN-BFSM-98-016, Affects of Opening 3/4-50-402 While 3/4-50-401 is Fully Open
4. PC/M 02-018, ICW Header Low Alarm Setpoint Change

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

An operable intake cooling water header consists of an intact header being supplied by at least one intake cooling water pump.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Verify Adequate Intake Cooling Water Header Flow:** a. Check alarm I 4/4, ICW HEADER A/B LO PRESS - OFF b. Check Intake Cooling Water Header Pressure - GREATER THAN 10 PSIG • PI-3-1619 • PI-3-1620	Perform the following: 1. Dispatch operator to investigate for intake cooling water system leakage. 2. **IF** starting an available intake cooling water pump will **NOT** overload an EDG, **THEN** start available intake cooling water pump(s) as follows: a) **IF** offsite power is **NOT** available **AND** diesel generator load is greater than 2250 KW, **THEN** shed smaller loads until diesel generator load is less than 2250 KW. b) Start available intake cooling water pump(s). c) Restart any loads which were shed to allow intake cooling water pump start. 3. **IF** leakage is found, **THEN** perform the following: a) Isolate affected portion of intake cooling water system. b) Start intake cooling water pumps and align valves as necessary to establish at least one operable intake cooling water header. 4. **IF** leakage is **NOT** found **AND** headers are split, **THEN** tie headers together.
6	**Verify Intake Cooling Water Header Pressure - LESS THAN OR EQUAL TO 35 PSIG** • PI-3-1619 • PI-3-1620	Perform the following: a. Dispatch operator to investigate for intake cooling water system blockage. b. **IF** blockage is found, **THEN** align valves and start intake cooling water pumps as necessary to establish at least one operable intake cooling water header.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	078		2.2.44
	Importance Rating	4.2		

Equipment Control: 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.

Proposed Question: RO Question # 54

Given the following conditions:

- Unit 3 is at 100% power.
- Instrument Air Header Pressure on PI-3-1444 is 85 psig and lowering slowly.
- Containment pressure is 0.3 psig and rising slowly.
- A Field Operator closes 3-40-337, Header Supply to Containment Isolation.

Which one of the following completes the statement below?

Normal letdown ___(1)___ expected to be lost and 3C loop pressurizer spray valve is expected to fail ___(2)___ .

A.	(1) is NOT (2) closed
B.	(1) is (2) open
C.	(1) is (2) closed
D.	(1) is NOT (2) open

Proposed Answer: C

A.	Incorrect. Part 1 is incorrect, but plausible if candidate believes instrument air pressure is restored, not considering that the containment header was isolated to stop the leak. Since pressure is now 85 psig rising, letdown was never effected and all systems will return to normal. Also, plausible if candidate believes normal letdown fails open thinking since letdown isolation valve LCV-3-460 fails open then so do the orifice valves. Part 2 is correct.		
B.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate believes that spray is a necessary for RCS pressure control and, therefore, would not fail closed or be rendered unavailable on a loss of air. Also plausible if candidate believes one spray valve fails closed and so the other must fail open to give a net balanced spray flow from pressure control which can be offset by cycling heaters.		
C.	Correct. Normal letdown isolation valves are air-to-open and fail-closed on loss of instrument air. Spray valves fail in the closed position on loss of air.		
D.	Incorrect. Both parts incorrect, but plausible if candidate thinks that spray and letdown are necessary controls and, therefore, would not fail closed or be rendered unavailable on a loss of air.		

Technical Reference(s)	3-ONOP-013		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.2 Subsequent Operator Actions (continued)

NOTE

1. With Instrument Air to containment less than 75 psig, a loss or partial loss of function may occur to the following:
   - Blowdown Isolation Valves
   - Letdown Isolation Valves
   - Pressurizer Spray Valves
   - Excess Letdown Isolation Valves
   - PORVs
2. See Attachment 4, Simplified Instrument Air Header Distribution Schematic.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**CHECK** for IA leak inside containment:	
	A. Report or indication of IA leak inside containment that requires complete isolation of IA to containment.	**A.** **GO TO** Section 3.2, Step 8.
	B. **VERIFY** all available IA compressors running.	
	C. IF the Unit 3 Reactor is **NOT** tripped, THEN manually **TRIP** Reactor.	
	D. **DIRECT** Field Operators to CLOSE 3-40-337, HEADER SUPPLY TO CONTAINMENT ISOLATION, to isolate leak containment air.	
	E. **CHECK** PI-3-1444 indicates pressure is increasing.	**E.** **DIRECT** Field Operator to OPEN 3-40-337, HEADER SUPPLY TO CONTAINMENT ISOLATION AND **GO TO** Section 3.2, Step 8.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	Topic and K/A #	103		A2.04
	Importance Rating	3.5		

Ability to (a) predict the impacts of the following malfunctions or operations on the containment system-and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations Containment evacuation (including recognition of the alarm)

Proposed Question: RO Question # 55

Given the following conditions:

- Unit 3 is in Mode 6 with refueling in progress.
- A spent fuel assembly is damaged in containment.
- 3-ONOP-033.3, Accidents Involving New or Spent Fuel, is entered.
- The Containment Evacuation alarm is actuated.

Which one of the following completes the statements below:

The sound of the containment evacuation alarm can be described as a ___(1)___ .

After the plant page is made to evacuate containment, the crew must next stop the ___(2)___ .

A.	(1) beeping tone (2) Containment Purge Fans
B.	(1) beeping tone (2) Normal Containment Cooler Fans
C.	(1) wailing tone (2) Containment Purge Fans
D.	(1) wailing tone (2) Normal Containment Cooler Fans

Proposed Answer: C

A.	Incorrect. Plausible because the audible SR counts sound off inside containment. Second part correct
B.	Incorrect. Plausible for same reason as option A and second part is plausible because the candidate may see it as an action that must be performed in accordance with 0-ADM-211 (NCC fans will be stopped to minimize the spread of airborne contamination inside containment).
C.	Correct. The siren is a wailing tone and containment purge fans are stopped and isolation valves are closed.
D.	Incorrect. First part is correct and second part is plausible as decribed in Option B

Technical Reference(s)	3-ONOP-033.3		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6900283 obj 7		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

CAUSES:
1. Flux has increased to alarm point
2. Detector loss of voltage

B9/1

B/U NIS A/B TROUBLE/ HI FLUX AT SHUTDOWN

DEVICE:
- NI-3-6649A-1, SR count
- NI-3-6649A-2, WR % pwr
- NI-3-6649B-1, SR count
- NI-3-6649B-2, WR % pwr

SETPOINT:
1/2 decade above count rate at shutdown
OR
Loss of voltage to detector

LOCATION:
N/A

ALARM CONFIRMATION

1. **CHECK** count trend on recorder for N31/N32 on console.
2. **CHECK** SR indicators for increased level since shutdown.
3. **CHECK** Gamma Metrics indication reading down scale.

OPERATOR ACTIONS

1. IF in Mode 6, THEN **PLACE** any or both of the two B/U SR NI HI FLUX at SHUTDOWN BLOCK SWITCHES to the BLOCK position to eliminate nuisance B9/1 AND Containment evacuation alarms caused by spiking.
   A. WHEN spiking is no longer present, THEN **PLACE** HI FLUX at SHUTDOWN BLOCK SWITCH to NORMAL.
2. **ENSURE** containment evacuation alarm for count rate increase above alarm setpoint.
3. **ANNOUNCE** containment evacuation over Page System.
4. IF a startup OR 3-PMI-028.3, RPI Hot Calibration, CRDM Stepping Test, and Rod Drop Test, is in progress, THEN **BLOCK** the alarm.
5. IF count rate has increased due to a planned evolution such as, heatup, boron dilution, etc., THEN **ADJUST** the High Flux at Shutdown alarm setpoint using 3-OSP-059.6, High Flux at Shutdown, to maintain a 1/2 decade above indicated source range count rate.
6. IF count rate has increased unexpectedly AND rods are withdrawn, THEN **TRIP** the reactor.
7. IF flux continues to increase, THEN **BORATE** using 3-ONOP-046.1, Emergency Boration.
8. **INVESTIGATE** for possible dilution/cooldown of RCS.
9. IF SR NI malfunctions, THEN **GO TO** 3-ONOP-059.6, Backup NIS (Gamma Metrics) Malfunction.

REFERENCES:
1. 5613-E-25, Sheet 101A
2. Tech Spec 3.9.2

4.0 IMMEDIATE ACTIONS

4.1 None

5.0 SUBSEQUENT ACTIONS

5.1 Inform the Control Room of the accident.

5.1.1 Evacuate all personnel from the area in which the accident occurred.

5.1.2 **IF** the accident involves spent fuel inside Containment, **THEN** perform the following:

1. Announce over the plant PA System:

   Attention all personnel in Unit 3 Containment, evacuate Unit 3 Containment.

2. Sound the Containment evacuation alarm.

3. Announce over the plant PA System:

   Attention all personnel in Unit 3 Containment, evacuate Unit 3 Containment.

4. Stop the Contmt Purge Air Supply Fan 3V-9.

5. Stop the Contmt Purge Exhaust Fan 3V-20.

Stop containment purge

6. Close the Contmt Purge Supply Isol. Valves POV-3-2600 and 2601.

7. Close the Contmt Purge Exhaust Isol. Valves POV-3-2602 and 2603.

8. Close the Contmt Inst Air Bleed Valves CV-3-2826 and CV-3-2819.

5.2 Accident Involving Spent Fuel

5.2.1 Accident Occurring in the Containment

1. Within 30 minutes of event, verify or place the Control Room HVAC in the recirculation mode using Attachment 2. [Commitment Step 6.3.1]

2. **IF** Control Room Ventilation did **NOT** isolate, **OR** Control Room Emergency Ventilation system (CREVS) is **NOT** operable, **THEN**:

   a. Notify SM/US to refer to TS 3.7.5, Control Room Emergency Ventilation System.

   b. **IF** Compensatory Filter Train is installed and its operation is required, **THEN** startup CREVS Compensatory Filter Train per 0-NOP-025, Control Room Ventilation.

3. Concurrently perform 3-ONOP-067, Radioactive Effluent Release.

4. Inform the Shift Manager to refer to 0-EPIP-20101, Duties of Emergency Coordinator, **AND** take any actions that may be required.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	002		K5.14
	Importance Rating	3.7		

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

Proposed Question: RO Question # 56

Given the following initial conditions:

- Unit 3 is at 50% power.
- All RCP currents indicate 600 amps.

Subsequently:

- 3A RCP current drops and stabilizes at 150 amps.

Which one of the following completes the statement below?

The reactor will automatically trip on _______ .

A.	RCS loop loss of flow
B.	4kV bus undervoltage
C.	OPΔT
D.	OTΔT

Proposed Answer: A

A.	Correct. Single loop loss of flow will trip the reactor to maintain minimum DNBR >1.3

B.	Incorrect. Plausible, if candidate confuses power equation and believes that 3A bus undervoltage causes 3A RCP amps to lower. Especially since it is the only RCP on the 3A bus. Candidate will also believe that the bus undervoltage reactor trip is similar to under-frequency trip where only one bus is required to makeup the logic.		
C.	Incorrect. Plausible, if candidate believes that heat being generated by fuel will raise temperature of the fuel when coolant flow is lowered. PCT is associated with power density and the OPDT trip		
D.	Incorrect. Plausible because OTDT trip does protect DNBR but combines a combination of parameters and is designed to protect for slower moving transients		
Technical Reference(s)	ADM-536, TS Basis		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

ATTACHMENT 2
Technical Specification Bases
(Page 11 of 193)

2.2.1 (Continued)

Reactor Coolant Flow

The Reactor Coolant Flow-Low Trip provides core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more Reactor Coolant Pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a Turbine Inlet Pressure at approximately 10% of full power equivalent), an automatic Reactor Trip will occur if the flow in more than one loop drops below 90% of loop design flow. Above P-8 (a power level of approximately 45% of RATED THERMAL POWER) an automatic Reactor Trip will occur if the flow in any single loop drops below 90% of design loop flow. Conversely, on decreasing power between P8 and the P-7 an automatic Reactor Trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low Trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	011		K4.05
	Importance Rating	3.7		

Knowledge of PRZ LCS design feature(s) and/or interlock(s) which provide for the following: PRZ level inputs to RPS

Proposed Question: RO Question # 57

Given the following initial conditions:

- Unit 3 is at 40% power.
- LT-3-461, PRZ LEVEL PROT / CONT channel failed.
- All bistables for the failed channel are tripped.
- The plant is stabilized and all systems are in automatic.

Subsequently:

PRZ LEVEL PROT / CONTROL channel:
- LI-3-459 is 90% and rising.
- LI-3-460 is 94% and rising.
- LI-3-461 is 85% and stable.

Which one of the following identifies a condition the crew is currently responding to?

A.	Reactor trip breakers opening
B.	Charging pumps tripping
C.	PRZ heaters tripping
D.	Backup PRZ heaters energizing

Proposed Answer: A

A.	Correct. Channels 459, 460, and 461 input to PRZ level trip. Logic required is 2 out of 3.

B.	Incorrect, but plausible if candidate believes that since charging pump speed is lowered to minimum with control systems in automatic that the next logical system design would be to automatically trip the pumps. Also plausible since charging pumps automatically trip on other automatic signals (e.g. safety injection).		
C.	Incorrect. Plausible since at 5% above program, trip interlock for backup heaters is actuated. However, the setpoint does not affect control group heaters.		
D.	Incorrect. Plausible because prior to EPU they would energize >5% from program.		
Technical Reference(s)	5613-T-L1 SH 2 ARP C 3/5		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902163 obj 7		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

CAUSES:
1. Rapid load change
2. PZR heater control malfunction
3. PZR spray valve malfunction
4. PZR safety or PORV leaking/open

A9/2

PZR
CONTROL
HI/LO PRESS

DEVICE:	SETPOINT:	LOCATION:
• PC-3-445C	HI 2300 psig	N/A
• PC-3-445B	LO 2185 psig	N/A

PROMPT ACTIONS

IF PZR pressure is less than 2235 psig AND the Pressurizer Pressure Control System has malfunctioned, THEN using manual control:

- **CLOSE** any open PORV valve or the associated block valve.
- **CLOSE** any open PZR spray valve.

ALARM CONFIRMATION

1. **CHECK** PI-3-445, PRESSURIZER PRESSURE on VPA for the following:
   - Equal to or greater than 2300 psig.
   - Equal to or less than 2185 psig.
2. **CHECK** PI-3-444, PRESSURIZER PRESSURE on VPA for the following:
   - Equal to or greater than 2300 psig.
   - Equal to or less than 2185 psig.

OPERATOR ACTIONS

REFER TO 3-ONOP-041.5, PZR Press Control Malfunction.

REFERENCES: FPL Drawing 5610-T-D-16B, Pressurizer Pressure Control

CAUSES:
1. Failed instrument
2. Load rejection
3. Pressure control failure
4. Rod control malfunction

A8/1

PZR PROTECTION HI PRESS

DEVICE:	SETPOINT:	LOCATION:
• PC-455A	2385 psig	N/A
• PC-456A	2385 psig	N/A
• PC-457A	2385 psig	N/A

ALARM CONFIRMATION

1. **CHECK** the following greater than or equal to 2385 psig at VPA:
   - PI-3-455
   - PI-3-456
   - PI-3-457
2. **CHECK** the following bistables ON at VPB:
   - BS-3-455A
   - BS-3-456A
   - BS-3-467A

OPERATOR ACTIONS

1. IF either of the following conditions exist,
   - Two or more press protection indicators are greater than 2385 psig.
   - Two or more bistables are ON.

   THEN:

   A. **TRIP** the reactor and turbine.

   B. **PERFORM** 3-EOP-E-0, Reactor Trip or Safety Injection.
2. IF an instrument has failed, THEN **REFER TO** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
3. **REFER TO** 3-ONOP-041.5, Pressurizer Pressure Control Malfunction.

REFERENCES:
1. FPL Drawing 5610-T-L1, Sheet 18, PZR Caused RX Trips and SI
2. Tech Spec 3.3.2, RX Trip System Instrumentation

CAUSES:
1. Level control malfunction
2. Load Rejection
3. Failed Instrument
4. Charging/Letdown mismatch

A8/3

PZR PROTECTION HI LEVEL

DEVICE:	SETPOINT:	LOCATION:
• LY-459A1 • LY-460A1 • LY-461A1	92%	N/A

ALARM CONFIRMATION

1. **CHECK** the following greater than or equal to 92% at VPA:
   - LI-3-459A
   - LI-3-460
   - LI-3-461
2. **CHECK** the following bistables ON at VPB:
   - BS-3-459A-1
   - BS-3-460A-1
   - BS-3-461A-1

OPERATOR ACTIONS

1. IF power is greater than P-7 AND <u>either</u> of the following exist:
   - Two or more level indicators are greater than 92%,
   - Two or more bistables are ON

   THEN:

   A. **TRIP** the reactor and turbine.

   B. **ENTER** 3-EOP-E-0, Reactor Trip or Safety Injection.
2. IF power is less than P-7 or an instrument has failure, THEN:

   A. **REFER TO** 3-ONOP-041.6, Pressurizer Level Control Malfunction.

   B. **REFER TO** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.

REFERENCES:
1. FPL Drawing 5610-T-L1, Sheet 18, PZR Caused RX Trips and SI
2. Tech Spec 3.3.1, RX Trip System Instrumentation
3. Tech Spec 3.3.3.3, Accident Monitoring Instrumentation.

CAUSES:
1. CVCS system malfunction
2. Instrument failure

C3/5

PZR HI LEVEL TRIP

DEVICE:
- LY-459A1
- LY-460A1
- LY-461A1

SETPOINT:
2 out of 3 at 92%, PWR greater than P-7

LOCATION:
N/A

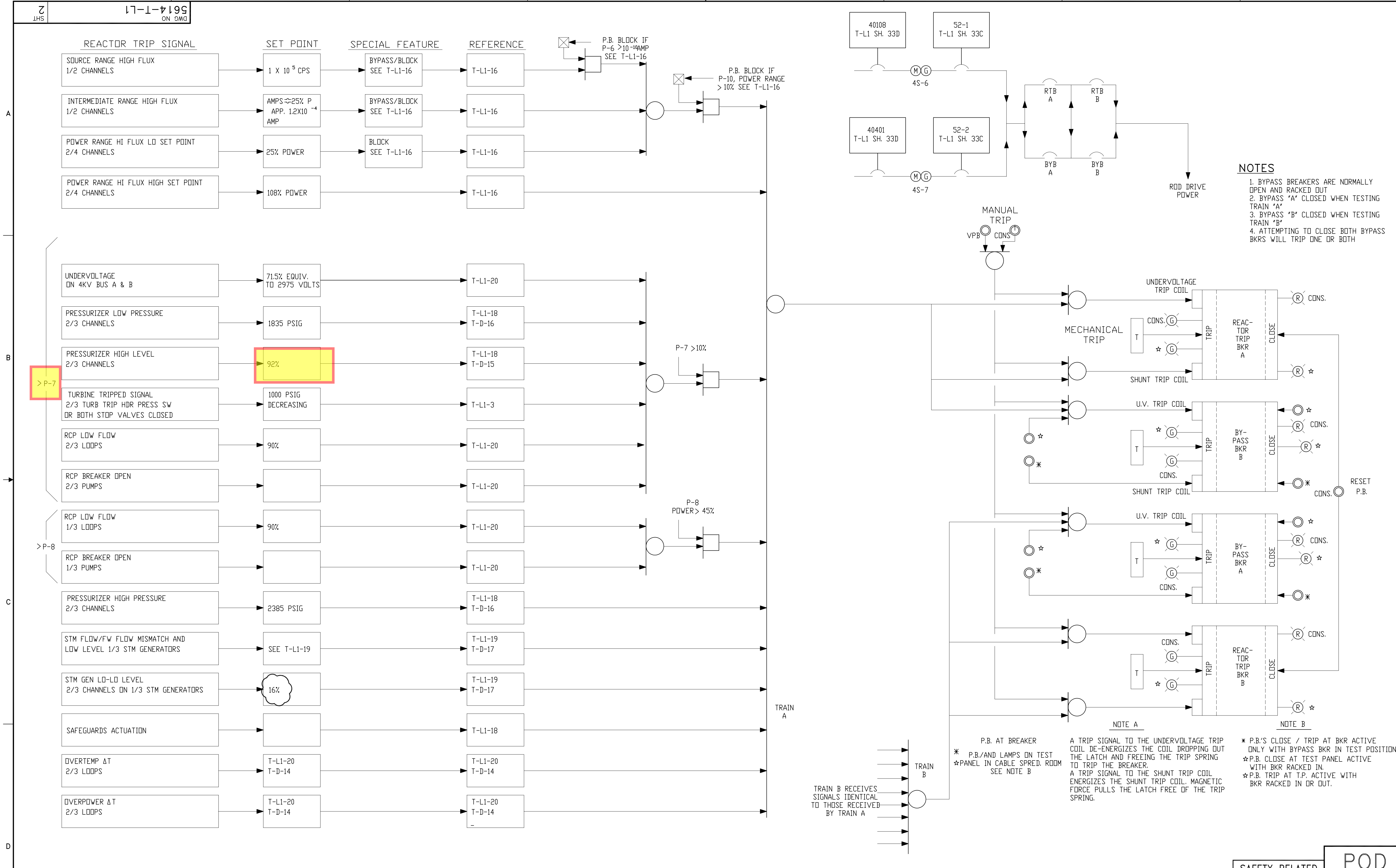
ALARM CONFIRMATION

1. **CHECK** reactor trip and bypass breakers OPEN on console and VPB.
2. **CHECK** two out of three PI-3-459A, 460, 461, PRESSURIZER LEVEL at or above 92% on VPA.
3. **CHECK** two out of three bistables LC459A1, LC460A1, LC461A1, PRZR HI LEVEL status lights LIT on VPB:

OPERATOR ACTIONS

1. **ENSURE** an automatic reactor trip has occurred.
2. **ENTER** 3-EOP-E-0, Reactor Trip or Safety Injection.

REFERENCES:
1. FPL Logic Diagram 5610-T-L1, Sheets 2 and 18
2. FPL Control System Diagram 5610-T-D-15
3. Tech Spec Sections 3/4.3.1, 3/4.3.3



SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP
1	07-16-13	ISSUED AS-BUILT PER CRN-015 (EC 247049) (ITOP 13-07-037).	RV	RH	–	PMB
0	3/19/13	THIS DRAWING SUPERSEDES DWG. 5610-T-L1 SH 2. ISSUED AS-BUILT PER EC 246859; INCORP. CRN-171 AND ENTERED INTO FPL SYSTEM. (ITOP 13-03-178).	RV	BB	–	PMB

FPL

TURKEY POINT NUCLEAR UNIT 4

LOGIC DIAGRAM

REACTOR TRIP SIGNALS

FLORIDA POWER & LIGHT

DRAWING NUMBER 5614-T-L1

SYS –

SHEET: 2

REV 1

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	014		A1.02
	Importance Rating	3.2		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: Control rod position indication on control room panels

Proposed Question: RO Question # 58

Given the following conditions:

- A load reduction is in progress in accordance with 3-GOP-100, Fast Load Reduction.
- Rod Control is in AUTO.
- Control bank C RPIs indicate 188 steps.
- Control bank D RPIs indicate 60 steps.
- Reactor power is 50%.

Which one of the following completes the statements below?

The Rod Insertion Limit Monitor compares C & D bank steps with ___(1)___ to determine if limits are exceeded.

The rod insertion limits ___(2)___ exceeded.

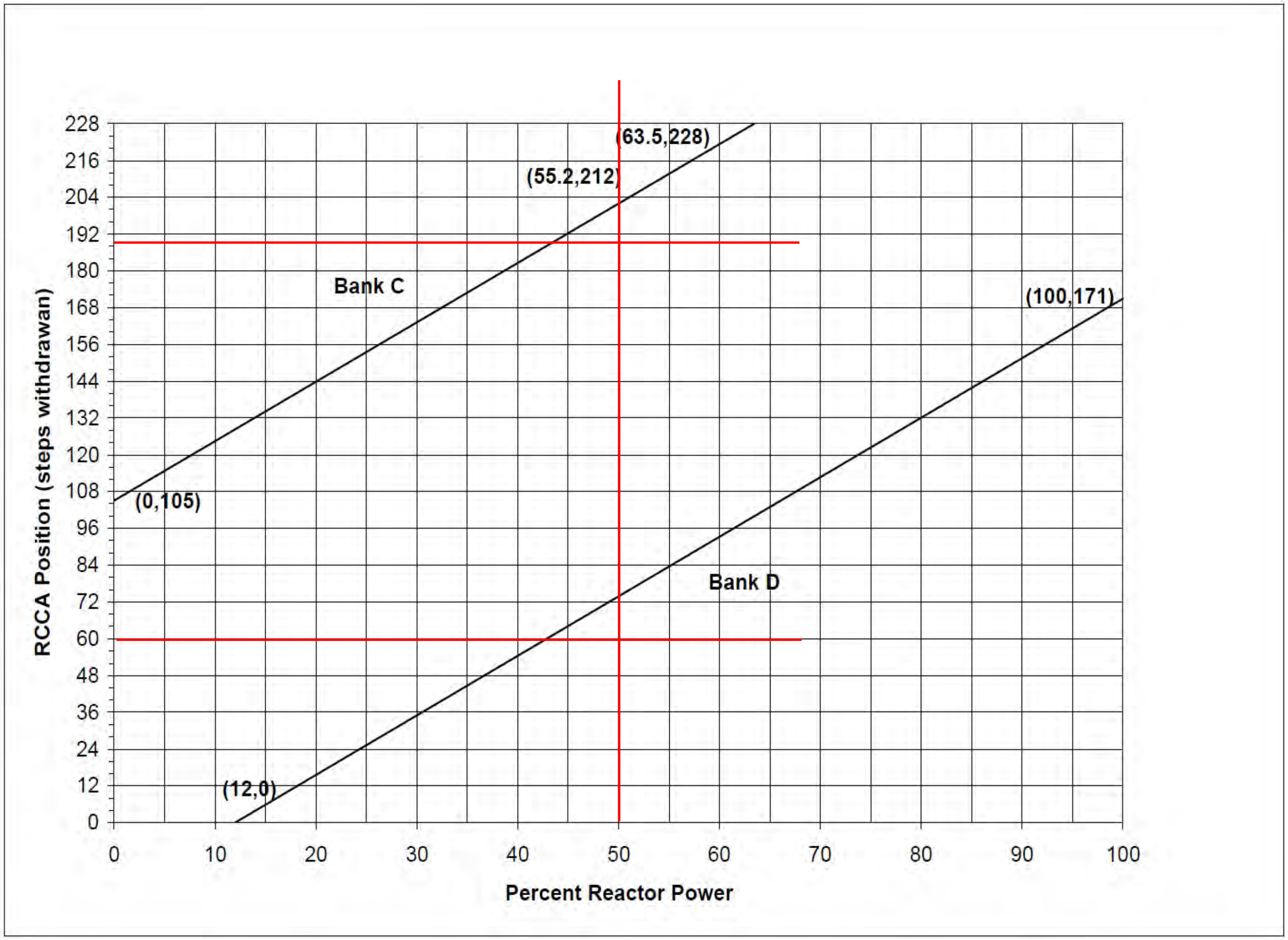
REFERENCE PROVIDED

A.	(1) median loop Tavg (2) are
B.	(1) median loop Tavg (2) are NOT
C.	(1) median loop ΔT (2) are
D.	(1) median loop ΔT (2) are NOT

Proposed Answer: C			
A.	Incorrect. Part 1 is incorrect, but plausible because median Tavg inputs into other alarms / control systems but not the RIL monitor. Part 2 is correct based on PCB.		
B.	Incorrect. Part 1 is incorrect. Part 2 is incorrect but plausible if candidate misuses PCB.		
C.	Correct. C & D bank position is compared to program median loop ΔT. The rod insertion limits are exceeded (below the curves in PCB).		
D.	Incorrect. Part 1 is correct. Part 2 is incorrect.		
Technical Reference(s)	3-ARP-097-CR.B 8/2 PCB sec VII fig A3		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y- PCB sec VII fig A3
Learning Objective:	6902106 obj 5		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	6	
	55.43		
Design, components, and function of reactivity control mechanisms and instrumentation.			
Comments:			

FIGURE A3

Turkey Point Unit 3 Cycle 28 Rod Insertion Limits vs Thermal Power
ARO = 228 Steps Withdrawn, Overlap = 100 Steps



CAUSES: Control bank A, B, C, or D inserted to or below its low low limit

B8/2

ROD BANK A/B/C/D EXTRA LO LIMIT

DEVICE:
Temperature Comparators:
- TC-409D
- TC-409E
- TC-409F
- TC-409L

SETPOINT:
Banks A and B - Fixed at 203 steps

Banks C and D - Variable, dependent on RCS loop ΔT

LOCATION:
Control Racks 22 and 28

NOTE

Expected alarm during reactor startup or shutdown when rods are below the low low insertion limit.

ALARM CONFIRMATION

1. **CHECK** Control Rod Position - Insertion Limit recorders on VPA.
2. **CHECK** RPI and step counters on console.

OPERATOR ACTIONS

1. **RESTORE** the control rods back above the low limit AND **RESTORE** shutdown margin by:
   A. **STOP** driving control rods in.
   B. **PERFORM** immediate boration equal to or greater than 16 gpm.
2. **CHECK** for load increase with **NO** rod movement.
3. **CHECK** for inadvertent dilution due to valve misalignment in CVCS System.
4. IF a control rod malfunction, THEN **REFER TO** the following as appropriate:
   - 3-ONOP-028, Reactor Control System Malfunction
   - 3-ONOP-028.1, RCC Misalignment
   - 3-ONOP-028.2, RCC Position Indication Malfunction
   - 3-ONOP-028.3, Dropped RCC
5. IF control rods are **NOT** above the rod insertion limit within one hour, THEN **PERFORM** emergency boration using 3-ONOP-046.1, Emergency Boration.

REFERENCES:
1. FPL Drawing 5610-T-D-12B, Sheet 1
2. Tech Spec Section 3.1.1.1, 3.1.3.6

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	017		K6.01
	Importance Rating	2.7		

Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors

Proposed Question: RO Question # 59

Given the following conditions:

- Unit 4 is at 100% power.
- A Core Exit Thermocouple fails.
- ANN A4/2 QSPDS INADEQUATE CORE COOLING, alarms.

Which one of the following completes the statement below?

QSPDS ___(1)___ automatically bypass the failed inputs.

ANN A4/2, can also be actuated by a failed ___(2)___.

A.	(1) will (2) Tavg module
B.	(1) will NOT (2) Tavg module
C.	(1) will (2) RVLMS sensor
D.	(1) will NOT (2) RVLMS sensor

Proposed Answer: D

A.	Incorrect. Failed inputs are not automatically bypassed by QSPDS. Plausible because other plant systems do automatically bypass failed channels. (e.g. TCS, median Tavg signal selector, Eagle-21, feedwater control…). Part 2 is incorrect but plausible if candidate believes that inadequate core cooling is caused by high RCS Tavg.
B.	Incorrect. Part 1 is correct. Part 2 is incorrect.
C.	Incorrect. Part 1 is incorrect. Part 2 is correct.
D.	Correct. Failed thermocouples are not automatically bypassed. A failed RVLMS sensor will actuate ANN A-4/2.

Technical Reference(s)	3-ARP-097.CR.A 4-NOP-042 4-NOP-042.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902103 Obj 8		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

CAUSES:
1. RCS leak/LOCA
2. Loss of RHR cooling
3. Pressure spike due to sampling of RCS loop A hot leg

A4/2

QSPDS INADEQUATE CORE COOLING

DEVICE:	SETPOINT:	LOCATION:
• Th Inputs TY-413,423,433	30°F RCS Subcooling	N/A
• Tc Inputs TY-410,420,430	30°F RCS Subcooling	N/A
• Press Inputs PY-404, 406		N/A
• RX Vessel Level	Void at any location in Head or Plenum	N/A
• Core Exit Thermocouples	700°F in MODES 1, 2, or 3 350°F in MODE 4 200°F in MODES 5 or 6 140°F in Reduced Inventory Ops 1200°F (Any 5 valid CETs above setpoint) 10°F superheated (CETs saturation margin low)	N/A

ALARM CONFIRMATION

1. **CHECK** A and B QSPDS displays for internal alarms.
2. **CHECK** QSPDS Setpoint Display Screen for correct CET setpoint of present mode.
3. **CHECK** the following for indication of a failed or spiking probe, or CET:
   - Reactor Vessel Level screen
   - Core Exist Thermocouple screen

OPERATOR ACTIONS

1. IF applicable when using the EOPs, subcooling drops to less than OR equal to 19°F, THEN **TRIP** the RCPs.
2. **REFER TO** 3-NOP-042, QSPDS - Inadequate Core Cooling Monitor.
3. **REFER TO** 3-OSP-204, Accident Monitoring Instrumentation Channel Checks, for temperature element locations and abandoned detector elements.
4. **REFER TO** 0-OSP-200.5, Miscellaneous Tests, Checks and Operating Evolutions, for a Defeated or Out of Service Annunciator.
5. IF on RHR, THEN **REFER TO** 3-ONOP-050, Loss of RHR.

REFERENCES:
Drawing 5613-M-3041, Reactor Coolant System
Drawing 5613-J-806 SH 2A1, Inadequate Core Cooling System Subcooled Margin Monitors
Drawing 5613-J-806 SH 4, Inadequate Core Cooling System Interconnection Diagram

4.0 NORMAL OPERATIONS

NOTE

The Unit 3 QSPDS System/Channel is placed in service per 3-OP-042.1, QSPDS - Inadequate Core Cooling Monitor Infrequent Operations.

4.1 Startup

None

4.2 Operation

4.2.1 Operation of QSPDS

NOTE

Unit 3 QSPDS Touch Screen Flat Panel Display (FPD) screens are listed in Attachment 1, QSPDS Alarms.

1. **SELECT** the desired Display as follows:
   - **TOUCH** the desired QSPDS FPD icons to activate that display.
   - **SELECT** major screens using the icons on the Directory screen.
   - **SELECT** sub level or related screens using the icons on the major screen.
   - **TOUCH** a point on the display to show the point ID.
   - **TOUCH** a point a second time to return the screen to normal.
2. IF any computer point becomes Bad OR Poor, THEN:
   - A. **NOTIFY** Unit Supervisor.
   - B. **EVALUATE** bypassing the point.
   - C. **EVALUATE** effect on QSPDS functionality and OPERABILITY.
   - D. **NOTIFY** the QSPDS System Engineer.

4.2.1 Operation of QSPDS (continued)

2. (continued)

E. **BYPASS** desired points per 3-OP-042.1, QSPDS - Inadequate Core Cooling Monitor Infrequent Operations.

4.3 Shutdown

None

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	029		A4.01
	Importance Rating	2.5		

Ability to manually operate and/or monitor in the control room: Containment purge flow rate

Proposed Question: RO Question # 60

Given the following conditions:

- Unit 3 is in Mode 5.
- A Containment purge is initiated in accordance with 3-NOP-053, Containment Purge System.

Which one of the following completes the statements below?

Containment Purge flow is monitored on ___(1)___ .

A high alarm on noble gas monitor 3-R-12 ___(2)___ cause the purge exhaust and supply fans to trip.

A.	(1) DCS (2) will
B.	(1) VPB (2) will
C.	(1) DCS (2) will NOT
D.	(1) VPB (2) will NOT

Proposed Answer: A

A.	Correct. Containment purge flow is monitored on DCS. Purge flow is stopped on a R-12 high alarm.

B.	Incorrect. Part 1 is incorrect, but plausible if candidate believes containment purge flow is located on VPB on the containment parameters recorder. Multiple flow instruments exist on VPB, however containment purge flow cannot be monitored there. Also plausible if candidate believes since containment purge valves and fans are on VPB, flow is located there too. Part 2 is correct.		
C.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate believes containment purge flow does not stop (candidate confuses R-12 with another process monitor). Also plausible when candidate assumes the containment purge isolates like other containment isolation points- valves closed with pumps running (e.g. RCDT pumps, ECC fans and RCPs).		
D.	Incorrect. Both parts incorrect plausible for same independent reasons above.		
Technical Reference(s)	3-ONOP-067		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902163 obj 3		(As available)
Question Source:	Bank	10006	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		13
	55.43		
Procedures and equipment available for handling and disposal of radioactive materials and effluents.			
Comments:Made 2x2. Added how monitored.			

FOLDOUT PAGE

1. Notify plant personnel of any potentially hazardous effluent release via the plant page system with page boost as follows:

   - Give specific information about the nature of the release
   - Give the location of affected plant areas
   - Warn personnel to remain clear

2 **IF** a Reactor Trip occurs **AND** any following PRMS alarms Actuate, **THEN** within 30 minutes of the alarm, manually align Control Room ventilation in the Emergency Recirculation Mode [Commitment Step 3.3.1]:

   - R-15, Condenser Air Ejector Monitor
   - R-19, Steam Generator Blowdown Monitor
   - R-20, CVCS Letdown Line Radioactivity Monitor

3. **IF** any PRMS high alarm occurs **AND** automatic actions are required, **THEN** verify the applicable automatic actions for the occurring PRMS HIGH ALARMS:

   a. R-11/12 HIGH ALARM

      1) Containment purge supply and exhaust valves - CLOSED

         - POV-3-2600
         - POV-3-2601
         - POV-3-2602
         - POV-3-2603

      2) Containment instrument air bleed valves - CLOSED

         - CV-3-2819
         - CV-3-2826

      3) Containment purge supply and exhaust fans - OFF

      4) Control Room ventilation in Recirculation alignment per ATTACHMENT 2

   b. R-14 HIGH ALARM

      1) RCV-014, Gas Decay Tank Discharge Valve - CLOSED

   c. R-17A/B HIGH ALARM

      1) RCV-3-609, CCW Head Tank Vent Valve - CLOSED

   d. R-18 HIGH ALARM

      1) RCV-018, Liquid Waste Discharge Valve - CLOSED

Facility:	WTSI Corporate		Question 60 original	
Vendor	WEC			
Exam Date:				
Exam Type:				

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Initial conditions:

- Unit 3 is in Mode 5.
- A Containment purge has been initiated in accordance with 3-NOP-053, Containment Purge System.
- The following fans are started:
  4V20, Unit 4 Containment Purge Exhaust Fan
  3V9, Unit 3 Containment Purge Supply Fan

Subsequently:

H 1/4, PRMS HI RADIATION, alarms due to PRMS-R-3-11, Particulate Radiation Monitor.

In accordance with 3-NOP-053, what is the effect of the R-11 alarm?

A. Both fans will trip. Containment isolation is achieved when the purge isolation valves close

B. Both fans will trip. Containment isolation is achieved if one purge isolation valve fails to close.

C. Only 3V9 will trip. Containment isolation is achieved by the purge isolation valves

D. Only 3V9 will trip. 4V20 must be stopped to achieve Containment isolation

Proposed Answer: C

Explanation (Optional):

A. Incorrect; only 3V9 will trip. Plausible; expect CVI signal to isolate Containment

B. Incorrect; only 3V9 will trip. Plausible; expect CVI signal to isolate Containment

C. Correct; only 3V9 trips and Containment isolated

D. Incorrect; 4V20 stopped to protect fan. Plausible; stopping purge exhaust fan part of CVI.

Techical Reference(s): 3-NOP-053 step 2.1.4 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: (As available)

Question Source: Bank 10006
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2010 Turkey Point

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	033		A2.01
	Importance Rating	3.0		

Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System ; and (b) based those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadequate SDM

Proposed Question: RO Question # 61

Given the following conditions:

- Unit 3 is in Mode 5.
- Spent Fuel Pit (SFP) level is 56' 11".
- SFP boron concentration is 2300 ppm.

Subsequently:

- A tube in the in-service Spent Fuel Pool Heat Exchanger breaks.
- A crew is dispatched to isolate the heat exchanger.

Which one of the following correctly completes the statement below?

The crew must take action to prevent Spent Fuel Pool ___(1)___ from violating the Tech Spec LCO by performing a ___(2)___ to the SFP in accordance with 3-NOP-033, Spent Fuel Pit Cooling System.

A.	(1) level (2) direct boration
B.	(1) level (2) primary water fill
C.	(1) boron concentration (2) direct boration
D.	(1) boron concentration (2) primary water fill

Proposed Answer: C			
A.	Incorrect. Part 1 is incorrect, but plausible when candidate recalls SFP cooling system is higher pressure than CCW. A leaking SFP HX tube in this incorrect model would cause SFP water to flow from the SFP to the CCW system (SFP level lowers). Part 2 is correct.		
B.	Incorrect. Part 1 is incorrect. Part 2 is incorrect, but plausible because the PWST is the quickest method to fill the SFP. Since the unit is also in a short Tech Spec action time, PWST would be chosen, however this is wrong because a further dilution of the SFP boron concentration would occur for the given situation.		
C.	Correct. CCW system pressure is higher than SFP cooling system pressure. A leak would cause the SFP to overfill. Direct boration is required (spreading dry boric acid).		
D.	Incorrect. Part 1 is correct. Part 2 is incorrect, but independently plausible per discussion above.		
Technical Reference(s)	3-ONOP-033.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902141 obj 7		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.2	**Subsequent Operator Actions (continued)**	
23.	**ISOLATE** SFP HX 3E208A.	
	A. **STOP** any running Spent Fuel Pit Cooling Pump.	
	B. **CLOSE** 3-820, SFP HX 3E208A OUTLET ISOLATION VALVE	
	C. **CHECK** SFP level stable.	**GO TO** Section 3.2, Step 24.
	D. **GO TO** Section 3.2, Step 25	
24.	**ISOLATE** SFP HX 3E208B.	
	A. **STOP** any running Spent Fuel Pit Cooling Pump.	
	B. **CLOSE** 3-927, SFP HX 3E208B OUTLET ISOLATION VALVE.	
	C. **CHECK** SFP level stable.	**NOTIFY** Shift Manager, **GO TO** Section 3.2, Step 26
25.	WHEN SFP level is stable, THEN **INITIATE** an ECO to isolate applicable SFP Heat Exchanger.	

REFUELING OPERATIONS

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.14 The following conditions shall apply to spent fuel storage:

a. The minimum boron concentration in the Spent Fuel Pit shall be 2300 ppm.

b. The combination of initial enrichment, burnup, and cooling time of each fuel assembly stored in the Spent Fuel Pit shall be in accordance with Specification 5.5.1.

APPLICABILITY: At all times when fuel is stored in the Spent Fuel Pit.

ACTION:

a. With boron concentration in the Spent Fuel Pit less than 2300 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 2300 ppm or greater.

b. With condition b not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEI LLANCE REQUIREMENTS

4.9.14.1 The boron concentration of the Spent Fuel Pit shall be verified to be 2300 ppm or greater in accordance with the Surveillance Frequency Control Program.

4.9.14.2 A representative sample of inservice Metamic inserts shall be visually inspected in accordance with the Metamic Surveillance Program described in UFSAR Section 16.2. The surveillance program ensures that the performance requirements of Metamic are met over the surveillance interval.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	035		2.4.6
	Importance Rating	3.7		

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: RO Question # 62

Given the following conditions:

- A 3C Steam Generator tube rupture occurs.
- A loss of offsite power occurs.
- The crew enters 3-EOP-E-3, Steam Generator Tube Rupture.
- ECCS flow has been terminated.
- Offsite power has been restored to the emergency buses.
- The crew is performing step 37, Evaluate RCP Status.

Which one of the following describes the preferred operation of the RCPs?

A.	Start the 3C RCP, then 3B RCP.
B.	Start only the 3B RCP.
C.	Start only the 3C RCP.
D.	Start only the 3A RCP.

Proposed Answer: B

A.	Incorrect. Plausible because starting both of the RCPs with PRZ spray capability makes sense for maximum PRZ pressure control.

B.	Correct. Westinghouse Background Document, E-3, RCP operation is preferred during recovery from a steam generator tube rupture to provide normal pressurizer spray and to ensure homogeneous fluid temperatures and boron concentrations. In addition to minimizing pressurized thermal shock and boron dilution concerns this also aids in cooling the ruptured steam generator. The procedure step states that RCP B is the preferred pump as it is best for sprays. If RCP 3B cannot be started then the procedure directs starting RCP 3C to provide normal spray. If neither 3B nor 3C can be started, then 3A RCP will be started for forced flow.
C.	Incorrect but plausible when candidate confuses 3C RCP with 3B RCP as the preferred pump.
D.	Incorrect. 3A RCP is plausible because it follows a natural order (A,B,C) and because it is the only RCP on 3A 4kV bus while 3B 4kv bus powers 3B and 3C RCPs.

Technical Reference(s)	3-EOP-E-3 step 37		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			NO
Learning Objective:	6902339 obj 4		(As available)
Question Source:	Bank	13154	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Seabrook
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments: Made EOP rev 3 / PTN specific.			

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

CAUTION

- If RCP seal cooling from Seal Injection and Thermal Barrier CCW flow has previously been lost, the affected RCP(s) shall **NOT** be started prior to a status evaluation.
- CCW System load requirement of 3-NOP-030, COMPONENT COOLING WATER SYSTEM, shall **NOT** be exceeded.

NOTE

RCPs should be run in order of priority (3B then 3C) to provide normal PRZ spray.

37. Check RCP Status

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. 3B RCP – RUNNING	**a.** Perform one of the following: * IF **NO** RCPs are running, THEN go to Step 37.d. * IF any RCPs are running, THEN go to Step 37.w.
b. Check remaining RCP(s) status:	
1) 3A RCP – **NOT** RUNNING	**1)** Stop 3A RCP.
2) 3C RCP – **NOT** RUNNING	**2)** Stop 3C RCP.
c. Check Auxiliary Spray – **NOT** REQUIRED	
1) Normal Spray – AVAILABLE	**1)** IF Normal Letdown in service, THEN establish Auxiliary Spray using Attachment 4.
2) Go to Step 38	

Facility:	WTSI Corporate		Question 62 original
Vendor	WEC		
Exam Date:			
Exam Type:			

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following plant conditions:

- A tube rupture has occurred in Steam Generator 1C2.
- Subsequently a loss of offsite power occurred.
- The crew has entered E-3, 3Steam Generator Tube Rupture.4
- ECCS flow has been terminated.
- Offsite power has been restored.
- The crew is performing step 38, 3Evaluate RCP Status.4

Which of the following describes the <u>preferred</u> course of action for operation of the Reactor Coolant Pumps, and why?

A. All available RCP2s should be started to ensure uniform boron concentration.

B. No RCP2s should be started. Starting any RCP will increase the rate of steam generator tube leakage.

C. NO RCP2s should be started. Starting any RCP may cause ruptured steam generator safety valve actuation.

D. ONLY RCP 1C2 should be started to provide pressurizer spray and minimize Pressurized Thermal Shock during cooldown.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. It is true that one of the reasons for RCP restart is to ensure uniform boron concentration however the preferred method is to start RCP 1C2 only.

B. Incorrect but plausible. It is true that starting an RCP while on natural circulation will increase the transfer of thermal energy into the steam generators. It is plausible that this could result in leakage through the ruptured steam generator tubes. The procedure includes a note describing this concern but does not prohibit restarting an RCP.

C. Incorrect but plausible. It is true that starting an RCP may cause a steam generator safety valve actuation. This would most likely occur with the specific steam generator associated with the RCP restarted. The procedure includes a note describing this concern but does not prohibit restarting an RCP.

D. Correct. Per Westinghouse Background Document, E-3, 3RCP operation is preferred during recovery from a steam generator tube rupture to provide normal pressurizer spray and to ensure homogeneous fluid temperatures and boron concentrations. In addition to minimizing pressurized thermal shock and boron dilution concerns this also aids in cooling the ruptured steam generator4. The procedure step states that RCP 1C is the preferred pump as it is 3best for sprays4. If RCP 1C2 cannot be started then the procedure directs starting all available RCP2s to provide normal spray.

Techical Reference(s):	E-3, “Steam Generator Tube Rupture”.		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		NO	
Learning Objective:	L1205I02, L1205I03		(As available)
Question Source:	Bank	13154	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Seabrook
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	055		K3.01
	Importance Rating	2.5		

Knowledge of the effect that a loss or malfunction of the CARS will have on the following: Main condenser

Proposed Question: RO Question # 63

Given the following conditions:

- Unit 3 is at 75% power and stable.
- Absolute condenser pressure is 2 in Hg.
- A steam leak occurs downstream of steam jet air ejector common steam supply valve, 3-30-020.

Which one of the following completes the statements below?

Absolute condenser pressure will ___(1)___ .

Turbine exhaust hood temperatures will ___(2)___ .

A.	RISE LOWER
B.	RISE RISE
C.	LOWER LOWER
D.	LOWER RISE

Proposed Answer: B

A.	Incorrect. Part 1 is correct. Part 2 is incorrect but plausible if candidate believes since vacuum rises and turbine load lowers then turbine speed lowers and turbine blading cools.
B.	Correct per the references. Condenser vacuum will decrease. Main generator megawatts will decrease, and turbine exhaust hood temp will rise. MORE WINDAGE ON TURBINE TIPS.
C.	Incorrect. Part 1 is incorrect, but plausible if candidate confuses absolute pressure on DCS with vacuum pressure on VPA (an indirect measure), believing that vacuum is improving. At 100% power, DCS reads absolute vacuum as 2" Hg, while Vertical Panel A vacuum is 28" Hg (~ 29.92 – 2). Part 2 is incorrect, but plausible because it is true when vacuum actually improves.
D.	Incorrect. Part 1 is incorrect. Part 2 is correct. This combination is plausible when the candidate assumes since vacuum improves, turbine load rises causing more heat at turbine blades.

Technical Reference(s)	5613-M-3014, Sheet 3 3-ONOP-014, Step 2.1.2, 3-ARP-097.CR E 5/2		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	10026	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		4
	55.43		
Secondary coolant and auxiliary systems that affect the facility.			
Comments: Changed stem & changed distractors to raise plausibility of distractors.			

CAUSES:
1. High temperature condition in turbine exhaust hood
2. Generator motoring
3. Instrument failure

E5/2

TURB EXHAUST HOOD HI TEMP

DEVICE:	SETPOINT:	LOCATION:
• 26/EHT-1	• 175°F	• N/A
• 26/EHT-2	• 26/EHT-2 Alarm function has been disabled by EC-TMD285375. Verify temperature by using these DCS points: A. 3TTEMP_17:T3474_A.PNT B. 3TCS4_18:T3474_B.PNT C. 3TTEMP_17:T3475_A.PNT D. 3TCS4_18:T3475_B.PNT	• N/A

NOTE

Auto spray action should have occurred at 142°F. Turbine trip will occur at 250°F.

ALARM CONFIRMATION

1. IF condition occurs during unit startup, THEN **CHECK** vacuum gauges PI-1612 and PI-2612 on VPA.

OPERATOR ACTIONS

1. **ENSURE** Exhaust Hood Spray Valves OPEN.
2. IF alarm occurs during unit startup, THEN **DISPATCH** operator to perform the following:
   A. Locally **CHECK** temperature on TI-3-3400A and TI-3-4301, EXHAUST HOOD TEMP less than 175°F.
   B. **CHECK** proper exhaust hood spray valve operation.
   C. IF required, THEN manually **INITIATE** exhaust hood spray.
   D. IF alarm is caused by lowering vacuum, THEN **PLACE** hogging jet in service.
3. IF alarm occurs after load reduction AND generator is motoring THEN:
   A. **ENSURE** unit is on startup transformer.
   B. **OPEN** the following:
      - Mid and East GCBs
      - Aux Transformer bkrs.
   C. **PERFORM** actions of Steps 2.A thru 2.D.

REFERENCES: FPL EWD 5610-E-29, Sh 23

1.0 PURPOSE

This procedure provides instructions to be followed when main condenser vacuum is low.

2.0 ENTRY CONDITIONS

2.1 Indications

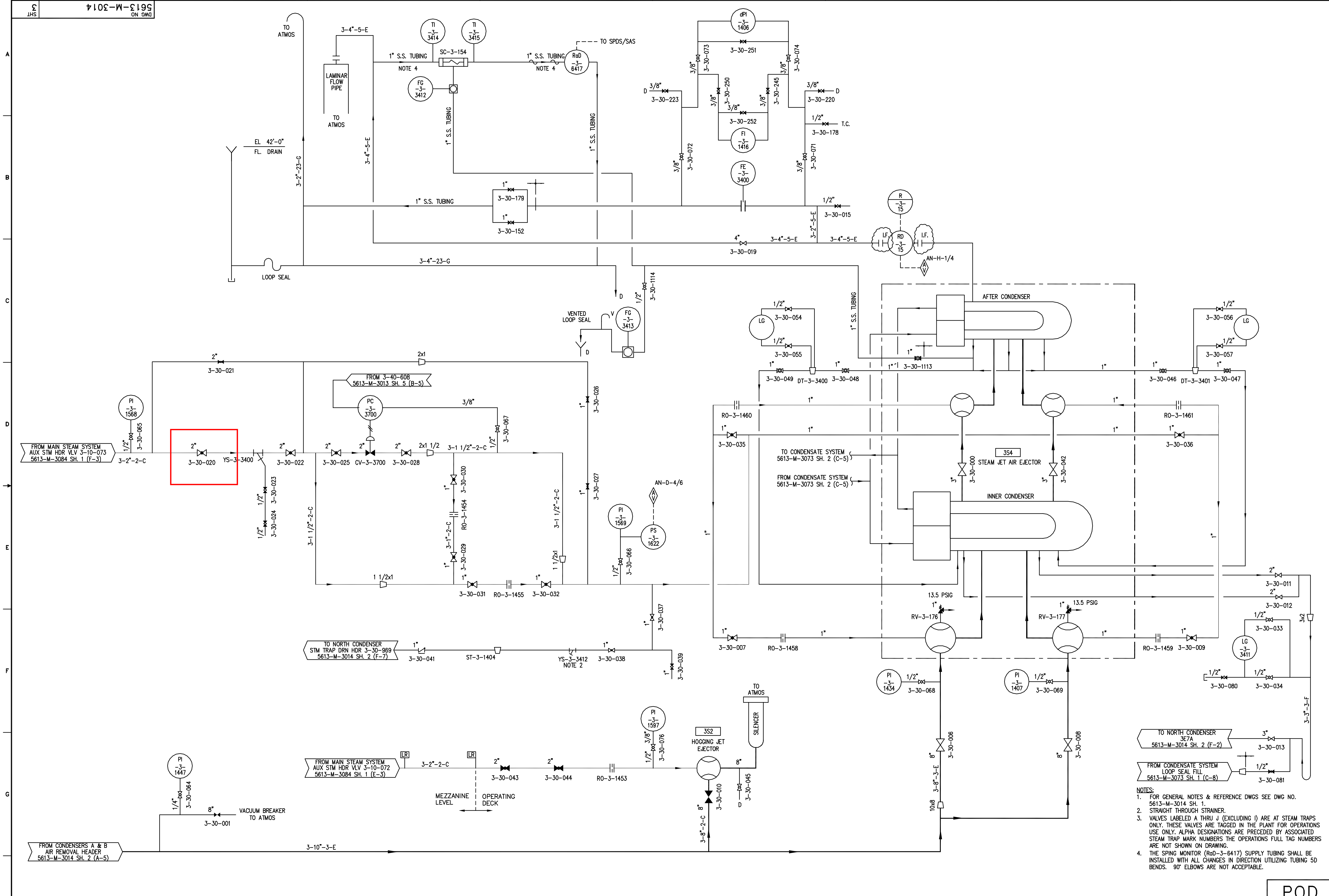
NOTE

Diverse indications of condenser vacuum should be used to validate the loss of vacuum including, DCS back pressure indication.

1. Main condenser vacuum decreasing.
2. Main generator load decreasing.
3. Main condenser air in-leakage greater than 30 scfm.

2.2 Alarms

1. E 5/3, CONDENSER LO VACUUM
2. I 3/5, CONDENSER WATER BOX LOW VACUUM
3. E 6/4, TCS TURB TRIPS



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5613-M-3014 SH. 1.
2. STRAIGHT THROUGH STRAINER.
3. VALVES LABELED A THRU J (EXCLUDING I) ARE AT STEAM TRAPS ONLY. THESE VALVES ARE TAGGED IN THE PLANT FOR OPERATIONS USE ONLY. ALPHA DESIGNATIONS ARE PRECEDED BY ASSOCIATED STEAM TRAP MARK NUMBERS THE OPERATIONS FULL TAG NUMBERS ARE NOT SHOWN ON DRAWING.
4. THE SPING MONITOR (RaD-3-6417) SUPPLY TUBING SHALL BE INSTALLED WITH ALL CHANGES IN DIRECTION UTILIZING TUBING 5D BENDS. 90° ELBOWS ARE NOT ACCEPTABLE.

POD

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4063 SH. 1 REV. 38

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
20	7/17/08	ISSUED AS-BUILT PER CRN M-12249 (PC/M 07-095).	RH	BB	BSG	PRB	14	3-16-00	ISSUED AS-BUILT PER CRN M-9929 (PC/M 99-046).	RH	RV	JK	JMM
19	10-3-05	ISSUED AS-BUILT PER PC/M 04-085 & INCORP. CRN M-11527	RH	RV	MCT	JTL	13	8-13-98	ISSUED AS-BUILT PER CRN-M-9412 (PC/M 98-027).	RH	RV	CMZ	JJC
18	11-16-04	ISSUED AS-BUILT PER PC/M 04-135.	RH	RRW	JK	JMM	12	8-6-98	ISSUED AS-BUILT PER PC/M 97-051.	RH	RV	SAG	JJC
17	8-28-02	ISSUED AS-BUILT PER CRN-M-10598 (PC/M 02-065).	RH	JPG	AAP	JTL	21	2-8-10	ISSUED AS-BUILT PER CRN-M-12800 (PC/M 07-055)	RH	RV	BSG	VMG
16	6-17-02	ISSUED AS-BUILT PER CRN-M-10465 (PC/M 00-016).	RH	RV	AGM	JTL	0	12-13-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	GFH	MD	AM/RAS	LRB
15	11-26-01	ISSUED AS-BUILT PER CRN-M-10464 (PC/M 00-016).	RH	RV	AAP	ASD	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-249.	—	—	—	—

FPL

TURKEY POINT NUCLEAR UNIT 3

P & I D

CONDENSER SYSTEM

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3014

SYS 014

SHEET 3

REV 21

5613-M-3014_SH 3

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Question 63 original

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		
Group #		
Topic & KA #		
Importance Rating:		

KA Statement

Proposed Question:

Unit 3 is at 100% power when the steam jet air ejector common steam supply valve, 3-30-020, is inadvertently closed.

Which one of the following describes the effect of closing valve 3-30-020?

A. Condenser vacuum will decrease
Turbine exhaust hood temperatures will decrease
Turbine efficiency will decrease

B. Condenser vacuum will decrease
Main generator megawatts will decrease
Turbine exhaust hood temperatures will increase

C. Tavg will increase
Turbine exhaust hood temperatures will decrease
Turbine efficiency will decrease

D. Tavg will increase
Main generator megawatts will increase
Turbine exhaust hood temperatures will increase

Proposed Answer: B

Explanation (Optional):

A. Incorrect because turbine exhaust hood temperatures will increase.
Plausible because condenser vacuum will decrease and turbine efficiency will decrease

B. Correct per the references. Condenser vacuum will decrease. Main generator megawatts will decrease. Turbine exhaust hood temperatures will increase

C. Incorrect because turbine exhaust hood temperatures will increase. Plausible because Tavg will increase and turbine efficiency will decrease

D. Incorrect because main generator megawatts will dencrease.
Plausible because Tavg will increase and Turbine exhaust hood temperatures will increase

Techical Reference(s): 5613-M-3014, Sheet 3
3-ONOP-014, Step 2.1.2,
3-ARP-097.CR E 5/2
Gen Physics, PWR Components Ch. 3 Heat Exchangers & Condensers Pages 13, 15, & 16
(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: (As available)

Question Source: Bank 10026
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2010 Turkey Point

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Level 2 because the operator must realize that closing valve 3-30-020 will stop the steam supply to the CARS which in turn will allow air to flow backwards through the SJAE into the condenser which will result in condenser vacuum decreasing and turbine exhaust hood temperature increasing. The operator must then relate the operation of the air ejector to the operation of the condenser and then predict those effects on turbine efficiency and megawatts

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	068		A3.02
	Importance Rating	3.6		

Ability to monitor automatic operation of the Liquid Radwaste System including:
Automatic isolation

Proposed Question: RO Question # 64

Given the following conditions:

- Discharge of a Waste Monitor Tank is in progress.
- H 1/4, PRMS HI RADIATION, is received.
- R-18 HI alarm is lit.
- R-18 FAIL/TEST light is NOT lit.

Which one of the following describes the action required by the crew?

A.	Manually trip the in-service Waste Monitor Tank discharge pump.
B.	Manually close RCV-018, Liquid Waste Effluent Isolation Valve.
C.	Verify that the in-service Waster Monitor Tank discharge pump automatically trips.
D.	Verify that RCV-018, Liquid Waste Effluent Isolation Valve automatically closes.

Proposed Answer: D

A.	Incorrect. Plausible because this would stop flow but incorrect because high radiation will isolate the flowpath with RCV-018
B.	Incorrect. Plausible because this isolates the flowpath but manual isolation is not required. The high radiation signal will isolate RCV-018

C.	Incorrect. Plausible because this would stop discharge flow and because if the candidate knows there is an automatic action but is unsure what that action is they could easily choose this response.		
D.	Correct. RCV-018 automatically closes on a high alarm.		
Technical Reference(s)	3-ARP-097.CR.H 1/4 3-ONOP-067		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	11	
	55.43		
Purpose and operation of radiation monitoring systems, including alarms and survey equipment.			
Comments:			

CAUSES:
1. High radiation in one of systems monitored by PRMS
2. PRMS system component failure

H1/4

PRMS
HI RADIATION

DEVICE:
- R-11
- R-12
- R-14
- R-15
- R-17A
- R-17B
- R-18
- R-19
- R-20

SETPOINT:
Variable with each PRMS channel

LOCATION:
N/A

ALARM CONFIRMATION

1. **CHECK** the following:
   - Countrate meter on each PRMS drawer in Rack QR-66
   - Alarm indicators on each drawer in Rack QR-66

OPERATOR ACTIONS

1. IF alarm is on R-11, R-12, R-14, R-17A/B, R-18, or R-20, THEN **REFER TO** 3-ONOP-067, Radioactive Effluent Release, for expected automatic actions.
2. IF alarm is on R-15 or R-19, THEN **REFER TO** 3-ONOP-071.2, Steam Generator Tube Leakage, for expected automatic actions.
3. IF alarm is on R-14, R-17A, R-17B, R-18, or R-19, THEN **CHECK** alarm valid as follows:
   A. **CHECK** FAIL/TEST light **NOT** LIT.
   B. **PUSH** FAIL/TEST light (meter reading of 288 or 289K)
   C. **PUSH** SOURCE CHECK light (should get meter increase).
   D. **PUSH** HIGH ALARM light to determine if meter level is above high alarm setpoint.
4. **ENSURE** required automatic actions.
5. IF alarm is on R-11, R-12, R-14, R-17A/B, R-18, OR R-20, THEN **REFER TO** 3-ONOP-067, Radioactive Effluent Release.
6. IF alarm is on R-15 OR R-19, THEN **REFER TO** 3-ONOP-071.2, Steam Generator Tube Leakage.
7. **REFER TO** TS 3.3.3, 3.4.6, and 3.9.13 for additional required actions.
8. IF alarm is on R-17A/B, THEN **ISOLATE** the Supplemental Cooling System by referring to 3-NOP-030.01, Unit 3 CCW Supplemental Cooling.

REFERENCES:
1. Tech Spec Sections 3.3.3, 3.4.6, and 3.9.13
2. PC/M 07-055, R-15 Steam Jet Air Ejector Monitor Replacement
3. EC 283225, Unit 3 CCW Supplemental Cooling

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C A U T I O N

If more than one high radiation event is occurring, the operator should prioritize actions to minimize offsite dose.

NOTES

- *Prioritization should include consideration of release rate, size of leak, isolable or NOT, etc.*
- *Step 3 RNO actions should be performed in the determined order of priority.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Check PRMS High Alarm - OFF**	Perform the following:
	• Check R-11 **AND** R-12 High Alarms - OFF	* **IF** R-11 **OR** R-12 High Alarm is ON, **THEN** go to Step 16.
	• Check R-17A **AND** R-17B High Alarms - OFF	* **IF** R-17A **OR** R-17B High Alarm is ON, **THEN** go to Step 29.
	• Check R-14 High Alarm - OFF	* **IF** R-14 High Alarm is ON, **THEN** go to Step 42.
	• Check R-18 High Alarm - OFF	* **IF** R-20 High Alarm is ON, **THEN** perform 3-ONOP-041.4, EXCESSIVE REACTOR COOLANT SYSTEM ACTIVITY, while continuing with this procedure.
	• Check R-20 High Alarm - OFF	* **IF** R-18 High Alarm is ON, **THEN** perform the following: a. Verify RCV-018 - Closed. b. **IF** a Liquid Release is in progress, **THEN** terminate the release. c. Inform the Shift Manager of R-18 alarm. d. Determine and correct the cause of the R-18 high alarm before commencing another liquid release.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	Topic and K/A #	079		K1.01
	Importance Rating	3.0		

Knowledge of the physical connections and/or cause-effect relationships between the SAS and the following systems: IAS

Proposed Question: RO Question # 65

Given the following conditions:

- Both units are at 100% power.
- ANN I-6/1 INSTRUMENT AIR HI TEMP/LO PRESS alarms on both units.
- Both units implement 3/4-ONOP-013, Loss of Instrument Air.
- Instrument air pressure on Unit 3 is 85 psig and slowly lowering.
- Instrument air pressure on Unit 4 is 90 psig and stable.
- NO additional Instrument Air Compressors can be started.

Which one of the following describes the required actions in accordance with 3/4-ONOP-013?

A.	Close Header Pressure Control Valve CV-4-1605 and then close CV-3-1605.
B.	Close Header Pressure Control Valve CV-3-1605 and then close CV-4-1605.
C.	Start available Service Air Compressors and open Service Air Supply to Unit 3 / Unit 4 tie valve 40-2059.
D.	Start available Service Air Compressors and open Units 1 and 2 Instrument / Service air supply valve.

Proposed Answer: C

A.	Incorrect. Plausible because both units are near the setpoint for cross-tie valves closing automatically, and the idea behind that is to try to stabilize IA pressure on the unaffected unit

B.	Incorrect. Plausible for the same reason as Option A, however candidate may believe Unit 3 isolates 1st based on the leak being on Unit 3 (lower pressure).		
C.	Correct. Start available Service Air Compressors and open Service Air Supply to Unit3/Unit4 tie valve 40-2059 will be performed IAW 3-ONOP-013.		
D.	Incorrect. Plausible because units 1 and 2 are the backup service air supply in the event of a failure of Unit 3 and 4 Air compressors.		

Technical Reference(s)	3-ONOP-013 step 2 RNO		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6900286 NO LP provided		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.2 Subsequent Operator Actions (continued)

NOTE

- 40-2059, SERVICE AIR BACKUP TO UNITS 3/4 INSTRUMENT AIR ISOLATION VLV, is located on mezzanine SW of 3A Heater Drain Tank.
- Refer to Attachment 3, 3CM and 4CM Elektronikon Control Panel, for clarification of controls and indications.

2. **START** all available Instrument Air Compressors.

   A. **REFER TO** Attachment 1, Manual Start of Instrument Air Compressors.

IF Instrument Air Compressors can **NOT** be started, THEN:

1. **ENSURE** Service Air System is in service per 0-NOP-013.01, Service Air System.
2. **ENSURE** both Service Air Compressors are operating.
3. At Shift Managers discretion:
   a. **OPEN** 40-2059, SERVICE AIR BACKUP TO UNITS 3/4 INSTRUMENT AIR ISOLATION VLV.
   b. **ENSURE** OPEN 3-40-308, INSTRUMENT AIR CROSSTIE HEADER UNIT 3 ISOLATION VALVE.
   c. **ENSURE** CLOSED 4-40-408, INSTRUMENT AIR CROSSTIE HEADER UNIT 4 ISOLATION VALVE.
4. **NOTIFY** Maintenance to restore an Instrument Air Compressor to service.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	Topic and K/A #	G1		2.1.36
	Importance Rating	3.0		

Conduct of Operations: Knowledge of procedures and limitations involved in core alterations.

Proposed Question: RO Question # 66

Which one of the following completes the statement below?

At a minimum, ___(1)___ Source Range Neutron Flux Monitor(s) with continuous visual indication in the control room and audible indication in the containment and control room, and one ___(2)___ channel with continuous visual indication in the control room shall be operable for core alterations.

A.	(1) one (2) Intermediate Range
B.	(1) two (2) Intermediate Range
C.	(1) one (2) Gamma Metrics
D.	(1) two (2) Gamma Metrics

Proposed Answer: C

A.	Incorrect. Part 1 is correct. Part 2 is incorrect but plausible given source range and intermediate ranges are tracked and plotted during a reactor startup (1/m plot).
B.	Incorrect. Part 1 incorrect, but plausible if candidate believes both source range NIs must be operable such as during a startup. Part 2 is incorrect.
C.	Correct, IAW 3-OP-038.1.
D.	Incorrect. Part 1 is incorrect. Part 2.is correct.

Technical Reference(s)	3-OP-038.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			No
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

INIT 5.1.2 (Cont'd)

____ b. **IF** not performed within the previous 7 days, **THEN** perform 3-SMI-059.10, Gamma Metrics Source Range Test and High Flux at Shutdown Adjustments. (N/A if not desired prior to entry into MODE 6)

____ (1) As applicable, record completion of the work order associated with the 7 day MODE 6 surveillance for Gamma Metrics, PMID 43489-01

____ c. Verify, as a minimum, one primary Source Range Neutron Flux Monitor with continuous visual indication in the Control Room and audible indication in the containment and Control Room, **AND** one of the remaining three Source Range Neutron Flux Monitors (one primary or one of the two backup monitors) with continuous visual indication in the Control Room are Operable.

CAUTION

Residual Heat Removal operability prior to Mode 6, Refueling, is required by Technical Specifications.

____ 3. Verify both RHR loops are operable, and at least one RHR loop is in operation with a flow rate of greater than or equal to 3000 gpm.

____ 4. Complete Attachment 1. [Commitment Step 2.3.4]

____ 5. Inform the Chemistry Department to initiate the required boron analyses using 0-NCOP-001.1, Primary Chemistry Shutdown Guidelines.

____ 6. Verify all log entries specified in Subsection 2.2 have been recorded.

Date/Time Completed:________________ / ________

PERFORMED BY (Print)	INITIALS
________________________	____________
________________________	____________
________________________	____________

REVIEWED BY:____

Shift Manager or SRO Designee

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	Topic and K/A #	G1		2.1.45
	Importance Rating	4.3		

Conduct of Operations: Ability to identify and interpret diverse indications to validate the response of another indicator.

Proposed Question: RO Question # 67

Given the following conditions:

- Unit 3 is at 100% power.
- A 3C loop Tcold fails HIGH.
- The following alarms are received:
    - B 3/6, OT∆T
    - B 4/5, RCS HI/LO TAVG
    - B 5/5, OT∆T/OP∆T ROD STOP
    - B 5/6, ∆T DEVIATION
    - J 7/4, EAGLE 21 TROUBLE
    - J 9/5, RTD CHANNEL FAILURE

Which one of the following completes the statements below?

Median Tavg on TR-3-408 will ___(1)___ .

3C loop ___(2)___ setpoint on VPA will indicate lower.

A.	(1) indicate higher (2) OT∆T
B.	(1) remain the same (2) OT∆T
C.	(1) indicate higher (2) OP∆T
D.	(1) remain the same (2) OP∆T

Proposed Answer: B			
A.	Incorrect. First part is incorrect, but plausible if candidate believes median tavg rises based on math logic. This is not the case since median tavg selector bypasses a failed Tcold input. Part 2 is correct.		
B.	Correct. Median Tavg selector deselects the failed Tcold input. The Tcold will still input into the OTΔT setpoint; with rising Tavg, setpoint will lower.		
C.	Incorrect. First part is incorrect. Plausible per A discussion. Second part is incorrect, but plausible if candidate confuses inputs to channels. Candidate may believe delta T is used to backup calorimetric power, therefore it must be an input to the Overpower setpoint.		
D.	Incorrect. First part is correct. Second part is incorrect. Plausible per C discussion.		
Technical Reference(s)	3-ARP-097.CR.B 5610-T-D-12B 5610-T-D-14		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			NO
Learning Objective:			(As available)
Question Source:	Bank	15327	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2011	Sequoyah
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments: Changed stem and distractors.			

CAUSES:

1. Protection channel OT∆T above OT∆T trip point on one or more loops
2. RTD channel failure in Eagle 21 System

B3/6

OT∆T

DEVICE:

- Loop A: TY-412C1
- Loop B: TY-422C1
- Loop C: TY-432C1

SETPOINT:
Variable

LOCATION:
Prot. Rack #1
Prot. Rack #11
Prot. Rack #14

ALARM CONFIRMATION

CHECK the following for each loop on VPA:

- Protection channel OT∆T
- Overtemp setpoint indicators

OPERATOR ACTIONS

1. **REDUCE** load to lower ∆T below setpoint.
2. IF reactor trip occurs, THEN **PERFORM** 3-EOP-E-0, Reactor Trip or Safety Injection.
3. **DETERMINE** cause of alarm by:
   A. **CHECK** ∆I meters on console.
   B. **USE** rods/boron to return ∆I to within limits.
   C. **CHECK** RPIs for rods out of alignment.
   D. **CHECK** High T_{avg} with low RCS pressure.
   E. **CHECK** core power distribution by running a flux map.
4. IF RTD channel failure in Eagle 21 System, THEN **PERFORM** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.

REFERENCES:

1. FPL Drawing 5610-T-D-14, Sheet 1
2. FPL Logic Drawing T-L-20, Sheet 1
3. Tech Spec Section 3.3.1
4. DCR TPI-92-299

CAUSES:
1. T_{AVG} above or below alarm setpoint
2. RTD channel failure in Eagle 21 System

B4/5

RCS
HI/LO TAVG

DEVICE:
Loop A:
- TY-412D1
- TY-412D2
- TY-412E

Loop B:
- TY-422D1
- TY-422D2
- TY-422E

Loop C:
- TY-432D1
- TY-432D2
- TY-432E

SETPOINT:
High - 581.5°F, nominal (PCB Section 1, Figure 1)
Low - 545°F
Low - Low - 543°F

LOCATION:
N/A

ALARM CONFIRMATION
1. **CHECK** T_{AVG}-T_{REF} recorder, TR-3-408 on console.
2. **CHECK** T_{AVG} indication on VPA.
3. **CHECK** T_{AVG} indication in DCS.
   A. **NAVIGATE** to Utilities → Misc Inputs Menu → MISCELLANEOUS ANALOG INPUTS 3 → TALPAPRO_A, TALPBPRO_A, TALPCPRO_A.
4. **CHECK** for excessive feedwater flow, fast load reduction, excessive steam flow, inadvertent boration/dilution, or rapid Xenon transient.

OPERATOR ACTIONS
1. IF High T_{AVG}, THEN:
   A. IF AUTO control rod malfunction, THEN **INSERT** controls rods manually.
   B. **CHECK** for dilution or Xenon transient.
2. IF Low T_{AVG}, THEN:
   A. **CHECK** for possible steam dump malfunction.
   B. IF a load rejection has occurred, THEN **ENSURE** STEAM DUMP TO CONDENSER CONTROL switch is OFF.
   C. IF AUTO Rod Control System failure, THEN:
      (1) **TAKE** manual control of rods.
      (2) **MATCH** T_{AVG} to T_{REF} for the existing load.
   D. IF Annunciator Panel B, Window 4/4, TAVG/TAVG-TREF DEVIATION, is in alarm, THEN **REFER TO** operator actions in that annunciator.
3. IF alarm is due to RTD channel failure in Eagle 21 System, THEN **PERFORM** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.

REFERENCES:
1. FPL Drawing 5610-T-D-14, Sheet 1
2. Tech Spec Sections 2.1.1, 3.1.1.4, 3.2.5
3. EC 246973, MS Pressure Lead/Lag/Eagle 21
4. EC 280037, Turkey Point Average Temperature Alarm Neutralization
5. FPL Drawing 5610-M-430-171, Sheet 4 & 8.
6. FPL Drawing 5610-M-401C-96, Sheet 90.

CAUSES:

1. Any OPΔT/OTΔT setpoint exceeded due to ΔI offset, high T_{avg}, low RCS pressure, rapid increase in T_{avg}, or high ΔT
2. Instrumentation failure

B5/5

OTΔT/
OPΔT
ROD STOP

DEVICE:

- TY-412B2
- TY-422B2
- TY-432B2
- TY-412C2
- TY-422C2
- TY-432C2

SETPOINT:

One out of three ΔT equal to OPΔT setpoint
OR
One out of three ΔT equal to OTΔT setpoint

LOCATION:

Protection Racks 1, 11, and 14

ALARM CONFIRMATION

1. **COMPARE** ΔT indications with OPΔT and OTΔT setpoint indications on VPA.
2. **CHECK** RTD channel failure in Eagle 21.
3. **CHECK** protection pressure channel failure low.
4. **CHECK** PR channel failures affecting I.

OPERATOR ACTIONS

1. IF two out of three coincidence for OPΔT or OTΔT, THEN **ENSURE** reactor is tripped.
2. IF a reactor trip has occurred, THEN **GO TO** 3-EOP-E-0, Reactor Trip or Safety Injection.
3. IF the alarm is valid, THEN manually **REDUCE** Reactor/Turbine power to clear the alarm.
4. IF the alarm is due to an instrument failure, THEN **PERFORM** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.

REFERENCES:

1. FPL Drawing 5610-T-D-14, Sheet 1
2. FPL Logic Diagram 5610-T-L1, Sheet 21
3. DCR-TPI-92-299
4. PC/M-03-048, OPDT/OTDT Turbine Runback Elimination

B5/6

ΔT DEVIATION

CAUSES:
1. Actual ΔT between loops greater than setpoint caused by RCS loop flow deviation, radial flux tilt, or rod misalignment
2. Instrumentation System failure

DEVICE:
Temperature comparators:
- TC-409A
- TC-409B
- TC-409C

SETPOINT:
4.5° ΔT between any two RCS loops

LOCATION:
Rack 22

ALARM CONFIRMATION

1. **COMPARE** ΔT channels on VPA.
2. **COMPARE** Power Range indications on console.
3. **COMPARE** RCS loop flow indications on VPA.
4. **CHECK** RPI indications on console for misaligned control rod.
5. **CHECK** CHANNEL DEVIATION light on Comparator and Rate drawer lit.
6. **CHECK** RTD channel failure in Eagle 21 System.

OPERATOR ACTIONS

1. IF due to slipped control rod, THEN **RESTORE** rod to bank position using 3-ONOP-028.1, RCC Misalignment.
2. IF due to a dropped control rod, THEN **RESTORE** rod to bank position using 3-ONOP-028.3, Dropped RCC.
3. **PERFORM** 3-OSP-059.10, Determination of Quadrant Power Tilt Ratio.
4. IF QPTR greater than 2%, THEN **PERFORM** 3-ONOP-059.9, Excessive Quadrant Power Tilt Ratio.
5. IF instrument failure was the cause of the deviation, THEN **PERFORM** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
6. IF other indications have **NOT** pinpointed the cause of deviation, THEN **PERFORM** a flux map.

REFERENCES:
1. FPL Drawing 5610-T-D-12B, Sheet 1
2. Tech Spec Sections 3.3.1 and 3.2
3. 5610-M-430-154, Sheet 3
4. 5610-M-430-213
5. 5610-M-401C-96, Sheet 92
6. 5610-J-844, Sheet 5E

J7/4

EAGLE 21 TROUBLE

CAUSES:
1. EPT
2. Input quality problem
3. Cabinet overtemp
4. SIR bus failure
5. TSP diagnostic failure
6. I/O board removed from slot
7. TSP DLH trouble
8. RTD trouble
9. Single 15V P/S failure
10. Loss of vital bus power
11. Channel bypass failure has occurred
12. EPT card manual partial trip (PT) bistable switch in trip position

DEVICE:	SETPOINT:	LOCATION:
Relays: • UY-001A-R1 • UY-011A-R11 • UY-014A-R14	N/A	N/A

ALARM CONFIRMATION

MONITOR Control Room parameters to determine affected channel.

OPERATOR ACTIONS

1. **GO TO** 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
2. **REQUEST** I&C determine cause of alarm with the Man Machine Interface.
3. **SUBMIT** a PWO AND **NOTIFY** I&C Supervisor of problem.
4. **REFER TO** Tech Spec 3/4.3.2 for operability concerns.

REFERENCES:
1. FPL Dwg 5613-M-430, Sh 297
2. FPL Dwg 5613-M-430, Sh 298
3. FPL Dwg 5613-M-430, Sh 299
4. PC/M 90-220
5. 5613-T-D-14A

CAUSES: T Hot or T Cold RTD output quality is bad

J9/5

RTD CHANNEL III FAILURE

DEVICE:	**SETPOINT:**	**LOCATION:**
TY-3-432	N/A	Protection Rack R1

ALARM CONFIRMATION

MONITOR Control Room parameters to determine affected RTD(s).

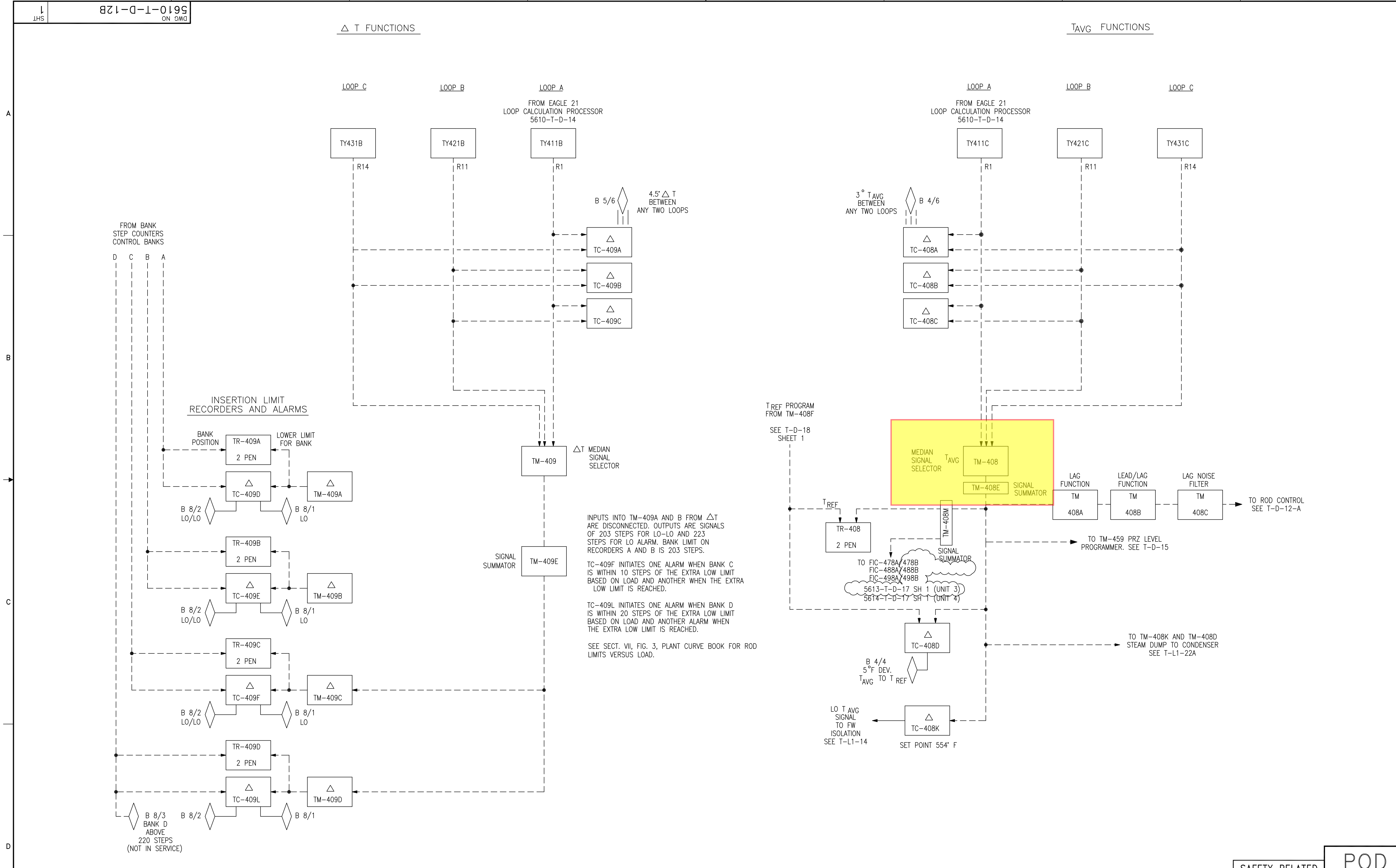
OPERATOR ACTIONS

NOTE

- If a T Hot RTD fails, the system will disregard its input if it exceeds a predetermined ∆T Hot from the T Hot Average.
- Only one T Hot RTD will be automatically removed from the system, a subsequent T Hot RTD failure will result in the T Hot Average equal to the failed RTD output.

1. **REQUEST** I&C determine cause of alarm using the Man Machine Interface.
2. **SUBMIT** a PWO AND **NOTIFY** I&C Supervisor of problem.

REFERENCES: FPL Dwg 5613-M-430, Sh 297



REV	DATE	REVISION	BY	CH	APP	APP
13	04-08-14	ISSUED AS-BUILT PER EC 246870 (246870-P002).	RV	BB	JPB	–
12	03-22-13	ISSUED AS-BUILT PER EC 246879 (ITOP 13-03-183).	RV	RH	JSM	PSB

REV	DATE	REVISION	BY	CH	APP	APP
11	3-17-10	ISSUED AS-BUILT PER CRN-E-18140 (PC/M 10-007).	RV	RH	–	LG
10	10-9-96	ISSUED AS-BUILT PER PC/M 95-170	RV	RH	JG	TS
9	5-11-95	ISSUED AS-BUILT FOR PC/M 94-120	RV	RH	BAW	JCW
8	12-30-94	ISSUED AS-BUILT FOR PC/M 94-133.	RV	RAS	–	JCW
7	6/27/93	ISSUED AS-BUILT FOR DCR TPE-93-277.	RH	RV	JJN	DBJ
6	5/31/91	ISSUED AS-BUILT FOR PC/M'S 90-220/221	RH	LH	ET	JG

SAFETY RELATED

POD

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

CONTROL SYSTEM DIAGRAM

T AVG CONTROL AND INSERTION LIMIT ALARMS

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-D-12B

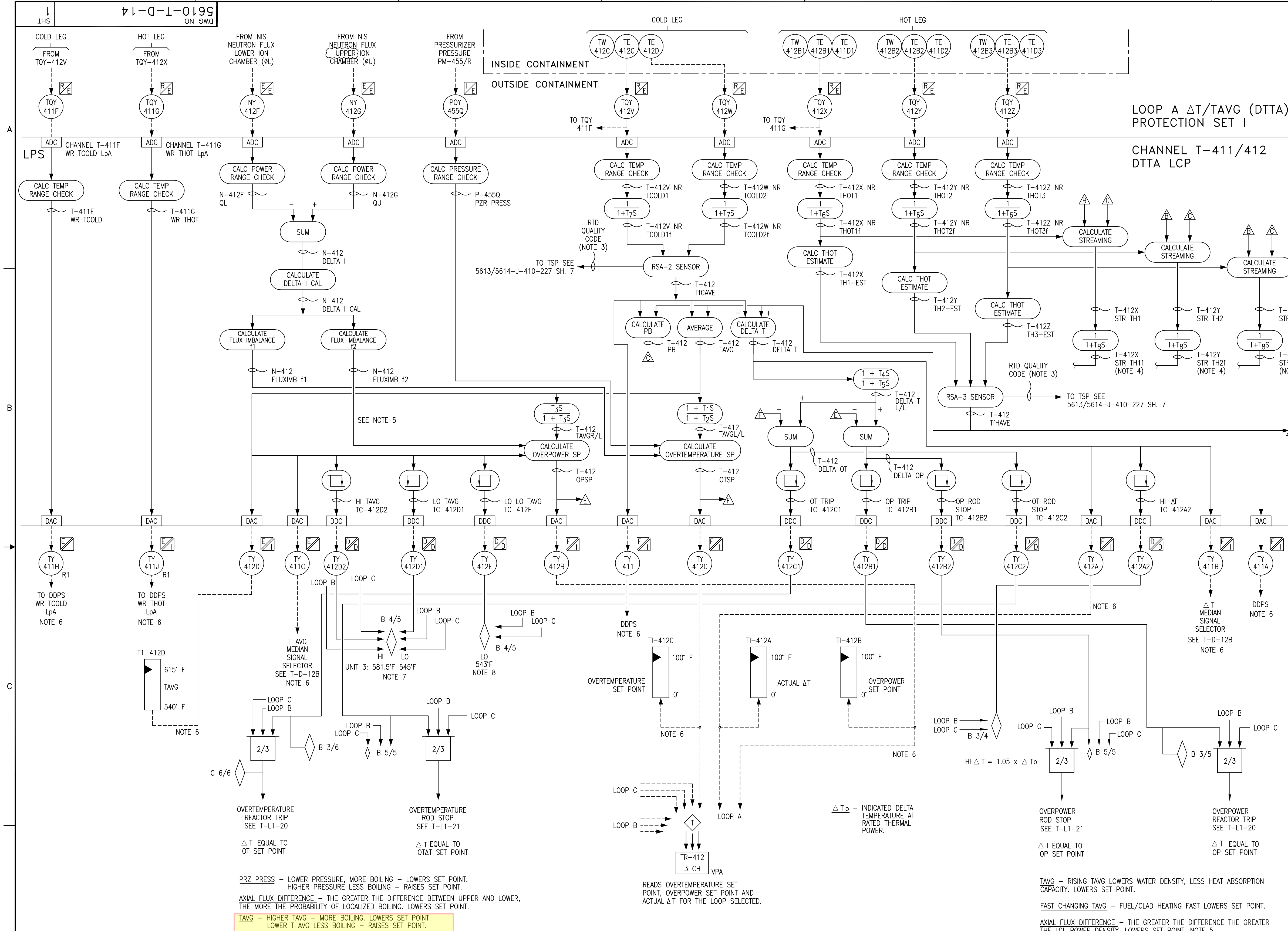
SYS –

REV 13

SHEET: 1

5610-T-D-12B_SH 1.DWG

5610-T-D-14 SHT 1



CHANNEL 1:

EAGLE-21 OUTPUT CHANNEL	DESCRIPTION	DCS POINT
TY-411H	TCOLD LOOP A WIDE RANGE	T412CD_A
TY-411J	THOT LOOP A WIDE RANGE	T411D123_A
TY-411A	THOT LOOP A AVERAGE TEMPERATURE	T411D_A
TY-411	TCOLD LOOP A AVERAGE TEMPERATURE	T412C_A
TY-412C	OVERTEMPERATURE SETPOINT LOOP A	OTSPLP_A_A
TY-412B	OVERPOWER SETPOINT LOOP A	OPSPLP_A_A
TY-412D	TAVE LOOP A PROTECTION	TALPAPRO_A
TY-411C	TAVE LOOP A CONTROL	TALPACON_A
TY-412A	DELTA T LOOP A PROTECTION	DTLPAPRO_A
TY-411B	DELTA T LOOP A CONTROL	DTLPACON_A

CHANNEL 2:

EAGLE-21 OUTPUT CHANNEL	DESCRIPTION	DCS POINT
TY-421H	TCOLD LOOP B WIDE RANGE	T422CD_A
TY-421J	THOT LOOP B WIDE RANGE	T421D123_A
TY-421A	THOT LOOP B AVERAGE TEMPERATURE	T421D_A
TY-421	TCOLD LOOP B AVERAGE TEMPERATURE	T422C_A
TY-422C	OVERTEMPERATURE SETPOINT LOOP B	OTSPLP_B_A
TY-422B	OVERPOWER SETPOINT LOOP B	OPSPLP_B_A
TY-422D	TAVE LOOP B PROTECTION	TALPBPRO_A
TY-421C	TAVE LOOP B CONTROL	TALPBCON_A
TY-422A	DELTA T LOOP B PROTECTION	DTLPBPRO_A
TY-421B	DELTA T LOOP B CONTROL	DTLPBCON_A

CHANNEL 3:

EAGLE-21 OUTPUT CHANNEL	DESCRIPTION	DCS POINT
TY-431H	TCOLD LOOP C WIDE RANGE	T432CD_A
TY-431J	THOT LOOP C WIDE RANGE	T431D123_A
TY-431A	THOT LOOP C AVERAGE TEMPERATURE	T431D_A
TY-431	TCOLD LOOP C AVERAGE TEMPERATURE	T432C_A
TY-432C	OVERTEMPERATURE SETPOINT LOOP C	OTSPLP_C_A
TY-432B	OVERPOWER SETPOINT LOOP C	OPSPLP_C_A
TY-432D	TAVE LOOP C PROTECTION	TALPCPRO_A
TY-431C	TAVE LOOP C CONTROL	TALPCCON_A
TY-432A	DELTA T LOOP C PROTECTION	DTLPCPRO_A
TY-431B	DELTA T LOOP C CONTROL	DTLPCCON_A

NUMBERS SHOWN ARE FOR LOOP A

LOOP A	LOOP B	LOOP C
412	422	432
411	421	431

NOTES:

1. FOR NOTES AND REFERENCES SEE 5610-T-D-1 SH. 2.
2. TAU VALVE SUBSCRIPT DESIGNATION TS CROSS REFERENCE IN 5613/4-J-841 SH. 21.
3. NOT ACCESSIBLE VIA MMI.
4. VALUES ARE ENTERED AS TUNING CONSTANTS IN THE RSA-3 SENSOR ALGORITHM VIA PARAMETER UPDATE.
5. CURRENTLY SET TO ZERO.
6. SEE TABLE FOR DCS POINTS.
7. ALARM VALUE IS SELETED BASED ON THE MAXIMUM T AVG REQUIRED BY LOCA CONTAINMENT ANALYSIS. RESET = TRIP-0.75°F (1% OF PROCESS RANGE). 581.5°F IS THE AVERAGE ALARM VALUE OF ALL THREE LOOPS.
8. LO TAVG TO STEAM BREAK PROTECTION SEE T-D-18B, SH. 1.

SAFETY RELATED

POD

REV	DATE	REVISION	BY	CH	APP	APP
20	11-23-04	REISSUED AS-BUILT PER PC/M 03-048	RV	BP	MCP	TS
28	03-23-15	ISSUED AS-BUILT PER EC 283491.	RH	BB	–	SB
27	07-15-14	ISSUED AS-BUILT PER EC 281974.	RH	BB	–	AFG
26	06-25-13	ISSUED AS-BUILT PER EC 246974 (ITOP 13-06-080)).	RV	RH	–	PMB

REV	DATE	REVISION	BY	CH	APP	APP
25	07-31-12	ISSUED AS-BUILT PER EC 246973 (PC/M 09-077).	RV	BB	–	PMB
24	10-12-10	ISSUED AS-BUILT PER EC 249489.	RV	RH	–	PDS
23	08-03-09	ISSUED AS-BUILT PER CRN-E-17987 (PC/M 09-070).	RV	BB	RRV	PDS
22	07-01-09	ISSUED AS-BUILT PER CRN-I-4976 (PC/M 09-005).	RV	RH	–	RS
21	6-8-05	ISSUED AS-BUILT PER PC/M 04-069.	RH	RV	MCP	PDS
12	1-20-94	REDRAWN & ISSUED AS-BUILT FOR DCR-TPI-93-515.	RV	GM	RAS	JCW

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

CONTROL SYSTEM DIAGRAM

OVERTEMPERATURE-OVERPOWER PROTECTION SYSTEM

FLORIDA POWER & LIGHT

DRAWING NUMBER 5610-T-D-14

SYS –

SHEET 1

REV 28

5610-T-D-14_SH 1.DWG

Facility: WTSI Corporate

Question 67 original

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following plant conditions:
- Unit 1 is at 100% RTP.
- The following alarms are received on 1-M-5A:
- TS-68-2M/N RC LOOPS T AVG/AUCT T AVG DEVN HIGH-LOW (A-6)
- TS-68-2A/B REACTOR COOLANT LOOPS T DEVN HIGH-LOW (B-6)
- TS-68-2P/Q REAC COOL LOOPS T REF T AUCT HIGH-LOW (C-6)
- NARROW RANGE RTD FAILURE LOOP 3 (E-6)

Which ONE of the following would identify both;

(1) the loop 3 RTD that has failed
AND
(2) an alternate indication which would confirm the cause of these alarms?

A. (1) Tcold failed HIGH
(2) Control Rods Insert

B. (1) Tcold failed HIGH
(2) OTT Setpoint rises

C. (1) Thot failed HIGH
(2) Control Rods Insert

D. (1) Thot failed HIGH
(2) OTT Setpoint rises

Proposed Answer: A

Explanation (Optional):

A. Correct. First part is correct. Any Tcold failure will cause the NR RTD FAILURE alarm to actuate. Second part is correct. Tcold failing high will cause Tave to be higher and will become the highest Tave (Auctioneered High Tave) and will be higher than Tref. Rods will insert to match Tave and Tref.

B. Incorrect. First part is correct (see item A). Second part is incorrect. Tcold failing high will causes Tave for Loop 3 to be high, which causes the OTT setpoint to lower (be closer to the actual T). This is plausible since the novice operator commonly confuses the direction of the setpoint as it relates to the direction of the failure.

C. Incorrect. First part is incorrect. Failure of both Thot RTDs is required to cause the NR RTD FAILURE alarm to actuate. This is plausible since only one Tcold failure is require for this alarm and it is a common alarm. Second part is correct. Thot failing high will cause Tave to be higher and will become the highest Tave and will be higher than Tref. Rods would insert to match Tave if both Thot RTDs failed.

D. Incorrect. First part is incorrect (see item C). Second part is incorrect. Thot failing high will cause Tave to be higher and will become the highest Tave (Auctioneered High Tave). This causes the OTT setpoint to lower (be closer to the actual T). This is plausible since the novice operator commonly confuses the direction of the setpoint as it relates to the direction of the failure.

Techical Reference(s): 1-AR-M5-A (A-6, B-6, C-6 and E-6) R37 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: OPL271AOP-I.02 Obj #8 and 9. (As available)

Question Source: Bank 15327

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 Sequoyah

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	Topic and K/A #	G1		2.1.8
	Importance Rating	3.4		

Conduct of Operations: Ability to coordinate personnel activities outside the control room.

Proposed Question: RO Question # 68

Given the following conditions:

- Unit 3 is at 100% power.
- 3-OSP-049.1, Reactor Protection System Logic Test, is being performed.

Which one of the following completes the statements below?

The operator at the controls ___(1)___ hold primary responsibilities for this surveillance.

Communications must be established between the control room and personnel at the ___(2)____ .

A.	(1) can NOT (2) MCC 3B and at the Cable Spreading Room
B.	(1) can NOT (2) Computer Room and at the Test Racks behind VPA
C.	(1) can (2) MCC 3B and at the Cable Spreading Room
D.	(1) can (2) Computer Room and at the Test Racks behind VPA

Proposed Answer: A

A.	Correct. IAW OP-AA-100-1000, the operator at the controls primary function is to monitor the unit and is prohibited from holding primary responsibility for this surveillance. Locations are correct IAW 3-OSP-049.1.
B.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible if candidate believes the surveillance is performed at the locations detailed as such is the case for other tests.
C.	Incorrect. Part 1 is incorrect, but plausible if candidate believes he can perform the test as one of the exceptions listed in the conduct of operations (e.g. surveillance effecting reactivity). Part 2 is correct.
D.	Incorrect. Both parts incorrect, but plausible per discussion above.

Technical Reference(s)	3-OSP-049.1 OP-AA-100-1000		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

INITIALS
CK'D VERIF

3.3 Inspect the NBFD Relays using the following attachments, as applicable to the trains to be tested **AND** mark the As Found state:

____ • Attachment 13 (Ref. 5613-M-430-146, Sh 3A)

____ • Attachment 14 (Ref. 5613-M-430-146, Sh 3B)

____ 3.4 **IF** DCS SOE timing is **NOT** available **OR** if Dewetron Recorder timing will be performed, **THEN** notify the Electrical Maintenance Department that a Dewetron Recorder is required to be connected to the Reactor Trip Breaker circuit as directed by Enclosure 1 for breaker opening timing. (Dewetron Recorder timing is not required if DCS opening time is used.)

____ 3.5 Check the Reactor Protection System for any abnormalities using one of the following methods **AND** correct, as necessary:

____ 3.5.1 Call up the DCS Reactor Protection SOE Group displays **AND** check for any abnormalities that may cause a reactor trip.

____ 3.5.2 Have I&C perform a visual check of the reactor trip relays in the reactor protection racks to verify none of the reactor trip relays are in the Tripped mode.

____ 3.6 Verify the Shift Manager has provided personnel adequate to man the Control Room, the Cable Spreading Room, and the 3B MCC Room when needed to perform the test, and that necessary communications between these stations can be established.

____ 3.7 Ensure Reactor Trip Breakers A and B are Closed.

3.8 **IF** performing the test in Mode 1, 2, or 3 with power level below P-7 (10%), **THEN** perform the following:

3.8.1 **IF** the turbine is capable of being Latched, **THEN** perform the following:

____ 1. Ensure the turbine is Latched.

2. Have I&C perform the following to simulate the left and right turbine stop valves being reset:

____ ____ a. At 3QR32 Rear (R), install a jumper wire across Terminals 1I30-3 and 1I30-4.

____ ____ b. At 3QR33R, install a jumper wire across Terminals 2I30-3 and 2I30-4.

____ c. At 3QR32 Front (F), verify relay 3-SL-X is Energized (tab not protruding).

____ d. At 3QR33F, verify relay 3-SR-X is Energized (tab not protruding).

ATTACHMENT 5
CONTROL ROOM CONDUCT
(Page 3 of 9)

3.2 Operator at the Controls (OATC):

1. At least one licensed control room operator shall be designated as the Operator at the Controls per unit. The OATC is directly responsible for monitoring nuclear safety within the control room. This operator shall maintain a broad perspective of activities being conducted to ensure effects on nuclear safety are known. Concerns shall be raised to the Control Room Supervisor and/or Shift Manager prior to allowing conditions to affect safe operations of the Unit. The OATC duties and responsibilities are as follows:

A. The majority of the time monitor controls and indications to ensure plant conditions are clearly understood and that important parameter changes are identified promptly.

B. Communications will be minimized to the extent that it does not detract from OATC duties.

C. Monitoring the unit operation and controls and responding as necessary to alarms and changing plant conditions including use of annunciator response procedures.

D. Remain focused on the main control panels and plant computer monitors.

E. Perform an area panel walk down of key control parameters on control panels and plant computer monitors at least every 15 minutes.

F. OATC ensure that a licensed operator performs an end to end control panel walk downs hourly.

G. Responsible for monitoring for the effects of primary reactivity manipulations on the unit (control rods, boration, dilution and turbine control system adjustments).

H. Utilize the plant process computer as necessary for monitoring, trending and alarm response.

I. May perform administrative duties and log reviews as necessary to support plant operation or monitoring (i.e. performance of leak rate calculation, checking a log trend or previously made narrative log entry).

ATTACHMENT 5
CONTROL ROOM CONDUCT
(Page 4 of 9)

3.2 Operator at the Controls (OATC): (continued)

1. (continued)

J. May NOT be engaged in activities which detract from overall focus on monitoring unit operation such as initiating CRs, accessing e-mail or other computer work that does not directly support unit monitoring or trending.

K. May NOT eat main meal while performing OATC duties however may consume a light snack or beverage.

L. With exception of the examples provided, may NOT perform primary responsibilities for testing, surveillances, call outs, etc., however may provide peer checks as required. The following examples are allowed for the OATC to perform:

(1) Instrument or Panel checks as part of shiftly or periodic surveillance requirements that specifically focus the OATC on the plant status.

(2) Surveillances that are required which result in Reactivity Manipulations.

M. The control room staff may be augmented as necessary to perform specific functions such as surveillance testing or involved plant evolutions. Additional control room personnel shall closely coordinate activities with the OATC.

N. The OATC responsibility may be rotated between licensed operators as necessary to maintain alertness, facilitate meals, back panel walk downs, restroom breaks, etc. The Control Room Supervisor shall be notified when the OATC responsibility is changed.

O. The majority of time must be spent monitoring the key plant parameters with other important parameters being monitored at least every 15 minutes. Unless involved in activities where OATC involvement is required by the Conduct of Operations (for example reactivity manipulations, peer checks or detailed panel reviews), key parameters as defined by 3.2.1.P should be monitored every 2-3 minutes.

ATTACHMENT 5
CONTROL ROOM CONDUCT
(Page 5 of 9)

3.2 Operator at the Controls (OATC): (continued)

1. (continued)

P. While operating, the operator "at the controls" should monitor the following key parameters as a high priority and at a frequency to assure a constant awareness of their value and trend:

- Rx Power
- RPV level (BWR)
- Steam generator pressure (PWR)
- RCS temperature
- Steam generator level (PWR)
- RCS pressure
- Steam flow / feed flow
- Pressurizer level (PWR)
- Main generator output (Mw)
- VCT level
- Main condenser vacuum
- Other critical system parameters as directed by unit risk.

ATTACHMENT 5
CONTROL ROOM CONDUCT
(Page 6 of 9)

3.2 Operator at the Controls (OATC): (continued)

1. (continued)

Q. While shutdown and defueled, the operator "at the controls" should monitor the following parameters as a high priority and at a frequency to assure a constant awareness of their value and trend:

- RCS inventory
- RCS temperature
- RCS pressure
- Shutdown cooling
- Nuclear Instrumentation
- Other critical system parameters as directed by unit risk.

2. During Planned Plant Maneuvers, regular monitoring of plant parameters is required to provide early warning to an unexpected plant response. In these cases, the OATC must be extra vigilant in ensuring that key parameters have established bands and that they are being monitored. The actual parameter monitoring of some parameters may be accomplished by another operator that is assigned to a particular evolution, panel(s) or parameter(s). The OATC should limit activities to supporting reactivity manipulations and other activities that directly support monitoring key parameters every 2-3 minutes.

3. During implementation of Off Normal or Emergency Operating Procedures, the restriction from performing equipment manipulations will normally be waived in order to ensure timely implementation of the strategy. However the OATC should still be designated as the operator primarily responsible for monitoring plant status in the following manner

A. The OATC will be responsible for ensuring critical parameters and control bands are monitored. The actual parameter monitoring of some parameters may be accomplished by another operator that is assigned to a particular evolution, panel(s) or parameter(s).

ATTACHMENT 5
<u>CONTROL ROOM CONDUCT</u>
(Page 7 of 9)

3.2 Operator at the Controls (OATC): (continued)

3. (continued)

B. In a transient condition the crew complement should be regularly monitoring plant conditions as a whole. As strict adherence to the panel and end to end walk downs is not expected.

C. The OATC shall remain in an area where his assigned critical parameters can be monitored. Activities that require manipulations outside of this area (i.e. back panels) should be deferred to other available licensed operators.

D. Operators that are not the OATC are the preferred choice for communications to field operators unless it directly supports activities being performed by the OATC

E. The OATC should not be directly involved in detail oriented activities such as Emergency Plan determination or implementation P&ID or loop and logic reviews, or developing reactivity plans.

4. The duties of the Operator at the Controls may be transferred to another licensed operator to support needed relief. The Operator at the Controls may be relieved following an appropriate exchange of responsibility. Turnover should include at a minimum:

A. Status of any recent reactivity changes completed or in progress.

B. Status change of control room alarms

C. Status of any evolutions or surveillances in progress

5. At least one Operator at the Controls shall be stationed within the designated area at all times when so required by existing plant conditions or Technical Specifications.

6. Training activities (such as practical factors and job performance measures required to be completed in the control room, and check-outs) may be performed as part of normal "Operator at the Controls" duties provided distractions are minimized.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	Topic and K/A #	G2		2.2.12
	Importance Rating	3.7		

Equipment Control: Knowledge of surveillance procedures.

Proposed Question: RO Question # 69

Given the following conditions:

- Both units are at 100% power.

Which one of the following completes the statements below?

In accordance with 3-OSP-075.1, Auxiliary Feedwater Train 1 Operability Verification, Train 1 of AFW ___(1)___ be declared operable with AFW feedwater control valves in manual.

When the test is complete, AFW flow controllers will be set to ___(2)___ .

A.	(1) will (2) 130 gpm
B.	(1) will (2) 135 gpm
C.	(1) will NOT (2) 130 gpm
D.	(1) will NOT (2) 135 gpm

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible if candidate assumes that as long as the flow setpoint is correct, then the required design basis flow will be provided on an automatic AFW actuation. This is true, however the controller must be placed in automatic to ensure the manual demand setpoint is not inadvertently operated and flow is not diverted from the opposite unit. Candidate may consider AFW like RHR for example where the system remains operable with controllers in manual as long as minimum flow is met. Part 2 is incorrect but plausible if candidate remembers pre-EPU setpoints.		
B.	Incorrect. Part 1 is incorrect and plausible per discussion above. Part 2 is correct.		
C.	Incorrect. Both incorrect and independently plausible per discussions above.		
D.	Correct, IAW 3-OSP-075.1.		

Technical Reference(s)	3-OSP-075.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902520 obj 3		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

7.0 **PROCEDURE**

INIT Date/Time Started: ____________________ / ________________

7.1 AFW Pump **A** Operability Test

7.1.1 Initial Conditions

____ 1. Applicable prerequisites in Section 3.0 are complete.

____ 2. The System Engineer and the Shift Manager have determined if the flow transmitters will be vented by I&C while the AFW System is running.

7.1.2 Procedure Steps

NOTES

- *Train 1 AFW is Inoperable on both Units 3 and 4 when:*
  - * *The T&T valve is tripped or is under manual control.*
  - * *An AFW flow controller is placed in manual or is not set for 135 gpm.*
- *Train 1 AFW may be declared Operable after completing Attachment 2.*

1. Inform the Shift Manager of the following requirements:

____ a. To declare Train 1 AFW inoperable on both Units 3 and 4 during performance of this procedure.

____ b. To enter a 72-hour action statement per Tech Spec 3/4.7.1.2.

NOTES

- *If oil level is low or the oil is cloudy, the Unit Supervisor should be contacted and corrective action taken prior to proceeding.*
- *With turbine running, governor oil level should be at the mid-level mark in the sight glass +/- 1/8 inch. Initiate an AR and WR for out of specification condition.*
- *When the turbine is secured, the AFW Pump is inoperable with governor oil level below the mid-level mark.*

2. Verify normal oil level and oil clarity in the following:

____ a. AFW Pump **A** lube oil sump (level within sightglass, approximately 3/4 full; approximately 1/2 full when the Auxiliary Oil Pump is running).

____ b. AFW Pump **A** governor oil sump (level above the top of the sightglass).

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	Topic and K/A #	G2		2.2.41
	Importance Rating	3.5		

Equipment Control: Ability to obtain and interpret station electrical and mechanical drawings.

Proposed Question: RO Question # 70

Given the following Main Steam System Drawing:

REFERENCE PROVIDED

Which one of the following completes the statements below?

PT-3-468, 3C SG Steam Header Pressure, provides the ___(1)___ pressure input to satisfy the steam break protection logic for the Steamline ΔP safety injection signal.

The Train 1 AFW steam supply can be traced to mechanical drawing 5613-M-3075 SH.1 location ___(2)___ .

A.	(1) high (2) C-2
B.	(1) low (2) C-2
C.	(1) high (2) G-2
D.	(1) low (2) G-2

Proposed Answer: A

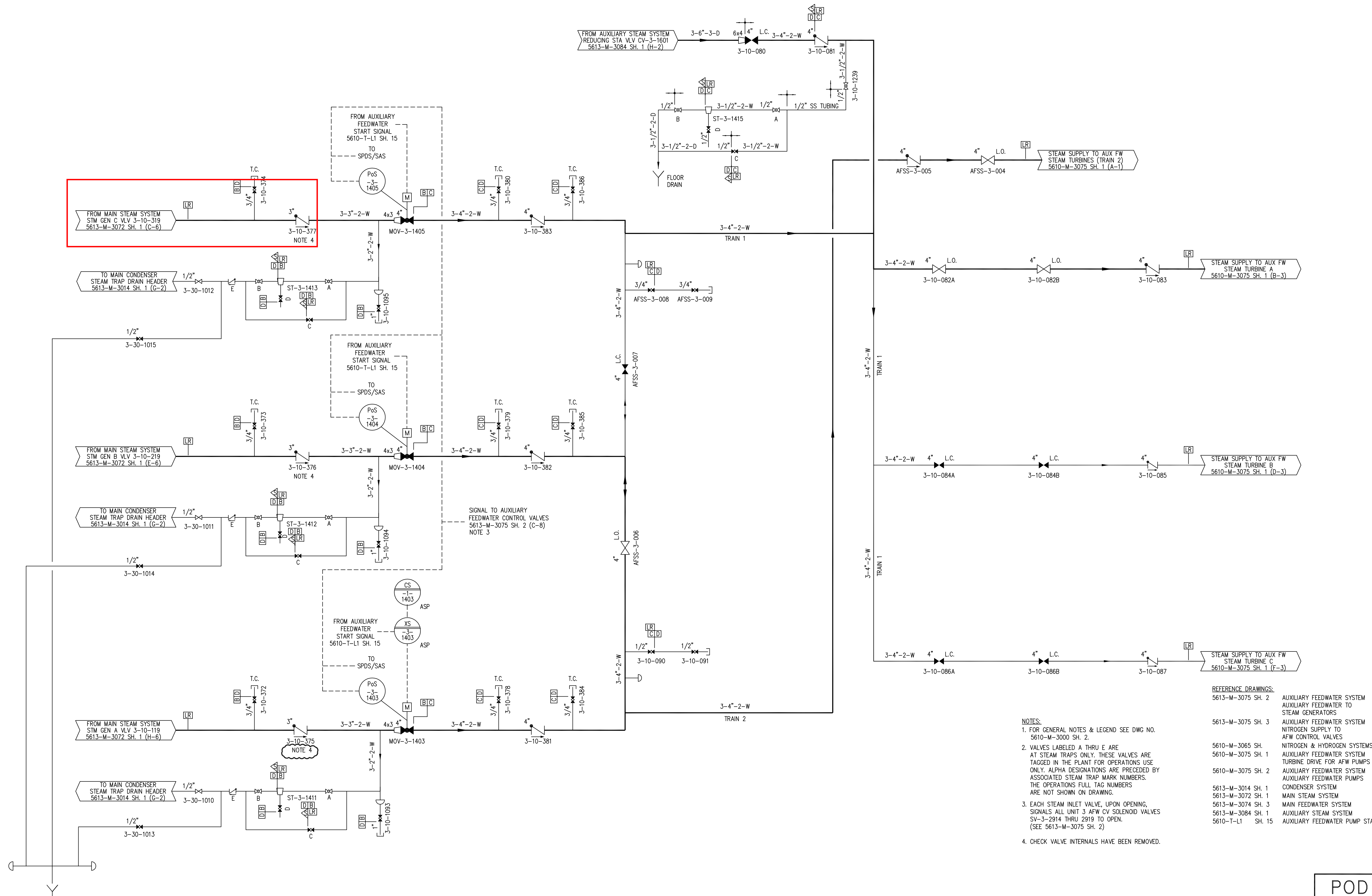
A.	Correct. PT-3-468 is the header input to the high side of the High Main Steamline DP SI signal. Location is correct per reference.

B.	Incorrect. Part 1 is incorrect, but plausible when the candidate confuses it with the low pressure input to the downstream break protection logic OR when candidate assumes the logic looks for a DP between headers. Part 2 is correct.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible when candidate confuses Train 1 steam supply with Train 2 supply. Train 2 steam is supplied by 3C S/G which traces to 5613-M-3075 SH.1 location G-2. Also plausible when candidate assumes Train 1 is supplied by A S/G.
D.	Incorrect. Both parts incorrect per discussion above.

Technical Reference(s)	5613-M-3072 SHT 1 5613-M-3075 SHT 1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y-5613-M-3072 SHT 1
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2.
2. VALVES LABELED A THRU E ARE AT STEAM TRAPS ONLY. THESE VALVES ARE TAGGED IN THE PLANT FOR OPERATIONS USE ONLY. ALPHA DESIGNATIONS ARE PRECEDED BY ASSOCIATED STEAM TRAP MARK NUMBERS. THE OPERATIONS FULL TAG NUMBERS ARE NOT SHOWN ON DRAWING.
3. EACH STEAM INLET VALVE, UPON OPENING, SIGNALS ALL UNIT 3 AFW CV SOLENOID VALVES SV-3-2914 THRU 2919 TO OPEN. (SEE 5613-M-3075 SH. 2)
4. CHECK VALVE INTERNALS HAVE BEEN REMOVED.

REFERENCE DRAWINGS:

5613-M-3075 SH. 2	AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER TO STEAM GENERATORS
5613-M-3075 SH. 3	AUXILIARY FEEDWATER SYSTEM NITROGEN SUPPLY TO AFW CONTROL VALVES
5610-M-3065 SH.	NITROGEN & HYDROGEN SYSTEMS
5610-M-3075 SH. 1	AUXILIARY FEEDWATER SYSTEM TURBINE DRIVE FOR AFW PUMPS
5610-M-3075 SH. 2	AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER PUMPS
5613-M-3014 SH. 1	CONDENSER SYSTEM
5613-M-3072 SH. 1	MAIN STEAM SYSTEM
5613-M-3074 SH. 3	MAIN FEEDWATER SYSTEM
5613-M-3084 SH. 1	AUXILIARY STEAM SYSTEM
5610-T-L1 SH. 15	AUXILIARY FEEDWATER PUMP START

THIS DRAWING SUPERSEDES DRAWINGS:
5610-M-1 SH. 2 REV. 5
5610-M-1302 REV. 5

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4061 SH. 4 REV. 30

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
8	9-5-95	INCORPORATED CRN-M-8346 FOR PC/M 95-099.	RH	RV	DJB	BD	12	03-29-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061).	JPG	RGR	—	JRH
17	5-13-10	ISSUED AS-BUILT PER PC/M 05-036.	RH	RV	BSG	ASD	11	11-24-98	ISSUED AS-BUILT PER CRN-M-9439 (PC/M 98-036).	RH	RV	JMM	JDL
16	8-30-05	ISSUED AS-BUILT PER PC/M 05-036. (PARTIAL)	RH	RV	AAP	RSV	10	6-3-97	ISSUED AS-BUILT PER PC/M 96-029 AND INCORPORATED CRN-M-8922.	RH	JK	JZ	JJC
15	8-15-05	ISSUED AS-BUILT PER PC/M 05-036. (PARTIAL)	RH	RRW	AAP	JTL	9	6-18-96	INCORPORATED CRN-M-8684 FOR PC/M 96-023.	RH	RV	PDK	BD
14	02-07-02	ISSUED AS-BUILT PER CRN-C-10939 (PC/M 00-016).	RH	RV	CMZ	JTL	0	5-14-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	GFH	MD	AM/RAS	LRB
13	05-09-01	ISSUED AS-BUILT PER CRN-M-10292 (PC/M 00-016).	RH	RV	AAP	JM	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-137.	—	—	—	—

POD

FPL

TURKEY POINT NUCLEAR UNIT 3

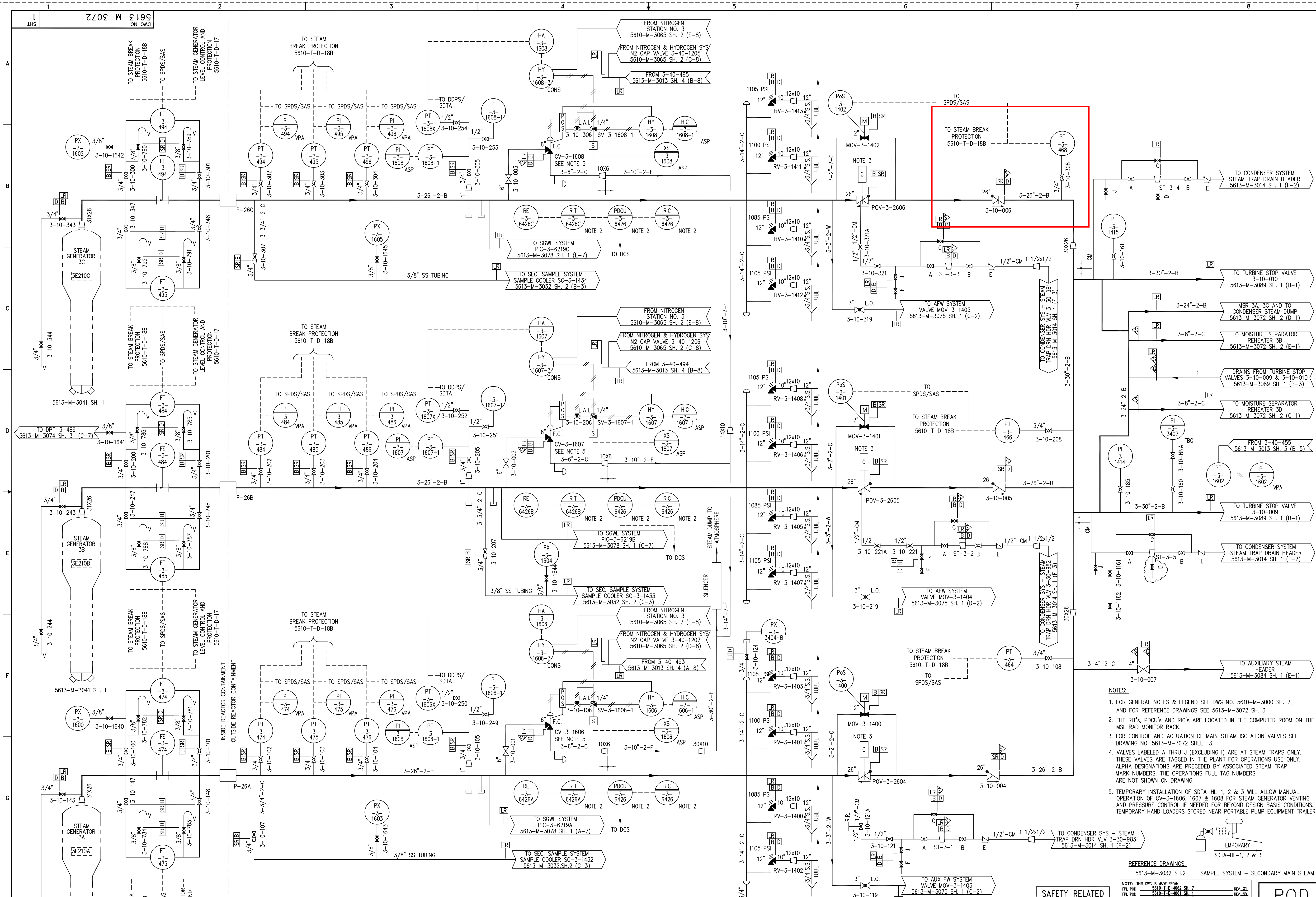
P & ID

AUXILIARY FEEDWATER SYSTEM
STEAM TO AUXILIARY FEEDWATER PUMP TURBINES

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3075 | SYS 075 | SHEET 1 | REV 17

5613-M-3075_SH 1.DWG



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2, AND FOR REFERENCE DRAWINGS SEE 5613-M-3072 SH. 3.
2. THE RIT's, PDCU's AND RIC's ARE LOCATED IN THE COMPUTER ROOM ON THE MSL RAD MONITOR RACK.
3. FOR CONTROL AND ACTUATION OF MAIN STEAM ISOLATION VALVES SEE DRAWING NO. 5613-M-3072 SHEET 3.
4. VALVES LABELED A THRU J (EXCLUDING I) ARE AT STEAM TRAPS ONLY. THESE VALVES ARE TAGGED IN THE PLANT FOR OPERATIONS USE ONLY. ALPHA DESIGNATIONS ARE PRECEDED BY ASSOCIATED STEAM TRAP MARK NUMBERS. THE OPERATIONS FULL TAG NUMBERS ARE NOT SHOWN ON DRAWING.
5. TEMPORARY INSTALLATION OF SDTA-HL-1, 2 & 3 WILL ALLOW MANUAL OPERATION OF CV-3-1606, 1607 & 1608 FOR STEAM GENERATOR VENTING AND PRESSURE CONTROL IF NEEDED FOR BEYOND DESIGN BASIS CONDITIONS. TEMPORARY HAND LOADERS STORED NEAR PORTABLE PUMP EQUIPMENT TRAILER.

REFERENCE DRAWINGS:

5613-M-3032 SH.2 SAMPLE SYSTEM - SECONDARY MAIN STEAM.

SAFETY RELATED

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4062 SH. 7 REV. 21
FPL POD 5610-T-E-4061 SH. 1 REV. 65
FPL POD 5610-T-E-4061 SH. 3 REV. 36

POD

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
37	10-4-11	ISSUED AS-BUILT PER EC-DCR 273958.	RH	BB	BSG	VMG	41	7-25-12	ISSUED AS-BUILT PER EC 242385 (PC/M 08-017).	RV	BB	DM	CM
36	10-30-10	ISSUED AS-BUILT PER EC 242400 (PC/M 08-034).	RV	YV	CP	JD	40	7-9-12	ISSUED AS-BUILT PER EC 246875.	RH	RV	PB	JD
45	7-15-13	ISSUED AS-BUILT PER EC 250185 (ITOP 13-07-032).	RH	RV	–	EB	39	06-29-12	ISSUED AS-BUILT PER EC 246875. (PARTIAL)	RV	RH	–	SB
44	2-6-13	ISSUED AS-BUILT PER EC 271032 (ITOP 13-02-001).	RV	AFG	BB	BAP	38	06-05-12	ISSUED AS-BUILT PER EC 246875. (PARTIAL)	RV	SB	–	CW
43	12-2-12	ISSUED AS-BUILT PER CRNs 007 & 011 (EC 242400). ITOP 12-12-003.	AFG	PSB	–	JD	0	1-16-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	TAM	MD	LRB/RAS	HAR
42	11-16-12	ISSUED AS-BUILT PER EC-DCR 250231.	RH	JPH	–	SB	——	——	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-90-512.	——	——	——	——

FPL

TURKEY POINT NUCLEAR UNIT 3

P & I D

MAIN STEAM SYSTEM

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3072 | SYS 072 | SHEET 1 | REV 45

30x42 E SIZE 02/26/93

5613-M-3072_SH 1.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	Topic and K/A #	G3		2.3.15
	Importance Rating	2.9		

Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO Question # 71

Given the following conditions:

- Unit 3 is raising power from 25% to 100%.
- Containment radiation monitors indicate rising trends.
- R-11/12, Containment Air Particulate / Gas Radiation Monitors are in alarm.

Which one of the following completes the statement below?

In accordance with 3-ONOP-067, Radioactive Effluent Release, to check the channel operability of R-11/12, the operator must __________.

A.	depress the CHECK SOURCE pushbutton and ensure that the readout rises slightly.
B.	depress the CHECK SOURCE pushbutton and ensure that the readout equals 288K or 289K.
C.	depress the FAIL/TEST pushbutton and ensure that the readout rises slightly.
D.	depress the FAIL/TEST pushbutton and ensure that the readout equals 288K or 289K.

Proposed Answer: A

A.	Correct. R11/12 does not have a fail/test button

B.	Incorrect. Plausible if candidate believes a check source will check channel operability which is true in some cases (R11/12), however the readout will not read 288k on a source check.		
C.	Incorrect. A second check of the monitor is to ensure the green FAIL light is off but it has nothing to do with pressing the button. Other process monitors have fail test buttons which may either cause a deflection or LED readout to indicate 288k or 289k.		
D.	Incorrect. Plausible because other process monitors have fail test buttons which may either cause a deflection or LED readout to indicate 288k or 289k.		
Technical Reference(s)	3-ONOP-067		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	X	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		11
	55.43		
Purpose and operation of radiation monitoring systems, including alarms and survey equipment.			
Comments: PTN internal bank.			

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

- *A PRMS source check on a channel with a HIGH Alarm may be inconclusive because the effect of the source may be insignificant to cause noticeable change in the readout.*
- *Step 2b is NOT applicable to R-3-11/12, R-3-15/19, or R-3-20 HIGH ALARM.*
- *Step 2C is NOT applicable to R-3-20 HIGH ALARM.*

2 **Check Affected PRMS Channel alarm - Alarm Setpoint Exceeded**

Perform the following:

- Notify the Shift Manager of problem with PRMS channel.
- Direct Radiation Protection Shift Supervisor to conduct radiological surveys to confirm validity of alarm.
- Direct Chemistry to perform sampling to confirm validity of alarm.
- Continue with procedure until affected systems are verified normal.

a. Check readout on affected channel - GREATER THAN OR EQUAL TO ALARM SETPOINT

~~b. Check channel operability as follows~~

~~1) Depress and hold FAIL/TEST pushbutton on affected PRMS Channel~~

~~2) Check readout - EQUAL TO 288K OR 289K~~

~~3) Release FAIL/TEST pushbutton~~

c. Check affected PRMS drawer responds to source check

d. Check for PRMS channel failure

- Check Fail indicator – OFF
- Display and recorder reading - **<u>NOT</u>** FAILED LOW
- For R-11/12 check RM-80 Green Monitor Light - ON

d. Perform Step 8.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	Topic and K/A #	G2		2.2.2
	Importance Rating	4.6		

Equipment Control: Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Proposed Question: RO Question # 72

Given the following conditions:

- Unit 3 is MOL.
- A reactor startup is in progress in accordance with 3-GOP-301, Hot Standby to Power Operation.
- Power is stabilized at 10^{-8} amps for critical data.

Which one of the following completes the statements below?

During the next rod withdrawal, the maximum startup rate allowed is ___(1)___ .

Assuming no operator action, once the maximum SUR is established, SUR will ___(2)___ .

A.	(1) 1 dpm (2) remain constant until the reactor trips
B.	(1) 0.5 dpm (2) remain constant until the reactor trips
C.	(1) 1 dpm (2) lower until reactor power stabilizes
D.	(1) 0.5 dpm (2) lower until reactor power stabilizes

Proposed Answer: C

A.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible because the candidate must know that during a mid-cycle startup, MTC and power coefficient are negative, countering the power increase. Without this understanding the candidate can assume that power keeps rising without additional action by the RCO.		
B.	Incorrect. Part 1 is incorrect, but plausible given 0.5 dpm SUR is a true limit when above the POAH. Part 2 is incorrect, but plausible per discussion above.		
C.	Correct. When the reactor reaches the point of adding heat, the plant Tavg, pressurizer pressure and level increase. Doppler and moderator feedback will limit the power increase and the plant will stabilize at a higher power level determined by the steam demand. MOL MTC will also counter the positive reactivity of a rod withdrawal.		
D.	Incorrect. Part 1 is incorrect, but plausible per discussion above. Part 2 is correct.		

Technical Reference(s)	Caution pg 52, 3-GOP-301		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	16356	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2012	Indian Point 2
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		1
	55.43		
Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.			
Comments: Modified stem and distractors. Made 2x2. Added SUR requirement below the POAH.			

4.0 **PRECAUTIONS/LIMITATIONS**

4.1 Criticality should be anticipated anytime when shutdown or control rod banks are being withdrawn or boron dilution is in progress.

4.2 All shutdown rods shall be fully withdrawn before the reactor is made critical.

4.3 Do not make the reactor critical with a moderator temperature coefficient of reactivity more positive than +5 pcm/°F (except as permitted for low power physics tests).

4.4 The approach to criticality shall be guided by plotting inverse count rate ratio versus control rod position. Observe the 1/m plot to assure criticality will not occur below the insertion limit for zero power.

4.5 Before withdrawing any rod bank from the fully inserted position, the group step counters and the rod position indicators for that bank shall meet the control rod position Acceptance Criteria in 3-OSP-201.1, RO Daily Logs.

4.6 When moving shutdown or control rod banks; the Group Step Counters, RPIs, and all Nuclear Instrumentation Channels shall be closely monitored to verify proper bank movement and bank overlap for control rods.

4.7 The Reactor Coolant System lowest operating loop temperature (Tavg) shall be greater than or equal to 541°F with Keff greater than or equal to 1.0.

4.8 All Reactor coolant loops shall be in operation prior to making the reactor critical, Mode 2. With less than 3 Loops in operation, restore all Loops to operable status or be in Hot Standby within six (6) hours.

4.9 Before transferring the Rod Control selector from Manual to Auto mode, the control rod banks shall be positioned as required to adjust Tavg within 1.0°F of Tref.

4.10 At power, all Rod Position Indicators and Power Range Nuclear Channels shall be periodically monitored for control rod misalignment and abnormal power distribution.

4.11 Every attempt should be made to maintain the Axial Flux Difference within the Operational Space to avoid; otherwise, unnecessary power reductions; reference 0–NOP–059.09, Operation Within the Axial Flux Difference Operational Space.

4.12 Control banks shall be maintained above the respective Rod Bank A-B-C or D Low Limit Alarm by maintaining the required RCS boron concentration.

4.13 When any control rod bank is below the Rod Bank A-B-C or D Extra Low Limit Alarm, then refer to T.S. 3.1.3.6, Control Rod Insertion Limits.

4.14 SUR should not be permitted to exceed a steady state value of 1.0 dpm below the POAH and 0.5 dpm above the POAH.

4.15 If the Steam Dump System is automatically armed by a load rejection and equilibrium conditions are re-established, the Steam Dump Control shall be reset by placing the steam dump to condenser Mode Selector switch to Reset.

4.16 The Steam Pressure Control Dump to Condenser Auto/Manual station shall have a zero output signal prior to placing the Steam Dump to Condenser Mode Selector in Manual.

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Question 72 original

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

The following plant conditions exist during a reactor start-up with the MSIVs closed:

- ReS Boron is 1000 PPM.
- Estimated Critical Position is Bank D at 100 steps
- Doppler Power Coefficient is -12 pcm/%
- Moderator Temperature Coefficient is -5 pcm/F
- Differential Rod Worth is 5 pcm/step

After recording critical data, the Reactor Operator pulls rods 10 steps to put the reactor on a 0.2 DPM startup rate. If no other operator action is taken to stabilize the plant at the POAH, the expected plant response will be:

A. Tavg, power level, and pressurizer level will all increase until the reactor trips at 10% power.

B. Tavg, power level, and pressurizer level will increase while the steam dumps open to stabilize the plant.

C. Tavg, power level, and pressurizer level will increase while the atmospherics open to stabilize the plant.

D. Tavg will increase which will add negative reactivity causing power to decrease, which will drive the reactor sub-critical.

Proposed Answer: C

Explanation (Optional):

A. Incorrect.

B. Incorrect.

C. When the reactor reaches the point of adding heat, the plant Tavg, pressurizer pressure and level increase. Doppler and moderator feedback will limit the power increase and the plant will stabile at a higher power level determined by the steam demand on the SG atmospherics.

D. Incorrect.

Techical Reference(s): 2-POP-1.3 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: I2LP-ILO-POP006 3 (As available)

Question Source: Bank 16356

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2012 Indian Point 2

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	Topic and K/A #	G4		2.4.37
	Importance Rating	3.0		

Emergency Procedures / Plan: Knowledge of the lines of authority during implementation of the emergency plan.

Proposed Question: RO Question # 73

Given the following conditions:

An imminent airborne threat has been confirmed by Plant Security.

Which one of the following plant individuals will become the Emergency Coordinator and who will relieve them?

	Emergency Coordinator	Provides Relief
A.	Shift Manager	Emergency Coordinator in TSC
B.	Security Shift Supervisor	Recovery Manager in the EOF
C.	Shift Manager	Recovery Manager in the EOF
D.	Security Shift Supervisor	Emergency Coordinator in TSC

Proposed Answer: A

A.	Correct, IAW EPIP-20101.
B.	Incorrect. SM makes initial declaration. Plausible because it is a security event and safeguards information related to security is not routinely disseminated. Recovery Manager is plausible because he/she is responsible for the overall recovery effort, but they are in the EOF
C.	Incorrect. SM will be relieved by the EC. Same explanation as Option B
D.	Incorrect. SM makes initial declaration. Second part correct and first part plausible as in option B

Technical Reference(s)	EPIP 20101 section 3.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	16272	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2011	Callaway
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			

5.1 General (Cont'd)

5.1.6 In any case where a **General Emergency** has been declared, the minimum protective action recommendation shall be: **Shelter all people within a 2 mile radius from the plant and 5 miles in the down wind sectors**.

5.1.7 The Emergency Coordinator responsibilities shall reside with the EC in the Control Room until they have been formally transferred to the EC in the TSC.

5.1.8 Once indications are available that an EAL threshold has been reached, the Shift Manager/Emergency Coordinator is required to classify the event within 15 minutes.

5.1.9 Emergency notification to State and Local counties is required within15 minutes of declaring an emergency.

5.1.10 Emergency notification to the NRC is required immediately following notification of State and Counties, but **NOT** later than 1 hour from the declaration of an emergency.

5.1.11 If, during the notification process, it becomes necessary to upgrade the emergency classification:

- Update the notification to reflect the higher emergency classification **AND** complete the update notifications within 15 minutes of the lesser emergency declaration.
- If the notification can **NOT** be updated **AND** completed within 15 minutes of the lesser emergency declaration, the ERO should make the declaration of the lesser emergency within 15 minutes of its declaration. The notification can contain the caveat that a change in classification was forthcoming.
- At no time will there be a restart of the notification clock.

Facility:	Turkey Point
Vendor	WEC
Exam Date:	
Exam Type:	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

An imminent airborne threat has been confirmed IAW OTO-SK-00002, Plant Security Event-Aircraft Threat.

Which ONE (1) of the following plant individuals will make the initial Radiological Emergency Response Plan (RERP) event declaration **AND** which plant individual will relieve this individual after the RERP organization becomes functional?

	Declares the Event	Relieves Individual Declaring Event
A.	Shift Manager (SM)	Emergency Duty Officer (EDO)
B.	Security Shift Supervisor (SSS)	Recovery Manager (RM)
C.	Shift Manager (SM)	Recovery Manager (RM)
D.	Security Shift Supervisor (SSS)	Emergency Duty Officer (EDO)

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. SM makes initial declaration.

C. Incorrect. SM will be relieved by the EDO.

D. Incorrect. SM makes initial declaration.

Techical Reference(s): (Attach if not previoously provided)

Proposed Reference to be provided to applicants during examination:

Learning Objective: (As available)

Question Source: Bank

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	Topic and K/A #	G4		2.4.5
	Importance Rating	3.7		

Emergency Procedures / Plan: Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Proposed Question: RO Question # 74

Given the following conditions:

- Unit 3 trips from full power due to a Startup Transformer failure.
- The Startup Transformer must be replaced.
- Safety Injection is NOT required.

Which one of the following completes the statements below:

The crew will transition from 3-EOP-E-0, Reactor Trip or Safety, to a(n) __(1)___ to cooldown the plant.

When performing non-accident EOPs, all Technical Specification surveillances ___(2)___ required to be complied with.

A.	(1) Functional Restoration Procedure (2) are NOT
B.	(1) Functional Restoration Procedure (2 are
C.	(1) Optimal Recovery Procedure (2) are NOT
D.	(1) Optimal Recovery Procedure (2) are

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect. Plausible when candidate interprets functional as restoring the function of normal 4kV bus power (off-site power) while cooling down the plant. Part 2 is incorrect, but plausible when candidate waives technical specifications during 3-EOP-ES-0.4 Natural Circulation Cooldown as they are waived in “accident” EOPs.
B.	Incorrect. Part 1 is incorrect and plausible per discussion above. Part 2 is correct.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect and plausible per discussion above.
D.	Correct, IAW 0-ADM-211.

Technical Reference(s)	0-ADM-211 pg 12 and 32		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	LP 6902320 Obj 2		(As available)
Question Source:	Bank	11683	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2009	Comanche Peak
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		10
	55.43		
Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Comments:			
Made 2x2. Added non-accident EOP tech spec requirements.			

4.0 INSTRUCTIONS

4.1 Introduction

1. Procedural guidance for mitigation of transients and accidents falls within the following groups of procedures:

   - Emergency Operating Procedures (EOPs)
   - Off-Normal (Abnormal) Operating Procedures (ONOPs/AOPs)

2. Emergency Operating Procedures

   A. EOPs direct operator actions to mitigate the effects of operational accidents, to recover the plant to a stable condition, and, for most events, to cool down and depressurize the RCS to a Cold Shutdown condition. The EOPs are developed from the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs), which use a two-column format. The EOPs are comprised of three distinct parts:

   (1) Optimal Recovery Procedures (E-series) are event-based EOPs. They diagnose and provide operator actions to respond to predefined events or combinations of events. These procedures are termed OPTIMAL because their actions are tailored for specific, predefined events, or event combinations. E-series EOPs are numbered using E-, ES-, and ECA-prefixes.

   (2) Function Restoration Procedures (FRPs) are symptom-based EOPs. They provide operator actions to respond to challenges to plant safety, as indicated by predefined critical safety functions (CSFs). FRPs are independent of the event in progress and are numbered using an FR-prefix followed by a one-letter designator identifying the associated CSF.

4.7 Procedure Adherence for Emergency and Off Normal (Abnormal) Procedures (continued)

7. Relationship between EOP/ONOP (AOP) and Technical Specifications.

A. EOP implementation during accident conditions may violate Technical Specifications; these violations were considered in the ARG and ERG development process and are permissible. The non-accident EOPs, (ES-0.1, ES-0.2, ES-0.3, and ES-0.4) and the non-shutdown event ARGs do **NOT** include actions that would intentionally violate Technical Specifications.

B. When performing accident EOPs, Technical Specification surveillance and monitoring requirements, for which the operator has responsibility and which would normally be performed as part of the evolution in progress, are suspended until the EOPs are completed, unless otherwise specified by the EOP or another procedure referenced by EOP. When performing non-accident EOPs, all Technical Specification requirements are to be complied with, to ensure plant operation is conducted in a safe manner with design features ready to respond should an accident occur.

Facility:	WTSI Corporate	
Vendor	WEC	Question 74 original
Exam Date:		
Exam Type:		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Which ONE (1) of the following procedure groups are the Optimal Recovery Guidelines at Comanche Peak?

A. EOP, EOS, Status Trees.

B. EOP, ECA, FRG.

C. ECA, FRG, Status Trees.

D. EOP, EOS, ECA.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because EOP and EOS procedures are correct, however, the Status Trees are part of the Functional Recovery Guidelines.

B. Incorrect. Plausible because EOP and ECA procedures are correct, however, the FRGs are part of the Functional Recovery Guidelines.

C. Incorrect. Plausible because the ECA procedures is correct, however, the FRGs and Status Trees are part of the Functional Recovery Guidelines

D. Correct. These are the three sets of procedures that make up the Optimal Recovery Guidelines.

Techical Reference(s): LO21.ERG.XG1.LN, Page 12 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: N

Learning Objective: LO21.ERG.XG1.OB04
LIST and DIFFERENTIATE between the three types of Optimal Recovery Guidelines. (As available)

Question Source: Bank 11683
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2009 Comanche Peak

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	Topic and K/A #	G4		2.4.3
	Importance Rating	3.7		

Emergency Procedures / Plan: Ability to identify post-accident instrumentation.

Proposed Question: RO Question # 75

Which one of the following identifies a Control Board Instrument that may be relied upon in a post-accident condition, and the required color of the instrument label, in accordance with 0-ADM-209, Equipment Tagging and Labeling?

A.	PI-3-444, Pressurizer Pressure; blue
B.	PI-3-444, Pressurizer Pressure; purple
C.	TI-3-410A, Loop A T-cold Wide Range; blue
D.	TI-3-410A, Loop A T-cold Wide Range; purple

Proposed Answer: D

A.	Incorrect since WR, not PRZ, RCS pressure instrumentation is required by 0-ADM-209 and tech specs. Also incorrect since WR Tcold *is* an Accident Monitoring instrument and 0-ADM-209, *Equipment Tagging and Labeling*, Definition 4.6, requires a purple label, not a blue label. Plausible because PI-3-444 is a control room color coded component IAWA 0-ADM-209. Also plausible because many safety-related instruments in the Control Room have blue labels.
B.	Incorrect and plausible per discussion above. Purple is the correct color code for post-accident instrumentation.
C.	Incorrect since WR Tcold *is* an Accident Monitoring instrument and 0-ADM-209, *Equipment Tagging and Labeling*, Definition 4.6, requires a purple label, not a blue label. Plausible because the 1st part is correct. Also plausible because many safety related instruments in the Control Room have blue labels.

D.	Correct. Per 0-ADM-209, *Equipment Tagging and Labeling*, Definition 4.6,Reg Guide 1.97, Common Markings - A fade resistant vinyl type tape colored purple which will enable Control Room Operators to identify instruments/indicators which may be relied upon in a Post Accident Condition. TI-3-410A is included in Attachment 3 Reg Guide 1.97 verification checklist.		
Technical Reference(s)	TS Tables 3.3.1, 3.3-2, 3.3-5		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			NO
Learning Objective:	LP 6900523, Obj. 3		(As available)
Question Source:	Bank	12856	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2011	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		7
	55.43		
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Comments:			

4.0 DEFINITIONS

4.1 ETI Tag (Equipment Temporary Identification Tag) - Pre-printed tags similar to Enclosure 1 with unique serial numbers for tracking purposes. ETI Tags without serial numbers are not to be used.

4.2 Equipment Temporary Identification (ETI Tag) Tacking Log - An index/log used to track the status of ETI Tags and completed tags/labels. The log shall be in a format similar to Attachment 6 and may be kept electronically in the G:\Ops\DeptShares\ETI Labeling directory.

4.3 Operator Aids - Information including sketches, notes, graphs, instructions, drawings, and other documents used to assist operators in performing assigned duties. For the purposes of administrative control, a Temporary Information Tag is not considered to be an Operator Aid.

4.4 Temporary Information Tags - Preprinted tags filled in with information of the status of equipment and precautions or instructions for its operation. These tags may be white, red, or green, depending on the application:

4.4.1 White/clear information tags (of any format) are normally used to provide general information.

4.4.2 Red or Green information tags are normally used on control switches for equipment with the breaker deenergized (indicating lights are off) to indicate the open/closed, on/off status of the equipment.

4.4.3 Tags should be attached to the switches or equipment to which they refer.

4.5 Operator Aid and Temporary Information Tag Log - A notebook maintained in the Control Room, which contains two indexes; Operator Aid Index and Temporary Information Tag Index (a form similar to Attachment 2).

4.6 Permanent Information - Information that appears on a medium not suitable to change and determined by the Assistant Operations Manager to be applicable indefinitely. An example would be notes or cautions produced on a Bakelite plate. Information posted in this format is not considered to be an operator aid in this procedure and does not provide direction to operate the plant.

4.7 Reg Guide 1.97, Common Markings - A fade resistant vinyl type tape colored purple which will enable Control Room Operators to identify instruments/indicators which may be relied upon in a Post Accident Condition.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirement specified in Table 4.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 2	1 1	2 2	1, 2 3*, 4*, 5*	1 9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1##, 2	2
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3
4. Source Range, Neutron Flux					
a. Startup	2	1	2	2#	4
b. Shutdown**	2	0	2	3, 4, 5	5
c. Shutdown	2	1	2	3*, 4*, 5*	9
5. Overtemperature ΔT	3	2	2	1, 2	13
6. Overpower ΔT	3	2	2	1, 2	13
7. Pressurizer Pressure-Low (Above P-7)	3	2	2	1	6
8. Pressurizer Pressure--High	3	2	2	1, 2	6
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	13
10. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen.	2/stm. gen.	1, 2	6
12. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen.	1 stm. gen. level coin-cident with 1 stm./feed-water flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6
13. Undervoltage--4.16 KV Busses A and B (Above P-7)	2/bus	1/bus on both busses	2/bus	1	12
14. Underfrequency-Trip of Reactor Coolant Pump Breaker(s) Open (Above P-7)	2/bus	1 to trip RCPs***	2/bus	1	11
15. Turbine Trip (Above P-7)					
a. Emergency Trip Header Pressure	3	2	2	1	12
b. Turbine Stop Valve Closure	2	2	2	1	12

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Safety Injection Input from ESF	2	1	2	1, 2	8
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2#	7
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	7
or					
Turbine Inlet Pressure	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
18. Reactor Coolant Pump Breaker Position Trip					
a. Above P-8	1/breaker	1	1/breaker	1	11
b. Above P-7 and below P-8	1/breaker	2	1/breaker	1	11
19. Reactor Trip Breakers	2	1	2	1, 2	8, 10
	2	1	2	3*, 4*, 5*	9
20. Automatic Trip and Interlock logic	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value within permissible calibration tolerance.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-3, either:

    1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-3 and determine within 12 hours that the affected channel is OPERABLE; or

    2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection					
a. Manual Initiation	2	1	2	1 2, 3, 4	17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1 2, 3, 4	14
c. Containment Pressure - High	3	2	2	1 2, 3	15
d. Pressurizer Pressure - Low	3	2	2	1 2, 3#	15
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1 2, 3#	15

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2. 3*	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements. (Manual S.I. initiation will not initiate Phase A Isolation).				
b. Phase "B" Isolation					
1) Manual Initiation	2	2 (Both buttons must be pushed simultaneously to actuate)	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15
c. Containment Ventilation Isolation					
1) Containment Isolation Manual Phase A or Manual Phase B	See Items 3.a.1 and 3.b.1 above for all Manual Containment Ventilation functions and requirements.				

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	16
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions requirements.				
4) Containment Radioactivity-High	2##	1	1	1, 2, 3, 4	16
4. Steam Line Isolation					
a. Manual Initiation (individual)	1/operating steam line	1/operating steam line	1/operating steam line	1, 2, 3	21
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
c. Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with: Steam Generator Pressure--Low	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3	15
	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3	15
or T_{avg}--Low	1/Loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	25
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Steam Generator Water Level -- High-High# # # #	3/steam generator	2/steam generator in any operating steam generator	2/steam generator in any operating steam generator	1, 2, 3	15
6. Auxiliary Feedwater# # #					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater### (Continued)					
b. Stm. Gen. Water Level-- Low-Low	3/steam generator	2/steam generator in any steam generator	2/steam generator	1, 2, 3	15
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Bus Stripping	1/bus	1/bus	1/bus	1, 2, 3	23
e. Trip of all Main Feed-water Pumps Breakers	1/breaker	(1/breaker) /operating pump	(1/breaker) /operating pump	1, 2	23
7. Loss of Power					
a. 4.16 kV Busses A and B (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4	18
b. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
Coincident with: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Loss of Power (Continued)					
c. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Degraded Voltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
8. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure	3	2	2	1, 2, 3	19
b. T_{avg} - Low	3	2	2	1, 2, 3	19
9. Control Room Ventilation Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 6**	16
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Containment Radioactivity--High	2	1	1	1, 2, 3, 4, 6**	16
d. Containment Isolation Manual Phase A or Manual Phase B	2	1	2	1, 2, 3, 4	17
e. Control Room Air Intake Radiation Level	2	1	2	All	24

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The accident monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

b. The provisions of Specification 3.0.4 are not applicable to ACTIONS in Table 3.3-5 that require a shutdown.

c. Separate Action entry is allowed for each Instrument.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
1. Containment Pressure (Wide Range)	2	1	1, 2, 3	31, 32
2. Containment Pressure (Narrow Range)	2	1	1, 2, 3	36
3. Reactor Coolant Outlet Temperature T_{HOT} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
4. Reactor Coolant Inlet Temperature T_{COLD} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
5. Reactor Coolant Pressure – Wide Range	2	1	1, 2, 3	31, 32
6. Pressurizer Water Level	2	1	1, 2, 3	31, 32
7. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator	1, 2, 3	31, 32
8. Reactor Coolant System Subcooling Margin Monitor	2(2)	1(2)	1, 2, 3	31, 32
9. PORV Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	33
10. PORV Block Valve Position Indicator	1/valve	1/valve	1, 2, 3	33
11. Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	32
12. Containment Water Level (Narrow Range)	2	1	1, 2, 3	36
13. Containment Water Level (Wide Range)	2	1	1, 2, 3	31, 32

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

Facility:	WTSI Corporate		Question 75 original
Vendor	WEC		
Exam Date:			
Exam Type:			

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Which ONE of the following choices identifies a Control Board Instrument required by Technical Specification 3.3.3.3, Accident Monitoring Instrumentation, and the required color of the instrument label in accordance with 0-ADM-209, Equipment Tagging and Labeling?

A. PI-3-444, Pressurizer Pressure; blue

B. PI-3-444, Pressurizer Pressure; purple

C. TI-3-410A, Loop A T-cold Wide Range; blue

D. TI-3-410A, Loop A T-cold Wide Range; purple

Proposed Answer: D

Explanation (Optional):

A. Incorrect since WR, not PZR, RCS pressure instrumentation is required by TS Table 3.3-5. Also incorrect since WR Tcold *is* an Accident Monitoring instrument and 0-ADM-201, *Equipment Tagging and Labeling*, Definition 4.6, requires a purple label, not a blue label. Plausible because TS Table 3.3-1, Reactor Trip System Instrumentation requires the Functional Units 7 & 8, Pressurizer Pressure.
Also plausible since TS Table 3.3-2, ESF Actuation System Instrumentation requires the Functional Unit 1d, Pressurizer Pressure. Also plausible because many safety-related instruments in the Control Room have blue labels.

B. Incorrect since WR, not PZR, RCS pressure instrumentation is required by TS Table 3.3-5. Plausible because the 2nd part is correct. Also plausible because TS Table 3.3-

1, Reactor Trip System Instrumentation, requires the Functional Units 7 & 8, Pressurizer Pressure. Also plausible since TS Table 3.3-2, ESF Actuation System Instrumentation, requires the Functional Unit 1d, Pressurizer Pressure.

C. Incorrect since WR Tcold *is* an Accident Monitoring instrument and 0-ADM-201, *Equipment Tagging and Labeling*, Definition 4.6, requires a purple label, not a blue label. Plausible because the 1st part is correct. Also plausible because many safetyrelated instruments in the Control Room have blue labels.

D. CORRECT. TS Table 3.3-5, Accident Monitoring Instrumentation, Instrument 4, requires WR Tcold. Per 0-ADM-201, *Equipment Tagging and Labeling*, Definition 4.6, Reg Guide 1.97, Common Markings - A fade resistant vinyl type tape colored purple which will enable Control Room Operators to identify instruments/indicators which may be relied upon in a Post Accident Condition.

Techical Reference(s): TS Tables 3.3.1, 3.3-2, 3.3-5 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: LP 6900523, Obj. 3 (As available)

Question Source: Bank 12856

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	Topic and K/A #	007		EA2.02
	Importance Rating			4.6

Ability to determine or interpret the following as they apply to a reactor trip: Proper actions to be taken if the automatic safety functions have not taken place

Proposed Question: SRO Question # 76

Given the following initial conditions:

- Unit 3 is at 7% power.
- An automatic reactor trip signal is generated, but the reactor fails to trip.
- 3-EOP-E-0, Reactor Trip or Safety Injection, is entered.
- The crew completes their immediate operator actions.
- The unit is at 6% power and lowering.

Which one of the following completes the statements below?

The SRO ___(1)___ direct the RO to perform a boration in accordance with 0-OP-046, CVCS Boron Concentration Control for the failure.

If an automatic SI fails to occur when required, the SRO will ___(2)___ .

A.	(1) will NOT (2) direct a transition back to 3-EOP-E-0, Reactor Trip or Safety Injection
B.	(1) will NOT (2) continue in current procedure while aligning equipment in Attachment 3 of 3-EOP-E-0
C.	(1) will (2) direct a transition back to 3-EOP-E-0, Reactor Trip or Safety Injection
D.	(1) will (2) continue in current procedure while aligning equipment in Attachment 3 of 3-EOP-E-0

Proposed Answer: B

A.	Incorrect. Part 1 is correct. Part 2 is incorrect but plausible because the candidate may believe a transition back to E-0 from FR-S.1 is warranted since this is done in other EOPs.
B.	Correct, boration will be performed IAW FR-S.1 body steps and SRO will remain in procedure and perform E-0 Attachment 3 IAW page 9 CAUTION.
C.	Incorrect. Part 1 is incorrect, but plausible if candidate believes they transitioned to 3-EOP-ES-0.1 and a normal boration is required. Also plausible if candidate believes a normal boration is performed in FR-S.1. EOPs also reference Normal Procedures (e.g. when placing PAHMs in service, AFW and EDG operation). Part 2 is incorrect.
D.	Incorrect. Both parts incorrect. Plausible per discussion above.

Technical Reference(s)	3-EOP-E-0 3-EOP-ES-0.1 3-EOP-FR-S.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and choose the appropriate procedural response in event of an ATWS containing additional complications of a safety injection actuation

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

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

> **NOTE**
>
> Step 1 through Step 4 are IMMEDIATE ACTION Steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Verify Reactor Trip** • Rod Bottom Lights – ON • Reactor Trip AND Bypass Breakers – OPEN • Rod Position Indicators – AT ZERO • Neutron flux – DECREASING	Manually trip reactor. IF reactor power is greater than 5% OR Intermediate Range Power is **NOT** stable or decreasing, THEN perform the following: **a.** Monitor Critical Safety Functions using 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES. **b.** Go to 3-EOP-FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION / ATWS, Step 1.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**Verify All Control Rods – FULLY INSERTED**	IF any Control Rod **NOT** fully inserted, THEN Emergency Borate for stuck control rods using 3-ONOP-046.1, EMERGENCY BORATION, while continuing with Step 4.
4.	**Check 4KV Power Status To Both Unit 3 And Unit 4**	
	a. Check 4A AND 4B 4KV Bus – BOTH DE-ENERGIZED	**a.** Go to Step 5.
	b. Check 3A AND 3B 4KV Bus – ONLY ONE ENERGIZED	**b.** Go to Step 5.
	c. Check Unit 3 Energized Bus – ENERGIZED FROM EDG	**c.** Go to Step 5.
	d. Go to Attachment 2	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Initiate Emergency Boration Of RCS:**	
	a. Verify SI – RESET	
	b. Verify Charging Pumps – AT LEAST ONE RUNNING IN MANUAL	
	c. Stop Makeup System	
	d. Manually start Boric Acid Pump 3A or 3B	**d.** Align Charging Pump suction to the RWST as follows: **1)** Hold closed LCV-3-115C Control switch. **2)** Direct an operator to open Breaker 30669 for LCV-3-115C. **3)** WHEN 30669 is open, THEN release LCV-3-115C Control switch. **4)** Go to Step 4.f.
	e. Open MOV-3-350, Emergency Boration Valve	**e.** Perform the following: **1)** Open FCV-3-113A, Boric Acid To Blender. **2)** Open FCV-3-113B, Blender Flow To Charging Pump. **3)** Locally open 3-356, Manual Emergency Boration Valve. **4)** WHEN 3-356, Manual Emergency Boration Valve is open, THEN close FCV-3-113B, Blender To Charging Pump. **5)** Continue with Step 4.f.
	f. Open HCV-3-121, Charging Flow To Regen Heat Exchanger	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

4. (continued)

g. Verify CV-3-310A, Loop A Charging Isolation – OPEN

RNO **g.** Open CV-3-310B, Loop C Charging Isolation.

h. Establish Emergency Boration flow:

- FI-3-110 – GREATER THAN 60 GPM
- FI-3-122A – GREATER THAN 45 GPM

RNO **h.** Perform one <u>or</u> more of the following as necessary to establish Emergency Boration flow:

- * Adjust operating Charging Pump(s) speed controller(s).
- * Start additional Charging Pumps.
- * Manually align valves.

5. Verify Containment Ventilation Isolation:

a. Verify Unit 3 Containment Purge Exhaust <u>AND</u> Supply Fans – OFF

b. Verify Containment Purge Supply <u>AND</u> Exhaust Isolation Valves – CLOSED:

- POV-3-2600
- POV-3-2601
- POV-3-2602
- POV-3-2603

RNO **b.** <u>IF</u> <u>any</u> Purge Valve can **NOT** be closed, <u>THEN</u> pull fuses for any open Purge Valves from behind VPB:

- XEP for POV-3-2600
- XLAG for POV-3-2601
- XEQ for POV-3-2602
- XLAH for POV-3-2603

c. Verify Containment Instrument Air Bleed Isolation Valves – CLOSED

- CV-3-2819
- CV-3-2826

RNO **c.** <u>IF</u> **<u>neither</u>** valve can be closed, <u>THEN</u> locally close:

- MPAS-3-005, Containment Air Bleed to Purge Air Return Line Isolation.
- 3-11-018A, Instrument Air Bleed Line Drain Isolation Valve, (reach rod, Aux Bldg Hallway outside P&V Room)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If an SI signal exists or occurs AND the reactor is subcritical, proper safeguards equipment alignment is required to be verified using Attachment 3 of 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.

6. Check If The Following Trips Have Occurred:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Reactor Trip	**a.** In 3B MCC Room, locally trip reactor as follows: • Open 3A and 3B Reactor Trip Breakers. • Open 3A and 3B Reactor Trip Bypass Breakers. • Open A/B MG Set Generator Output Breakers. • Open A/B MG Set Motor Input Breakers
b. Turbine Trip	**b.** Locally trip turbine at Turbine Front Standard.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	Topic and K/A #	015		2.4.31
	Importance Rating			4.1

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: SRO Question # 77

Given the following conditions:

- Unit 3 is heating up in Mode 4.
- RCS pressure is 330 psig at 280F.
- Shortly after starting 3A RCP, the following is observed:
  - ANN A 7/5, RCP TROUBLE, alarms.
  - #1 seal ΔP is 140 psid.
  - #2 seal ΔP is 70 psid.
  - #3 seal ΔP is 120 psid.
  - Controlled Bleed Off (CBO) is 0.4 gpm.

Which one of the following describes the correct response?

The SRO will address a ___(1)___ seal failure and the CBO flow condition in accordance with ___(2)___ .

A.	(1) #1 (2) 3-GOP-103, Power Operation to Hot Standby
B.	(1) #2 (2) 3-GOP-103, Power Operation to Hot Standby
C.	(1) #1 (2) 3-ONOP-041.1, Reactor Coolant Pump Off-Normal
D.	(1) #2 (2) 3-ONOP-041.1, Reactor Coolant Pump Off-Normal

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible if candidate believes since #1 seal dP is high and CBO flow is lower than normal, therefore the #1 seal dP seal faces are abnormally tight. Part 2 is incorrect, but plausible because overall RCP operation guidance is provided in the GOP, but the required guidance to address the low CBO flow condition is found in the ONOP.		
B.	Incorrect. Part 1 is correct. Part 2 is incorrect.		
C.	Incorrect. Part 1 is incorrect. Part 2 is correct.		
D.	Correct. DP should be evenly distributed across each seal. #2 seal DP is low, indicating seal failure. CBO flow is also low. 3-ONOP-041.1 is the correct guidance.		
Technical Reference(s)	3-ARP-097.CR.A 1/4 3-ARP-097.CR.G 2/2 3-NOP-041.01A 3-ONOP-041.1, steps 34-36		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		5
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Comments:			
10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and choose the appropriate procedural response in event of an RCP seal failure. Systems knowledge is required for diagnosing the correct seal failure, but knowledge of the PTN procedure hierarchy is required to correctly answer the question			

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

6. **CHECK** DCS display DIFFERENTIAL SEAL STAGE PRESSURE indicating greater than:
   (1) 1192 psid (Hi) OR less than 298 psid (Low) for DP1 or indicating differential pressure outside 60% expected band.
   (2) 1117.5 psid (Hi) OR less than 372.5 psid (Low) for DP2 and DP3 or indicating differential pressure outside 50% expected band.
7. **CHECK** DCS display CONTROL BLEED OFF TEMPERATURE indicating greater than 195°F.
8. **CHECK** DCS display RCP SHAFT VIBRATION HI.

OPERATOR ACTIONS

1. **CHECK** associated CONTROL BLEED OFF CONTROL VALVE, CV-3-303A, B, C, OPEN
   A. IF Control Valve is CLOSED, THEN **OPEN** Control Valve
      (1) **MONITOR** Seal Parameters.
   B. IF Control Valve is CLOSED AND will **NOT** open, THEN **GO TO** 3-ONOP-041.1, REACTOR COOLANT PUMP OFF NORMAL
2. IF any of the following parameters - CBO flow, CBO temperature, Seal DP, P2, or P3 pressure - are changing OR have changed unexpectedly, THEN **REFER TO** 3-ONOP-041.1, Reactor Coolant Pump Off Normal.
3. IF Charging/Seal injection is lost, THEN **RESTORE** per 3-ONOP-047.1, Loss of Charging Flow in Modes 1 Through 4.

REFERENCES:
1. 5613-M-3041, Sheet 3, Reactor Coolant System – Reactor Coolant Pumps
2. EC 280399, Unit 3 RCP Seals Upgrade Project

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

ENCLOSURE 1 provides expected Controlled Bleed Off (CBO) flows for normal and off-normal conditions.

2	**Monitor CBO Flows**	
	a. Check RCS Pressure - GREATER THAN 2000 psig	a. Read NOTE prior to Step 4, and go to Step 4
	b. Check CBO flow - GREATER THAN 1.5 gpm	b. Go to Step 14

NOTE

A seal differential pressure of greater than 1490 psid is an indication that one seal stage has failed, and a second Seal Stage is degrading. RCP operation with only the #1 Seal or #2 Seal failed is acceptable.

3	**Monitor DCS To Determine RCP Seal Differential Pressures**	
	a. All RCPs – ALL stages less than 2000 psid	a. Go to Step 22
	b. All RCPs – ALL stages less than 1700 psid	b. Go to Step 24
	c. All RCPs – ALL stages less than 1490 psid	c. Go to Step 26
	d. Go to Step 5	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

17 **Check For Indications Of A Pressure Breakdown Device Blockage**

a. Check all of the following conditions exist on the same RCP

- RCP CBO flow – LESS THAN 1.5 gpm
- Any Seal Stage differential pressure – GREATER THAN 1300 psid
- RCDT level stable with no significant change in level trend

RESPONSE NOT OBTAINED: Notify Engineering of low CBO flow, and Continue with Step 19.

18 **Go To Step 24**

NOTE

If any RCP #3 Seal has failed, CBO flow will reduce and may go to zero, Seal Leak Off flow will increase to the RCDT, and due to the limited flow capacity of the Seal Leak Off line, some flow may go out the top of the Seal Cartridge to Containment atmosphere and to the Containment Sump. The RCDT and the Containment Sump should be monitored when indications of a #3 Seal failure has occurred.

19 **Check For Indications Of A #3 Seal Failure**

a. Check if ALL of the following conditions exist on <u>any</u> RCP

- RCP CBO flow LESS THAN 0.5 gpm
- RCP CBO isolation valve Open
- P3 pressure LESS THAN 100 psig
- P2 pressure GREATER THAN 1000 psig

RESPONSE NOT OBTAINED:

a. Notify SM and Engineering of indications of a degrading #3 Seal.

b. Monitor RCDT level and Containment Sump level for increased leakage

RESPONSE NOT OBTAINED:

b. Monitor the RCDT level and the Containment Sump level for potential increased leakage.

c. Return to Step 14.

4.1.1 Starting 3A Reactor Coolant Pump (continued)

NOTE

DCS indications should be verified in addition to local indications to avoid future additional containment entries to comply with 3-GOP-301, Hot Standby to Power Operations Prerequisites.

11. WHEN CCW flows have been adjusted to all three RCPs, THEN ensure the following CCW flows to RCPs are within their specified range.

A. FI-3-626, RCP Thermal Barrier Flow 63 - 84 gpm.

B. FI-3-677, RCP Bearing CCW Flow 465 - 510 gpm for Normal Operations OR 414 - 435 gpm with RHR in service.

12. IF RCS cold leg temperature is less than or equal to 275°F AND **NO** RCPs are RUNNING, THEN **CHECK** Steam Generator secondary water temperature less than 10°F above RCS temperature in 3A, 3B, and 3C Steam Generators using Section 5.5.

NOTE

RCS pressure range is 325 to 350 psig for solid plant condition.

13. **ENSURE** RCS pressure is greater than or equal to 325 psig

14. **ENSURE** 3A RCP CBO isolation valve CV-3-303A is OPEN

15. **CHECK** 3A RCP CBO flow, as indicated on FR-3-154A, within limits shown in Attachment 3, Control Bleed Off (CBO) Normal Operating Range.

16. On DCS, **CHECK** each seal stage dP is approximately 1/3 of the total seal dP.

17. On DCS, **DETERMINE** if dP across each seal stage is greater than 40.5 psid

A. IF any seal stage dP is less than or equal to 40.5 psid, THEN **CONSULT** Engineering prior to continuing.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	Topic and K/A #	026		AA2.03
	Importance Rating			2.9

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition

Proposed Question: SRO Question # 78

Given the following conditions:

- Unit 3 is at 100% power.
- ANN H8/6, CCW HEAD TANK HI/LO LEVEL alarms.
- ANN A4/6, VCT HI/LO LEVEL alarms.
- CCW Head Tank level is 0%.
- LI-3-115, VCT level is 81% and rising.
- Tavg is rising slowly.
- Control rods are automatically inserting.
- CCW Surge Tank level stabilizes with MOV-3-832, Component Cooling Water Surge Tank Makeup, fully opened.

Which one of the following completes the statement below?

The SRO will direct isolating the ___(1)___ in accordance with ___(2)___ .

A.	(1) Seal Water Return Heat Exchanger (2) 3-ONOP-030, Component Cooling Water Malfunction
B.	(1) Non-Regen Heat Exchanger (2) 3-ONOP-030, Component Cooling Water Malfunction
C.	(1) Seal Water Return Heat Exchanger (2) 3-ONOP-046.4, Malfunction of Boron Concentration Control System
D.	(1) Non-Regen Heat Exchanger (2) 3-ONOP-046.4, Malfunction of Boron Concentration Control System

Proposed Answer: A

A.	Correct. Seal Water Heat Exchanger is at a lower pressure (VCT pressure ~ 30 psig) than CCW (~128 psig) so CCW will leak into the heat exchanger and ultimately to the VCT, which would potentially dilute the RCS, causing rods to insert. 3-ONOP-030 must be entered to isolate the leaking HX.
B.	Incorrect. Part 1 is incorrect, but plausible if candidate believes the letdown side pressure through the non-regen HX is lower than CCW pressure. Part 2 is correct.
C.	Incorrect. Part 1 is incorrect, but plausible if candidate believes the letdown line is providing to much flow to the VCT and must be isolated. This is incorrect because CCW head tank level is lowering. Part 2 is plausible because it is an action taken if the candidate chooses this procedure, it is a legitimate action.
D.	Incorrect. Part 1 is incorrect, but plausible if the candidate believes the primary water makeup to the CVCS system must be isolated given VCT level and tavg are rising (signs of an uncontrolled dilution). Part 2 is plausible because it is an action taken if the candidate chooses this procedure, it is a legitimate action.

Technical Reference(s)	3-ARP-097.CR H8/6 & A4/6 3-ONOP-030 5613-M-3030 SH 2		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		5
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Comments:			

10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and choose the appropriate procedural response in event of a CCW leak. Knowledge of plant response is required to determine the affected component but the SRO must determine the correct AOP among 2 plausible choices based upon plant conditions

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

FOLDOUT PAGE
For Procedure 3-ONOP-030

TOTAL LOSS OF CCW FLOW

1) Manually **TRIP** the reactor.
2) **CONFIRM** reactor trip using the EOP network.
3) **STOP** all RCPs.
4) **ISOLATE** Letdown and Excess Letdown.
5) **ESTABLISH** one Charging Pump running at maximum speed, and **DISPATCH** operator to establish emergency cooling water to one of the remaining two Charging Pumps per Attachment 1.
6) **MONITOR** RCS pressure closely while running Charging Pump at maximum speed.
7) WHEN Attachment 1 is COMPLETE, THEN **OPERATE** Charging Pump supplied with emergency cooling to maintain RCP seal cooling.

LOSS OF CCW TO ANY COMPONENT

IF Component Cooling Water flow to any component cooled by CCW is lost, THEN **SHUT DOWN** the affected component.

CHARGING PUMP EMERGENCY COOLING CRITERIA

IF Cooling Water is **NOT** available to Charging Pumps, THEN **OPERATE** Charging Pump at maximum speed until cooling is restored from CCW System or per Attachment 1.

CCW PUMP STOPPING CRITERIA

IF any Component Cooling Water Pump is cavitating, THEN **STOP** the affected Component Cooling Water Pumps, and **PLACE** in PULL-TO-LOCK.

REACTOR TRIP CRITERIA

IF tripping a RCP is required, THEN manually **TRIP** the reactor prior to STOPPING the RCP.

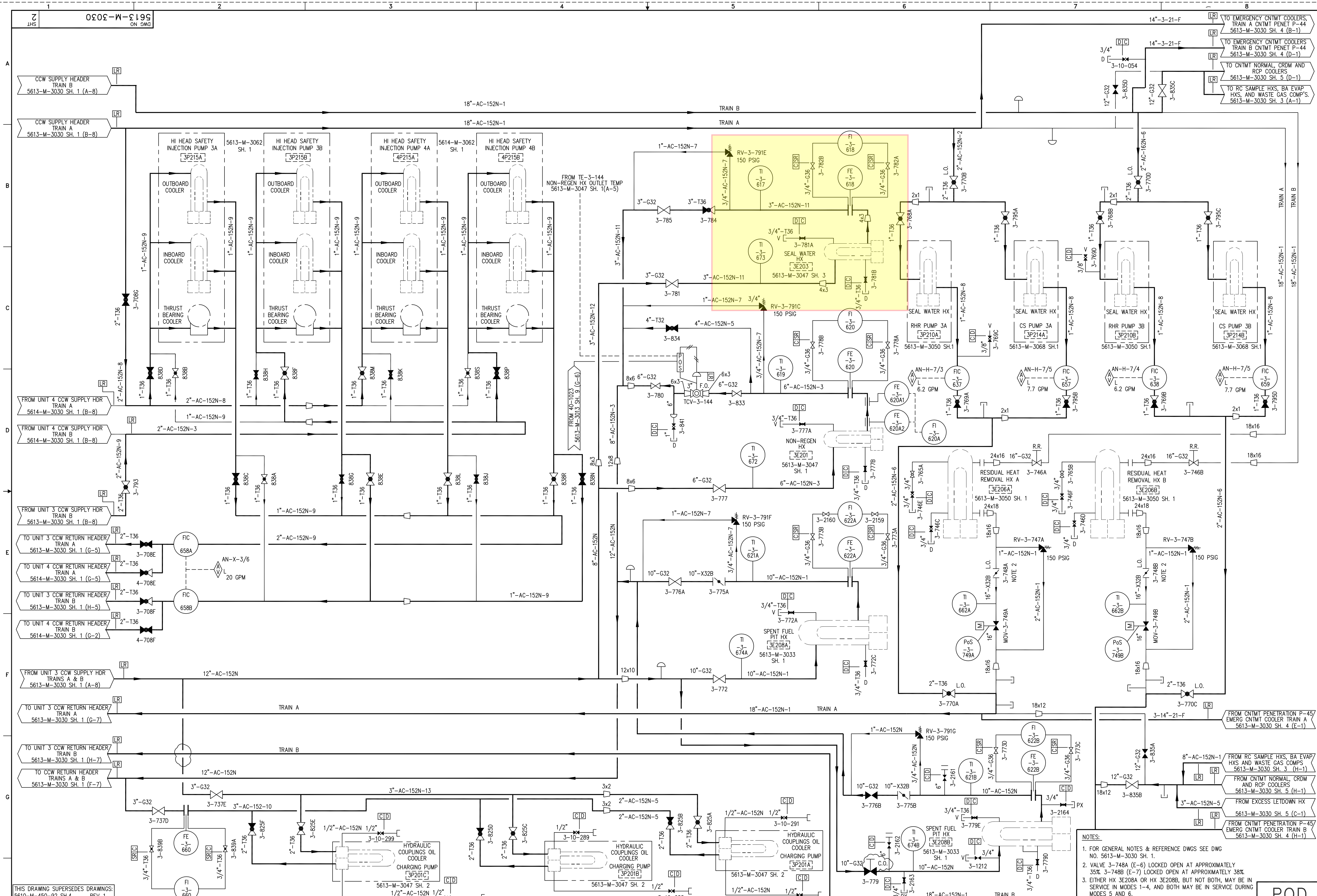
RCP STOPPING CRITERIA

IF any RCP bearing temperature annunciator alarm actuates AND its associated motor bearing temperature is greater than 195°F, THEN **TRIP** reactor and **STOP** the affected RCPs.

CCW PUMPS, HEAT EXCHANGERS, AND FLOWS/LOADS

CCW System operation once CCW System Header has been restored shall be within the operating restrictions of 3-NOP-030 summarized as follows:

- N-1 CCW Pumps (where N = number of CCW HXs aligned to CCW)
- All CCW HXs in service when RHR in service
- With only two CCW HXs in service AND both RHR HXs aligned to CCW, **PLACE** two CCW Pumps in PULL-TO-LOCK.
- Maximum five out of six CCW Heat Loads.



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5613-M-3030 SH. 1.
2. VALVE 3-748A (E-6) LOCKED OPEN AT APPROXIMATELY 35%. 3-748B (E-7) LOCKED OPEN AT APPROXIMATELY 38%.
3. EITHER HX 3E208A OR HX 3E208B, BUT NOT BOTH, MAY BE IN SERVICE IN MODES 1-4, AND BOTH MAY BE IN SERVICE DURING MODES 5 AND 6.

POD

THIS DRAWING SUPERSEDES DRAWINGS:

5610-M-450-92 SH.4	REV. 1
5610-M-450-92 SH.1	REV. 3
5610-M-450-57 SH.5	REV. 14
5610-M-450-57 SH.6	REV. 21
5610-M-450-92 SH.2	REV. 5
5610-M-450-57 SH.4	REV. 13
5610-M-450-92 SH.5	REV. 3
5610-M-450-92 SH.6	REV. 5
5610-M-450-57 SH.1	REV. 13
5610-M-450-57 SH.2	REV. 21
5610-M-372	REV. 3

SAFETY RELATED

NOTE: THIS DWG IS MADE FROM: FPL POD 5610-T-E-4512 SH.1 REV. 88

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
13	3-08-12	ISSUED AS-BUILT PER EC 246920 AND INCORP. CRN-180 (PARTIAL) .	RH	SB	–	JD	17	5-23-13	ISSUED AS-BUILT PER CRN-289 (EC 246920) ITOP 13-04-129.	RH	BB	–	RDP
12	1-23-12	ISSUED AS-BUILT PER EC 242212 (PC/M 06-018) & INCORP CRN M-12967.	RH	RV	–	EB	16	6-13-12	ISSUED AS-BUILT EC-DCR 276175.	RV	RH	–	AC
11	10-19-09	ISSUED AS-BUILT PER PC/M 03-049.	RV	RH	LG	SB	15	6-11-12	ISSUED AS-BUILT PER EC 247046.	RH	GR	–	AZ
10	4-17-09	ISSUED AS-BUILT PER PC/M 06-103.	RH	RV	ASD	PRB	14	3-16-12	ISSUED AS-BUILT PER EC 246920 (PC/M 08-171) AND INCORP. CRN-223.	RH	RV	–	JD
9	3-20-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061).	JPG	TEL	—	JRH	0	4-23-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	RB	MD	RAS SVM	LRB
8	12-17-93	ISSUED AS-BUILT FOR DCR-TPM-93-453.	SMM	MD	LIK	BD	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-136.	—	—	—	—

TURKEY POINT NUCLEAR 14

P & I D

COMPONENT COOLING WATER SYSTEM

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER	SYS
5613-M-3030	030
SHEET 2	REV 17

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	Topic and K/A #	062		2.4.2
	Importance Rating			4.6

Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: SRO Question # 79

Given the following conditions:

- Unit 3 is at 100% power.
- 3C ICW pump is OOS.
- 3A2 CWP is OOS.

Subsequently:

- Intake cooling water header pressure is 6 psig.
- 3A1 Travelling Screen DP is 12 inches of water.

Which one of the following completes the statement below?

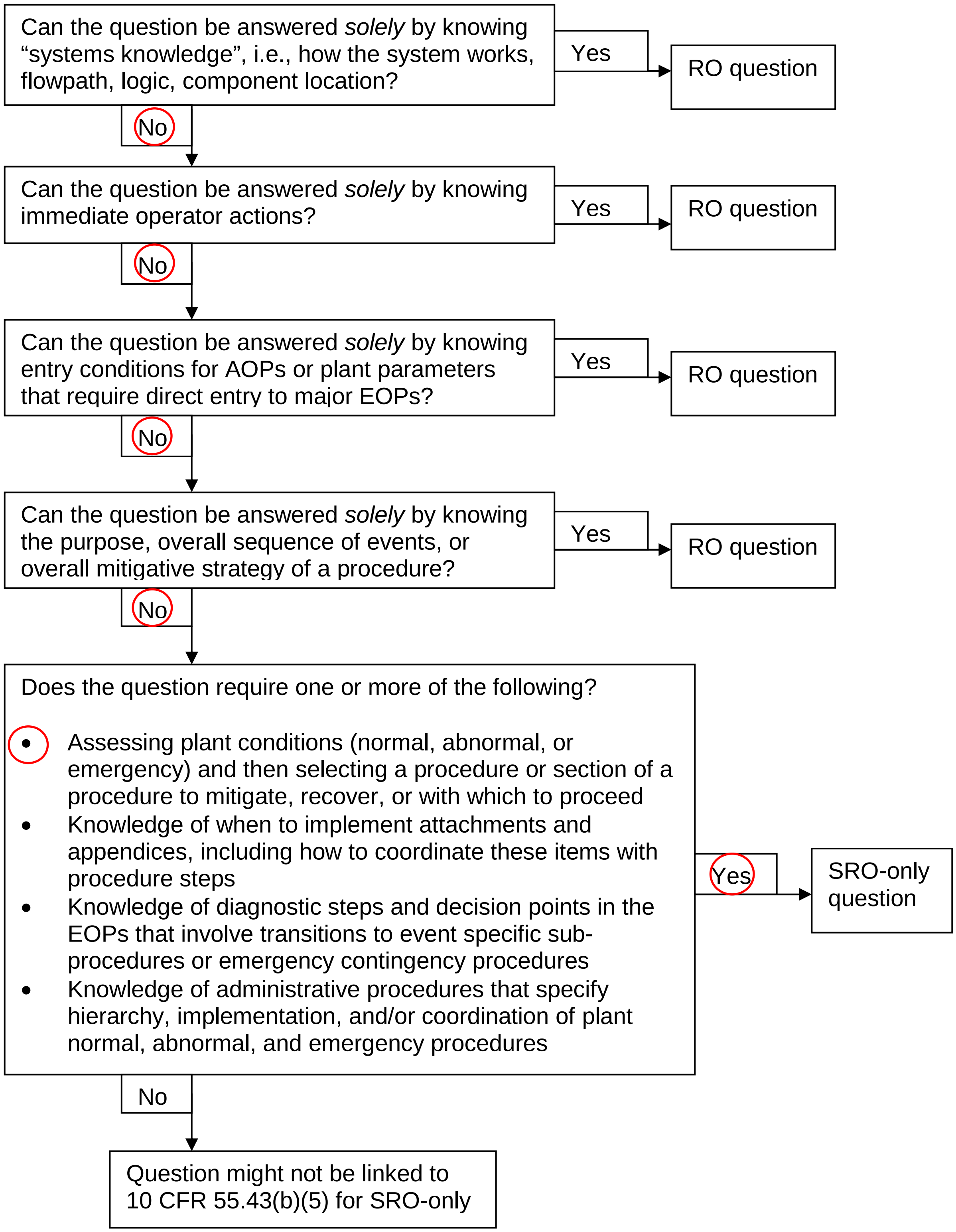
The SRO will direct the foldout page actions of ___(1)___ and will direct a reactor trip when ___(2)___ .

A.	(1) 3-ONOP-019, Intake Cooling Water Malfunction (2) TPCW Supply Header temperature is 110°F
B.	(1) 3-ONOP-019, Intake Cooling Water Malfunction (2) Turbine Bearing temperature is 185°F
C.	(1) 3-ONOP-011, Screen Wash System/Intake Malfunction (2) TPCW Supply Header temperature is 110°F
D.	(1) 3-ONOP-011, Screen Wash System/Intake Malfunction (2) Turbine Bearing temperature is 185°F

Proposed Answer: B

A.	Incorrect. Part 1 is correct . Part 2 is incorrect, but plausible if candidate confuses the required action with the ONOP requirement to reduce load to maintain TPCW supply header temperature less than 110°F. Candidate may also confuse with T.S. LCO: The ultimate heat sink shall be OPERABLE with an average supply water temperature less than or equal to 104°F.		
B.	Correct. ICW Pressure is not greater than 10 psig and trip criteria is met when bearing temperatures are not less than 180°F.		
C.	Incorrect. Part 1 incorrect but plausible because if candidate believes that the screen DP is high or above the limit causing blockage at the HX and thereby reducing the HX effectiveness. Part 2 is incorrect.		
D.	Incorrect. Part 1 is incorrect. Part 2 is correct.		
Technical Reference(s)	3-ONOP-019 3-ARP-097.CR.I 4/4 3-ARP-097.CR.I 3/3 3-ARP-097.CR.E 2/2		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	69002770702	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		5
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Comments:			
PTN Bank 69002770702 modified part 2 and modified stem. 10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and choose the appropriate procedural response in event of a failure in the ICW header. The SRO candidate must determine that entry conditions are met for one of 2 plausible procedures, and must also understand the decision points requiring entry to E-0			

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



FOLDOUT PAGE FOR 3-ONOP-019

1. **TRIP CRITERIA**

   - Component Cooling Water temperature as read on TI-3-607A and TI-3-607B cannot be maintained less than 120°F.
   - Turbine or Generator bearing temperatures cannot be maintained less than 180°F.

2. **MINIMUM FLOW REQUIREMENTS FOR CCW HXs**

   While isolating a CCW/ICW strainer, ICW flow less than minimum required through the CCW HXs can be tolerated without entry into Technical Specification Action 3.0.3, provided flow is restored to the minimum allowable, as determined by 3-NOP-019, Intake Cooling Water System, in less than 5 minutes by reopening the strainer isolation valves. If flow is below the minimum allowable value for greater than 5 minutes, then entry into Technical Specification Action 3.0.3 is started at the point where flow first fell below the minimum value. [Reference 3.1.4]

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

An operable intake cooling water header consists of an intact header being supplied by at least one intake cooling water pump.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Verify Adequate Intake Cooling Water Header Flow:** a. Check alarm I 4/4, ICW HEADER A/B LO PRESS - OFF b. Check Intake Cooling Water Header Pressure - GREATER THAN 10 PSIG • PI-3-1619 • PI-3-1620	Perform the following: 1. Dispatch operator to investigate for intake cooling water system leakage. 2. **<u>IF</u>** starting an available intake cooling water pump will **<u>NOT</u>** overload an EDG, **<u>THEN</u>** start available intake cooling water pump(s) as follows: a) **<u>IF</u>** offsite power is **<u>NOT</u>** available **<u>AND</u>** diesel generator load is greater than 2250 KW, **<u>THEN</u>** shed smaller loads until diesel generator load is less than 2250 KW. b) Start available intake cooling water pump(s). c) Restart any loads which were shed to allow intake cooling water pump start. 3. **<u>IF</u>** leakage is found, **<u>THEN</u>** perform the following: a) Isolate affected portion of intake cooling water system. b) Start intake cooling water pumps and align valves as necessary to establish at least one operable intake cooling water header. 4. **<u>IF</u>** leakage is **<u>NOT</u>** found **<u>AND</u>** headers are split, **<u>THEN</u>** tie headers together.
6	**Verify Intake Cooling Water Header Pressure - LESS THAN OR EQUAL TO 35 PSIG** • PI-3-1619 • PI-3-1620	Perform the following: a. Dispatch operator to investigate for intake cooling water system blockage. b. **<u>IF</u>** blockage is found, **<u>THEN</u>** align valves and start intake cooling water pumps as necessary to establish at least one operable intake cooling water header.

Item: 1.1.25.77.7.2

69002770702;

Question 79 original

Given the following conditions:

- Unit 3 is operating at 100% power
- The 3A and 3B ICW pumps are running
- Annunciator I-4/4 (ICW HEADER A/B LO PRESS) is actuated
- PI-3-1619 (A ICW header pressure indicator) reads 7 psig and is slowly decreasing
- PI-3-1620 (B ICW header pressure indicator) reads 14 psig and steady
- No other annunciators are currently actuated

Under these conditions, operators should enter ___(1)___ and, if the condition cannot be corrected, they must ___(2)___ .
(Reference provided)

A) (1) 3-ONOP-019 (Intake Cooling Water Malfunction)
(2) apply Technical Specification 3.0.3 and take action within one hour to be in hot standby within the next six hours

B) (1) 3-ONOP-019 (Intake Cooling water Malfunction)
(2) restore the system to operable status within 72 hours or be in hot standby within the next six hours

C) (1) 3-ONOP-011 (Screen Wash System/Intake Malfunction)
(2) apply Technical Specification 3.0.3 and take action within one hour to be in hot standby within the next six hours

D) (1) 3-ONOP-011S(creen Wash System/Intake Malfunction)
(2) restore the system to operable status within 72 hours or be in hot standby within the next six hours

CORRECT or INCORRECT feedback for item: 1.1.25.77.7.2

A. Incorrect since 3.0.3 does not apply. Loop B is at normal pressure, so, given a lack of other problems, it is still OPERABLE. With no "ICW PUMP TRIP" annunciators, 3.0.3 would not apply for # of ICW pumps. Plausible because the 1st part is correct. Also plausible because the low pressure annunciator is common to both loops. .

B. CORRECT. The combination of the "annunciator ICW HEADER A/B LO PRESS (I 4/4) and PI-3-1619, A HEADER ICW PRESS, is indicating 7 psig and decreasing slowly" meet the Symptoms of 3-ONOP-019. If not correct, this is reason to declare the loop INOPERABLE and apply TS 3.7.3, Action c.

C. Incorrect since 3-ONOP-011 is not the correct procedure, 3-ONOP-019 is. Since annunciator TRAVELING SCREEN HI DP (I 3/3) is not given, and since no local reports of abnormal screen DP were given, then 3-ONOP-011 does not apply. Plausible because 3-ONOP-011 is a potential precursor to 3-ONOP-019 and because it is mentioned in 3-ONOP-019, Step 1 CAUTIONS and Step 2:

Per 3-ONOP-019, Step 2 (Page 5):

D. Incorrect since 3-ONOP-011 is not the correct procedure, 3-ONOP-019 is. Since annunciator TRAVELING SCREEN HI DP (I 3/3) is not given, and since no local reports of abnormal screen DP were given, then 3-ONOP-011 does not apply. Plausible because 3-ONOP-011 is a potential precursor to 3-ONOP-019 and because it is mentioned in 3-ONOP-019, Step 1 CAUTIONS and Step 2. Also plausible because the 2nd part is correct.

Item Classification: Application
Item difficulty: 0.00
Keywords: 076 A2.02
Item weight: 1

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	Topic and K/A #	065		2.4.30
	Importance Rating			4.1

Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: SRO Question # 80

Given the following conditions:

- Unit 3 is at 100% power.
- A loss of instrument air (IA) is in progress.
- The crew is unable to restore IA pressure and initiates a manual reactor trip.
- RCS pressure is 2150 psig.
- The crew initiates a controlled cooldown to 470°F.
- During the cooldown, SI actuates on Main Steam Line High DP.
- The Shift Manager decides to declare an Unusual Event

Which one of the following describes the time the SRO is required to notify the NRCOC in accordance with LI-AA-102-1001, Regulatory Reporting?

A.	immediately
B.	15 minutes
C.	1 hour
D.	4 hours

Proposed Answer: C

A.	Incorrect. Plausible since some event notifications are required to be made immediately. Also, candidate may believe that the NRC resident must be notified immediately given the conditions. This event is resent plant OE.

B.	Incorrect. Plausible since some event notifications are required to be made in 15 minutes. State notifications are required within 15 minutes.		
C.	Correct. 1 hr notification is required		
D.	Incorrect. Plausible since some event notifications are required to be made in 4 hours. ECCS actuation requires a 4 hour notification.		
Technical Reference(s)	LI-AA-102-1001, Regulatory Reporting		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		1
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Comments:			
10CFR55.43(b) item 1 is met because the SRO must determine reporting requirements based upon a given event. Reporting requirements for abnormal and emergency events are a condition of the facility license			

ATTACHMENT 1
REPORTABLE EVENTS
(Page 1 of 8)

Declaration of an Emergency Class (See NUREG 1022 Section 3.1.1)	
1 Hour Report § 50.72(a)(1)(i) "The declaration of any of the Emergency Classes specified in the licensee's approved Emergency Plan."	
Plant Shutdown Required by Technical Specifications (See NUREG 1022 Section 3.2.1)	
4 Hour Report § 50.72(b)(2)(i) "The initiation of any nuclear plant shutdown required by the plant's Technical Specifications."	**60 Day LER § 50.73(a)(2)(i)(A)** "The completion of any nuclear plant shutdown required by the plant's Technical Specifications."
Operation or Condition Prohibited by Technical Specifications (See NUREG 1022 Section 3.2.2)	
	60 Day LER § 50.73(a)(2)(i)(B) "Any operation or condition which was prohibited by the plant's Technical Specifications except when: (1) The Technical Specification is administrative in nature; (2) The event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or (3) The Technical Specification was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event."

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	Topic and K/A #	E04		EA2.1
	Importance Rating			4.3

Ability to determine and interpret the following as they apply to the (LOCA Outside Containment) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: SRO Question # 81

Given the following initial conditions:

- Reactor trip and safety injection occurred on Unit 4.
- The crew enters 4-EOP-ECA-1.2, LOCA Outside Containment.

Subsequently:

- MOV-4-843A and MOV-4-843B, SI to Cold Leg Isolation valves, are closed to isolate the leak.
- 4A Charging Pump is running.
- RCS pressure is 1700 psig and rising slowly.
- PZR level is 10% and rising.
- SG levels are 16% and rising.
- CET subcooling is 34°F and stable.

Which one of the following identifies the required procedure transition and EOP strategy?

Transition to...

A.	4-EOP-E-1, Loss of Reactor or Secondary Coolant, and then to 4-EOP-ES-1.1, SI Termination.
B.	4-EOP-E-1, Loss of Reactor or Secondary Coolant, and then to 4-EOP-ES-1 .2, Post LOCA Cooldown and Depressurization
C.	4-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation, and then to 4-EOP-ES-1 .2, Post LOCA Cooldown and Depressurization

D.	4-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation, and then to 4-EOP-E-1, Loss of Reactor or Secondary Coolant
Proposed Answer: A	
A.	Correct. With RCS pressure rising the transition is to E-1. With the leak isolated by closing SI valves, the next logical procedure transition is to ES-1.1
B.	Incorrect. Procedure transition to E-1 is correct, and ES-1.2 is plausible because the current RCS pressure is not high enough for transition to ES-1.1 but since it is rising and the leak is isolated, the crew will not need ES-1.2
C.	Incorrect. Plausible because if the RCS pressure had been dropping after the valve closure, then the transition would be to ECA-1.1. ECA-1.2 only has two transitions, and ECA-1.1 is one of them. RCS pressure rising slowly precludes the need for ECA-1.1
D.	Incorrect, Plausible because if the RCS pressure had been dropping after the valve closure, then the transition would be to ECA-1.1. The most likely transition from ECA-1.1 would be to E-1

Technical Reference(s)	4-EOP-ECA-1 .2, LOCA Outside Containment E-1, Loss of Reactor or Secondary coolant		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			NO
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2011	Watts Bar
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Comments:
Changed context, conditions, and answer options to require a strategy beyond transition to one procedure 10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and choose the appropriate procedural response in event of a LOCA outside containment that has been isolated. Based upon plant conditions presented after action has been taken, the SRO must determine the appropriate procedure transition and based on action in that procedure, determine a subsequent procedure

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

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

20.	**Check Auxiliary Building Radiation – NORMAL**
	• Check Plant Vent Process Radiation Monitor, R-14
	• Check Auxiliary Building Area Radiation Monitors
	• Check Plant Vent SPING-4 Monitor
	• Direct RP to survey the following for abnormal radiation:
	  • Pipe & Valve Room
	  • Electrical Penetration Rooms
		Evaluate cause of abnormal conditions.
		
		IF the cause is a loss of RCS inventory outside Containment,
		THEN go to 3-EOP-ECA-1.2, LOCA OUTSIDE CONTAINMENT, Step 1.

21.	**Check PRT Conditions – NORMAL**	Evaluate cause of abnormal conditions.
22.	**Verify SI – RESET**	Reset SI.
23.	**Verify Containment Isolation Phase A AND Phase B – RESET**	Reset Phase A and Phase B.
24.	**Verify Instrument Air To Containment**	
	a. CV-3-2803, Instrument Air Containment Isolation – OPEN	**a.** Manually open valve.
	b. PI-3-1444, Instrument Air Pressure – GREATER THAN 95 PSIG	**b.** Restore Instrument Air pressure using 0-ONOP-013, LOSS OF INSTRUMENT AIR, while continuing this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**Check If Break Is Isolated:**	
	a. RCS pressure – INCREASING	**a.** Go to 3-EOP-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.
	b. Go to 3-EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1	

End of Section 3.0

FOLDOUT PAGE
For Procedure 3-EOP-E-1

1. ADVERSE CONTAINMENT CONDITIONS

a. IF either condition listed below occurs, THEN use [Adverse Containment Setpoints]:
- * Containment atmosphere temperature ≥ 180°F, OR
- * Containment radiation levels ≥ $1.3x10^5$ R/hr

b. WHEN Containment atmosphere temperature returns to less than 180°F, THEN Normal Setpoints can again be used.

c. WHEN Containment radiation levels return to less than $1.3x10^5$ R/hr, THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.

2. RCP TRIP CRITERIA

a. IF all conditions listed below occur, THEN trip all RCPs:
1) High head SI Pumps – AT LEAST ONE RUNNING AND SI FLOWPATH VERIFIED
2) RCS subcooling – LESS THAN 19°F[41°F]
3) Controlled RCS cooldown is **NOT** initiated

b. IF Phase B actuated, THEN trip all RCPs

3. SI TERMINATION CRITERIA

IF all conditions listed below occur, THEN go to 3-EOP-ES-1.1, SI TERMINATION, Step 1:

a. RCS subcooling based on Core Exit TCs – GREATER THAN 19°F[GREATER THAN ADVERSE VALUE in table below]

SI TERMINATION ADVERSE SUBCOOLING VALUE	
RCS PRESSURE (PSIG)	**ADVERSE SUBCOOLING VALUE**
< 2485 AND ≥ 2000	35 °F
< 2000 AND ≥ 1500	45 °F
< 1500 AND ≥ 1000	55 °F
< 1000 AND ≥ 500	110 °F
< 500	160 °F

b. Total feed flow to intact S/Gs – GREATER THAN 400 GPM, OR
Narrow Range Level in at least one intact S/G – GREATER THAN 7%[27%]

c. RCS pressure – GREATER THAN 1625 PSIG[1950 PSIG] AND STABLE OR INCREASING

d. PRZ level – GREATER THAN 7%[48%]

e. Charging capability – AVAILABLE

4. SECONDARY INTEGRITY CRITERIA

IF any S/G pressure is decreasing in an uncontrolled manner OR has completely depressurized, AND that S/G has **NOT** been isolated, THEN go to 3-EOP-E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.

5. E-3 TRANSITION CRITERIA

IF any S/G level increases in an uncontrolled manner OR any S/G has abnormal radiation,
THEN manually start SI Pumps and go to 3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE, Step 1

6. COLD LEG RECIRCULATION SWITCHOVER CRITERIA

IF RWST level decreases to less than 155,000 gallons, THEN go to 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1

7. RECIRCULATION SUMP BLOCKAGE

IF SI Pump flow AND SI Pump OR RHR Pump amps become erratic OR abnormally low after recirculation is established,
THEN transition to 3-EOP ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1

8. CST MAKEUP WATER CRITERIA

IF CST level decreases to less than 12%, THEN add makeup to CST using 3-NOP-018.01, CONDENSATE STORAGE TANK (CST)

9. LOSS OF OFFSITE POWER OR SI ON OTHER UNIT

IF SI has been reset AND subsequently either offsite power is lost OR SI actuates on the other unit,
THEN restore safeguards equipment, and at least one Computer Room Chiller to required configuration.
Refer to Attachment 1 for essential loads.

10. RHR SYSTEM OPERATION CRITERIA

IF RHR flow less than 1100 gpm, THEN RHR Pumps shall be shut down within 44 minutes of initial start signal.

11. LOSS OF CHARGING CRITERIA

IF Charging capability has been lost, AND High-Head SI Pumps are running at shutoff head,
THEN rotate High-Head SI Pumps as necessary to maintain continuous run time of any pump less than 30 minutes while maintaining at least one High-Head SI Pump running.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**Try To Identify And Isolate Break:**	
	a. Verify SI – RESET	**a.** Reset SI.
	b. Close RHR Discharge To Cold Leg Isolation valves: • MOV-4-744A • MOV-4-744B	
	c. RCS pressure – STABLE OR DECREASING	**c.** Go to Step 3.
	d. Open RHR Discharge To Cold Leg Isolation valves: • MOV-4-744A • MOV-4-744B	
	e. Close SI To Cold Leg Isolation valves: • MOV-4-843A • MOV-4-843B	**e.** Locally close valve(s).
	f. RCS pressure – STABLE OR DECREASING	**f.** Go to Step 3.
	g. Open SI To Cold Leg Isolation Valves: • MOV-4-843A • MOV-4-843B	
	h. Contact RP for survey of Auxiliary Building to determine source of high radiation	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**Check If Break Is Isolated:**	
	a. RCS pressure – INCREASING	**a.** Go to 4-EOP-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.
	b. Go to 4-EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1	

End of Section 3.0

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Question 81 original

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following:

- During performance of ECA-1 .2, 3LOCA Outside Containment,4 the crew determines RCS pressure is rising after RHR Train B cold leg injection valve 1-FCV-63-94 is closed.
- The crew then stops and locks out RHR pump 1 B-B and closes its suction valve.

Which ONE of the following identifies the required procedure transition?

A. ES-1.1, 3SI Termination4

B. E-1, 3Loss of Reactor or Secondary Coolant4

C. ECA-1 .1, 3Loss of Emergency Coolant Recirculation4

D. ES-i .2, 3Post LOCA Cooldown and Depressurization4

Proposed Answer: B

Explanation (Optional):

A. Incorrect, Plausible because ES-1. us a sub-procedure in the LOCA series of emergency procedures and would be a transition that could be required subsequent to the E-1 transition depending on the RCS pressure trend.

B. Correct, the RCS pressure rising indicates that the leak has been terminated and with the RCS pressure rising the transition to E-I is directed by the step in ECA-1.1.

C. Incorrect, Plausible because if the RCS pressure had been dropping after the valve closure, then the transition would be to ECA-1.1

D. Incorrect, Plausible because ES-1.2 is a sub-procedure in the LOCA series of emergency procedures and would be a transition that could be required subsequent to the E-1 transition depending on the RCS pressure trend.

Techical Reference(s): ECA-1 .2, LOCA Outside Containment, Revision 0005
WOG ECA-1 .2 Background, Revision 2
E-1, Loss of Reactor or Secondary coolant, Revision 0016
(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: 3-OT-ECA0101
08. Given a set of plant conditions, use procedures ECA-1.1 and 1.2 to identify any required procedure transition.
(As available)

Question Source: Bank 13677
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2011 Watts Bar

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

10 CFR Part 55 Content: 55.41
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			2
	Topic and K/A #	060		2.2.40
	Importance Rating			4.7

Equipment Control: Ability to apply technical specifications for a system.

Proposed Question: SRO Question # 82

Given the following conditions:

- Both Units are at 100% power.

On Unit 3:
- The crew is preparing to release a Gas Decay Tank.
- R-14, Plant Vent Radiation Monitor alarm setpoint cannot be adjusted properly and is declared inoperable.
- RAD-6304, Plant Vent SPING Radiation Monitor, is inoperable.

On Unit 4:
- Fuel shuffle is in progress in the Spent Fuel Pool.

Which one of the following completes the statements below?

Tech Spec LCO(s) ___(1)___ is / are applicable, and the most time restrictive action required is ___(2)___ .

REFERENCE PROVIDED

A.	(1) 3.3.3.1 (Unit 4) (2) initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours
B.	(1) 3.3.3.1 (Unit 4) (2) immediately suspend spent fuel manipulations in the Unit 4 Spent Fuel Pool
C.	(1) 3.3.3.1 (Unit 4) and 3.3.3.3 (Both units) (2) initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours

D.	(1) 3.3.3.1 (Unit 4) and 3.3.3.3 (Both units) (2) immediately suspend spent fuel manipulations in the Unit 4 Spent Fuel Pool
Proposed Answer: D	
A.	Incorrect. Part 1 is incorrect, but plausible because error can be made when candidate confuses the U4 SFP vent flowpath (U3 vents from its own SFP vent stack and U4 vents out of the common Plant Vent. Part 2 is incorrect but plausible when only 3.3.3.1 is considered.
B.	Incorrect. Part 1 is incorrect per discussion above. Part 2 is correct.
C.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible when candidate focuses soley on Unit 3 requirements, dismissing Unit 4.
D.	Correct. Both LCO actions must be entered and the most restrictive action is to immediately stop the fuel shuffle on Unit 4.

Technical Reference(s)	Tech Spec 3.3.3.1 and 3.3.3.3		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y- Tech Spec 3.3.3.1 and 3.3.3.3
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.		
Comments:		

10CFR55.43(b) item 2 is satisfied because the SRO must determine which LOCs are applicable and also determine the appropriate actions in accordance with TS/ODCM. Knowledge required is below the line

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)

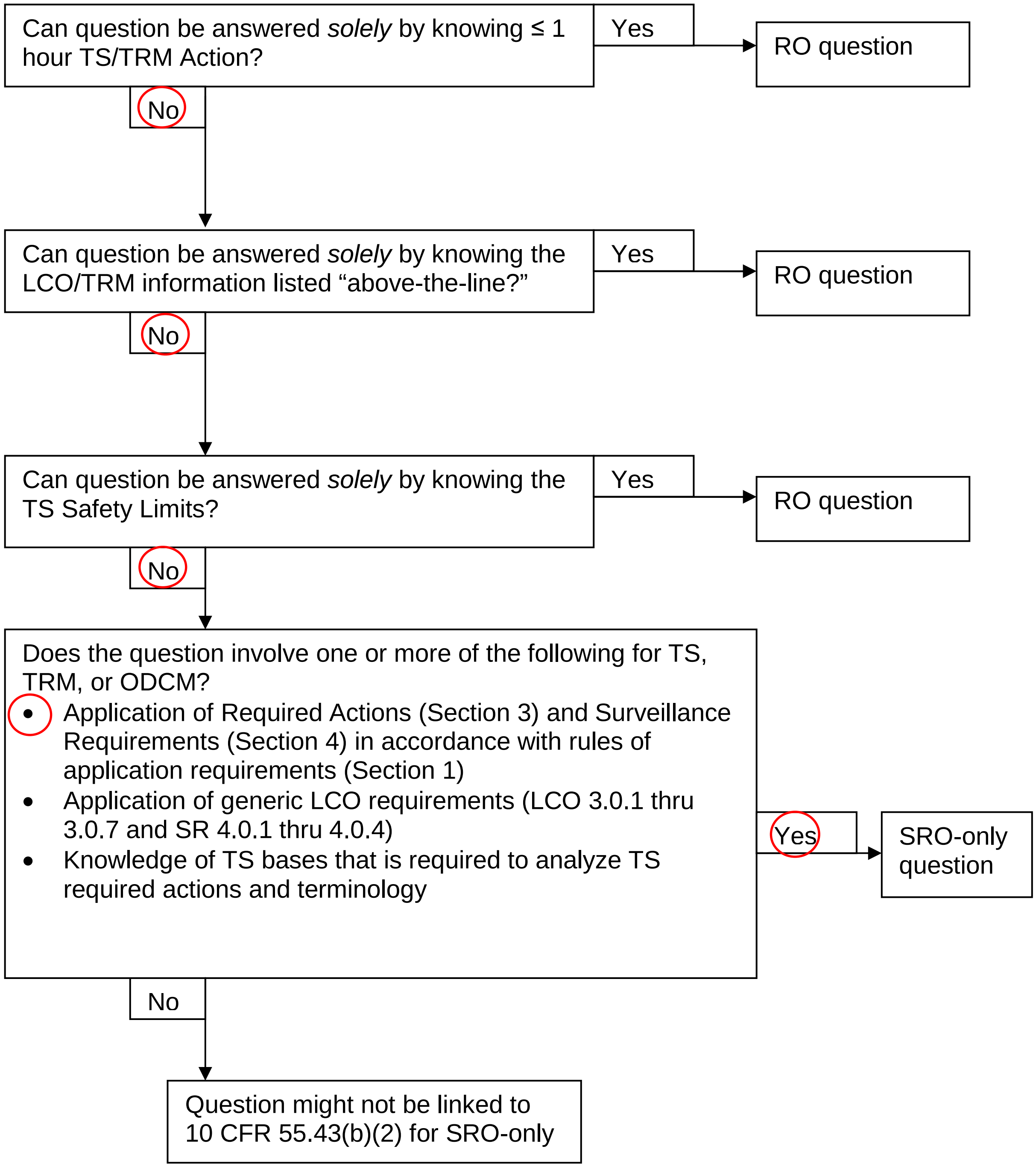


TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			2
	Topic and K/A #	061		AA2.06
	Importance Rating			4.1

Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Required actions if alarm channel is out of service

Proposed Question: SRO Question # 83

Given the following conditions:

- Unit 3 is in Mode 4 during a plant heatup.
- RI-3-6311A, Containment High Range Radiation Monitor fails.
- RI-3-6311B is unaffected.

Which one of the following identifies the action required in accordance with Technical Specifications?

Declare the channel inoperable and...

REFERENCE PROVIDED

A.	continue RCS heatup to Mode 3.
B.	remain in Mode 4 until it is restored to operable status.
C.	initiate the preplanned alternate method of monitoring the parameter, within 72 hours. Heatup to Mode 3 may continue.
D.	restore the inoperable channel(s) to operable status within 7 days, or make a special report to the NRC.

Proposed Answer: A

A.	Correct. One Channel is required by TS for accident monitoring
B.	Incorrect. There is no action entered for one inoperable channel so there is no reason to apply TS 3.0.4 for mode change with inoperable equipment

C.	Incorrect. Plausible because this would be the correct action for both channels out of service		
D.	Incorrect. Plausible because this is action taken for other accident monitoring system instrumentation such as gammametrics.		
Technical Reference(s)	TS table 3.3-5		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y- Tech Spec 3.3.3.1
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	2	
Facility operating limitations in the technical specifications and their bases.			
Comments:			
10CFR55.43(b) item 2 is satisfied because the SRO must determine the appropriate action for an inoperable radiation monitor as it applies to a planned mode change . The SRO must apply knowledge of the generic LCO requirements to determine the correct answer			

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

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High (Particulate or Gaseous (See Note 1.))	1	1*	All*	Particulate $\leq 6.1\times10^{5}$CPM Gaseous See Note 2.	26 for MODES 1, 2, 3, 4 or 27 for MODES 5 and 6
b. RCS Leakage Detection Particulate Radio-activity or Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	26
2. Spent Fuel Storage Pool Areas					
a. Unit 3 Radioactivity – High Gaseous	1	1	**	$<5.5\times10^{-2}\ \frac{\mu Ci}{cc}$	28
b. Unit 4 Radioactivity – High Gaseous#	1	1	**	$<2.8\times10^{-2}\ \frac{\mu Ci}{cc}$ (SPING) or $<1.0\times10^{6}$CPM (PRMS)	28

AMENDMENT NOS. 260 AND 255

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

* During CORE ALTERATIONS or movement of irradiated fuel within the containment comply with Specification 3/4.9.13.

** With irradiated fuel in the spent fuel pits.

\# Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint $= \frac{(3.2 \times 10^4)}{(\quad F \quad)}$ CPM,

$$\text{Where } F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment sump level monitoring system is OPERABLE,

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours,

3) A Reactor Coolant System water inventory balance is performed at least once per 8*** hours except when operating in shutdown cooling mode, and

4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.****

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).

*** Not required to be performed until 12 hours after establishment of steady state operation.

**** Instrument Air Bleed Valves may be opened intermittently under administrative controls.

TABLE 3.3-4 (Continued)

ACTION STATEMENTS (Continued)

ACTION 27 - In MODES 5 or 6 (except during CORE ALTERATION or movement of irradiated fuel within the containment): With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement perform the following:

1) Obtain and analyze appropriate grab samples at least once per 24 hours, and

2) Monitor containment atmosphere with area radiation monitors.

Otherwise, isolate all penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

During CORE ALTERATION or movement of irradiated fuel within the containment: With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements, comply with ACTION statement requirements of Specification 3.9.9 and 3.9.13.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, immediately suspend operations in the Spent Fuel Pool area involving spent fuel manipulations.

TABLE 4.3-3
RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere Radioactivity--High	SFCP	SFCP	SFCP	All
b. RCS Leakage Detection				
1) Particulate Radio-activity	SFCP	SFCP	SFCP	1, 2, 3, 4
2) Gaseous Radioactivity	SFCP	SFCP	SFCP	1, 2, 3, 4
2. Spent Fuel Pool Areas				
a. Unit 3 Radioactivity--High Gaseous	SFCP	SFCP	SFCP	*
b. Unit 4 (Plant Vent) Radioactivity--High Gaseous# (SPING and PRMS)	SFCP	SFCP	SFCP	*

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.

\# Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 16 detector thimbles when used for recalibration and check of the Excore Neutron Flux Detection System and monitoring the QUANDRANT POWER TILT RATIO*, and at least 38 detector thimbles when used for monitoring $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABLITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO*, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO*, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and $F_{xy}(Z)$.

---

* Exception to the 16 detector thimble requirement of monitoring the QUADRANT POWER TILT RATIO is acceptable when performing Specification 4.2.4.2 using two sets of four symmetric thimbles.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The accident monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

b. The provisions of Specification 3.0.4 are not applicable to ACTIONS in Table 3.3-5 that require a shutdown.

c. Separate Action entry is allowed for each Instrument.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
1. Containment Pressure (Wide Range)	2	1	1, 2, 3	31, 32
2. Containment Pressure (Narrow Range)	2	1	1, 2, 3	36
3. Reactor Coolant Outlet Temperature T_{HOT} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
4. Reactor Coolant Inlet Temperature T_{COLD} (Wide Range)	2-2 Detectors per Channel	1-2 Detectors per Channel	1, 2, 3	31, 32
5. Reactor Coolant Pressure – Wide Range	2	1	1, 2, 3	31, 32
6. Pressurizer Water Level	2	1	1, 2, 3	31, 32
7. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator	1, 2, 3	31, 32
8. Reactor Coolant System Subcooling Margin Monitor	2(2)	1(2)	1, 2, 3	31, 32
9. PORV Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	33
10. PORV Block Valve Position Indicator	1/valve	1/valve	1, 2, 3	33
11. Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve	1, 2, 3	32
12. Containment Water Level (Narrow Range)	2	1	1, 2, 3	36
13. Containment Water Level (Wide Range)	2	1	1, 2, 3	31, 32

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 31 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 33 Close the associated block valve and open its circuit breaker.

ACTION 34 With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) Either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 35 DELETED

ACTION 36 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 37 With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 30 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			2
	Topic and K/A #	069		AA2.01
	Importance Rating			4.3

Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Loss of containment integrity

Proposed Question: SRO Question # 84

Given the following conditions:

- Unit 3 is in Mode 1.
- A containment entry is in progress.
- The gasket on the inner air lock door is damaged.
- There is a gouge across the entire sealing surface approximately 1/8 inch deep and 1/2 inch wide.
- Air flow could be heard through the gouge when pressure is equalized across the inner door.
- When personnel are exiting the hatch, the outer door fails to close.
- No corrective maintenance has started.

Which one of the following identifies (1) the applicable Technical Specification LCO(s), and (2), the MOST restrictive action required?

REFERENCE PROVIDED

A.	(1) 3.6.1.3 ONLY (2) Restore the air lock door within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
B.	(1) 3.6.1.3 ONLY (2) Restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
C.	(1) 3.6.1.1 AND 3.6.1.3 (2) Restore containment integrity within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

D.	(1) 3.6.1.1 AND 3.6.1.3 (2) Restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.			
Proposed Answer: C				
A.	Incorrect. Plausible because 3.6.1.3 does apply but in the surveillance requirements for 3.6.1.1, verifying an operable air lock is required. Also, the action is correct except for substituting the word 'air lock' for the words 'containment integrity'			
B.	Incorrect. Plausible because this is the action statement for 3.6.1.3 item B for inoperable air lock			
C.	Correct. Both LCO Actions must be entered due to both hatch doors failing. A loss of personnel hatch (air lock) leads to loss of containment integrity requiring TS action 3.6.1.1.			
D.	Incorrect. This action would be required for an inoperable airlock under action B of TS 3.6.1.3 but it would not be the most restrictive action			

Technical Reference(s)	TS 3.6.1.1 TS3.6.1.3		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y- Tech Spec 3.6.1.1 & 3.6.1.3
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2
Facility operating limitations in the technical specifications and their bases.		

Comments:
10CFR55.43(b) item 2 is satisfied because the SRO must determine the appropriate action for a loss of containment integrity including determination of technical specification LCOs and appropriate action required that is greater than 1 hour

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

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 31 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 33 Close the associated block valve and open its circuit breaker.

ACTION 34 With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) Either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 35 DELETED

ACTION 36 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 37 With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 30 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT INTEGRITY shall be demonstrated:

a. In accordance with the Surveillance Frequency Control Program by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their closed positions;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

---

* Exception may be taken under Administrative Controls for opening of valves and airlocks necessary to perform surveillance, testing requirements and/or corrective maintenance. In addition, Specification 3.6.4 shall be complied with.

** Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the measured overall integrated containment leakage rate exceeding 1.0 L_a within one hour, initiate action to be in HOT STANDBY within the next 6 hours, and COLD SHUTDOWN within the following 30 hours. Restore the overall integrated leakage rate to less than 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the required test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, or during the performance of containment air lock surveillance and/or testing requirements, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

   1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

   2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

   3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Following each closing, at the frequency specified in the Containment Leakage Rate Testing Program, by verifying that the seals have not been damaged and have seated properly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

b. By conducting overall air lock leakage tests in accordance with the Containment Leakage Rate Testing Program.

c. In accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			1
	Group #			2
	Topic and K/A #	E03		2.4.20
	Importance Rating			4.3

Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes.

Proposed Question: SRO Question # 85

Given the following conditions:

- A LOCA has occurred on Unit 3.
- The SRO transitions to 3-EOP-E-1, Loss of Reactor or Secondary Coolant.
- RCS pressure is 900 psig.
- RWST level is 180,000 gallons.
- ALL Charging Pumps are tripped.
- Phase B isolation inadvertently actuated and has NOT been reset.
- AFW is operating as required.

Which one of the following identifies (1) the subsequent procedure entry required, and (2) the operation of the RCPs once transition is performed?

A.	(1) 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization (2) ONE RCP will be started for mixing of the RCS and to prevent void formation in the head
B.	(1) 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization (2) RCPs may NOT be started until a status evaluation has been performed.
C.	(1) 3-EOP-ES-1.3, Transfer to Cold Leg Recirculation (2) ONE RCP will be started for mixing of the RCS and to prevent void formation in the head
D.	(1) 3-EOP-ES-1.3, Transfer to Cold Leg Recirculation (2) RCPs may NOT be started until a status evaluation has been performed.

Proposed Answer:	B		
A.	Incorrect. Plausible because part 1 is correct and part 2 is normally correct except that with a loss of seal injection and phase B actuation, the RCPs may not be restarted without a status evaluation IAW 3-EOP-ES-1.2		
B.	Correct. 3-EOP-ES-1.2 will be entered based on given conditions and RCPs may NOT be started prior to evaluation.		
C.	Incorrect. Plausible because ES-1.3 is the transition for LOCAs where RWST level is below 155,000 gallons. Also plausible in part 2 same reason as Option A		
D.	Incorrect. Plausible for part 1 same reason as option C. Part 2 is correct.		
Technical Reference(s)	3-EOP-E-1 3-EOP-ES-1.2 caution prior to step 13		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		5
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Comments:			
10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and choose the appropriate procedural response in event of a LOCA, including strategy for operation of RCPs under abnormal conditions			

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

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17.	**Check If RCS Cooldown And Depressurization Is Required**	
	a. SI System – ALIGNED IN THE RWST INJECTION MODE	**a.** IF SI System has already been aligned for Cold OR Hot Leg Recirculation, THEN go to Step 19.
	b. RCS pressure – GREATER THAN 275 PSIG[575 PSIG]	**b.** Perform the following: **1)** IF RHR Pump flow greater than 1100 gpm, THEN go to Step 18. **2)** IF RHR Pump flow less than or equal to 1100 gpm, THEN go to 3-EOP-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1.
	c. RHR Pump flow – LESS THAN 1100 GPM	**c.** Go to Step 18.
	d. Go to 3-EOP-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1	
18.	**Check If Transfer To Cold Leg Recirculation Is Required**	
	a. SI System – ALIGNED IN THE RWST INJECTION MODE	**a.** IF SI System has already been aligned for Cold OR Hot Leg Recirculation, THEN go to Step 19.
	b. RWST level – LESS THAN 155,000 GALLONS	**b.** Return to Step 16.
	c. Go to 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1	

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	Topic and K/A #	004		2.1.30
	Importance Rating			4.0

Conduct of Operations: Ability to locate and operate components, including local controls.

Proposed Question: SRO Question # 86

Given the following conditions:

- Unit 3 is at 100% power when a loss of all charging flow occurs.
- 3-ONOP-047.1, Loss of Charging flow in Modes 1-4, is entered.
- PZR level is 55% and lowering.
- VCT level is 2% and lowering.
- LCV-3-115B, Emergency Make up to Charging Pumps, is closed and will not open.

Which one of the following completes the statements below?

Tech Spec 3.0.3 ___(1)___ applicable.

If a charging pump is subsequently restored, the Unit Supervisor will direct the crew to ___(2)___ restore charging pump suction in accordance with Attachment 2, Establish Charging Flow.

A.	(1) is (2) manually
B.	(1) is NOT (2) manually
C.	(1) is (2) locally
D	(1) is NOT (2) locally

Proposed Answer: C

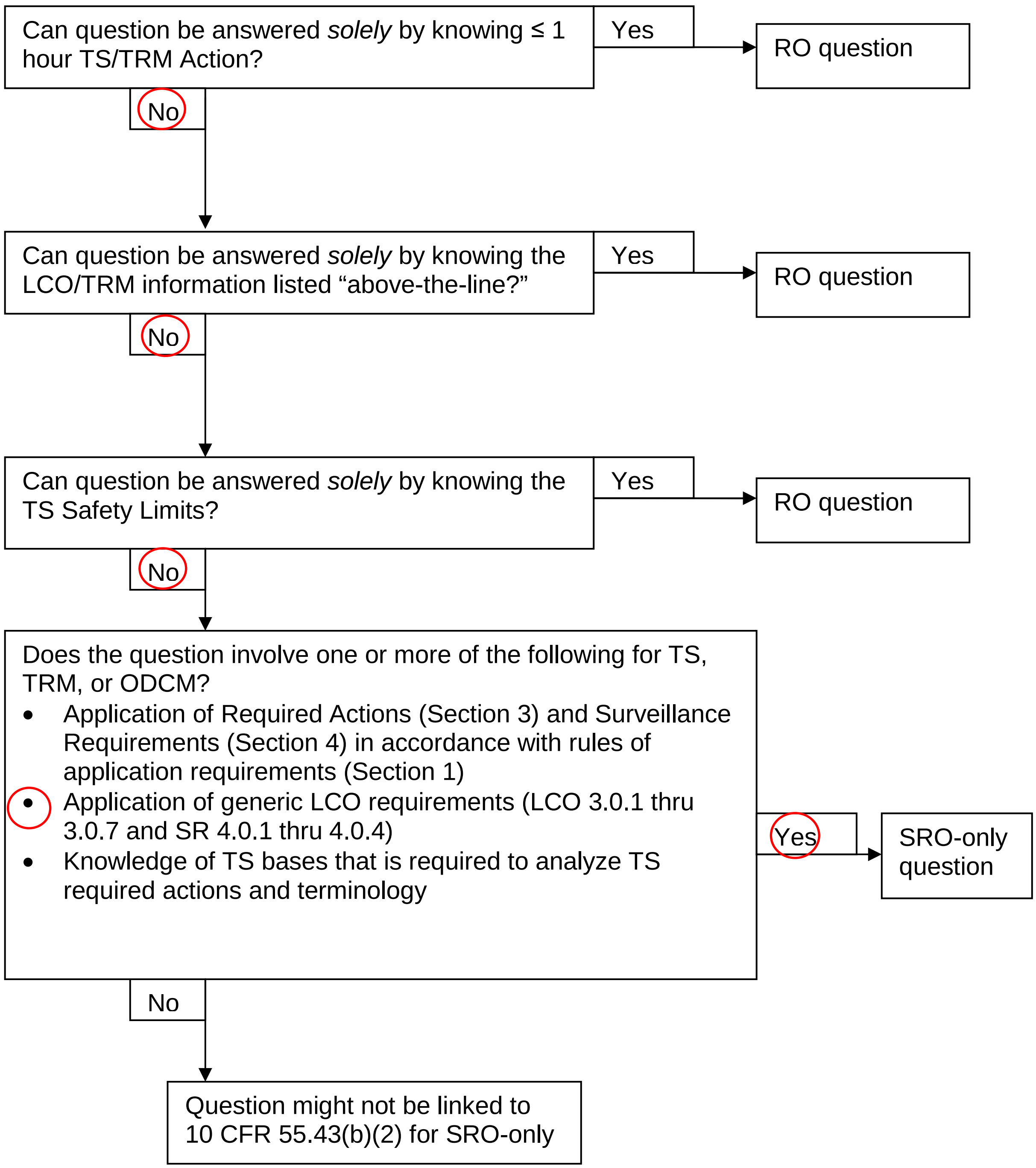
A.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible because the operators will initially try to align manually IAW Attachment 2.		
B.	Incorrect. Part 1 is incorrect, but plausible when candidate focuses on the fact that seal inject / charging flow to the RCPs can be lost for up to 24 hrs before RCP trip is required, therefore a rapid shutdown is NOT required. Also plausible since one charging pump can be inoperable with no LCO impact. Part 2 is incorrect, plausible per discussion above.		
C.	Correct. Part 1 is correct. Part 2 is correct. 3.0.3 for no charging pumps operable. Local action required since VCT level is <4% and LCV-3-115b suction from RWST can NOT be opened.		
D.	Incorrect. Plausible for same reasons as options A and B		

Technical Reference(s)	3-ONOP-047.1, Attachment 2 TS 3.1.2.1 TS 3.1.2.3 T.S. 3.0.3	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		N
Learning Objective:		(As available)

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

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	2
Facility operating limitations in the technical specifications and their bases.		
Comments:		
10CFR55.43(b) item 2 is satisfied because the SRO must determine the appropriate action for inoperable charging pumps where TS 3.0.3 must be applied, and procedure response relating to re-establishing charging flow		

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 2
Page 1 of 1

ESTABLISH CHARGING FLOW

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Verify VCT Level, LT-3-115, Greater Than 4% AND LCV-3-115C Open**	Perform the following: a. Verify LCV-3-115B Open. b. **IF** unable to open LCV-3-115B, **THEN** locally Open 3-358, RWST Emer Makeup to Chrg Pumps LCV-3-115B Bypass.
2.	**Adjust All Functional Charging Pumps Speed Control Hand-Auto Station Demand Meters To 20 To 25%**	
3.	**Open Charging Flow To Regen Hx, HCV-3-121**	
4.	**Open Loop A Charging Isolation, CV-3-310A**	
5.	**Start Functional Charging Pumps As Necessary To Restore Pressurizer Level**	
6.	**Adjust Charging Pump Speed Controllers To Restore Pressurizer Level To Program**	

ATTACHMENT 3
(Page 1 of 1)

NATURAL CIRCULATION INDICATIONS

The following conditions support or indicate natural circulation flow:

- RCS subcooling based on core exit TCs - GREATER THAN 19°F[73°F].
- S/G pressures - STABLE OR DECREASING
- RCS hot leg temperatures - STABLE OR DECREASING
- Core exit TCs - STABLE OR DECREASING
- RCS cold leg temperatures - WITHIN 30°F OF SATURATION TEMPERATURE FOR INTACT S/G PRESSURE

FINAL PAGE

FOLDOUT PAGE FOR 3-ONOP-047.1

1. ADVERSE CONTAINMENT CONDITIONS

IF either of the conditions listed below occur, **THEN** use [Adverse Containment Setpoints]:

* Containment atmosphere temperature greater than or equal to 180°F

OR

* Containment radiation levels greater than or equal to $1.3x10^5$ R/hr

WHEN Containment atmosphere temperature returns to less than 180°F,
THEN Normal Setpoints can again be used.

WHEN Containment radiation levels return to less than $1.3x10^5$ R/hr,
THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.

2. 3-EOP-E-0 TRANSITION CRITERIA

IF PZR level is 10% below program **OR** can **NOT** be maintained above 7%, **THEN** perform the following:

- Trip the Reactor and Turbine.
- Initiate Safety Injection **AND** Phase A Containment Isolation.
- Go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

3. 3-ONOP-041.7 TRANSITION CRITERIA

IF PRZ level can **NOT** be maintained above 7% with the plant in Mode 3 (less than 1000#), or Mode 4, **THEN** go to 3-ONOP-041.7 SHUTDOWN LOCA[Mode 3 (Less than 1000 psig) or Mode 4].

4. RESTORATION OF CHARGING

IF charging capability is restored any time during the performance of this procedure, **THEN** perform the following, if desired:

a. Reestablish letdown using ATTACHMENT 1.
b. Reestablish charging using ATTACHMENT 2.
c. Establish pressurizer level on program.
d. Go to plant procedure appropriate for plant conditions.

5. TECH SPEC MONITORING

Monitor Tech Spec 3.1.2.2 and 3.1.2.3 during the performance of this procedure.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 70 hours or be in at least HOT STANDBY and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F within 8 hours; restore at least two charging pumps to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REOUI REMENTS

4.1.2.3.1 The required charging pumps shall be demonstrated OPERABLE by testing pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	Topic and K/A #	013		A2.04
	Importance Rating			4.2

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; Loss of instrument bus

Proposed Question: SRO Question # 87

Given the following conditions:

- Unit 4 tripped from 100% power.
- 4B DC Bus voltage is zero.
- 4P08 Vital Instrument Bus Inverter has failed.

Which one of the following describes (1) the effect on Unit 4 and (2) the most restrictive Technical Specification (TS) actions required?

A.	(1) 4B ESF Load Sequencer and AFW actuation for bus stripping are lost (2) Comply with actions of TS 3.3.2, ESF System Actuation Instrumentation
B.	(1) 4B ESF Load Sequencer and AFW actuation for bus stripping are lost (2) Comply with actions of TS 3.0.3
C.	(1) Startup Transformers are inoperable due to loss of power to protection circuits (2) Comply with actions of TS 3.8.1.1, AC Sources
D.	(1) Startup Transformers are inoperable due to loss of power to protection circuits (2) Comply with actions of TS 3.0.3

Proposed Answer: B

A.	Incorrect. Plausible because the equipment affected by the loss of power is contained in TS 3.3.2, but with the current conditions, 3.0.3 will apply. Also plausible because of unit 3 to unit 4 differences. 3P06 and 3P07 failures put Unit 3 in 3.0.3 , 3P08 and 3P09 put unit 4 in 3.0.3.

B.	Correct. 4P08 will require compliance with 3.0.3 due to loss of AFW actuation signal from bus stripping IAW 4-ONOP-003.8.
C.	Incorrect. Plausible because the Unit 4 Startup Transformer is lost and because TS 3.8.1.1 does apply, but it is not the most restrictive TS action required.
D.	Incorrect. Part 1 plausibility as Option C and part 2 is correct

Technical Reference(s)	TS LCO 3.0.3 TS 3.3.2 TS 3.8.1.1 4-ONOP-003.8		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		2
Facility operating limitations in the technical specifications and their bases.			
Comments:			
10CFR55.43(b) item 2 is satisfied because the SRO must determine the appropriate action for loss of an instrument bus where TS 3.0.3 must be applied, specifically when other technical specification action statements also apply			

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

ENCLOSURE 1
(Page 1 of 3)

CONTROL ROOM FUNCTIONS AND INDICATIONS LOST ON FAILURE OF VITAL INSTRUMENT PANEL 4P08

FUNCTIONS, OPERATING

Loss of Auto control of C Feedwater Control Valve, FCV-4-498, shifts to Backup Controller
PRMS MONITORS due to loss of power
    Lose S/G Blowdown, causes Steam Generator level to increase
    Loss of power to R-11 and R-12, initiates Control Room **AND**
    Containment Ventilation Isolation
    Lose Liquid/Gas Release
Steam dump to condenser valves receive trip open signal but no Arming Signal.
Lose Auxiliary Feedwater Train 2 Controllers (3)
4C Charging Pump controller locks up as is
Disarms AMSAC due to loss of PT-446 (after six minute time delay)
Lose automatic operation of PORV PCV-4-456 (if OMS in normal OPS)
Loss of 4B Diesel Load Sequencer, 4C23B denergized
Loss of Train 2 AFW Valves, CV-4-2830, 2831, 2832 Fail closed.

NOTE

With Vital Panel 4P08 deenergized, 4B sequencer is out of service resulting in the following Tech Spec implications:

(1) AFW actuation signals from bus stripping on 4B 4KV bus will NOT be generated, placing the unit in Tech Spec 3.0.3. (Tech Spec 3.3.2, Table 3.3-2, Functional Unit 6.d)

(2) Loss of Power signals are lost via the 4B bus sequencer, placing the unit in Tech Spec 3.0.3 (Tech Spec 3.3.2, Table 3.3-2, Functional

(3) Bus stripping will NOT automatically occur, 4B EDG will NOT automatically close in on the bus and is out of service; actions of Tech Spec 3.8.1.1 apply.

INDICATORS

FI-4-110	Emerg Borate Flow
TI-4-116	VCT Temperature
PI-4-117	VCT Pressure
PI-4-156A	A RCP P2 Seal Pressure
TI-4-432B	C Loop OVPWR ΔT
TI-4-432A	C Loop ΔT
TI-4-432C	C Loop OVTEMP ΔT
TI-4-432D	C Loop Temp Avg
PI-4-457	Pzr Pressure

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-2 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-3.

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value within permissible calibration tolerance.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-3, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-3 and determine within 12 hours that the affected channel is OPERABLE; or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-2.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection					
a. Manual Initiation	2	1	2	1 2, 3, 4	17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1 2, 3, 4	14
c. Containment Pressure - High	3	2	2	1 2, 3	15
d. Pressurizer Pressure - Low	3	2	2	1 2, 3#	15
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1 2, 3#	15

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Steam Generator Pressure--Low	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2. 3*	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	25
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements. (Manual S.I. initiation will not initiate Phase A Isolation).				
b. Phase “B” Isolation					
1) Manual Initiation	2	2 (Both buttons must be pushed simultaneously to actuate)	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
3) Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15
c. Containment Ventilation Isolation					
1) Containment Isolation Manual Phase A or Manual Phase B	See Items 3.a.1 and 3.b.1 above for all Manual Containment Ventilation functions and requirements.				

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation (Continued)					
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	16
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions requirements.				
4) Containment Radioactivity-High	2##	1	1	1, 2, 3, 4	16
4. Steam Line Isolation					
a. Manual Initiation (individual)	1/operating steam line	1/operating steam line	1/operating steam line	1, 2, 3	21
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20
c. Containment Pressure--High-High	3	2	2	1, 2, 3	15
Coincident with: Containment Pressure--High	3	2	2	1, 2, 3	15

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with: Steam Generator Pressure--Low	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3	15
	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3	15
or T_{avg}--Low	1/Loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	25
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Steam Generator Water Level -- High-High# # # #	3/steam generator	2/steam generator in any operating steam generator	2/steam generator in any operating steam generator	1, 2, 3	15
6. Auxiliary Feedwater# # #					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

TABLE 3.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater### (Continued)					
b. Stm. Gen. Water Level-- Low-Low	3/steam generator	2/steam generator in any steam generator	2/steam generator	1, 2, 3	15
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Bus Stripping	1/bus	1/bus	1/bus	1, 2, 3	23
e. Trip of all Main Feed-water Pumps Breakers	1/breaker	(1/breaker) /operating pump	(1/breaker) /operating pump	1, 2	23
7. Loss of Power					
a. 4.16 kV Busses A and B (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4	18
b. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
Coincident with: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Loss of Power (Continued)					
c. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Degraded Voltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
8. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure	3	2	2	1, 2, 3	19
b. T_{avg} - Low	3	2	2	1, 2, 3	19
9. Control Room Ventilation Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 6**	16
b. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Containment Radioactivity--High	2	1	1	1, 2, 3, 4, 6**	16
d. Containment Isolation Manual Phase A or Manual Phase B	2	1	2	1, 2, 3, 4	17
e. Control Room Air Intake Radiation Level	2	1	2	All	24

TABLE 3.3-2 (Continued)

TABLE NOTATION

\# Trip function may be blocked in this MODE below the Pressurizer Pressure Interlock Setpoint of 2000 psig.

\# # Channels are for particulate radioactivity and for gaseous radioactivity.

\# # # Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

\# # # # Steam Generator overfill protection is not part of the Engineered Safety Features Actuation System (ESFAS), and is added to the Technical Specifications only in accordance with NRC Generic Letter 89-19.

* Trip function may be blocked in this MODE below the T_{avg}--Low Interlock Setpoint.

** Only during CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST or TRIP ACTUATING DEVICE OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 16 - With less than the Minimum Channels OPERABLE requirement, comply with the ACTION statement requirements of Specification 3.3.3.1 Item 1a of Table 3.3-4.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-2 (Continued)

TABLE NOTATION (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. Both channels of any one load center may be taken out of service for up to 8 hours in order to perform surveillance testing per Specification 4.3.2.1.

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 22 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 8 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, comply with Specification 3.0.3.

ACTION 24 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the control room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 25 - With number of OPERABLE channels one less than the Total number of channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the tripped condition within 6 hours. For subsequent required DIGITAL CHANNEL OPERATIONAL TESTS the inoperable channel may be placed in bypass status for up to 4 hours.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	Topic and K/A #	061		A2.04
	Importance Rating			3.8

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

Proposed Question: SRO Question # 88

Given the following conditions:

- Unit 3 has experienced a LOCA.
- RCS pressure is 1250 psig.
- Containment temperature is 187°F.
- SG narrow range levels are off-scale low.
- Tavg is 552°F
- All AFW pumps are tripped and will NOT restart.
- The crew transitions out of 3-EOP-E-0, Reactor Trip or Safety Injection.

Which one of the following describes the condition of the Unit and the action required?

A.	RCS bleed and feed requirements are met. Enter 3-EOP-FR-H.1, Loss of Secondary Heat Sink, trip RCPs and initiate Bleed and Feed.
B.	RCS bleed and feed requirements are NOT met. Enter 3-EOP-FR-H.1 and attempt to establish Standby Feedwater flow to at least ONE SG.
C.	RCS bleed and feed requirements are NOT met. Enter 3-EOP-FR-H.1 and attempt to establish Main Feedwater flow to at least ONE SG.
D.	Secondary Heat Sink is NOT required. Enter 3-EOP-E-1, Loss of Reactor or Secondary Coolant, and verify the event in progress.

Proposed Answer: A

A.	Correct. With adverse containment conditionspresent (6.5 psig will equate to greater than 180°F) the criteria for bleed and feed is <27% narrow range in SGs.

B.	Incorrect. Plausible because if a candidate does not see that adverse conditions are met, this is the first strategy in FR-H.1
C.	Incorrect. Plausible because if SSGFP cannot be started, this is the next strategy in FR-H.1
D.	Incorrect. Plausible because the event is a LOCA and the first step in E-1 is to check the RCS uncoupled from SGs by determining whether RCS pressure is >non-faulted SG pressures. With RCS at 1250 psig, it is higher than SG pressure, which will be approximately 1000 psig.

Technical Reference(s)	3-EOP-E-0 3-EOP-FR-H.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and choose the appropriate procedural response in event of a loss of secondary heat sink, including the applicable strategy for restoration of secondary heat sink

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

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	**Verify Proper AFW Flow**	
	a. Check Narrow Range Level in at least one S/G – GREATER THAN 7%[27%]	**a.** Verify AFW flow greater than 400 gpm. IF AFW flow less than 400 gpm, THEN manually start pumps and align valves to establish greater than 400 gpm flow. IF total feed flow from all sources greater than 400 gpm can **NOT** be established, THEN perform the following: **1)** Monitor Critical Safety Functions using 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES. **2)** Go to 3-EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, Step 1.
	b. Maintain feed flow to S/G until Narrow Range Levels between 21%[27%] and 50%	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE Foldout page is required to be monitored throughout this procedure.	
➔ **2.**	**Check If Bleed And Feed Is Required**	
	a. Two S/G Wide Range Levels – LESS THAN 10% [Narrow Range Level in all S/Gs – LESS THAN 27%]	**a.** Observe CAUTION prior to Step 3, and go to Step 3.
	b. Stop all RCPs	
	c. Observe CAUTION prior to Step 13, and go to Step 13	

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	Topic and K/A #	064		2.4.9
	Importance Rating			4.2

Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: SRO Question # 89

Given the following conditions:

Core offload is in progress on Unit 3.
3D 4kv bus is OOS on a clearance.
3B RHR pump is in service.
RCS temperature is 155°F.

Subsequently:

The switchyard is de-energized.
3A EDG is manually loaded onto the 3A 4kV bus after a 10 minute delay.
3B EDG fails to start.
Switchyard expected time of restoration is 2 hours.

Which one of the following completes the statement below?

The SRO will declare an __(1)__ in accordance with 0-EPIP-20101 and will reinitiate core cooling by __(2)__ .

REFERENCE PROVIDED

A.	(1) Unusual Event (2) loading an RHR pump on an EDG in accordance with 3-ONOP-004, Loss of Offsite Power
B.	(1) Unusual Event (2) establishing a secondary heat sink with a SSGFP in accordance 3-ONOP-050, Loss of RHR

C.	(1) Alert (2) loading an RHR pump on an EDG in accordance with 3-ONOP-004, Loss of Offsite Power
D.	(1) Alert (2) establishing a secondary heat sink with a SSGFP in accordance 3-ONOP-050, Loss of RHR

Proposed Answer: A

A.	Correct. IAW 3-ONOP-050 Step 1 NOTE: If loss of RHR is due to a loss of off-site power capability, power and RHR flow should be restored utilizing 3-ONOP-004, LOSS OF OFFSITE POWER
B.	Incorrect. Part 1 is correct Part 2 is incorrect, but plausible if candidate believes that since there was a delay in restoring power to an emergency bus and that RHR is lost during that time, ONOP-050 loss of RHR takes precedence and will provide immediate relief to cool the core by dumping steam from SGs. Step 13 of 3-ONOP-050 establishes a secondary heat sink by feeding with the SSGFP.
C.	Incorrect. Part 1 is incorrect, but plausible if candidate believes the ALERT criteria for loss of power is 10 minutes vice 15 minutes or if candidate believes that since switchyard power is lost for 2 hr, then an ALERT is warranted..
D.	Incorrect. Plausible for same reasons as options B and C

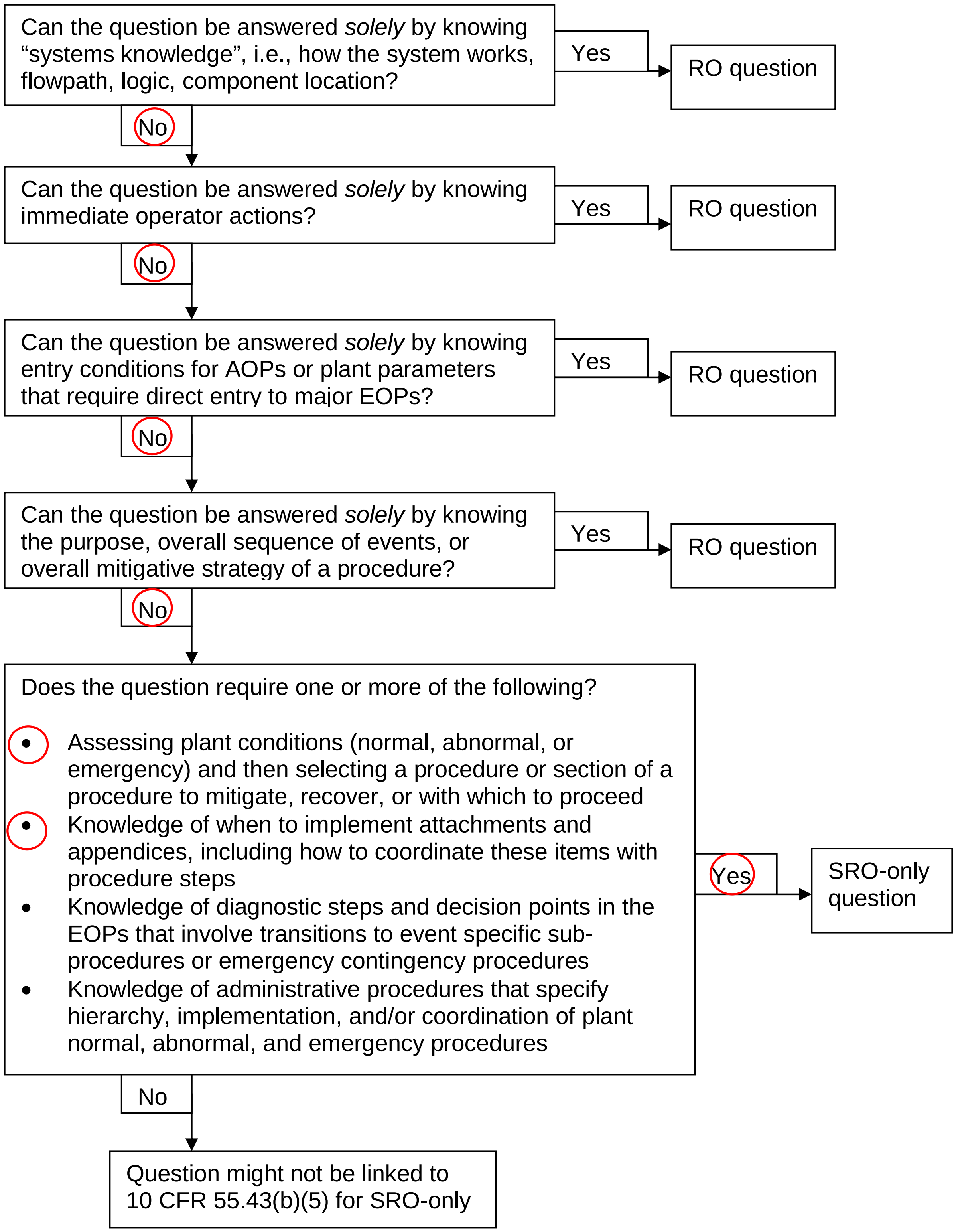
Technical Reference(s)	0-EPIP-20101 3-ONOP-050 3-ONOP-004	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		Y- 0-EPIP-20101 F668
Learning Objective:		(As available)

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

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

10 CFR Part 55 Content:	55.41	
	55.43	5
Facility operating limitations in the technical specifications and their bases.		
Comments:		
10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and choose the appropriate procedural response in event of a loss of off-site power while the unit is refueling. The SRO must determine the appropriate emergency classification as well as the appropriate recovery priority for recovery		

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C A U T I O N

If leakage from the RHR system is discovered, the leak should be isolated using 3-ONOP-041.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE.

NOTES

- *Oscillations in flow or motor amps may be indicative of RHR pump cavitation.*
- *If loss of RHR is due to a loss of off-site power capability, power and RHR flow should be restored utilizing one of the following:*
- *3-ONOP-004, LOSS OF OFFSITE POWER*

OR

  - *3-ONOP-004.10, LOSS OF OFFSITE POWER WHILE ON BACKFEED.*

OR

  - *3-ONOP-004.14, LOSS OF ALL AC POWER WHILE IN MODE 5, 6, OR DEFUELED, Attachment 17, Loss of All AC Recovery On Station Blackout Tie.*

OR

  - *3-ONOP-004.15, LOSS OF ALL AC POWER IN MODE 3 (LESS THAN 1000 PSIG) OR MODE 4, Attachment 6, Loss of All AC Recovery On Station Blackout Tie.*
- *During an Extended Loss of AC Power (ELAP), this procedure should be used for reference only.*
- *During a Loss of Power (excluding ELAP), this procedure should be used to establish containment closure and alternate cooling if RHR flow remains unavailable.*
- *The foldout page shall be monitored during the performance of this procedure.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check If RHR Pumps Should Be Stopped**	
	a. RCS level - GREATER THAN 10% PRESSURIZER COLD CAL	a. **<u>IF</u>** RCS Draindown Level Instrumentation is not available or RCS draindown level is LESS than 23%, **<u>THEN</u>** stop the running RHR pump **<u>AND</u>** go to 3-ONOP-041.8, Shutdown LOCA (Mode 5 or 6).
	b. RHR pumps - ANY RUNNING	b. Go to Step 2.
	c. RHR pumps - NOT CAVITATING • Amps Stable at normal value • Flow Stable at normal value	c. Stop RHR pumps.

General Emergency	Site Area Emergency	Alert	Unusual Event	Recognition Category	
SG1 Prolonged Loss of All Off-site and All On-Site AC Power to Emergency Busses. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: 1. a. Loss of all off-site and all on-site AC power to 3[4]A 4KV and 3[4]B 4KV Buses. **AND** b. **EITHER** of the following: • Restoration of at least one emergency bus in less than 4 hours is not likely. • CSF Status Tree for Core Cooling - Orange Conditions Met.	**SS1** Loss of All Off-site and All On-Site AC Power to Emergency Busses for 15 Minutes or Longer. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Loss of all Off-Site and all On-Site AC power to 3[4]A 4KV and 3[4]B 4KV Buses for 15 minutes or longer.	**SA5** AC Power Capability to Emergency Busses Reduced to a Single Power Source for 15 Minutes or Longer Such That Any Additional Single Failure Would Result in Unit Blackout. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. a. AC power capability to 3[4]A 4KV and 3[4]B 4KV Buses reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in unit blackout.	**SU1** Loss of All Off-Site AC Power to Emergency Busses for 15 Minutes or Longer. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Loss of all off-site AC power to 3[4]A 4KV and 3[4]B 4KV Buses for 15 minutes or longer.	**AC POWER**	**S – SYSTEM MALFUNCTIONS**
SG2 Automatic Trip and All Manual Actions Fail to Shutdown the Reactor AND Indication of an Extreme Challenge to the Ability to Cool the Core Exists. **Operating Mode Applicability: 1, 2** Threshold Value: 1. a. An automatic trip failed to shutdown the reactor. **AND** b. All manual actions or local Actions do not shutdown the reactor as indicated by CSF Status Tree for Subcriticality - Red Path Conditions Met **AND** c. **EITHER** of the following exist or have occurred due to continued power generation: • CSF Status Tree for Core Cooling - Red Path Conditions Met • CSF Status Tree for Heat Sink - Red Path Conditions Met	**SS2** Automatic Trip Fails to Shutdown the Reactor AND Manual Actions Taken in the Control Room are NOT Successful in Shutting Down the Reactor. **Operating Mode Applicability: 1, 2** Threshold Value: 1. a. An automatic trip failed to shutdown the reactor. **AND** b. Manual actions taken in the Control Room do not shutdown the reactor as indicated by CSF Status Tree Subcriticality - Red Path Conditions Met. **GUIDANCE BOX FOR SS2** Manual scram (trip) actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. (NEI 99-01, Rev. 5, SS2 Basis, Page 143)	**SA2** Automatic Trip Fails to Shutdown the Reactor AND the Manual Actions Taken in the Control Room are Successful in Shutting Down the Reactor **Operating Mode Applicability: 1, 2** Threshold Value: 1. a. An automatic trip failed to shutdown the reactor. **AND** b. Manual actions taken in the Control Room successfully shutdown the reactor as indicated by CSF Status Tree for Subcriticality – Red Conditions NOT Met. **GUIDANCE BOX FOR SA2** Manual scram (trip) actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor. (NEI 99-01, Rev. 5, SA2 Basis, Page 138)	**SU8** Inadvertent Criticality. **Operating Mode Applicability: 3, 4** Threshold Value: 1. UNPLANNED sustained positive startup rate observed on nuclear instrumentation.	**FAILURE OF R P S / CRITICALITY**	
	SS3 Loss of All Vital DC Power for 15 Minutes or Longer. **Operating Mode Applicability: 1, 2, 3, 4** Threshold Value: ***Note:*** *The Emergency Coordinator should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.* 1. Less than 105 VDC on all of the following Vital DC busses for 15 minutes or longer. • 3D01, 3D23 • 4D01, 4D23	**DEFINITION BOX** **IMMINENT –** Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where IMMINENT timeframes are specified, they shall apply. **UNPLANNED –** A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.		**DC POWER**	

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			1
	Topic and K/A #	073		A2.02
	Importance Rating			3.2

Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure

Proposed Question: SRO Question # 90

Given the following conditions:

- E-Waste Gas Decay Tank release is in progress.
- R-14, Plant Vent Gaseous Monitor, FAIL indicator light illuminates with high indication.
- Initial sample results for release of the tank were acceptable in accordance with 0-NCOP-004, Preparation of Gas Release Permits.

Which one of the following completes the statements below?

The release ___(1)___ terminated and ___(2)___.

REFERENCE PROVIDED

A.	(1) will be automatically (2) R-14 must be returned to operability prior to reinitiating the release
B.	(1) must be manually (2) R-14 must be returned to operability prior to reinitiating the release
C.	(1) will be automatically (2) compensatory actions are required to reinitiate the release with R-14 inoperable
D.	(1) must be manually (2) compensatory actions are required to reinitiate the release with R-14 inoperable

Proposed Answer: C

A.	Incorrect. Part 1 is correct. Part 2 is incorrect, plausible if the student assumes the first independent sample to start the discharge counts towards the 2 required to recommence the discharge. Also plausible because loss of some channels / non-conservative setpoints require suspension of releases through associated pathways.		
B.	Incorrect. Part 1 is incorrect, but plausible since some PRMS (e.g. SPINGs) are NOT interlocked with automatic isolation valves. Also plausible if candidate assumes channel failure has no impact on the system (e.g. since the release requires the lineup to be performed locally, no control room automatic actions will occur and the operator must be dispatched to stop the release). Part 2 is incorrect, but plausible per discussion above.		
C.	Correct. Failure of R-14 will automatically cause RCV-014 to close. The ODCM allows for continued release provided action 3.1.1 is performed.		
D.	Incorrect – Part 1 is incorrect, but plausible per discussion above. Part 2 is correct.		
Technical Reference(s)	ODCM Table 3.1-1 3-ONOP-067		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y - ODCM Table 3.1-1
Learning Objective:	6900150 EO12		(As available)
Question Source:	Bank	12870	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2011	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		4
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.			
Comments:			
10CFR55.43(b) item 4 is satisfied because the SRO must determine the appropriate action for a failed radiation detector that controls gaseous radioactive releases			

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1 Radioactive Gaseous Effluent Monitoring Instrumentation, Operability / Functionality and Alarm/Trip Setpoints, (Cont'd)

TABLE 3.1-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE / FUNCTIONAL	APPLICABILITY	ACTION
1. GAS DECAY TANK SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor)	1	*	3.1.1
b. Effluent System Flow Rate Measuring Device	1	*	3.1.2
2. Condenser Air Ejector Vent System			
a. Noble Gas Activity Monitor (SPING or PRMS)	1	#	3.1.3
b. Iodine Sampler	1	##	3.1.6
c. Particulate Sampler	1	##	3.1.6
d. Effluent System Flow Rate Measuring Device	1	##	3.1.2
e. Sampler Flow Rate Measuring Device	1	##	3.1.5

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability / Functionality and Alarm / Trip Setpoints, (Cont'd**)**

TABLE 3.1-1 (Cont'd)
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE / FUNCTIONAL	APPLICABILITY	ACTION
3. Plant Vent System (Include Unit 4's Spent Fuel Pool)			
a. Noble Gas Activity Monitor (SPING or PRMS)	1	*	3.1.3
b. Iodine Sampler	1	*	3.1.4
c. Particulate Sampler	1	*	3.1.4
d. Effluent System Flow Rate Measuring Device	1	*	3.1.2
e. Sampler Flow Rate Measuring Device	1	*	3.1.5
4. Unit 3 Spent Fuel Pit Building Vent			
a. Noble Gas Activity Monitor	1	*	3.1.3
b. Iodine Sampler	1	*	3.1.4
c. Particulate Sampler	1	*	3.1.4
d. Sampler Flow Rate Measuring Device	1	*	3.1.5

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability / Functionality and Alarm / Trip Setpoints (Cont'd)

TABLE 3.1-1 (Cont'd)
TABLE NOTATION

* At all times.
# Applies during Mode 1, 2, 3, and 4.
Applies during Mode 1, 2, 3, and 4 when primary to secondary leakage is detected.

ACTION 3.1.1 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, **and**

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 3.1.2 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 3.1.3 - With the number of channels OPERABLE / FUNCTIONAL less than required by the Minimum Channels OPERABLE / FUNCTIONAL requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours. These monitors may have Technical Specification requirements and action statements.

ACTION 3.1.4 With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, effluent releases via the affected pathway may continue provided continuous sample collection with auxiliary equipment as required by Table 3.2-1 is installed within 4 hours of the channel being declared non-functional, and analyzed at least weekly.

Facility: Turkey Point

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

The following conditions exist:

- A Waste Gas Decay Tank (WGT) E release was in progress.
- R-14, Plant Vent Gaseous Monitor, FAIL indicator light illuminates with indication pegged low.
- Initial sample results for release of the tank were acceptable in accordance with 0-NCOP-004, Preparation of Gas Release Permits

Which ONE of the following identifies (1) if RCV-14 will automatically close and (2) in accordance with the Offsite Dose Calculation Manual (ODCM), the MINIMUM required actions to recommence the release with this failure?

A. (1) Release will automatically terminate.
(2) The WGT E release may be recommenced ONLY after Chemistry performs ONE additional sample and ONE additional calculation.

B. (1) Release will automatically terminate.
(2) The WGT E release may be recommenced ONLY after Chemistry performs TWO independent samples and TWO independent calculations.

C. (1) Release will NOT automatically terminate.
(2) After the release has been locally terminated, the WGT E release may be recommenced ONLY after Chemistry performs ONE additional sample and ONE additional calculation.

D. (1) Release will NOT automatically terminate.
(2) After the release has been locally terminated, the WGT E release may be recommenced ONLY after Chemistry performs TWO independent samples and TWO independent calculations.

Proposed Answer: B

Explanation (Optional):

A. Incorrect due to the release will not automatically terminate if the rad monitor fails low. Plausible because the release will terminate automatically if the rad monitor fails high. One independent sample is plausible if the student assumes the first independent sample to start the discharge counts towards the 2 required to recommence the discharge.

B. Incorrect due to the release will not automatically terminate if the rad monitor fails low. Plausible because the release will terminate automatically if the rad monitor fails high. Also plausible because 2nd part is correct.

C. Incorrect - One independent sample is plausible if the student assumes the first independent sample to start the discharge counts towards the 2 required to recommence the discharge.
Also plausible because 1st part is correct.

D. Correct

Techical Reference(s): (Attach if not previoously provided)

Proposed Reference to be provided to applicants during examination:

Learning Objective: (As available)

Question Source: Bank

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam:

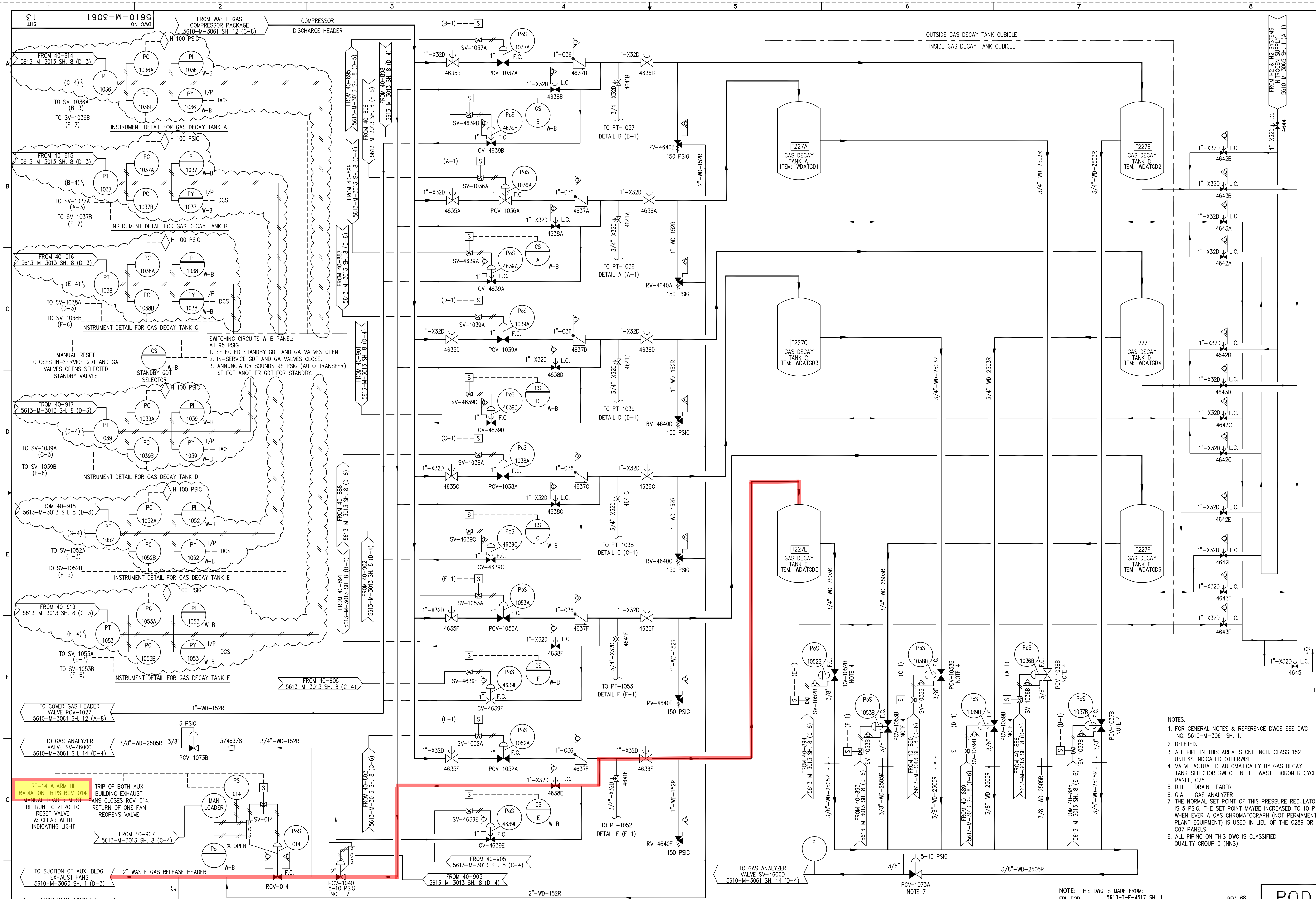
Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:



NOTES:
1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5610-M-3061 SH. 1.
2. DELETED.
3. ALL PIPE IN THIS AREA IS ONE INCH, CLASS 152 UNLESS INDICATED OTHERWISE.
4. VALVE ACTUATED AUTOMATICALLY BY GAS DECAY TANK SELECTOR SWITCH IN THE WASTE BORON RECYCLE PANEL, C25.
5. D.H. – DRAIN HEADER
6. G.A. – GAS ANALYZER
7. THE NORMAL SET POINT OF THIS PRESSURE REGULATOR IS 5 PSIG. THE SET POINT MAYBE INCREASED TO 10 PSIG WHEN EVER A GAS CHROMATOGRAPH (NOT PERMAMENT PLANT EQUIPMENT) IS USED IN LIEU OF THE C289 OR C07 PANELS.
8. ALL PIPING ON THIS DWG IS CLASSIFIED QUALITY GROUP D (NNS)

NOTE: THIS DWG IS MADE FROM: FPL POD 5610-T-E-4517 SH. 1 REV. 68

POD

REV	DATE	REVISION	BY	CH	APP	APP
8	2-6-13	ISSUED AS-BUILT PER EC 242033 (ITOP 12-12-20).	RH	RV	BB	BAP
7	4-5-00	ISSUED AS-BUILT PER CRN-I-3843 (PC/M 99-038).	RV	RH	BP	BAP
6	11-14-97	ISSUED AS-BUILT PER CRN-M-9135 (PC/M 97-036).	RH	RV	RSV	JJC
5	9-7-94	ISSUED AS-BUILT FOR PC/M 94-054 AND INCORPORATED CRN-I-3000.	RH	RV	HLM	BD
4	2-25-94	ISSUED AS-BUILT FOR DCR-TPM-93-512.	SMM	MD	WMC	JZ

REV	DATE	REVISION	BY	CH	APP	APP
3	04-01-93	ISSUED AS-BUILT FOR PC/M 93-002.	RKS	AB	LIK	JRH
2	03-05-93	ISSUED AS-BUILT FOR DCR-TPM-93-075.	RKS	INITIALS ON FILE		
9	5-29-14	ISSUED AS-BUILT PER EC-DCR 281736.	RH	BB	–	AD
0	05-26-92	THIS DWG REDRAWN FROM 5610-M-3061 SH. 4 IN ACCORDANCE WITH P&ID RECONSTITUTION PROJECT SCOPE AND ISSSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-092-031.	GFH	SAM	MD AM	LRB

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

P & I D

WASTE DISPOSAL SYSTEM
GAS
WASTE DECAY TANKS

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5610-M-3061

SHEET 13

SYS 061

REV 9

5610-M-3061_SH 13.DWG

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			2
	Topic and K/A #	001		A2.17
	Importance Rating			3.8

Ability to (a) predict the impacts of the following malfunction or operations on the CRDS- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rod-misalignment alarm

Proposed Question: SRO Question # 91

Given the following conditions:

- An unplanned load reduction is in progress on Unit 3.
- ANN B 9/3, SHUTDOWN ROD OFF TOP/DEVIATION, alarms and locks in.
- Control Bank D, rod H-12, is 20 steps above the remainder of control bank D.
- Control rod H-12 is mechanically bound at its current position.

Which one of the following completes the statements below?

The Rod Deviation Monitor ___(1)___ operable.

In accordance with Tech Specs, a comparison of the Demand Position Indication System and the Analog Rod Position Indication System must be performed at a minimum of once per ___(2)___ hours.

A.	(1) is NOT (2) 12
B.	(1) is NOT (2) 4
C.	(1) is (2) 12
D.	(1) is (2) 4

Proposed Answer: B

A.	Incorrect. Plausible because part 1 is correct. Also part 2 is required action for a misaligned rod that is not mechanically bound
B.	Correct. Rod Deviation Monitor is inoperable because the rod misalignment cannot be corrected so the alarm cannot be reset. For a mechanically bound control rod, shutdown is required.
C.	Incorrect. Plausible because the Rod Deviation Monitor will still measure rod positions, but with the alarm locked in, an additional alarm will not be received. Also because the TS actions are for a misaligned rod that is not mechanically bound.
D.	Incorrect. Plausible because the Rod Deviation Monitor will still measure rod positions, but with the alarm locked in, an additional alarm will not be received. Second part is correct.

Technical Reference(s)	TS 3.1.3.1 3-ARP-097.CR-B 9/3		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		2
Facility operating limitations in the technical specifications and their bases.			
Comments: 10CFR55.43(b) item 2 is satisfied because the SRO must determine the appropriate surveillance requirements for a misaligned control rod, greater than 1 hour action requirement			

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

CAUSES:

1. Any shutdown bank rod below 218 steps when control bank B above 35 steps
2. Difference in RPI between any two rods in the same bank greater than 12 steps or 24 steps while moving
3. RPI malfunction

B9/3

SHUTDOWN ROD OFF TOP/ DEVIATION

DEVICE:
Software

SETPOINT:
Any S/D rod below 218 steps and control bank B greater than 35 steps
OR
Deviation of 12 (24 moving) steps between any two rods in the same bank

LOCATION:
DCS

ALARM CONFIRMATION

1. **CHECK** RPI indications for rod deviations in any bank.

OPERATOR ACTIONS

1. IF rod is misaligned, THEN **REFER TO** 3-ONOP-028.1, RCC Misalignment.
2. IF alarm is locked in, THEN **NOTIFY** Unit Supervisor that the Rod Deviation Monitor should be declared inoperable.
3. **REFER** to Tech Spec 4.1.3.1.1, 4.1.3.2.1.

REFERENCES: Tech Spec Sections 3.1.3 and 3.10.5
PC/M 09-006

d. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand position by more than the Allowed Rod Misalignment of Specification 3.1.3.1, POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the Allowed Rod Misalignment of Specification 3.1.3.1, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the inoperable rod while maintaining the rod sequence and insertion limits of Specification 3.1.3.6; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

a) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the power range neutron flux high trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.d.3.c and 3.1.3.1.d.3.d below are demonstrated, and

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

c) A power distribution map is obtained from the movable incore detectors and F_Q (Z) and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the Allowed Rod Misalignment of the group step counter demand position in accordance with the Surveillance Frequency Control Program (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction in accordance with the Surveillance Frequency Control Program.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			2
	Topic and K/A #	045		A2.08
	Importance Rating			3.1

Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam dumps are not cycling properly at low load, or stick open at higher load (isolate and use atmospheric reliefs when necessary)

Proposed Question: SRO Question # 92

Given the following conditions:

- Unit 3 calorimetric power on DCS is 99.98% and stable on the hourly.

Subsequently:

- A steam dump to condenser valve fails open.
- The 5 minute calorimetric is 101% power.
- The hourly calorimetric is 100.05% power.
- The 8-hr calorimetric is 99.99% power.

Which one of the following identifies (1) the effect on the main generator and (2) the NRC Operations Center (NRCOC) notification requirement if applicable?

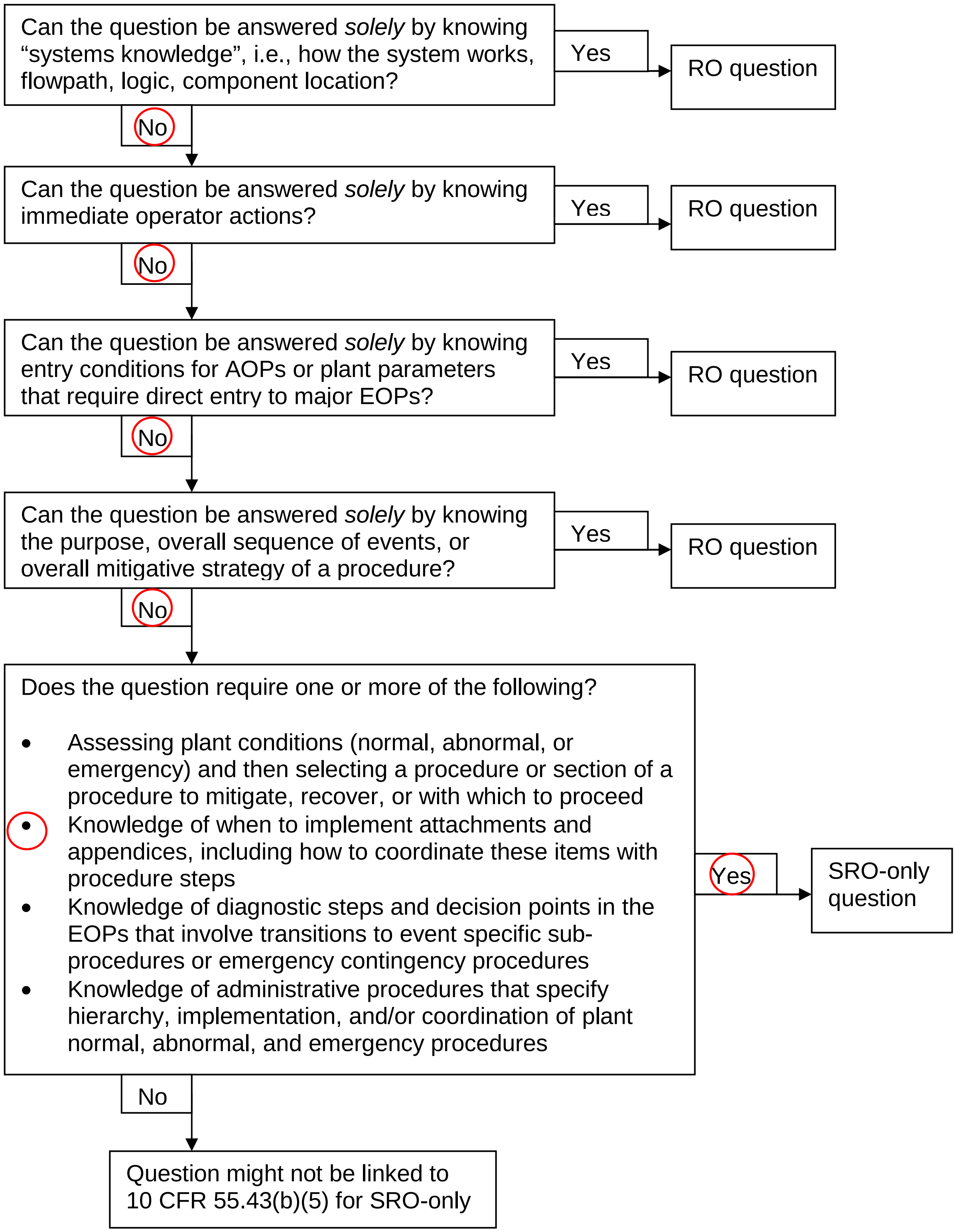
A.	(1) main generator load will not change (2) NRCOC must be notified
B.	(1) main generator load will lower (2) NRCOC must be notified
C.	(1) main generator load will not change (2) NRCOC is NOT required to be notified
D.	(1) main generator load will lower (2) NRCOC is NOT required to be notified

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible if candidate believes than since the net amount of steam coming from the SGs in unchanged (basically some of the steam is being diverted from the turbine to the condenser) therefore generator load remains unchanged. Part 2 is incorrect, but plausible if candidate believes NRCOC notification is required since the hourly calorimetric is >100 %. NRCOC notification is required when 8-hr calorimetric power exceeds rated thermal power.
B.	Incorrect. Part 1 is correct. Part 2 is incorrect.
C.	Incorrect. Part 1 is incorrect. Part 2 is correct.
D.	Correct, IAW 3-GOP-301 enclosure 4.

Technical Reference(s)	Enclosure 4, 3-GOP-301		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		5
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Comments:			
10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and determine whether the NRC must be notified based upon those conditions			

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



ENCLOSURE 4
(Page 1 of 2)

MAINTAINING REACTOR POWER BELOW 100 PERCENT TECH SPEC LIMIT

During full power operation, reactor power should be maintained below 100 percent using the following instructions based on 0-ADM-200, Operations Management Manual:

1. Reactor Power Shall be maintained as follows:

   a. Reactor power shall **NOT** be intentionally raised above 2644 MWth. For steady state full power operation, ensure the indicated Reactor Power is between 2640.0 and 2643.9 MWth (99.85 to 99.99%) on the DCS Calorimetric 1 Hour Average. This is to ensure that the 8 hour average power level does **NOT** exceed licensed power limit (LPL) of 2644 MWth. Maintaining the hourly calorimetric below 2643.9 MWth, ensures the 8 hour calorimetric average is below 2643.9 MWth (NRC requirement).

   b. Routine monitoring of alternate power indications shall be used as a tool to ensure the hourly indicated Reactor Power remains less than or equal to 99.99% and 2643.9 MWth. Alternate power indications to be monitored include, but are **NOT** limited to, RCS DeltaT, Tave- Tref, MWe, turbine inlet pressure, turbine valve position, circulating water temperature, feed flow, and condenser vacuum. [Section 6.2 Commitment 4 - CAPR]

      (1) If Alternate Power Indications exceed 99.99%, Operations and Engineering should evaluate the condition.

   c. The term “steady state” implies that temperatures, pressures, and flows are stable such that the nominal value of reactor power remains stable, subject to statistical uncertainties and normal fluctuations (e.g., feedwater oscillations for PWRs).

   d. Operating reactor compliance with the Licensed Power Limit (LPL) is demonstrated by the following process:

      (1) No actions are allowed that would intentionally raise core thermal power above the LPL for any period of time. Small, short-term fluctuations in power that are **NOT** under the direct control of a license reactor operator (secondary-side control valve oscillations for PWRs) are **NOT** considered intentional.

      (2) Closely monitor thermal power during steady state power operation with the goal of maintaining the one-hour thermal power average at or below the LPL. If the core thermal power average for a 1-hour period is found to exceed the LPL, take timely action to ensure that thermal power is less than or equal to LPL.

      (3) The core thermal power average for a shift is **NOT** to exceed the LPL. For the purpose of this guidance, a shift can be no longer than 8 hours.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			2
	Topic and K/A #	017		2.4.30
	Importance Rating			4.1

Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: SRO Question # 93

Given the following conditions:

- Unit 3 experiences a reactor trip and SI due to a SBLOCA.
- 3-EOP-FR-C.2, Response to Degraded Core Cooling, is entered.
- All CETs are 735°F and rising.
- RVLMS head indicates full.

Which one of the following identifies (1) the current Emergency Classification, and (2) the MAXIMUM amount of time to Notify the State of Florida?

REFERENCE PROVIDED

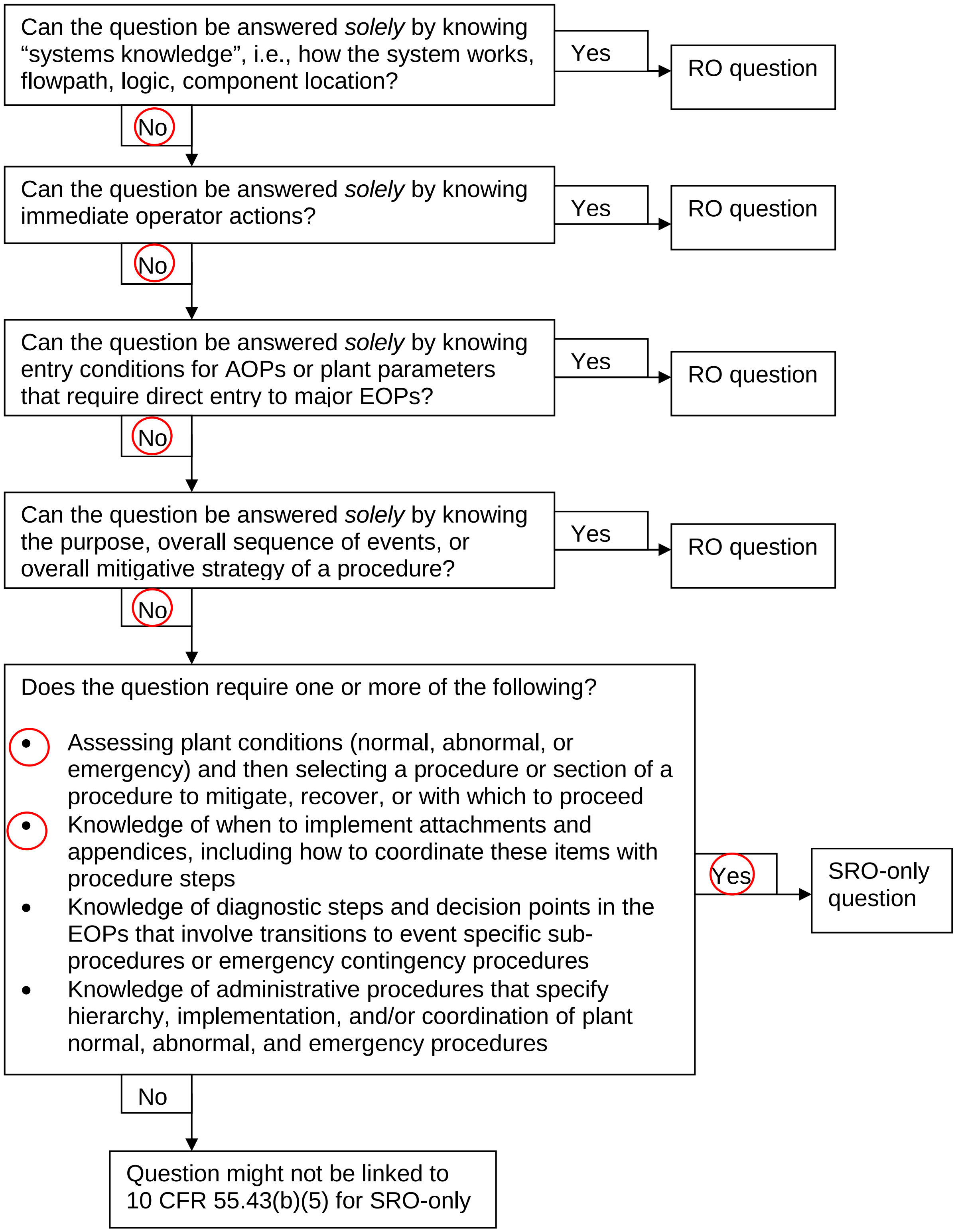
A.	(1) Site Area Emergency (2) 15 Minutes
B.	(1) Site Area Emergency (2) 60 minutes
C.	(1) General Emergency (2) 15 minutes
D.	(1) General Emergency (2) 60 minutes

Proposed Answer: A

A.	Incorrect. Plausible because CETs indicate 2 fission product barriers lost, which is a SAE. Candidate may overlook Containment Radiation, which is a loss of the containment barrier as well. Part 2 is correct

B.	Incorrect. Plausible for same reason as option A but part 2 is the maximum amount of time to notify the NRC		
C.	Correct, IAW 0-EPIP-20101 (F668- EAL classification table)		
D.	Incorrect. Part 1 correct but part 2 is time limit for NRC notification. State is required in 15 minutes		
Technical Reference(s)	0-EPIP-20101 FPB table and section 5.1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Yes- 0-EPIP-20101 F668
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		5
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Comments:			
10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and determine the appropriate emergency classification, as well as state notification requirements			

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



FISSION PRODUCT BARRIER TABLE WORKSHEET
(APPLICABILITY: Modes 1, 2, 3, & 4 ONLY)

FUEL CLAD BARRIER		REACTOR COOLANT SYSTEM BARRIER		PRIMARY CONTAINMENT BARRIER	
1. Critical Safety Function Status		**1. Critical Safety Function Status**		**1. Critical Safety Function Status**	
1. CSF Status Tree for Core Cooling Red Conditions Met. (3/4 FR-C.1 Required)	1. CSF Status Tree for Core Cooling Orange Conditions Met. (3/4 FR-C.2 Required) **OR** 2. CSF Status Tree for Heat Sink Red Conditions Met. (3/4 FR-H.1 Required)	**GUIDANCE BOX FOR SAFETY FUNCTION STATUS IN ALL THREE BARRIERS** **IF** directed to perform any mitigating step in applicable Function Restoration Procedures, **THEN** conditions have been met.	1. CSF Status Tree for Integrity Red Conditions Met. (3/4 FR-P.1 Required) **OR** 2. CSF Status Tree for Heat Sink Red Conditions Met. (3/4 FR-H.1 Required)	Not Applicable	1. CSF Status Tree for Containment Red Conditions Met. (3/4 FR-Z.1 Required)
OR		**OR**		**OR**	
2. Primary Coolant Activity Level		**2. RCS Leak Rate**		**2. Containment Pressure**	
1. Coolant activity greater than: 300 uCi/gm Dose Equivalent I-131. **GUIDANCE BOX** With Letdown in service, the R-3[4]-20 dose rate threshold of 2.5 R/hr is equivalent to the primary coolant activity level threshold of 300 μCi/gm DEQ I-131 in determining loss of the Fuel Clad Barrier. (Calculation PTN-9FJF-01-027, Rev. 0) See also SU4, Fuel Clad Degradation.	Not Applicable	1. RCS leak rate greater than available makeup capacity as indicated by a loss of RCS subcooling based on core exit TCs - LESS THAN 19°F[73°F]. **GUIDANCE BOX** See also SU5, RCS Leakage	1. RCS leak rate indicated by greater than maximum charging with Letdown isolated. **GUIDANCE BOX** Isolation of Letdown is to distinguish between RCS leakage and CVCS leakage and is performed when procedurally required.	1. A containment pressure rise followed by a rapid unexplained drop in containment pressure. **OR** 2. Containment pressure or sump level response not consistent with LOCA conditions.	1. Containment pressure greater than 55 psig and rising. **OR** 2. 4% H_2 in Containment **OR** 3.a. Containment Pressure greater than 20 psig. **AND** b. Less than one full train of depressurization equipment operating.
OR		**OR**		**OR**	
3. Core Exit Thermocouple Readings		**3. Not Applicable**		**3. Core Exit Thermocouple Reading**	
1. Core exit thermocouples reading greater than 1200ºF. **GUIDANCE BOX** At least five (5) Core Exit Thermocouples must exceed the threshold per F-0.	1. Core exit thermocouples reading greater than 700 ºF. **GUIDANCE BOX** At least five (5) Core Exit Thermocouples must exceed the threshold per F-0.	Not Applicable	Not Applicable	**GUIDANCE BOX** At least five (5) Core Exit Thermocouples must exceed the threshold per F-0. RVLMS Plenum indicating 0% indicates potential core uncovery.	1.a. Core exit thermocouples in excess of 1200 ºF. **AND** b. FR-C.1 **NOT** effective within 15 minutes. **OR** 2.a. Core exit thermocouples in excess of 700 ºF. **AND** b. RVLMS indicates head voids. **AND** c. FR-C.2 **NOT** effective within 15 minutes.
OR		**OR**		**OR**	
4. Reactor Vessel Water Level		**4. SG Tube Rupture**		**4. SG Secondary Side Release with P-to-S Leakage**	
Not Applicable	1. RVLMS (QSPDS) 0% Plenum Indicated. **GUIDANCE BOX** RVLMS Plenum indicating 0% indicates potential core uncovery.	1. RUPTURED SG results in a SI Actuation	Not Applicable	1. RUPTURED SG is also FAULTED outside of Containment. **OR** 2.a. Primary-to-Secondary leak rate greater than 10 gpm. **AND** b. UNISOLABLE steam release from affected SG to the environment.	Not Applicable
OR		**OR**		**OR**	
5. Not Applicable		**5. Not Applicable**		**5. CNTMT Isolation Failure or Bypass**	
Not Applicable	Not Applicable	Not Applicable	Not Applicable	1.a. Failure of all valves in any one line to close . **AND** b. Direct downstream pathway to the environment exists after containment isolation signal.	Not Applicable
OR		**OR**		**OR**	
6. Containment Radiation Monitoring		**6. Containment Radiation Monitoring**		**6. Containment Radiation Monitoring**	
1. CHRRM reading greater than 5.4 E+3 R/hr	Not Applicable	1. Containment Mezzanine radiation monitor RI-3-1401B [RI-4-1404B] reading greater than 10 mR/hr.	**GUIDANCE BOX** This threshold value indicates the release of reactor coolant to the containment assuming the instantaneous release and dispersal of the reactor coolant associated with **normal operating concentrations (i.e. within T.S. limits for RCS Activity).** This indication is not valid with a fuel clad barrier challenge. Radiation from adjacent piping and components containing elevated reactor coolant activity may cause radiation monitoring instrumentation inside containment to rise.	Not Applicable	1. CHRRM reading greater than 2.2 E+4 R/hr.
OR		**OR**		**OR**	
7. Emergency Coordinator Judgment		**7. Emergency Coordinator Judgment**		**7. Emergency Coordinator Judgment**	
1. Any condition in the opinion of the Emergency Coordinator that indicates Loss OR Potential Loss of the Fuel Clad Barrier.		1. Any condition in the opinion of the Emergency Coordinator that indicates Loss OR Potential Loss of the RCS Barrier.		1. Any condition in the opinion of the Emergency Coordinator that indicates Loss OR Potential Loss of the Containment Barrier.	
☐ **FUEL CLAD BARRIER**		☐ **REACTOR COOLANT SYSTEM BARRIER**		☐ **PRIMARY CONTAINMENT BARRIER**	
☐ **LOSS**	☐ **POTENTIAL LOSS**	☐ **LOSS**	☐ **POTENTIAL LOSS**	☐ **LOSS**	☐ **POTENTIAL LOSS**

ALERT
☐ **FA1** ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS

UNUSUAL EVENT
☐ **FU1** ANY Loss or ANY Potential Loss of Containment.

SITE AREA EMERGENCY
☐ **FS1** Loss or Potential Loss of ANY two Barriers

GENERAL EMERGENCY
☐ **FG1** Loss of ANY two Barriers **AND** Loss or Potential Loss of the Third Barrier

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			1
	Topic and K/A #	G1		2.1.4
	Importance Rating			3.8

Conduct of Operations: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, no-solo operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: SRO Question # 94

Given the following conditions:

- The date is 8/5/2016.
- Both Units are in Mode 1.
- Shift complement is at MINIMUM allowed by Technical Specifications.
- One of the ROs becomes ill and must be transported to the hospital.
- TWO potential replacements are identified for call-in.
- BOTH replacements have been assigned to OPS Support for the last year.
- The last times they were on shift are as follows:

Operator A	Operator B
12 hours on June 24 BOP	12 hours on June 21 RCO
12 hours on May 23 BOP	12 hours on June 20 RCO
12 hours on May 22 BOP	12 hours on June 19 RCO
12 hours on April 19 BOP	12 hours on June 18 RCO
12 hours on April 18 BOP	12 hours on May 5 WCC

Which one of the following completes the statements below?

In accordance with Technical Specifications, action must be taken to ensure the RO is replaced within a maximum of ___(1)___ .

Operator ___(2)___ will be selected as the replacement.

REFERENCE PROVIDED

A.	(1) 2 hours (2) A
B.	(1) 2 hours (2) B

C.	(1) 1 hour (2) A
D.	(1) 1 hour (2) B

Proposed Answer: A

A.	Correct. Operator A has an active license. Operator B has 4 12 hour shifts in the previous quarter, so his license has gone inactive. 5 12 hour shifts per quarter are required. TS table 6.2-1 requires action to be initiated immediately and a replacement to be in place in less than 2 hours.
B.	Incorrect. Operator B is inactive. Plausible because Operator B has stood the 4 most recent shifts, and also because to reactivate a license, 40 hours under instruction is required. If the applicant confuses the time requirements, Operator B will possible be chosen.
C.	Incorrect. Incorrect but plausible time provided in this Distractor.
D.	Incorrect. Same as C, plausible for reasons given in B and C.

Technical Reference(s)	TS Table 6.2-1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y - TS Table 6.2-1
Learning Objective:			(As available)
Question Source:	Bank	11824	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2008	North Anna

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2

Conditions and limitations in the facility license

Comments:
10CFR55.43(b) item 2 is satisfied because the SRO must determine the action required when minimum crew composition is not met. Also 10CFR55.43(b) item 1 because minimum crew composition is a condition of the facility license

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

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
NPS	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

NPS - Nuclear Plant Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Nuclear Plant Supervisor from the control room while a unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Nuclear Plant Supervisor from the control room while both units are in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position may be filled by the Nuclear Plant Supervisor or an individual with a Senior Operator license who meets the 1985 NRC Policy Statement on Engineering Expertise on Shift.

Facility: Turkey Point

Vendor WEC

Exam Date:

Exam Type:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following:
• The date is 6/5/2008.
• Both Units are in Mode 1.
• Shift complement is at MINIMUM allowed by Technical Specifications.
• One of the Reactor Operators becomes ill and must be transported to the hospital.
• TWO potential replacements are identified for call-in.
• BOTH replacements have been assigned to OPS Support for the last year.
The last times they were on shift are as follows:

Operator A	Operator B
• 12 hours on March 24 BOP	12 hours on March 21 RO
• 12 hours on February 23 BOP	12 hours on March 20 RO
• 12 hours on February 22 BOP	12 hours on March 19 RO
• 12 hours on January 19 BOP	12 hours on March 18 RO
• 12 hours on January 18 BOP	12 hours on January 5 WCC

Which ONE of the following identifies the Technical Specification requirement to replace the Reactor Operator, and which operator will be selected as the replacement?

A. Action must be taken to ensure the Reactor Operator is replaced within 2 hours; Operator A will be selected.

B. Action must be taken to ensure the Reactor Operator is replaced within 2 hours; Operator B will be selected.

C. Action must be taken to ensure the Reactor Operator is replaced within 1 hour; Operator A will be selected.

D. Action must be taken to ensure the Reactor Operator is replaced within 1 hour; Operator B will be selected.

Proposed Answer: A

Explanation (Optional):

A. Correct. Operator A has an active license. Operator B has 4 12 hour shifts in the previous quarter, so his license has gone inactive. 5 12 hour shifts per quarter are required. TS 5.2.2 requires action to be initiated immediately and a replacement to be in place in less than 2 hours.

B. Incorrect. Operator B is inactive. Plausible because Operator B has stood the 4 most recent shifts, and also because to reactivate a license, 40 hours under instruction is required. If the applicant confuses the time requirements, Operator B will possible be chosen.

C. Incorrect. Incorrect but plausible time provided in this Distractor.

D. Incorrect. Same as C, plausible for reasons given in B and C.

Techical Reference(s): (Attach if not previoously provided)

Proposed Reference to be provided to applicants during examination:

Learning Objective: (As available)

Question Source: Bank

Modified Bank (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			2
	Topic and K/A #	G2		2.2.19
	Importance Rating			3.4

Equipment Control: Knowledge of maintenance work order requirements.

Proposed Question: SRO Question # 95

Which one of the following completes the statements below?

In accordance with 0-ADM-701, Control Of Plant Work Activities, a(n) ___(1)___ shall be designated as a liason between the Shift Manager and the the troubleshooting team for troubleshooting sensitive or load threatening equipment.

When the plant is in a 3.0.3 Tech Spec action or a load threatening condition, the Shift Manager ___(2)___ allowed to authorize work without formal planning and prior to obtaining QC approval.

A.	(1) engineering supervisor (2) is NOT
B.	(1) senior reactor operator (2) is NOT
C.	(1) engineering supervisor (2) is
D.	(1) senior reactor operator (2) is

Proposed Answer: D

A.	Incorrect. Part 1 is incorrect, but plausible since engineering support is required in a variety of plant work activities described in ADM-701. Part 2 is incorrect, but plausible since normally the Work Authorization process must be followed.
B.	Incorrect. Part 1 is correct. Part 2 is incorrect, but plausible per discussion above.
C.	Incorrect. Part 1 is incorrect, but plausible per above discussion. Part 2 is correct.

D.	Correct, IAW 0-ADM-701.		
Technical Reference(s)	0-ADM-701 Attachment 4		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		3
Facility licensee procedures required to obtain authority for design and operating changes in the facility.			
Comments:			
10CFR55.43(b) item 3 is satisfied because the SRO must recognize requirements for performing maintenance activities on plant equipment, in this case when a plant shutdown in accordance with TS 3.0.3 is required			

4.2 Immediate Corrective Action and Emergency Work (A Priority PWO)

1. Immediate corrective action and emergency work may be performed without a formal preplanned PWO.

A. Immediate corrective action shall be taken with the intent of placing the plant in a safe condition, whenever a condition poses a threat to plant safety or to the health and safety of the public; when it could result in major equipment and material damage; or when it could, if **NOT** corrected, produce defects of significantly greater consequences than those immediately resulting from the condition. A follow-up action must be taken to provide the appropriate documentation for the immediate actions taken (e.g., Work Order, Action Request, Temporary Configuration Change or EC).

B. When the plant is in a load threatening condition or in an action statement of Technical Specifications Section 3.0.3, the Shift Manager may authorize work to start without formal planning and prior to obtaining QC approval. To immediately commence work, the Shift Manager shall originate a PWO in accordance with this procedure, Section 4.3, using a blank PWO form similar to Attachment 1, Nuclear Plant Work Order, and Attachment 2, Journeyman's Work Report Nuclear Plant Work Order; assign the priority as A, and sign the Shift Manager Start Permission block. Maintenance is then authorized to take immediate actions necessary to secure the plant's condition.

(1) Maintenance may proceed directly with work without further planning or approvals, but should attempt to contact QC for notification of the A priority work.

(2) All actions taken shall be thoroughly documented on the Journeyman's Work Report Section of the PWO per this procedure.

(3) A priority work may continue for up to eight hours without formal PWO planning and normal work controls.

(4) Job planning after work is completed is **NOT** required; however, the PWO should be immediately forwarded to the applicable Planning Group for review.

(5) Completed A priority PWOs shall be reviewed by QC within 48 hours of job completion.

3.0 RESPONSIBILITIES (continued)

7. (continued)

C. Authorize the start of work and acknowledge the completion of work on Power Block related PWOs in accordance with this procedure.

D. Shall designate a SRO as the liaison between the Shift Manager and the troubleshooting team for troubleshooting sensitive or load threatening equipment. The troubleshooting plan shall be reviewed by the SRO prior to the implementation of the plan. Completed troubleshooting activities shall have an independent assessment by the SRO and applicable Maintenance supervision to ensure proper development of subsequent troubleshooting steps. All changes to the troubleshooting plan, for sensitive or load threatening equipment, shall be reviewed by the SRO prior to implementation. The SRO shall keep the Shift Manager informed of the troubleshooting plan of action, progress, and subsequent changes to the plan.

8. Planning Supervisor/Fin Team Leader/Designated FIN Team Senior Reactor Operator (SRO)

A. Review and sign all Safety Related (SR) and Quality Related (QR) work control documents generated to implement a PC/M (MSP) and any changes thereto prior to implementation. The review shall include proper parts Procurement Classification (PC) levels, as well as, proper planning from design documents.

B. Review all SR and QR work orders for content and quality to ensure that the correct part/material PC levels are being installed in SR and QR Systems in accordance with 0-ADM-047, Identification and Control of Safety Related and Quality Related Parts, Materials and Components, as follows:

(1) Safety Related (SR) systems shall only have PC 1 or PC 2 level material installed.

a. PC 2 material shall have a Dedication Package for that application.

b. PC 3 or PC-4 material may only be installed in a SR Host Component with an approved Procurement Engineering (PE) evaluation.

(2) Quality Related (QR) systems shall have only PC 3 level material (or better) installed.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			2
	Topic and K/A #	G2		2.2.43
	Importance Rating			3.3

Equipment Control: Knowledge of the process used to track inoperable alarms.

Proposed Question: SRO Question # 96

A required annunciator must be defeated in accordance with 0-OSP-200.5, Miscellaneous Tests, Checks, and Operating Evolutions, Attachment 1.

Which one of the following identifies the maximum time the (1) annunciator may be defeated prior to performing a PCR to the ARP, and (2) allowed for an out of service annunciator prior to performing a 10CFR50.59 applicability/screening review?

A.	(1) 7 days (2) 7 days
B.	(1) 7 days (2) 60 days
C.	(1) 60 days (2) 7 days
D.	(1) 60 days (2) 60 days

Proposed Answer: B

A.	Incorrect. Plausible because 7 days is correct for the temporary change process but 60 days are allowed for 10CFR50.59 screening
B.	Correct, IAW 0-OSP-200.5 attachment 1.
C.	Incorrect. Both times are correct but reversed, making this plausible for the candidate that confuses the times.
D.	Incorrect. Plausible because 60 days is correct for a 10CFR50.59 screening, but incorrect because only 7 days is allowed prior to the additional documentation of the TC process

Technical Reference(s)	0-OSP-200.5 attachment 1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:	6902042, obj 4		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		3
Facility licensee procedures required to obtain authority for design and operating changes in the facility.			
Comments:			
10CFR55.43(b) item 3 is satisfied because the SRO must recognize requirements for remove ng an annunciator from service, including the requirement for performing a 10CFR50.59 screening			

7.11.2 (Cont'd)

5. Verify that each new defeated/OOS annunciator (since the previous performance of this subsection) that is **NOT** the result of a planned activity has been screened per OP-AA-108-1000, Operator Burdens Program Management. Determine if condition represents a potential operator burden and then perform the following as necessary:

   a. Open Work Request/Work Order

   b. Select Attributes

   c. Right Click Carrot under attribute column

   d. Select Operator Burden

   e. Right Click Carrot under Value Column

   f. Select appropriate code.

6. **IF** an annunciator has been defeated/OOS for 60 days, **THEN** ensure a 10 CFR 50.59, Applicability Determination/Screening per Block 7 of Attachment 1, has been performed.

7. **IF** the 10 CFR 50.59, Applicability Determination/Screening shows that the defeated/OOS annunciator may adversely affect the safe operation of the unit(s), **THEN** notify the Assistant Operations Manager for resolution.

NOTES

- *The maximum time an annunciator may be administratively defeated is 7 days, unless Substep 7.11.2.8 is completed.*

8. **IF** an annunciator can **NOT** be returned to service within 7 days, **THEN** complete Block 8 of Attachment 1.

9. **IF** an annunciator can **NOT** be returned to service within 7 days, **THEN** verify that Attachment 1, Section 8 has the appropriate tracking method for restoration.

10. **WHEN** the annunciator is returned to service, **THEN** complete Blocks 9 and 10 of Attachment 1.

7.11.3 Document the result of the determination on the QA Record Page.

7.11.4 Verify all log entries specified in Subsection 2.2 have been recorded.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			3
	Topic and K/A #	G3		2.3.14
	Importance Rating			3.8

Radiation Control: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: SRO Question # 97

Given the following conditions:

- A fast load reduction to 60% power was performed on Unit 3.
- ANN H 1/4, PRMS HI RADIATION, alarms.
- R-3-20, RCS Letdown Radiation Monitor, is in high alarm.
- RP reports that letdown piping is 300 mR/hr at 1 foot.
- Dose Equivalent Iodine-131 is approximately 5 microcuries per gram.
- The crew enters 3-ONOP-041.4, Excessive Reactor Coolant System Activity.

Which one of the following completes the statement below?

The R-3-20 alarm ___(1)___ an EAL threshold.

According to Technical Specifications, ___(2)___ .

REFERENCE PROVIDED

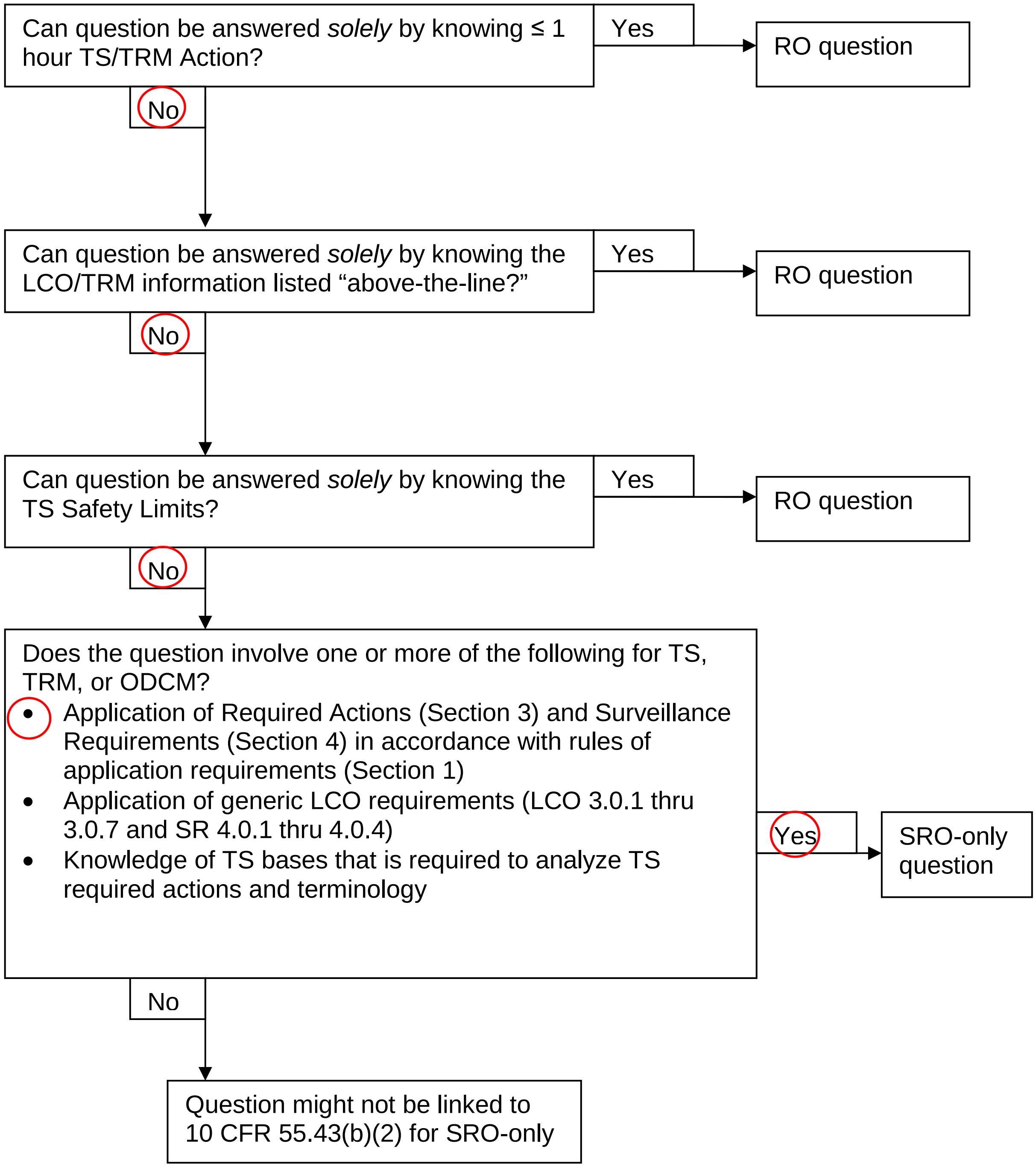
A.	(1) is (2) a plant shutdown and cooldown must be performed immediately
B.	(1) is (2) plant operation may continue up to 48 hours with increased RCS sampling frequency
C.	(1) is NOT (2) a plant shutdown and cooldown must be performed immediately
D.	(1) is NOT (2) plant operation may continue up to 48 hours with increased RCS sampling frequency

Proposed Answer: B			
A.	Incorrect. Part 1 is correct. Part 2 is incorrect but plausible if the candidate believes a shutdown is required IAW action 3.4.8.b - shutdown requirement when greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131 for greater than or equal to 48 hours during one continuous time interval.		
B.	Correct. 3-ONOP-041.4 describes R-20 as an EAL threshold and DEI-131 requires action 3.4.8.b.		
C.	Incorrect. Part 1 is incorrect, but plausible when candidate considers other process monitors do NOT have EAL thresholds. Candidate may also believe letdown will be isolated to maintaining ALARA conditions. This is performed in other procedures (e.g. when going on cold leg recirc, guidance has the operator close shield doors before dose rates get to high). Part 2 is correct.		
D.	Incorrect. Part 1 is incorrect. Part 2 is correct.		

Technical Reference(s)	TS 3.4.8 3-ONOP-041.4		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Y- TS 3.4.8
Learning Objective:			(As available)
Question Source:	Bank	11718	
	Modified Bank	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2008	McGuire
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		4
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.			
Comments:			

Modified conditions to make plant specific to PTN. Modified part 1 for threshold. 10CFR55.43(b) item 4 is satisfied because a high RCS activity and high radiation exist and the SRO must determine the appropriate action required under the conditions presented. 10CFR55.43(b) item 2 is also satisfied because technical specification action greater than 1 hour is applicable

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.25 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 447.7 microcuries per gram DOSE EQUIVALENT XE-133.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, verify DOSE EQUIVALENT I-131 is less than or equal to 60 microcuries per gram once per 4 hours.

b. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 60 microcuries per gram, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT I-131 to within the 0.25 microcuries per gram limit. Specification 3.0.4 is not applicable.

c. With the specific activity of the reactor coolant greater than 0.25 microcuries per gram DOSE EQUIVALENT I-131 for greater than or equal to 48 hours during one continuous time interval, or greater than 60 microcuries per gram DOSE EQUIVALENT I-131, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

d. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133, operation may continue for up to 48 hours while efforts are made to restore DOSE EQUIVALENT XE-133 to within the 447.7 microcuries per gram limit. Specification 3.0.4 is not applicable.

e. With the specific activity of the reactor coolant greater than 447.7 microcuries per gram DOSE EQUIVALENT XE-133 for greater than or equal to 48 hours during one continuous time interval, be in HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performing the sampling and analysis described in Table 4.4-4.

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.0 OPERATOR ACTIONS

3.1 <u>Immediate Actions</u>

None

3.2 <u>Subsequent Actions</u>

<u>NOTE</u>

- R-3-20 Alarm is an EAL Threshold. Refer to 0-EPIP-20101
- R-3-20 may alarm if RCS letdown flow is raised above 60 gpm.

1. **CONFIRM** R-3-20, REACTOR COOLANT LETDOWN Monitor high alarm as follows:

   A. **REQUEST** RP perform a survey of the letdown line in various areas to confirm that the letdown piping is the source of the high radiation levels.

   <u>OR</u>

   B. **REQUEST** Chemistry perform a radiochemical analysis of the RCS for fission product concentration and gross activity, to determine if fission product concentration is rising OR a crud burst is occurring.

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Question 97 original

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		
Group #		
Topic & KA #		
Importance Rating:		

KA Statement

Proposed Question:

Given the following:

A load reduction from 100% to 60% was performed on Unit 1 in the last 30 minutes due to a Feedwater Control problem.
The following alarms are received:

- **1EMF-48, REACTOR COOLANT HIGH RAD**
- **1EMF-18, REACTOR COOLANT FILTER 1A**

Chemistry sample indicates that the high activity is due to failed fuel.
Dose-Equivalent Iodine-131 is approximately 5 microcuries per gram.
The crew enters AP/18, High Activity in Reactor Coolant.

Which ONE (1) of the following describes the action(s) that will be performed in accordance with AP/18, and identifies the required Technical Specification actions?

REFERENCE PROVIDED

A. Raise Letdown flow to 120 GPM;
Plant shutdown and cooldown to less than 500F must be performed

B. Raise Letdown flow to 120 GPM;
Plant operation may continue with increased NC SYSTEM sampling frequency.

C. Ensure Mixed Bed Demin is in service and evaluate use of Cation Bed Demin; Plant shutdown and cooldown to less than 500F must be performed.

D. Ensure Mixed Bed Demin is in service and evaluate use of Cation Bed Demin; Plant operation may continue with increased NC SYSTEM sampling frequency.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Letdown flow is raised only for crud burst. Failed Fuel is indicated by iodine activity. TS shutdown is required if I-131 DE was above the curve in figure 3.4.16-1 or if operation at this level continued for 48 hours.

B. Incorrect. Letdown flow is raised only for crud burst. Failed Fuel is indicated by iodine activity, as described by conditions presented.

C. Incorrect. TS shutdown is required if I-131 DE was above the curve in figure 3.4.16-1 or if operation at this level continued for 48 hours. This condition is above TS steady state limit but below the transient limit on the curve

D. Correct.

Techical Reference(s): AP/18 Rev 2 and Basis Document
TS 3.4.16 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: Yes

Learning Objective: (As available)

Question Source: Bank 11718
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2008 McGuire

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

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			1
	Topic and K/A #	G1		2.1.36
	Importance Rating			4.1

Conduct of Operations: Knowledge of procedures and limitations involved in core alterations.

Proposed Question: SRO Question # 98

Given the following conditions:

- At 0000 on 7/31/2016, a Unit 4 shutdown to MODE 3 is commenced.
- At 0630 on 7/31/2016, Unit 4 enters MODE 3.
- At 2200 on 8/01/2016, the first reactor vessel head stud is de-tensioned.

Which one of the following completes the statements below?

In accordance with Technical Specifications, the EARLIEST time that fuel movement may commence is ___(1)___ .

The basis for this time requirement is to ensure that ___(2)___ .

A.	(1) 0630 on 08/03/16 (2) the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in the safety analysis
B.	(1) 0630 on 08/03/16 (2) the heat load assumptions specified in the safety analysis are met, so that boiling may be prevented in the Spent Fuel Pool
C.	(1) 2200 on 08/04/16 (2) the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in the safety analysis
D.	(1) 2200 on 08/04/16 (2) the heat load assumptions specified in the safety analysis are met, so that boiling may be prevented in the Spent Fuel Pool

Proposed Answer: A

A.	Correct. See TS 3/4.9.3 and ADM-536, page 106 of 112
B.	Incorrect. Plausible because the time is correct and because the basis is incorrect but heat load in the SFP is a concern for time to boil.
C.	Incorrect. Plausible because the time is 72 hours from Mode 6 entry and because the basis is correct
D.	Incorrect. Plausible because the time is 72 hours from Mode 6 entry and because the basis is incorrect but heat load in the SFP is a concern for time to boil.

Technical Reference(s)	TS 3/4.9.3 ADM-536		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank	12835	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2011	Turkey Point
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43		7

Fuel handling facilities and procedures.

Comments:

10CFR55.43(b) item 7 is satisfied because the SRO must determine when refueling activities may commence based on plant shutdown time. 10CFR55.43(b) item 2 is also applicable because the SRO is required to know the technical specification basis for the time required prior to refueling

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 ↓

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

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

ATTACHMENT 2
Technical Specification Bases
(Page 182 of 193)

3/4.9.2 (Continued)

A normal refueling consists of 2 CORE ALTERATION sequences: unloading the core, and reloading the core, typically with a suspension of CORE ALTERATIONS in between. The core unload sequence begins with control rod unlatching, followed by removal of upper internals, followed by unloading fuel assemblies to the SFP. The core reload sequence consists of reloading fuel assemblies from the SFP, followed by upper internals installation, followed by latching control rods. Therefore, if T.S. 4.9.2.c is complied with following the ANALOG CHANNEL OPERATIONAL TEST performed within 8 hours prior to start of control rod unlatching, then the ANALOG CHANNEL OPERATIONAL TEST need **NOT** be performed within 8 hours prior to the start of core reload. Otherwise, comply with T.S.4.9.2.b within 8 hours prior to the start of core reload.

3/4.9.3 Decay Time

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses, and ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in 10 CFR 50.67 and RG 1.183.

REFUELING OPERATIONS

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

Facility: WTSI Corporate

Vendor WEC

Exam Date:

Exam Type:

Question 98 original

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	Topic & KA #		
	Importance Rating:		

KA Statement

Proposed Question:

Given the following:
Date Time Activity
12/31/2011 0000 A Unit 4 Shutdown to MODE 3 is commenced.
12/31/2011 0630 Unit 4 enters MODE 3.
12/31/2011 1320 Unit 4 enters MODE 4.
12/31/2011 2210 Unit 4 enters MODE 5.
01/01/2012 2200 The first Reactor Vessel Head Stud is detensioned.
01/03/2012 0100 The Reactor Vessel Head is removed.
Which ONE of the following is (1) the EARLIEST time to commence fuel movement in accordance with Technical Specifications, and (2) the basis for the time requirement?

A. (1) 01/03/12 at 0630
(2) Ensures the heat load assumptions specified in the safety analysis are met to prevent boiling in the Spent Fuel Pool.

B. (1) 01/03/12 at 0630
(2) Ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in the safety analysis.

C. (1) 01/04/12 at 2200
(2) Ensures the heat load assumptions specified in the safety analysis are met to prevent boiling in the Spent Fuel Pool.

D. (1) 01/04/12 at 2200
(2) Ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in the safety analysis.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because the time is correct and because the basis is incorrect but heat load in the SFP is a concern for time to boil.

B. CORRECT. See TS 3.9.3 and ADM-536, page 106 of 112

C. Incorrect. Plausible because the time is 72 hours from Mode 6 entry and because the basis is incorrect but heat load in the SFP is a concern for time to boil.

D. Incorrect. Plausible because the time is 72 hours from Mode 6 entry and because the basis is correct

Techical Reference(s): TS 3.9.3
ADM-536 (Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: NO

Learning Objective: - (As available)

Question Source: Bank 12835
Modified Bank (Note changes or attach parent)
New

Question History: Last NRC Exam: 2011 Turkey Point

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

10 CFR Part 55 Content: 55.41
55.43 7

Fuel handling facilities and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			4
	Topic and K/A #	G4		2.4.26
	Importance Rating			3.6

Emergency Procedures / Plan: Knowledge of facility protection requirements, including fire brigade and portable firefighting equipment usage.

Proposed Question: SRO Question # 99

Given the following conditions:

Unit 3 is in Mode 3.
A fire alarm has been received in Fire Zone 086, Unit 3 Main and Startup Transformer Area.
The fire alarm has been confirmed by field operators.

Which one of the following completes the statements below?

The ___(1)___ Train is credited for Safe Shutdown for a fire in this area.

The ___(2)___ is responsible for calling in off-site assistance, if required, in fighting the fire

A.	(1) 3A (2) Shift Manager
B.	(1) 3A (2) Security Shift Supervisor
C.	(1) 3B (2) Shift Manager
D.	(1) 3B (2) Security Shift Supervisor

Proposed Answer: C

A.	Incorrect. Plausible because in a non-safety related fire zone, either train could be credited based upon the equipment that could potentially be affected. Also because part 2 is true.		
B.	Incorrect. Plausible for part 1 same reason as option A and part 2 is plausible because security is responsible for granting access and escorting the outside assistance to the fire area		
C.	Correct, IAW 0-ONOP-016.10.		
D.	Incorrect. Part 1 correct and part 2 plausible for same reason as option B		
Technical Reference(s)			(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		1
Facility operating limitations in the technical specifications and their bases.			
Comments:			
10CFR55.43(b) item 1 is satisfied because the SRO must know Appendix R requirements for safe shutdown of the facility and the person responsible for directing off-site response is applicable to 10CFR55.43(b) item 5			

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

FIRE ZONES 086
UNIT 3 MAIN TRANSFORMER AND STARTUP TRANSFORMER

(Fire Area OD-086)

Attachment A

Fire Zone 086 Unit 3 Response

(Page 1 of 1)

NOTES

- Foldout Page shall be monitored during performance of this procedure.
- 3B Train is credited for safe shutdown in this area.

1.0 Feedwater System

1.1 Within **10 minutes** of a failure of a Feedwater Regulating Valve (FRV) to isolate feedwater from a Unit 3 Steam Generator, **PERFORM** the following to prevent Steam Generator overfill.

1.1.1 In the Control Room, **PLACE** the following Control Switches for the following breakers to the STOP position to DE-ACTIVATE its associated initiation of the AFW Auto Start Circuit:

- 3AA03, 3A SGFP
- 3AC14, 3B SGFP

1.1.2 In the Control Room, **TRIP** the following breakers:

- 3AA03, 3A SGFP
- 3AC14, 3B SGFP

NOTE

Performance of the following step will de-energize 3D Bus resulting in loss of 3C ICW and 3C CCW.

2.0 Switchgear 3B Availability

2.1 Within **15 minutes**, from the Control Room, **OPEN** Breaker 3AB19, TIE BREAKER TO 3D BUS, to ensure availability of the 3B Bus and 3B EDG.

3.0 LCV-3-460, LETDOWN LINE STOP VALVE, Failure

3.1 IF LCV-3-460, LETDOWN LINE STOP VALVE, fails to close when required THEN **CLOSE** the following valves from the Control Room:

- CV-3-200A, Letdown Containment Isolation Valve
- CV-3-200B, Letdown Containment Isolation Valve
- CV-3-200C, Letdown Containment Isolation Valve

CAUTION

Safe shutdown actions need to be performed for the area of the fire. Safe shutdown actions for alarms from secondary smoke do NOT need to be performed.

NOTES

- *A single fire in a given fire zone may require both units to be shutdown or placed in safe condition.*
- *Fires or explosions inside the RCA involving Radiological Hazards should be monitored accordingly. Alarming (audible and/or visual) dosimetry should be used on Fire Brigade Members for monitoring direct Radiological Exposure. The air sampler, located in the fire locker in the Aux Bldg Hallway, should be used as needed, to monitor airborne activity.*
- *The Fire Brigade Leader is required to notify the Shift Manager/Emergency Coordinator when vital equipment is in jeopardy or the fire can NOT be readily extinguished.*
- *Erroneous indication; multiple spurious equipment malfunctions or failures; loss of off-site power coincident with fire or notification from Fire Brigade Leader are the determinates for initiating safe shutdown manual actions.*
- *Large – area fires or explosions involving multiple fire areas or zones may require mitigation or recovery strategies beyond those prescribed for NFPA 805 fire scenarios. Guidance for response to circumstances associated with loss of large areas due to fire or explosions is provided in EDMG-1, Guidline for Responding to Large Area Fire or Explosion Involving Multiple Fire Zones.*

5.3 The Shift Manager/Emergency Coordinator shall:

5.3.1 **IF** determined necessary, **THEN** initiate safe shutdown manual actions for the area containing the fire.

5.3.2 **IF** Fire/Explosion is in the Radiation Control Area (RCA), **THEN** notify Radiation Protection **AND** request radiation monitoring support.

5.3.3 Determine if the fire/explosion meets emergency classification criteria by referring to 0-EPIP-20101, Duties of Emergency Coordinator.

5.3.4 **IF** the fire/explosion meets emergency classification criteria, **THEN** implement the following procedures:

1. 0-EPIP-20101, Duties of Emergency Coordinator
2. 0-EPIP-20132, Technical Support Center (TSC) Activation and Operation

NOTES

- *Use Attachments 2 and 3 when communicating with off-site assistance as to the location of the fire and fire protection equipment.*
- *Attachments 2 and 3 are located in Lotus Notes and are not included in the hard copy of this procedure. If Lotus Notes is not available, hard copies of these attachments are available in the Control Room.*
- *Off-site assistance will have the latest copy of these attachments.*

5.3.5 Contact additional fire support as needed (phone numbers are located in Emergency Response Directory)

5.3.6 Notify Security if off-site fire support is needed.

5.3.7 Direct the Fire Brigade Leader to search the area for injured victims when conditions permit.

5.4 Direct Maintenance to perform Manual Actions to Mitigate the Consequences of Fire Damper Closure Attachment for the applicable fire zone, immediately following extinguishment of the fire.

5.5 **<u>IF</u>** fuse pullers are required, **<u>THEN</u>** obtain them from the nearest 0-ONOP-105, Communication Box:

5.5.1 <u>Location</u>

1. Main Control Room Unit 4
2. 4B 4160 Swgr Room
3. 4A 4160 Swgr Room
4. 4A and B LC Room
5. 4C and D LC Room
6. 4B MCC Room
7. Main FW Platform Unit 4
8. AFW Train 2 Platform Unit 4
9. U-4 Turbine Deck near exciter housing
10. Tech Support Center
11. 4B EDG Control Room
12. U-4 Containment Personnel Hatch
13. U-4 P and V Room
14. U-4 Charging Pump Room
15. U-4 RHR Pump Room Mezzanine Level

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			4
	Topic and K/A #	G4		2.4.8
	Importance Rating			4.5

Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question: SRO Question # 100

Given the following conditions:

- A Steam Generator Tube Rupture has occurred on Unit 3.
- The crew is performing 3-EOP-E-3, Steam Generator Tube Rupture, when a loss of vital instrument bus 3P07 occurs.

In accordance with 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage, which one of the following identifies the use of 3-ONOP-003.7, Loss of 120 Volt Vital Bus 3P07, while performing 3-EOP-E-3?

A.	Continue in 3-EOP-E-3. Perform actions of 3-ONOP-003.7 only after exiting the EOP network.
B.	Stabilize the unit in 3-EOP-E-3. Perform actions to restore 3P07 in accordance with 3-ONOP-003.7 that will not interfere with the actions of 3-EOP-E-3, and then return to 3-EOP-E-3.
C.	At US discretion, perform steps of 3-ONOP-003.7 concurrently with the performance of 3-EOP-E-3 to mitigate both events.
D.	The US must obtain Shift Manager authorization prior to implementation of any part of 3-ONOP-003.7 while 3-EOP-E-3 is in effect.

Proposed Answer: C

A.	Incorrect. Plausible because in procedure hierarchy, EOPs take precedence over ONOPs.
B.	Incorrect. Plausible because this is similar to concurrent use, and there are words in this option that are true, but E-3 would not be suspended for ONOP steps unless specifically directed by E-3

C.	Correct, IAW ADM-211, the US (procedure director) shall determine how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress.		
D.	Incorrect. The US is the procedure director in EOPs and he/she has the authority to implement procedures at his/her discretion under the rules of ADM-211.		
Technical Reference(s)	ADM-211 section 4.6 item 10 (pg 23 and 24 of 47)		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			N
Learning Objective:			(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		5
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Comments:			
10CFR55.43(b) item 5 is satisfied because the SRO must assess the conditions presented and determine when it is appropriate to perform actions of a specific AOP when the EOPs are in use			

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

4.6 Generic Procedure Usage Issues (continued)

10. Concurrent Procedure Use:

A. The US shall determine how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress.

B. More than one EOP shall **NOT** be performed concurrently unless directed by the EOP in effect.

EXAMPLE

RCP tripping criteria from E-0 Fold Out Page or ONOP-041.1 should **NOT** be performed while in FR-S.1.

C. Concurrent procedure use is usually **NOT** permitted during performance of Immediate Actions Steps. The only exception is the performance of action(s) that shall be performed promptly after a reactor trip as directed by an ONOP (AOP). The specified action(s), as determined by the ONOP, will be performed after verifying that the appropriate IOA step has been completed in E-0. This ensures the E-O IOAs are performed in sequence.

EXAMPLES

- ONOP-041.1, Reactor Coolant Pump Off Normal, directs a Reactor Trip followed by an RCP Trip. In this situation, the operator should perform the manual Reactor Trip, and verify the reactor is tripped by successfully completing the substeps of Step 1 of E-0 then trip the affected RCP.
- ONOP-47.1, Loss of Charging in Modes 1 Through 4, directs a Reactor Trip followed by manually initiating Safety Injection and Phase A Containment Isolation. In this situation, the operator(s) should perform the manual Reactor Trip, and verify Reactor Trip, Turbine Trip, and emergency power to 4kV busses by successfully completing Steps 1 through 3 of E-0 and then manually initiate SI and Phase A Containment Isolation.

D. When any EOP is in effect, ONOPs (AOPs) or ARPs may be performed in the discretion of the US or SM, only if they do **NOT** interfere with the actions called for in the EOPs and if their implementation is necessary to help mitigate the consequences of the event. Actions of the ONOP (AOP) or ARP should be performed in parallel with EOPs.

Facility: **Turkey Point Nuclear (PTN) – Units 3 and 4** Scenario No.: **1** Op Test No.: **2016-301**

Examiners: ______________________ Operators: ______________________ (SRO)

______________________ ______________________ (RCO)

______________________ ______________________ (BOP)

Initial Conditions:	The plant is at 75% power (MOL). Online risk is green. B train is protected on both units.		
Turnover:	The 3A RHR pump and 3A1 Circulating Water pump are OOS.		

Event	Malf. No.	Event Type*	Event Description
1	TFH1TU59	I-RCO I-SRO (TS)	LT-3-459, PZR Level Transmitter, Fails High
2	TAKPXA2 TAKPXB1 TAKPXB2	R-RCO R-SRO N-BOP	3A2 Intake Screen Blockage (Load reduction required)
3	TVS1MWED	I-BOP I-SRO (TS)	FT-3-474, 3A S/G Steam Flow Transmitter, Drifts High
4	TFH1TV44	I-RCO I-SRO (TS)	PT-3-444, PZR Pressure Transmitter, Fails Low
5	TFFVP6A	I-BOP I-SRO	3A Condensate Pump Sheared Shaft
6	TVHHCLB	M-RCO M-BOP M-SRO	Large Break LOCA
7	TFQ6A2BF	P-RCO	3B RHR Pump Fails To Auto Start
8	TFCVVSO5 TFCVOSV6	P-BOP	CV-3-2826/2819, Containment Isolation IA Bleed Valves, Fail To Auto Close

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (P)ost Trip

Scenario Summary

Event 1

Shortly after taking the watch, LT-3-459, PZR Level Control Transmitter, fails high causing charging flow to reduce to the minimum and Pressurizer level to start trending down. The US will enter 3-ONOP-041.6, Pressurizer Level Control Malfunction, and direct the RCO to place Pressurizer Level Control Transfer Switch to position 3, CH.2&3. Once Pressurizer level is stabilized the US will enter 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels. The RCO will verify the Pressurizer Level Control Transfer Switch has been switched to position 3 and Pressurize level control is in automatic.

Event 2

After the US evaluates Tech Specs for LT-3-459, the Intake Screen differential level will start rising on all running Circulating Water Pumps. The US will enter 3-ONOP-011, Screen Wash System/Intake Malfunction. The screen differential level on the 3A2 screen will require the 3A2 Circulating Water Pump be secured. Prior to securing the 3A2 CWP Circulating Water Pump Reactor power must be reduced to less than 60%. The crew will reduce power to less than 60% using 3-GOP-100, Fast Load Reduction, and then secure the 3A2 Circulating Water Pump.

Event 3

After the crew stops the 3A2 Circulating Water Pump FT-3-474, 3A S/G Steam Flow Transmitter, will drift high. The BOP will take manual control of the 3A S/G level and restore the 3A S/G level to normal. The US will enter 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, direct the BOP to select an operable channel, and restore 3A S/G level control to automatic.

Event 4

Once the US completes the Tech Spec evaluation for FT-3-474, Pressurizer Pressure Control Transmitter PT-3-444 fails low. The PZR Sprays will close and all PZR Heaters will turn on causing Pressurizer pressure to rise. The US will enter 3-ONOP-041.5, PZR Press Control Malfunction. The RCO will take manual control of PC-3-444J, PZR Press Controller, and restore PZR pressure to normal.

Event 5

After the crew restores Pressurizer Pressure the crew will start the 3C Condensate pump and secure the 3A Condensate pump due to a sheared shaft on the 3A Condensate Pump.

Event 6

Once the crew completes swapping condensate pumps a Large Break LOCA will occur. The crew will manually trip the Reactor and enter 3-EOP-0, Reactor Trip Or Safety Injection. When RCP Trip Criteria are the met the RCO will trip the RCPs.

Event 7

When SI actuates the 3B RHR pump fails to auto start. The RCO will start the 3B RHR pump following the Immediate Operator Actions of 3-EOP-E-0.

Event 8

When Phase A actuates, CV-3-2826 and CV-3-2819, IA Bleed Valves, fail to auto close. While performing 3-EOP-E-0 Attachment 3, Prompt Action Verifications the BOP manually closes CV-3-2826 however 3-CV-3-2819 is failed open and will not close in auto or manual.

Scenario Summary

The crew will transition from 3-EOP-0 to 3-EOP-E-1, Loss of Reactor or Secondary Coolant. During or shortly after the transition to 3-EOP-E-1 the crew will be required to go to 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition due to a RED path on the Integrity Status Tree. They will verify RHR flow greater than 1100 gpm, and then return to 3-EOP-E-1.

The scenario may be terminated after the crew transitions from 3-EOP-FR-P.1 to 3-EOP-1 at the Lead Evaluator's discretion once all Critical Tasks have been evaluated.

Event		Critical Tasks
6/8	**CT1**	**Start 3B RHR Pump** During a Large Break LOCA start at least one RHR pump to provide core cooling to avoid transition to 3-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation. *Safety Significance* -- Failure to manually start at least one low-head ECCS pump prior to the transition to a contingency procedure constitutes misoperation or incorrect crew performance in which the crew does not prevent degraded emergency core cooling system capacity that may lead to or prolong core uncovery.
6/7	**CT2**	**Close CV-3-2826** During a Large Break LOCA close containment isolation valves such that at least one valve is closed on each critical Phase A penetration before whichever of the following occurs first: • The completion of 3-EOP-0 Attachment 3. • Within 60 minutes of the Phase A actuation signal. *Safety Significance* -- Failure to perform the critical task leads to an unnecessary release of fission products to the auxiliary building, increasing the potential for release to the environment and reducing accessibility to vital equipment within the auxiliary building. High radiation in the auxiliary building can lead to increased doses to personnel.

Facility: **Turkey Point Units 3 & 4** Scenario No.: **2** Op Test No.: **2016-301**

Examiners: ____________ Operators: ____________ (SRO)

____________ ____________ (RCO)

____________ ____________ (BOP)

Initial Conditions:		The plant is at 100% power (BOL). Online risk is green. B train is protected on both units.	
Turnover:		The 3A RHR pump and 3A1 Circulating Water pump are OOS.	
Event.	**Malf. No.**	**Event Type***	**Event Description**
1	TFB1LTLV	I-RCO I-SRO	LT-3-115 VCT Level Transmitter Fails Low
2	TVUTPMPA TFL10101	C-RCO C-SRO	3A Heater Drain Pump (Turbine Runback) Rods Fail To Auto Insert
3	TFN1CP22	I-BOP I-SRO (TS)	N-42 Loss Of Instrument Power
4	N/A	R-RCO R-SRO N-BOP	3B S/G Feedwater pump High Vibration (Fast Load Reduction required)
5	TFS1M5EA	I-BOP I-SRO (TS)	FT-3-494 3C S/G Steam Flow Transmitter Fails As Is
6	TVFAHDR1 TVSBVL14 TAFK144 TAFK244 TAFK344	M-RCO M-BOP M-SRO	Common Main Feed Header Break Common Loss Of Suction To All AFW Pumps
7	TFU10005	P- BOP	Main Turbine Fails To Automatically Trip
8	TFHV55CC	P-RCO	PCV-3-455C Fails To Open. (PZR PORV)
*	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (P)ost Trip		

SCENARIO SUMMARY

Event 1

Shortly after the crew takes the shift LT-3-115 VCT Level Transmitter Fails Low which causes auto makeup to start. The crew responds using the 3-ONOP-046.4, Malfunction of Boron Concentration Control System. The RCO manually stops auto makeup.

Event 2

Once the crew stabilizes VCT level the 3A Heater Drain Pump trips which causes an automatic Turbine Runback. The crew will enter 3-ONOP-089, Turbine Runback. During the runback Control Rods fail to auto insert. The RCO will manually insert rods to maintain Tave $\pm$ 3°F. When the Runback is complete the RCO will borate as needed to clear Rod Lo Limit and Axial Flux alarms, and the BOP will reset the Steam Dumps. When the plant is stable the crew will enter 3-ONOP-028, Reactor Control System Malfunction for the failure of Rods to auto insert.

Event 3

Once the crew resets the Steam Dumps and completes the required actions for the Reactor Control System Malfunction the N-42 Instrument Power Fuse Blows. The crew will enter 3-ONOP-059.8, Power Range Nuclear Instrumentation Malfunction. The BOP will defeat or bypass the affect functions of N-42 as directed by the ONOP.

Event 4

After the crew completes the actions of 3-ONOP-059.8 Engineering reports High Vibration on the 3B SGFW pump. The SM directs the crew to start a 3-GOP-100 Fast Load Reduction to secure the 3B SGFW pump.

Event 5

When the crew starts the down power FT-3-494, 3C S/G Steam Flow Transmitter will be failed as is. The BOP will take manual control of the 3C S/G level and restore the 3C S/G level to normal. The US will enter 3-ONOP-049.1, direct the BOP to select an operable channel, and restore 3C S/G level control to automatic.

Event 6

After the crew reduces power by 5 to 10% and completes the actions for the failed Steam Flow channel a Main Feed Water Header break occurs. The crew responds to the reactor trip using 3-EOP-E-0, Reactor Trip or Safety Injection. During the loss of Main Feed Water there’s also a loss of the suction piping to all AFW pumps. The crew will transition to 3-EOP-FR-H.1, Response To Loss Of Secondary Heat Sink, and initiate Feed and Bleed.

Event 7

During 3-EOP-E-0, The Main Turbine fails to automatically trip. The BOP will take compensatory action to trip the Turbine manually.

Event 8

When the crew attempts to initiate Feed and Bleed one of the PZR PORVs fails to open so the crew will open all RCS Vent Valves

The scenario is terminated once the RCS Vent Valves are or at the Lead Evaluator's discretion once all critical tasks have been evaluated.

Event	CRITICAL TASKS	
6	CT1	**Manually Trip the Main Turbine** Manually trip the main turbine before any RCS cold leg temperature decreases by more than 100°F. *Safety Significance* - Failure to trip the main turbine causes an excessive rate of RCS cooldown, well beyond the conditions typically analyzed in the FSAR. The excessive cooldown rate creates large thermal stresses in the reactor pressure vessel and causes rapid insertion of a large amount of positive reactivity. Thus, failure to manually trip the Main Turbine under the postulated conditions can result in challenges to the Integrity and Subcriticality CSFs.
6	CT2	**Initiate Bleed-And-Feed** Initiate bleed-and-feed cooling in accordance with 3-EOP-FR-H.1 within 36 minutes following an automatic or manual Reactor trip due to a loss of Feed Water. (0-ADM-232,Time Critical Operator Actions in the PTN PSA Model) *Safety Significance* - Failure to initiate RCS bleed and feed before the RCS saturates at a pressure above which the high-head ECCS pumps can inject results in significant and sustained core uncovery. If RCS bleed is initiated so that the RCS is depressurized below the shutoff head of the high-head ECCS pumps, then core uncovery is prevented or minimized.

Facility: **Turkey Point Nuclear (PTN) – Units 3 and 4** Scenario No.: 3 Op Test No.: **2016-301**

Examiners: ____________________ Operators: ____________________ (SRO)

____________________ ____________________ (RCO)

____________________ ____________________ (BOP)

Initial Conditions:	The plant is at 10^{-8} Amps power (BOL). Online risk is green. B train is protected on both units.		
Turnover:	No equipment is OOS		

Event	Malf. No.	Event Type*	Event Description
1	N/A	R-RCO R- SRO N-BOP	Raise Power to 3%
2	TFN1IACF	I-BOP I-SRO (TS)	N35 Loss Of Compensating Voltage
3	TAB1SCLC V8CG30OF TCB1SCLC	C-RCO C-SRO	3C Charging Pump Speed Controller Air Leak
4	TVS1SR2O	C-BOP C-SRO	PT-3-1607, 3B S/G Steam Dump To Atmosphere Pressure Transmitter, drifts high
5	TFC1SOL	I-SRO (TS)	PS-3-2007, Containment Pressure Channel Fails High
6	TFN1IBFH TFL2RTAB TFL4AF	I-RCO I-SRO	N-36, Intermediate Range Nuclear Instrument fails high The Rx fails to automatically trip
7	TFLIA44 TFLIA84	P-RCO	2 Stuck Rods
8	TVSBVL14	M-RCO M-BOP M-SRO	3B S/G Faulted Inside Containment
9	TFFVV87M TFFVV89	P-BOP	POV-3-487, Feedwater Bypass Isolation, and FCV-3-489, Feedwater Bypass, Valves Fail To Isolate.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (P)ost Trip

SCENARIO SUMMARY

NOTE

Allow 30 minutes for the crew to brief raising power before entering the control room to brief raising power from 10^{-8} amps to 3%.

Event 1

After the crew takes the shift the RCO will start raising Rx power by withdrawing Control Rods per 3-GOP-301, Hot Standby to Power Operation. The BOP will manually adjust Feedwater Bypass Flow Control valves, FCV-3-479/489/499, to maintain Steam Generator levels

Event 2

After the crew levels power at ~ 3% N35 loses compensating voltage. The US will enter 3-ONOP-059.7, Intermediate Range Nuclear Instrumentation and direct the BOP to place N35 in bypass.

Event 3

After the actions of 3-ONOP-059.7 are complete an air leak will develop on the 3C Charging pump speed controller causing the controller to fail to maximum output. The US will enter 3-ONOP-041.6, Pressurizer Level Control Malfunction and direct the RCO to start the 3B charging pump and secure the 3C charging pump.

Event 4

After the charging pumps are swapped PT-3-1607, 3B S/G Steam Dump To Atmosphere Pressure Transmitter, will drift high causing CV-3-1607, 3B Steam Dump To Atmosphere, to slowly open lowering the 3B S/G pressure. The BOP will place CV-3-1607 in manual and reduce demand to stabilize 3B S/G pressure.

Event 5

After the plant is stabilized, PS-3-2007, Containment Pressure Channel fails high. The US will enter 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channel, and review Tech Specs for the failed pressure channel.

Event 6

The US complete a review of Tech Specs for PS-3-2007, Intermediate Range Nuclear Instrumentation Channel N36 fails high and the Rx fails to auto trip. The RCO will manually trip the Reactor. The US will enter and direct the actions 3-EOP-E-0, Reactor Trip or Safety Injection.

Event 7

When the Rx is tripped the RCO will report 2 control rods failed to fully insert. After the crew completes the Immediate Operator Actions they will transition to 3-EOP-ES-0.1, Reactor Trip Response. The RCO will start a boration for the 2 stuck rods.

Event 8

After the boration is started a steam leak will develop on the 3B S/G inside containment. The crew will return to 3-EOP-E-0, verify SI actuates, and isolate Aux Feed Water to the 3B S/G per the Foldout page of 3-EOP-E-0.

Event 9

When SI actuates POV-3-487, 3B S/G Feedwater Bypass Isolation valve, 3B S/G Feedwater Bypass is failed as is, and FCV-3-489 will leak by. The BOP will manually close POV-3-487 per Attachment 3 of 3-EOP-E-0. The crew will complete the actions of 3-EOP-E-0. About the time the crew is ready to transition to 3-EOP-E-2, Faulted Steam Generator Isolation, a red path will develop on the RCS Integrity Status Tree. The US will transition to 3-EOP-FR-P.1, Response To Imminent Pressurized Thermal Shock Condition.

The scenario is terminated once the crew transitions to 3-EOP-FR-P.1 or at the Lead Evaluator's discretion once all critical tasks have been evaluated.

Event	CRITICAL TASKS	
6	CT1	**Manually Trip The Rx** Trip reactor manually within one minute when automatic trip signal fails. (0-ADM-232 Attachment 2, Time Critical Operator Actions in the PTN PSA Model) *Safety Significance:* Failure to manually trip constitutes an incorrect performance that “necessitates the crew taking compensating action that would complicate the event mitigation strategy” and demonstrates the inability of the crew to “recognize a failure or an incorrect automatic actuation of an ESF system or component.”
8	CT2	**Stop AFW Flow To Faulted SG** During a MSLB inside Containment stop AFW flow to the faulted SG within 10 Minutes. (0-ADM-232 Attachment 1, Time Critical Operator Actions) *Safety Significance:* Failure to isolate a Faulted SG that can be isolated causes challenges to the Critical Safety Functions that may not otherwise occur. Failure to isolate flow could result in an unwarranted Orange or Red Path condition on Integrity and/or Subcriticality (if cooldown is allowed to continue uncontrollably). Additionally, Termination of AFW flow to faulted SG is necessary to limit mass and energy releases into containment to prevent exceeding design pressure.

Facility: **Turkey Point Nuclear (PTN) – Units 3 and 4** Scenario No.: **4** Op Test No.: **2016-301**

Examiners: ____________ Operators: ____________ (SRO)

____________ ____________ (RCO)

____________ ____________ (BOP)

Initial Conditions:	The plant is at 100% power (MOL). Online risk is green. B train is protected on both units.
Turnover:	The 3A RHR pump and 3A1 Circulating Water pump are OOS.

Event	Malf. No.	Event Type*	Event Description
1	TVS1M6WD	I-BOP I-SRO (TS)	PT-3-495 3C S/G Pressure Transmitter Fails High
2	HNACSPECIFIC_VRCPSEAL1CTVV HNACSPECIFIC_VRCPSEAL2CTVV	R-RCO R-SRO N-BOP	3C RCP Degraded Seals (Rapid S/D Required)
3	TCE6DG8C	C-RCO C-BOP C-SRO (TS)	3P08 Loss Of Power (Power Restored)
4	HNACSPECIFIC_VRCPSEAL1CTVV / EAL2CTVV	C-RCO C-SRO	3C RCP Seal Failure, (Rx Trip Required)
5	TFP1S3GC TFE2Z51S	M-RCO M-BOP M-SRO	Loss of All AC
6	TFQ6X1BF	P-BOP	3B 4Kv Bus Stripping Relay Failure
7	TVHHCLC	M-RCO M-BOP M-SRO	Small Break LOCA
8	TFL3S12D	P-RCO P-BOP	MOV-3-843B HHSI Discharge to Cold Leg fails to auto open. (Slave Relay Failure)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (P)ost Trip

Scenario Summary

Event 1

After the crew takes the shift PT-3-495 3C S/G Pressure slowly fails high. The BOP will take manual control of 3C S/G level, and restore level to normal. The crew will use the ARP or 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, to select operable channels and restore 3C S/G level control to automatic. The US will enter 3-ONOP-049.1 to verify all required actions are complete and to determine which bistables need to be tripped.

Event 2

Once equipment is restored to automatic, the 3C RCP seals will start degrading. The US will enter 3-ONOP-041.1, Reactor Coolant Pump Off-Normal and commence a unit shutdown using 3-GOP-100, Fast Load Reduction.

Event 3

After the crew starts the Load reduction Vital AC Bus 3P08 loses power. The Crew will enter 3-ONOP-003.8, Loss of Vital AC Bus 3P08. The crew will dispatch an operator to 3P08 to attempt to restore power. The 3A and 3B S/G level controllers shift to manual. The 3C S/G level controller shifts to manual on the Backup Controller. The BOP will select operable control channels for the 3A and 3B S/G and then restore automatic control. The operator dispatched to the restore power will report the Main Breaker for 3P08 will not close. Electrical Maintenance estimates it will take 2 hours to replace the breaker.

Event 4

After the US completes the Tech Spec review for the loss of 3P08, the 3C RCP Seal degrades to the point that requires a Reactor trip and stopping of the 3C RCP. The RCO will trip the Reactor, verify the Reactor is tripped, stop 3C RCP, close 3C RCP CBO Isolation Valve CV-3-303C, and close PCV-3-455A, PZR Spray Valve Loop C. The crew will enter 3-EOP-E-0, Reactor Trip or Safety Injection and complete the Operator Immediate Actions.

Event 5

After the RCO completes tripping the 3C RCP, a Loss of AC Power will occur. The 3A and 3B Emergency Diesel Generators will start but neither will energize its respective 4KV bus. The crew will enter 3-EOP-ECA-0.0, Loss of All AC Power.

Event 6

The 3A 4KV Bus is locked out so the US will direct the BOP perform Attachment 2, 3B 4KV Bus Stripping. The BOP will open the 3B ICW pump, 3B CCW pump, 3C CCW pump, 3B Load Center, and 3D Load Center breakers to complete bus stripping. Once Bus Stripping is complete the 3B EDG will automatically energize the 3B 4KV Bus.

Event 7

Once the 3B 4KV Bus is energized the crew will transition back to 3-EOP-E-0. Shortly after the transition a Small Break LOCA will occur.

Scenario Summary

Event 8

When SI actuates, the slave relay which opens MOV-3-843B fails to actuate. The RCO may open MOV-3-843B any time after SI actuates. If the RCO doesn't open MOV-3-843B the BOP will open it during the performance of 3-EOP-E-0 Attachment 3, Prompt Action Verifications.

When the crew is ready to transition to 3-EOP-E-1, Loss Of Reactor Or Secondary Coolant, they will notice the Integrity Critical Safety Function Status Tree is RED and will transition to 3-EOP-FR-P.1, Response To Imminent Pressurized Thermal Shock Condition.

The scenario is terminated after the crew transitions to 3-EOP-FR-P.1 and determines a soak is required, or at the Lead Evaluator's discretion once all critical tasks have been evaluated.

Event	Scenario Critical Tasks	
6	CT1	**Re-energize 3B 4KV Bus** Following a Loss Of All AC, complete bus stripping and restore power to the 3B 4KV bus prior to actuating SI and within 30 minutes of the loss of power. *Safety Significance*: The failure to energize an AC emergency bus in a timely manner constitutes a misoperation or incorrect crew performance in which the crew does not prevent a degraded emergency power capacity. The 30 minute time limit is based minimizing DC bus battery depletion and the requirement to manually load a de-energized DC bus battery charger onto the operating EDG. (0-ADM-232, Attachment 1, Time Critical Operator Actions)
7	CT2	**Open MOV-3-843B** During a SBLOCA establish at least one train of HHSI flow prior to completing 3-EOP-E-0 Attachment 3 and within 30 minutes the HHSI pump starting. *Safety Significance*: Failure to establish at least one train HHSI flow constitutes a misoperation or incorrect crew performance in which the crew does not prevent "degraded emergency core cooling system (ECCS) capacity." The 30 minute time limit is based on the requirement to limit the time the pump is operating at shutoff head to less than 30 minutes. (0-ADM-231 Attachment 1, Time Critical Operator Actions)

Facility: **Turkey Point Nuclear (PTN) – Units 3 and 4** Scenario No.: **5** Op Test No.: **2016-301**

Examiners: ______________ Operators: ______________ (SRO)
______________ ______________ (RCO)
______________ ______________ (BOP)

Initial Conditions:	The plant is at 60% power (MOL). Online risk is green. B train is protected on both units.
Turnover:	The 3A RHR pump and 3A1 Circulating Water pump are OOS.

Event	Malf. No.	Event Type*	Event Description
1	TVF1M87D	I-BOP I-SRO (TS)	FT-3-487 3B S/G Feed Water Flow Transmitter Drifts High
2	TFCMM2H3 TFKV609O	C-RCO C-SRO	R-3-17B CCW Hx Radiation Monitor Fails High & RCV-3-609 CCW Surge Tank Vent Fails To Auto Close
3	TAKD032	C-BOP C-SRO	3A TPCW Pump Cavitation
4	TFH1TV60	I-RCO I-SRO (TS)	LT-3-460 Pressurizer Level Fails Low
5	TFC1DOR TFC1DOR2	R-RCO R-SRO N-BOP	3A & 3B CRDM Fans Trip (Fast Load Reduction required)
6	TVHHSGA	M-RCO M-BOP M-SRO	3A SGTR with LOOP
7	TFP1S3GC TFP8D6MT TFP8D6BT	P- BOP	Control Room HVAC Fails To Align on SI
8	TFHV55CO	P-RCO	PCV-3-445C PZR PORV, Fails To Close During E-3 Depressurization

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Scenario #5

Event 1

Shortly after the crew takes the shift FT-3-487 3B S/G Feed Water Flow transmitter drifts high. The BOP will take manual control of 3B S/G level and restore level to normal. The crew may use the ARP or 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection, to select operable channels and restore 3B S/G level control to automatic. The US will enter 3-ONOP-049.1 to verify all required actions are complete and to determine which bistables need to be tripped.

Event 2

After the actions of Event 1 are complete CCW Surge Tank Radiation Monitor R-17B fails high. CCW surge tank vent valve RCV-3-609 fails to close on the high radiation signal. The US will enter 3-ONOP-067, Radioactive Effluent Release, to verify the failure and direct the RCO to manually close the valve.

Event 3

After the actions of Event 2 are complete the crew will respond to a TPCW low pressure alarm. The BOP will report signs of cavitation and swap TPCW pumps. The US may enter 3-ONOP-008, Turbine Plant Cooling Water Malfunction to verify all required actions are complete.

Event 4

After the crew swaps TPCW pumps Pressurizer Level transmitter LT-3-460 will fail low. The PZR Heaters will trip and letdown will isolate. The US will enter 3-ONOP-041.6, Pressurizer Level Control Malfunction. The RCO will select an operable channel, re-establish normal letdown flow, and restore PZR heaters to automatic. The US will also enter 3-ONOP-049.1 to verify all required actions are complete and to determine which bistables need to be tripped.

Event 5

After the actions of Event 4 are complete the 3A CRDM Fan Trips and a few minutes later the 3B CRDM Fan Trips. The crew will commence a shutdown using 3-GOP-100, Fast Load Reduction.

Event 6

After a 5 to 10% downpower a SGTR develops over a 5 minute period on the 3A S/G. The crew will take actions to maximize Charging and to isolate Letdown. When the leakage exceeds the CVCS capacity, the US will order the RCO to trip the Reactor and enter to 3-EOP-E-0, Reactor Trip Or Safety Injection. When the Generator trips a Loss Of Offsite Power occurs. Both Emergency Diesel Generators will start and energize their respective 4KV buses. When the Ruptured S/G Isolation Criteria are met the BOP or RCO will isolate Aux Feed Water flow to the 3A S/G.

Event 7

When SI actuates Control Room Ventilation fails to align for recirc. The BOP will manually open Emergency Inlet Dampers D-2 and D-3 per 3-EOP-E-0 Attachment 3, Prompt Action Verifications.

Event 8

The crew will transition from 3-EOP-E-0 to 3-EOP-E-3, Steam Generator Tube Rupture. The crew will isolate the 3A S/G, cooldown the RCS, Establish Charging Flow, and stop the RHR pumps. When the cooldown is complete the RCO will open PZR PORV PCV-3-455C to depressurize the RCS (PCV-3-456 is failed close). When the depressurization is complete PCV-3-455C will fail to close so the RCO will close block valve MOV-3-536 to stop the depressurization.

The scenario is terminated after the crew completes the depressurization per 3-EOP-E-3, or at the Lead Evaluator's discretion once all critical task have been evaluated.

Event	Scenario Critical Tasks	
6	CT1	**Isolate the Ruptured S/G** During a Steam Generator Tube Rupture, isolate the ruptured S/G before a the ruptured Steam Generator pressure drops below 450 psig to prevent transition to 3-EOP-ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired. *Safety Significance:* Isolation of the ruptured steam generator minimizes release of radioactivity from this generator. In addition, isolation is necessary to establish a pressure differential between the ruptured and non-ruptured steam generators in order to cool the RCS and stop primary-to secondary leakage. If any ruptured S/G cannot be isolated from at least one intact S/G, the operator is directed to go to 3-ECA-3.1, SGTR With Loss Of Reactor Coolant -Subcooled Recovery Desired.
6	CT2	**Control Initial RCS Cooldown** During a Steam Generator Tube Rupture dump steam from intact S/Gs at maximum rate to achieve Core Exit TCs less than required temperatures based on the lowest ruptured S/G pressure without causing a transition to 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, or 3-EOP-ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired. *Safety Significance*: A SGTR mitigation strategy leading to a transition from 3-EOP-E-3 to a contingency procedure constitutes an incorrect performance requiring the crew to take additional compensatory actions that complicate the event mitigation strategy. With a SGTR, there exists a breach of the RCS fission-product and Containment barriers which allows radioactive RCS inventory to leak into the SG and associated piping. Without controlling the cooldown, the primary-to-secondary leakage is not stopped. This continued leakage results in a larger release of radioactivity to the environment affecting the safety of the public.

Event		Scenario Critical Tasks
6	CT3	**Limit RHR Time On Recirculation** When a RHR Pump starts and is operating at shutoff head, limit the operating time at shutoff head with minimum flow recirculation to no more than 44 minutes. (0-ADM-232, Time Critical Operator Action Program–Attachment 1) *Safety Significance:* Failure to secure the RHR Pumps operating at shutoff head leads to pump overheating and adverse vibration which would constitutes incorrect crew performance in which the crew does not prevent a degradation of the emergency core cooling system (ECCS) capacity.
6	CT4	**Control Initial RCS Depressurization** During a Steam Generator Tube Rupture depressurize the RCS to the ruptured S/G pressure without causing a transition to 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, or 3-EOP-ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired. *Safety Significance:* A SGTR mitigation strategy leading to a transition from 3-EOP-E-3 to a contingency procedure constitutes an incorrect performance requiring the crew to take additional compensatory actions that complicate the event mitigation strategy. With a SGTR, there exists a breach of the RCS fission-product and Containment barriers which allows radioactive RCS inventory to leak into the SG and associated piping. Without controlling the cooldown, the primary-to-secondary leakage is not stopped. This continued leakage results in a larger release of radioactivity to the environment affecting the safety of the public.

Site:	**Turkey Point Units 3 and 4 (PTN)**
Title:	**L-16-1 NRC EXAM SCENARIO 1**
LMS #:	**NRC 21**
LMS Rev Date:	**6/6/16** **Rev #: 0**
SEG Type:	☐ **Training** ☒ **Evaluation**
Program:	☐ **LOCT** ☒ **LOIT** ☐ **Other**
Duration:	**120 minutes**

Developed by:	**Brian Clark** Instructor/Developer	**6/13/16** Date
Reviewed by:	**Tim Hodge** *Instructor (Instructional Review)*	**6/22/16** *Date*
Validated by :	**Rocky Schoenhals** *SME (Technical Review)*	**6/22/16** *Date*
Approved by:	**Mark Wilson** *Training Supervision*	**6/22/16** *Date*
Approved by:	**Rocky Schoenhals** *Training Program Owner (Line)*	**6/22/16** *Date*

SCENARIO REFERENCES		
DOC NO.	**TITLE**	**REV**
	PTN TECHNICAL SPECIFICATIONS	298
3-EOP-E-0	REACTOR TRIP OR SAFETY INJECTION	12
3-EOP-E-1	LOSS OF REACTOR OR SECONDARY COOLANT	8
3-EOP-FR-P.1	RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION	3
3-GOP-100	FAST LOAD REDUCTION	12
3-ONOP-011	SCREEN WASH SYSTEM/INTAKE MALFUNCTION	7
3-ONOP-041.5	PRESSURIZER PRESSURE CONTROL MALFUNCTION	0A
3-ONOP-041.6	PRESSURIZER LEVEL CONTROL MALFUNCTION	2
3-ONOP-049.1	DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNEL	4

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
Terminal Objective	Given this simulator scenario and resources normally found in the Control Room, the operating crew will perform Control Room operations IAW approved plant procedures in order to maintain the integrity of the plant and the health and safety of the public.
Enabling Objectives:	Given this simulator scenario and resources normally found in the Control Room, operate in accordance with approved plant procedures, Operations Department Instructions, and management expectations: 1. (ALL) Demonstrate personnel SAFETY awareness in interactions with plant staff and outside agencies. 2. (ALL) Demonstrate ALARA awareness in interactions with plant staff and outside agencies. 3. (ALL) Exchange correct information using 3-point communication/Repeat-backs with Control Room personnel and other plant staff. 4. (ALL) Inform plant personnel and System of plant conditions, as needed. 5. (US) Employ timely and concise crew briefs where appropriate. 6. (ALL) Maintain awareness of plant status and control board indication. 7. (ALL) Correctly diagnose plant situations. 8. (ALL) Solve operational problems as they arise. 9. (RCO/BOP) Manipulate plant controls properly and safely. 10. (ALL) Demonstrate self-checking using STAR and peer checks(when required) 11. (US) Demonstrate command and control of the crew. 12. (US) Coordinate the input of crew members and other plant staff. 13. (US) Utilize the input of crew members and other plant staff. 14. (ALL) Demonstrate conservative decision making. 15. (ALL) Demonstrate teamwork. 16. (ALL) Respond to plant events using procedural guidance (OPs/ONOPs/EOPs) as applicable in accordance with rules of usage. 17. (RCO/BOP) Implement any applicable procedural immediate operator actions without use of references. 18. (SRO) Maintain compliance with Tech Specs. 19. (ALL) Identify/enter applicable Tech Spec action statements. 20. (ALL) Respond to annunciators using ARPs (time permitting). 21. (ALL) Maintain written communication, logs, and documentation as needed to permit post-event reconstruction. Continued on the next page:

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
	While addressing the following events: 1. LT-3-459, PZR Level Transmitter, Fails High 2. 3A2 Intake Screen Blockage (Load reduction required) 3. FT-3-474, 3A S/G Steam Flow Transmitter, Drifts High 4. PT-3-444, PZR Pressure Transmitter, Fails Low 5. 3A Condensate Pump Sheared Shaft 6. Large Break LOCA 7. 3B RHR Pump Fails To Auto Start 8. CV-3-2826/19, Containment Isolation IA Bleed Valves, Fail To Auto Close
Prerequisites:	None
Training Resources:	PTN Unit 3 Plant Simulator
Development References:	• TR-AA-220-1003, Initial NRC and Audit Exam Process • TR-AA-230-1003, SAT Development • TR-AA-230-1007, Conduct of Simulator Training and Evaluation • 0-ADM-232, Time Critical Action Program • OP-AA-100-1000, Conduct Of Operations • OP-AA-103-1000, Reactivity Management • 0-ADM-200, Operations Management Manual • 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage • WCAP-17711-NP, Pressurized Water Reactor Owners Group Westinghouse Emergency Response Guideline Revision 2-Based Critical Tasks
Protected Content:	N/A
Evaluation Method:	Performance Mode
Operating Experience:	None
Risk Significant Operator Actions:	1. **Stop All RCPs:** Following a Phase B stop all RCPs within 13 minutes to prevent damage to the RCP due to overheating caused by the loss of CCW.

TASKS ASSOCIATED WITH SIMULATOR EXERCISE GUIDE

SRO TASK #	TASK TITLE
02009002300	INVESTIGATE SCREEN D/P ALARM
02028033500	AUTHORIZE UNIT TRIP
02041057300	RESPOND TO PRESSURIZER LEVEL CONTROL CHANNEL MALFUNCTION
02063008500	VERIFY S. I. OPERATION
02073030300	INVESTIGATE CONDENSATE SYSTEM ALARMS
02089026300	AUTHORIZE FAST LOAD REDUCTION
02200021500	RESPOND TO A LOSS OF COOLANT ACCIDENT
02200044500	RESPOND TO STEAM GENERATOR HIGH LEVEL

RO TASK	TASK TITLE
01009002300	INVESTIGATE SCREEN D/P ALARM
01010005100	SHUTDOWN CIRCULATING WATER PUMPS
01028015100	ADJUST POWER LEVEL
01041027100	ADJUST PRESSURIZER PRESSURE MANUALLY USING THE MASTER CONTROLLER (444-J)
01041057300	RESPOND TO PRESSURIZER LEVEL CONTROL CHANNEL MALFUNCTION
01046007100	BORATE THE RCS VIA THE BLENDER
01063008500	VERIFY SAFETY INJECTION OPERATION
01073030300	INVESTIGATE CONDENSATE SYSTEM ALARMS
01074011300	CONTROL STEAM GENERATOR LEVEL MANUALLY WITH MAIN FEED REGULATING VALVES
01089026300	RESPOND TO/ADJUST TURBINE DURING FAST LOAD REDUCTION
01200001500	RESPOND TO UNIT TRIP
01200021500	RESPOND TO A LOSS OF COOLANT ACCIDENT
01200044500	RESPOND TO STEAM GENERATOR HIGH LEVEL

UPDATE LOG:	
NOTES:	Place this form with the working copies of lesson plans and/or other materials to document changes made between formal revisions. For fleet-wide training materials, keep electronic file of this form in same folder as approved materials. Refer to TR-AA-230-1003 SAT Development for specific directions regarding how and when this form shall be used.

Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / REVIEWER	DATE / DATE
0-0	Initial Revision	Revised for L-16-1 NRC Exam	2108338	Note 5 / Note 5	Note 5 / Note 5
0-1					
0-2					
0-3					
0-4					
0-5					

1. Individual updating lesson plan or training material shall complete the appropriate blocks on the Update Log.
2. Describe the change to the lesson plan or training materials.
3. State the reason for the change (e.g., reference has changed, typographical error, etc.)
4. Preparer enters name/date on the Update Log and obtains Training Supervisor approval.
5. Initial dates and site approval on cover page.

SCENARIO SUMMARY

Initial Conditions

The plant is at 75% power (MOL). Online risk is green. B train is protected on both units.

Equipment OOS

The 3A RHR pump and 3A1 Circulating Water pump are OOS.

Event 1

Shortly after taking the watch, LT-3-459, PZR Level Control Transmitter, fails high causing charging flow to reduce to the minimum and Pressurizer level to start trending down. The US will enter 3-ONOP-041.6, Pressurizer Level Control Malfunction, and direct the RCO to place Pressurizer Level Control Transfer Switch to position 3, CH.2&3. Once Pressurizer level is stabilized the US will enter 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels. The RCO will verify the Pressurizer Level Control Transfer Switch has been switched to position 3 and Pressurize level control is in automatic.

Event 2

After the US evaluates Tech Specs for LT-3-459, the Intake Screen differential level will start rising on all running Circulating Water Pumps. The US will enter 3-ONOP-011, Screen Wash System/Intake Malfunction. The screen differential level on the 3A2 screen will require the 3A2 Circulating Water Pump be secured. Prior to securing the 3A2 CWP Circulating Water Pump, Reactor power must be reduced to less than 60%. The crew will reduce power to less than 60% using 3-GOP-100, Fast Load Reduction, and then secure the 3A2 Circulating Water Pump.

Event 3

After the crew stops the 3A2 Circulating Water Pump, FT-3-474, 3A S/G Steam Flow Transmitter, will drift high. The BOP will take manual control of the 3A S/G level and restore the 3A S/G level to normal. The US will enter 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, direct the BOP to select an operable channel, and restore 3A S/G level control to automatic.

Event 4

Once the US completes the Tech Spec evaluation for FT-3-474, Pressurizer Pressure Control Transmitter, PT-3-444 fails low. The PZR Sprays will close and all PZR Heaters will turn on causing Pressurizer pressure to rise. The US will enter 3-ONOP-041.5, PZR Press Control Malfunction. The RCO will take manual control of PC-3-444J, PZR Press Controller, and restore PZR pressure to normal.

Event 5

After the crew restores Pressurizer Pressure, the crew will start the 3C Condensate pump and secure the 3A Condensate pump due to a sheared shaft on the 3A Condensate Pump.

Event 6

Once the crew completes swapping condensate pumps, a Large Break LOCA will occur. The crew will manually trip the Reactor and enter 3-EOP-E-0, Reactor Trip Or Safety Injection. When RCP Trip Criteria are met, the RCO will trip the RCPs.

SCENARIO SUMMARY

Event 7

When SI actuates, the 3B RHR pump fails to auto start. The RCO will start the 3B RHR pump following the Immediate Operator Actions of 3-EOP-E-0.

Event 8

When Phase A actuates, CV-3-2826 and CV-3-2819, IA Bleed Valves, fail to auto close. While performing 3-EOP-E-0 Attachment 3, Prompt Action Verifications the BOP manually closes CV-3-2826 however 3-CV-3-2819 is failed open and will not close in auto or manual.

The crew will transition from 3-EOP-E-0 to 3-EOP-E-1, Loss of Reactor or Secondary Coolant. During or shortly after the transition to 3-EOP-E-1, the crew will be required to go to 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, due to a RED path on the Integrity Status Tree. They will verify RHR flow greater than 1100 gpm, and then return to 3-EOP-E-1.

The scenario may be terminated after the crew transitions from 3-EOP-FR-P.1 to 3-EOP-E-1, or at the Lead Evaluator's discretion once all Critical Tasks have been evaluated.

CRITICAL TASKS		
Event #		**Description**
6/7	**CT1**	**Start 3B RHR Pump** During a Large Break LOCA start at least one RHR pump to provide core cooling to avoid transition to 3-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation. *Safety Significance* -- Failure to manually start at least one low-head ECCS pump prior to the transition to a contingency procedure constitutes misoperation or incorrect crew performance in which the crew does not prevent degraded emergency core cooling system capacity that may lead to or prolong core uncovery.
6/8	**CT2**	**Close CV-3-2826** During a Large Break LOCA close containment isolation valves such that at least one valve is closed on each critical Phase A penetration before whichever of the following occurs first: • The completion of 3-EOP-0 Attachment 3. • Within 60 minutes of the Phase A actuation signal. *Safety Significance* -- Failure to perform the critical task leads to an unnecessary release of fission products to the auxiliary building, increasing the potential for release to the environment and reducing accessibility to vital equipment within the auxiliary building. High radiation in the auxiliary building can lead to increased doses to personnel.

SEQUENCE OF EVENTS	
EVENT #	**DESCRIPTION**
1.	LT-3-459, PZR Level Transmitter, Fails High
2.	3A2 Intake Screen Blockage (Load reduction required)
3.	FT-3-474, 3A S/G Steam Flow Transmitter, Drifts High
4.	PT-3-444, PZR Pressure Transmitter, Fails Low
5.	3A Condensate Pump Sheared Shaft
6.	Large Break LOCA
7.	3B RHR Pump Fails To Auto Start
8.	CV-3-2826, Containment Isolation IA Bleed Valves, Fail To Auto Close CV-3-2819, Containment Isolation IA Bleed Valves, Fail Open

SIMULATOR SET UP INSTRUCTIONS	
Check	**Action**
_____	Restore IC-16 (75% MOL) or equivalent IC.
_____	Unfreeze the Simulator.
_____	Stop the 3A1 Circ Water Pump
_____	Open & execute lesson file L-16-1 N1.lsn
_____	Ensure the following lesson steps are triggered: • SETUP - 3A RHR PUMP OOS • SETUP - 3A1 CWP OOS • EVENT 7 SETUP - 3B RHR PUMP FAILS TO AUTO START • EVENT 8 SETUP - IA BLEED VLVS FAIL TO CLOSE
_____	Place 3A RHR pump in PTL and hang ECO tag.
_____	Place 3A1 CWP in stop and hang ECO tag.
_____	Verify the trend for 3A1 Screen on the TWS DP Recorder is clear.
_____	Ensure Rod Group Step Counters have completed stepping out.
_____	Allow the plant to stabilize.
_____	Acknowledge any alarms and freeze Simulator.
_____	Ensure B train is protected train on VPA.
_____	Perform the SIMULATOR OPERATOR CHECKLIST or equivalent.
_____	Place TURNOVER SHEETS on RO's desk or give to the Lead Evaluator.

BRIEFINGS

- Shift turnover information is attached to the back of this guide.
- Ensure all applicants are prior briefed on Appendix E of NUREG 1021, Policies and Guidelines For Taking NRC Examinations.
- Conduct a Crew Pre-brief to cover turnover information. Shift turnover information is attached to the back of this guide.

US: ______________________________

RCO: ______________________________

BOP: ______________________________

SCENARIO NOTE

0-ADM-211, Emergency and Off-Normal Operating Procedure Usage, Prudent Operator Actions.
If redundant stand-by equipment is available and ready, the operator is permitted to start the redundant equipment for failed or failing operating equipment. Immediate follow up of applicable ARPs and ONOPs (AOPs) shall occur as required.

Critical Tasks are highlighted in red.	**Simulator Operator Actions** are highlighted in blue.	**Operator Verifiable Actions** are Highlighted in green.

EVENT 1 - LT-3-459, PZR LEVEL TRANSMITTER, FAILS HIGH		
3-ONOP-041.6, PRESSURIZER LEVEL CONTROL MALFUNCTION.		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE Ensure the Simulator is in **RUN** before the crew enters the Simulator.	
		US: Conducts shift turnover.
	BOOTH OPERATOR When directed by the Lead evaluator **trigger EVENT 1 - LT-3-459 FAILS HIGH**	RCO: Observes LT-3-459 failed high
		BOP: • Acknowledges A8/3, A9/3, G1/1 • Reviews ARP and recommends entry into 3-ONOP-041.6, Pressurizer Level Control Malfunction.
		US: Directs 3-ONOP-041.6 response.
		RCO: • Check Pressurizer level indicators LI-3-459A, LI-3-460 AND LI-3-461 • Selects ch 2 & 3 PZR level control (Position 3) • Maintains PZR level on program per 3-ONOP-041.6, Enclosure 1 • May place Master Charging Pump Controller, LC-3-459G in manual • May Start or Stop one charging pump as required. • Place LR-3-459 Channel Select Pressurizer Level Recorder to position 2 or 3. **Steps 5.1 - 5.4**
		US: Marks Steps 5.5 – 5.7 N/A

EVENT 1 - LT-3-459, PZR LEVEL TRANSMITTER, FAILS HIGH		
3-ONOP-041.6, PRESSURIZER LEVEL CONTROL MALFUNCTION.		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: WHEN desired place LC-3-459G in Automatic **Step 5.8**
		US: Perform actions required by 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels. **Step 5.9**

EVENT 1 - LT-3-459, PZR LEVEL TRANSMITTER, FAILS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNELS		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
		US: Enters and directs actions of 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, for response
		RCO: • Verify LT-459 failure by comparison LT-3-460/461 and known plant parameters and conditions • Verify no off-normal conditions exist on LT-3-460/461 • Verify ch 2 & 3 PZR level control in (Position 3) • Verify LR-3-459 Channel Selected to Pressurizer Level Recorder to position 2 or 3 • Verify PZR level control function is returned to automatic. **Steps 5.1 – 5.5**
		US Reviews TECH Specs • Tech Spec 3.3-1 Functional Unit 9 not met. - Action 13, inoperable channel must be placed in the tripped condition within 6 hours.
		US: Marks steps 5.7 – 5.11 N/A.
		US: Identify Bistables which need to be tripped. **Step 5.12**
		US: Marks steps 5.13 – 5.16 N/A.

EVENT 1 - LT-3-459, PZR LEVEL TRANSMITTER, FAILS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNELS		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR WCC/I&C: Acknowledge the report. If asked I&C would like to be present when bi-stables are tripped. They will be in the control room in about 1 hour.	US: Initiate a Plant Work Order AND notify the I&C Supervisor. **Step 5.17**
	LEAD EVALUATOR After the PZR level control is restored to auto and the US completes a review of Tech Specs, proceed to the next event at the Lead Evaluators discretion	US: • Conducts crew brief.

EVENT 2 - 3A2 INTAKE SCREEN BLOCKAGE		
3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the Lead evaluator **trigger EVENT 2 - INTAKE SCREEN BLOCKAGE**	
	NOTE The crew may enter 3-ONOP-011 prior to receiving the alarm. **BOOTH OPERATOR** If asked to rake the grizzly screen report the rake is stuck. Maintenance support has been requested. **NOTE** The crew may swap ICW pumps.	BOP: • Addresses Alarm Response for I3/3 • CHECKs Intake Well DPs greater than 25" water at VPA. • Dispatches an Operator to check the following: - Traveling Screens with HI DP are running in FAST. - Operating Traveling Screens Spray Wash is maintaining the screen clear of debris. - Recommends entering 3-ONOP-011, Screen Wash System / Intake Malfunction.
		US: Enters and directs the actions of 3-ONOP-011.
		US: Reviews Foldout page with the crew. • Circulating Water Pump Stopping Criteria • Fast Load Reduction Criteria • Reactor Trip Criteria • Loss of Intake Cooling Water • Plant Management Notification • Shift Manager Evaluation of Intake Screen Effectiveness • Amertap Screen Debris Monitoring **Foldout Page**

EVENT 2 - 3A2 INTAKE SCREEN BLOCKAGE		
3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When dispatched to locally start traveling screens, wait 2 minutes and then **trigger EVENT 2 LOA - START ALL PUMPS AND ALL SCREENS IN FAST** **BOOTH OPERATOR** When the screens start, verify **EVENT 2 - REDUCE SCREEN BLOCKAGE triggers.** **BOOTH OPERATOR** If asked all traveling screen equipment is operating normally.	BOP: Direct ANPO to: • Check Traveling Screens Operating Properly • Check Traveling Screen Spray Wash Pumps Operating Properly **Steps 1 & 2**
	BOOTH OPERATOR When dispatched to locally check ICW to CCW and TPCW heat Exchanger or strainers, wait 1-3 min, report all ICW/ CCW ICW / TPCW Strainers < 1 ΔP. If a specific value is requested use simulator drawings **COMMON SERVICES/INTAKE COOLING** and **COMMON SERVICES/TURBINE PLANT COOLING** to report actual values.	BOP: Directs SNPO: • Maintain Intake Cooling Water Flow To Component Cooling Water Heat Exchangers. • Maintain Intake Cooling Water To The Turbine Plant Cooling Water Heat Exchangers **Steps 3 & 4**
	BOOTH OPERATOR When I3/3,Traveling Screen High ΔP alarms, verify **EVENT 2 - STABILIZE SCREEN BLOCKAGE triggers** Monitor screen Δ level on the TWS DP recorder. If needed use the active malfunction summary page to adjust **TAKPXA2** to maintain screen Δ level less than 30 inches.	BOP: Check If Conditions Returned To Normal. **(NO, Go to Step 7)** **Step 5**
	BOOTH OPERATOR Acknowledge request for additional support.	BOP: • Contact WCC and Maintenance for additional support.

EVENT 2 - 3A2 INTAKE SCREEN BLOCKAGE		
3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR If asked to locally check level drop behind Traveling Screens, use the Simulator Panel drawings to report the same level displayed on the control room chart recorders.	BOP: Check If One Circulating Water Pump Should Be Stopped. • Locally check that level drop behind Traveling Screens is greater than 3 feet. • VPA ΔP greater than 36 inches. **Step 7**
		US: • Read Caution: - Three Circulating Water Pumps shall remain in service if Unit Load is greater than 60 percent. • Recognize that a Fast Load reduction will be required to secure the 3A2 Circulating Water Pump. **Step 8**
	BOOTH OPERATOR Acknowledge notifications If Chemistry is asked, request the crew maintain current blowdown flow.	US: Notify Plant Management.

EVENT 2 - 3A2 INTAKE SCREEN BLOCKAGE		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: • Directs actions to reduce Rx power from 50% per 3-GOP-100. • Completes Attachment 3 • Brief the crew per Attachment 4 **Steps 1-2**
		US: Reviews Foldout page with crew. • 3-EOP-E-0 Transition Criteria • Notify Chemistry Department • Boration Stop Criteria • Restore Blender to AUTO **FOLDOUT PAGE**
	BOOTH OPERATOR Acknowledge notifications.	BOP: Notify The Following Of Fast Load Reduction • System Dispatcher • Plant personnel using the Page Boost • Chemistry to start RCS sampling is required according to Tech Spec Table 4.4-4. **Step 3**

EVENT 2 - 3A2 INTAKE SCREEN BLOCKAGE		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Begin Boration For Initial Tavg Effect • Set the Boric Acid Totalizer to total boric acid volume value determined on Attachment 3. • Place the Reactor Makeup Selector Switch to BORATE. • Place the RCS Makeup Control Switch to START. • Adjust FC-3-113A, Boric Acid Flow Controller to achieve 40 gpm boric acid flow as indicated on FR-3-113. • WHEN Tavg begins to lower from the boration, THEN, adjust FC-3-113A, Boric Acid Flow Controller to load reduction value from Attachment 3. **Step 4**
		US: Determine Turbine Load Reduction in MW CNTRL **Step 5**
		BOP: Initiate Turbine Load Reduction in MW CNTRL • Select MW CNTRL • Set TARGET power level – MW VALUE from Attachment 3 • Set RAMP RATE – MW/M VALUE FROM Attachment 3. • Check T_{avg} has lowered 1° to 2°F from the initial value prior to boration. • Depress GO • Ensure FC-3-113A, Boric Acid Flow Controller, has been adjusted to the load reduction boration rate. Go to Step 10 **Step 6**

EVENT 2 - 3A2 INTAKE SCREEN BLOCKAGE		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Monitor Load Reduction • Adjusts power reduction rate to maintain T_{avg}/T_{ref} within limits of Attachment 3. • Monitors S/G level control to ensure feed reg valves properly maintain level control in automatic. • Refer to Enclosure 1 for expected alarms. **Step 10**
	BOOTH OPERATOR Respond as SNPO. If asked, idle Charging Pump ready for start.	RCO: • Maintain pressurizer level to ensure that automatic pressurizer level control maintains level on program. • If needed starts 2nd Chg Pp and places 2nd orifice in service. • Adjusts boration rate to maintain Tavg/Tref within limits of Attachment 3. • Refer to Enclosure 1 for expected alarms. **Step 10**
		RCO: Monitor Boration Rate • Monitor for excessive rod movement by monitoring TR-3-409D, Rod Position Bank D. • Determine if Insertion Limit and Bank D position are converging at a rate that will cause rod insertion limit alarms. • Adjust power reduction rate as needed to control rod insertion • Increase boration rate and/or total amount as necessary to limit control rod insertion **Step 11**

EVENT 2 - 3A2 INTAKE SCREEN BLOCKAGE		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: • Monitor Annunciator B 8/1, ROD BANK LO LIMIT – CLEAR • Monitor B 8/2 ROD BANK A/B/C/D EXTRA LO LIMIT – CLEAR **Steps 12-13**
		US: Have SM refer to the following procedures: • 0-EPIP-20101, DUTIES OF EMERGENCY COORDINATOR • 0-ADM-115, NOTIFICATION OF PLANT EVENTS **Step 14**
	<u>BOOTH OPERATOR</u> Acknowledge notifications If Chemistry is asked, request the crew maintain current blowdown flow.	RCO: Energize Pressurizer Backup Heaters **Step 15**
	<u>BOOTH OPERATOR</u> When directed to close 3-30-002 and 3-30-004 per NOP-3-010, wait 1 – 2 minutes, then **trigger LOA - CLOSE 3-30-002 AND 3-30-004.** Report when complete. **<u>BOOTH OPERATOR</u>** When dispatched to monitor the 3A2 CWP discharge MOV, report it fully closed after the pump stopped. All checks are SAT.	BOP: When power is reduce to <60%, stops the 3A2 CWP In Accordance With 3-NOP-010, Circulating Water System.
	<u>LEAD EVALUATOR</u> Once power the 3A2 CWP is shutdown, or at the Lead Evaluators discretion, proceed to the next event.	

EVENT 3 – FT-3-474, 3A S/G STEAM FLOW TRANSMITTER, DRIFTS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNELS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the lead evaluator, **trigger EVENT 4 - FT-3-474 DRIFTS HIGH**	BOP: Recognizes and reports FT-3-474 is failing high. • Takes manual control of 3A S/G level control valve FCV-3-478. • Restores 3A S/G level to normal.
	NOTE The crew may use direction in the ARP to select alternate input signals and return 3A S/G level control to automatic before enter 3-ONOP-049.1.	RCO: Addresses Alarm Response for C4/1, C5/1, C6/1 and C7/1. • Ensures BOP takes Prompt Actions - Take manual control of level. - Return SG levels to normal. • Checks if alarm is due to instrument failure, then refers to 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
		US: Enters and directs actions of 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, for response
	BOOTH OPERATOR If dispatched to reset AMSAC, wait 3 to 5 minutes and then **trigger EVENT 5 LOA - RESET AMSAC.** Report when complete.	BOP: • Verify FT-3-474 failure by comparison to FT-3-475 and known plant parameters and conditions. • Verify no off-normal conditions exist on FT-3-475 • Place 3A S/G Steam Flow Control transfer switch to FT-3-475. (Yellow) • Place 3A S/G Feed Water Flow Control transfer switch to FT-3-476.(Yellow) • When 3A S/G level is returned to normal place FCV-3-478 level control valve in auto. **Steps 5.1- 5.5**

EVENT 3 – FT-3-474, 3A S/G STEAM FLOW TRANSMITTER, DRIFTS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNELS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR Acknowledge notifications and request for additional support. If asked, I&C would like to be resent when the bi-stables are tripped. They will be in the control room as soon as their work package is ready, about an hour.	US Reviews Tech Specs • Tech Spec 3.3-1 Functional Unit 12 not met. - Action 6, inoperable channel must be placed in the tripped condition within 6 hours. • Tech Spec 3.3-2 Functional Unit 1f and 4d not met. - Action 15, inoperable channel must be placed in the tripped condition within 6 hours. **Step 5.6**
		US: Marks steps 5.7 – 5.11 N/A.
		US: Identify Bistables which need to be tripped. **Step 5.12**
		US: Marks steps 5.13 – 5.16 N/A.
		US: Initiate a Plant Work Order AND notify the I&C Supervisor. **Step 5.17**
	LEAD EVALUATOR After S/G level control is restored to auto and the Tech Spec review is complete, at the Lead Evaluators discretion, proceed to the next event.	US • Conducts crew brief.

EVENT 4 – PT-3-444, PZR PRESSURE TRANSMITTER FAILS LOW		
3-ONOP-041.5, PRESSURIZER PRESSURE CONTROL MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 4 - PT-3-444 FAILS LOW**	
		RCO: Reports PZR pressure rising due to PI-3-444 failing low.
	NOTE The RCO will most likely notice this failure before Alarm A9/2 comes in. The crew may use the ARP for guidance.	BOP: Reviews ARP for A9/2, PZR CONTROL HI/LO PRESS • Check PZR pressure less than 2235 psig **(NO)** • Check PI-3-445/444, PZR pressure greater than 2300 psig or Less than 2185 psig. • Refer to 3-ONOP-041.5, PZR Press Control Malfunction.
		US: Directs the Action of 3-ONOP-041.5, Pressurizer Pressure Control Malfunction.
		US: Reviews the Foldout Page with the crew. • Failed Instrument Isolation • 3-EOP-E-0 Transition Criteria • PORV Isolation/Leaking PORV Identification • Open/Leaking PZR Safety Valve Identification • Spurious Actuation Of CV-3-311, Auxiliary Spray Valve

EVENT 4 – PT-3-444, PZR PRESSURE TRANSMITTER FAILS LOW		
3-ONOP-041.5, PRESSURIZER PRESSURE CONTROL MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: • Check PT-3-444 - NOT FAILED **(NO)** - Verify PCV-3-455C OR MOV-3-536 CLOSED. - Take manual control of PC-3-444J, PZR PRESS CONTROL. • Check PT-3-445 - NOT FAILED **Step 1**
		RCO: • Check PORVs Closed • PZR pressure normal or trending to normal. • Check PZR Safety Valves Closed • Check PZR Pressure Stable Or Increasing • Check Pressurizer Pressure Above Normal Value **(NO – Go to Step 10)** **Step 2 - 6**
		RCO: Check Pressurizer Pressure Low Or Decreasing **(NO- Go to Step 20)** **Step 10**
		RCO: Check RCS Pressure Stable **Step 20**
	<u>BOOTH OPERATOR</u> Acknowledge notifications and request for support. **<u>Lead Evaluator</u>** After PZR Pressure is stable, then proceed with the next event.	RCO: Check If Automatic Pressure Control Can Be Established **(NO)** • Notify the Instrument and Controls Department. • Continue efforts to establish Automatic Pressure Control. • Return to Step 20 **Step 21**

EVENT 5 – 3A CONDENSATE PUMP SHEARED SHAFT		
3-NOP-073, CONDENSATE SYSTEM		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 5- 3A CONDENSATE PUMP SHEARED SHAFT.**	
		RCO: Reviews Alarm G8/3, COND PUMP A LO FLOW
	BOOTH OPERATOR If dispatched to check the 3A Condensate pump, wait 2- 3 minutes and then report you see nothing wrong with the pump.	BOP: • Check Condensate Pump flow indication on DCS. • Monitor feed pump suction pressure. • Report low amps on 3A Condensate pump and high amps on the 3B Condensate pump.
		US: Direct BOP to start the 3C Condensate pump and Secure the 3A Condensate pump per 3-NOP-073.
	BOOTH OPERATOR If asked, the 3C Condensate Pump is ready for a start. After the pump is started, report a SAT start.	BOP: Dispatch an operator to verify the 3B Condensate pump is ready to start per 3-NOP-073. • Start the 3C Condensate Pump. • Secure the 3A Condensate Pump.
	Lead Evaluator After the Condensate pumps are swapped, proceed to the next event.	US • Notifies WCC to initiate PWO and I&C for troubleshooting. • Conducts crew brief.

EVENT 6 – LARGE BREAK LOCA		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the Lead Evaluator **trigger EVENT 6 - LB LOCA**	
		RCO: Responds to various alarms PZR LO Pressure and level Alarms A9/2, A9/3. • Reports lowering Pressurizer and pressure and level. • Maximizes Charging • Isolates Letdown • When PZR cannot be maintained recommends a Reactor trip.
	NOTE Steps 1 - 4 of 3-EOP-E-0 are Immediate Operator Actions (IOAs). The board operators will call out the high level steps of the IOAs as each step is completed from memory.	US: Directs RCO to manually trip the Reactor.
		RCO: • Manually trips Reactor.
		RCO/BOP: Perform IOAs of 3-EOP-E-0.
		RCO: Verify Reactor Trip **STEP 1**
		BOP: Verify Turbine **STEP 2**
		BOP: Verify Power To Emergency 4 KV Buses **STEP 3**
		RCO: Checks If SI Is Actuated **STEP 4**
		US: Reviews Steps 1 - 4 of 3-EOP-E-0 with the crew.

EVENT 6 – LARGE BREAK LOCA		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE The crew may take Prudent Operator Actions to start the 3B RHR pump and/or close Containment Phase A Isolation valve CV-3-2826 or they may wait until directed by 3-EOP-E-0 Attachment 3. 3-CV-3-2819 is failed open and will not close in auto or manual.	
CT1	**Start 3B RHR Pump** During a Large Break LOCA, start at least one RHR pump to provide core cooling prior to transition to 3-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation.	RCO: • Starts the 3B RHR pump
CT2	**Close CV-3-2826** During a Large Break LOCA, close containment isolation valves such that at least one valve is closed on each critical Phase A penetration before whichever of the following occurs first: • The completion of 3-EOP-0 Attachment 3 • Within 60 minutes of the Phase A actuation signal.	BOP: • Close CV-3-2826
		US: Reviews FOP for 3-EOP-E-0 • Adverse Cntmt **(Met)** • RCP Trip Criteria **(Met)** – Trips RCPs once met. • Faulted S/G Isolation • Ruptured S/G Isolation • AFW Sys Operation Criteria • CST Makeup Water Criteria • RHR System Operation Criteria • Loss of Offsite Power or SI on the Other Unit • Loss of Charging Criteria **FOLDOUT PAGE**

EVENT 6 – LARGE BREAK LOCA		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE The actions of Attachment 3 are listed beginning on page 37.	BOP: • Continues with ATTACHMENT 3 to complete The Prompt Action Verifications. **STEP 5**
		RCO: • Check AFW Pumps – AT LEAST TWO RUNNING **STEP 6**
		RCO: • Verify AFW Valve Alignment – PROPER EMERGENCY ALIGNMENT **STEP 7**
		RCO: Verify Proper AFW Flow: • Check Narrow Range Level in at least one S/G – GREATER THAN 7%[27%] • Maintain feed flow to S/G until Narrow Range Levels between 21%[27%] and 50% **STEP 8**
		RCO: All RCP Thermal Barrier Alarms – CLEAR **(NO)** • Trip RCPs (RCPs tripped per Foldout Page.) • Check All RCP CBO temperatures – LESS THAN 260°F • SI – RESET • Start one Charging Pump at minimum speed for Seal Injection. • Adjust HCV-3-121, Charging Flow To Regen Heat Exchanger, to maintain proper Seal Injection flow **STEP 9**

EVENT 6 – LARGE BREAK LOCA

3-EOP-E-0, RX TRIP OR SAFETY INJECTION

TIME	EVALUATOR ACTIVITIES & NOTES	EXPECTED STUDENT RESPONSE
		RCO: • Check RCS Temperatures: – Check RCPs – ANY RUNNING **(NO)** – Check RCS Cold Leg temperatures stable between 545°F and 547°F or trending down to 547°F **(NO)** • IF T_{COLD} is decreasing, THEN perform the following: – Stop dumping steam. – IF cooldown continues AND is due to excessive feed flow, then reduce total feed flow to 400 gpm until Narrow Range Level greater than 7%[27%] in at least one S/G. – IF cooldown continues AND is due to excessive steam flow, THEN close Main Steamline isolation and Bypass valves. **STEP 10**
		RCO: Check PRZ PORVs, Spray Valves And Excess Letdown Isolated: **STEP 11**
		RCO: Check If RCPs Should Be Stopped: • RCPs – ANY RUNNING **(NO)** **STEP 12**
		RCO: Check If S/Gs Are Faulted: **(NO)** **STEP 13**
		RCO: Check If S/G Tubes Are Ruptured: **(NO)** **STEP 14**

EVENT 6 – LARGE BREAK LOCA		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Check If RCS Is Intact **(NO)** US: • Perform the following: – Monitor Critical Safety Functions using 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES. – INTEGRITY Critical Safety Function Status Tree is RED – Go to 3-EOP-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION **STEP 15**
		US: Reviews the Fold Page of 3-EOP-FR-P.1 • ADVERSE CONTAINMENT CONDITIONS
		RCO: • Check RCS Pressure – GREATER THAN 275 PSIG [575 PSIG] **(NO)** • Check RHR flow greater than1100 gpm.
		US: • Returns to 3-EOP-E-0, step 15. • Go to 3-EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1

EVENT 6 – LB LOCA		
3-EOP-E-1, Loss Of Reactor Or Secondary Coolant		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR If dispatched to start the B Standby SGFP, wait 3 minutes and then **trigger START B STANDBY SGFP**.	US: • Conducts EOP transition brief. • Directs 3-EOP-E-1 response.
		US: Reviews FOP for 3-EOP-E-1 with the crew. • Containment Adverse **(YES)** • RCP Trip Criteria - **Tripped** • SI Termination Criteria • Secondary Integrity Criteria. • E-3 Transition Criteria • Cold Leg Recirculation Switchover Criteria. (met < 155k) • Recirculation Sump Blockage. • CST Makeup Water Criteria. • Loss of Offsite Power or Unit 4 SI • RHR Sys Operation Criteria • Loss Of Charging Criteria **FOLDOUT PAGE**
		RCO: Check If RCPs Should Be Stopped **(tripped)** **STEP 1**
		RCO: Check If S/Gs Are NOT Faulted. **STEP 2**

EVENT 6 – LB LOCA		
3-EOP-E-1, Loss Of Reactor Or Secondary Coolant		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Check Intact S/G Levels: • Any Narrow Range Level Greater Than 7%[27%]. - Maintain total feed flow greater than 400 gpm until Narrow range Level greater the 7% [27%] in at least one S/G. • Control feed flow to maintain Narrow Range Level between 21%[27%] and 50%. • Narrow Range Level Less Than 50%. **STEP 3**
	BOOTH OPERATOR Acknowledge the request for Chemistry and HP support	RCO: Check Secondary Radiation: **STEP 4**
		RCO: Checks PRZ PORVs And Block Valves: **STEP 5**
		RCO: Check SI – RESET **STEP 6**
		RCO: Resets Containment Isolation Phase A and Phase B. **STEP 7**
		RCO: Verify Instrument Air To Containment **STEP 8**
		RCO: Check Power Supply To All Charging Pumps - ALIGNED TO OFFSITE POWER **STEP 9**

EVENT 6 – LB LOCA		
3-EOP-E-1, Loss Of Reactor Or Secondary Coolant		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Check If Charging Flow Has Been Established. • Charging pumps - AT LEAST ONE RUNNING **(NO)** • Establish desired charging by performing ATTACHMENT 2, steps 3 through 5. - Place RCS Makeup Control Switch in STOP - Start additional Charging pumps if needed. - Adjust Charging Flow To Regen Heat Exchanger, HCV-3-121, to maintain proper seal injection flow. - Verify charging pump suction auto transfers to RWST. - Notify Unit Supervisor That Attachment 2 Is Complete. **STEP 10**
		US: Check if SI Flow Should Be Terminated **(NO)** **STEP 11**
		RCO: Check if Containment Spray should be stopped. **(NO)** **STEP 12**
		RCO: Check If RHR Pumps Should Be Stopped. **(NO)** **STEPS 13**

EVENT 6 – LB LOCA		
3-EOP-E-1, Loss Of Reactor Or Secondary Coolant		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO\BOP: Check RCS And S/G Pressures • Pressure in all S/Gs – STABLE OR INCREASING • RCS pressure STABLE OR DECREASING **STEP 14**
	BOOTH OPERATOR If directed to Stop 4A and 4B EDG, acknowledge request. **BOOTH OPERATOR** If dispatched to place any stopped EDGs in standby, acknowledge request.	BOP: Check If Diesel Generators Should Be Stopped: • Stop 3A and 3B EDG by placing its Normal Stop/Normal Start switch in NORMAL STOP position. • Direct Unit 4 RCO to stop any unloaded diesel generator by placing its Normal Stop/Normal Start switch in NORMAL STOP position. • Dispatch Operator to place any stopped EDGs in standby using 3/4-OP-023, EMERGENCY DIESEL GENERATOR **STEP 15**
	BOOTH OPERATOR If dispatched to unlock and close cold leg recirc breakers, wait 2 to 3 minutes then **trigger LOA - ENERGIZE TRAIN A SI RECIR MOVS and LOA - ENERGIZE TRAIN B SI RECIR MOVS**	US: Initiate Evaluation Of Plant Status **STEP 16**
The scenario may be terminated after the crew transitions from 3-EOP-FR-P.1 to 3-EOP-E-1 at the Lead Evaluator's discretion once all Critical Task have been evaluated.		
***** END OF SCENARIO *****		

EVENT 8 – CV-3-2826/2819, CONTAINMENT ISOLATION IA BLEED VALVES, FAIL TO AUTO CLOSE		
3-EOP-E-0 ATTACHMENT 3, PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Check Load Centers Associated With Energized 4 KV Buses – ENERGIZED **STEP 1**
		BOP: Verify Feedwater Isolation: **STEP 2**
	NOTE The crew may have taken Prudent Operator Actions to close CV-3-2826 after SI Actuated. 3-CV-3-2819 is failed open and will not close in auto or manual.	BOP: Check If Main Steam Lines Should Be Isolated **STEP 3**
CT2	**Close CV-3-2826** During a Large Break LOCA, close containment isolation valves such that at least one valve is closed on each critical Phase A penetration before whichever of the following occurs first: • The completion of 3-EOP-0 Attachment 3 • Within 60 minutes of the Phase A actuation signal.	BOP: Verify Containment Isolation Phase A Valve White Lights On VPB – ALL BRIGHT **(NO)** • Manually actuate Containment isolation Phase A. • Close CV-3-2826 **STEP 4**

EVENT 7 – 3B RHR PUMP FAILS TO AUTO START		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
CT1	**<u>NOTE</u>** The crew may have taken Prudent Operator Actions to start the 3B RHR pump after SI actuated. **<u>Start 3B RHR Pump</u>** During a Large Break LOCA start at least one RHR pump to provide core cooling prior to transition to 3-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation.	BOP: Verify Pump Operation: • At least two High-Head SI Pumps – RUNNING • Both RHR Pumps – RUNNING **(NO)** • Starts the 3B RHR pump **STEP 5**
		BOP: Verify Proper CCW System Operation: • CCW Heat Exchangers – THREE IN SERVICE • CCW Pumps – ONLY TWO RUNNING • CCW Headers – TIED TOGETHER • MOV-3-626, RCP Thermal Barrier CCW Outlet – OPEN **(NO Phase B)** **STEP 6**
		BOP: Verify Proper ICW System Operation: • Verify ICW Pumps – AT LEAST TWO RUNNING • Verify ICW To TPCW Heat Exchanger – ISOLATED: • Check ICW Headers – TIED TOGETHER **STEP 7**
		BOP: Check Emergency Containment Coolers – ONLY TWO RUNNING **STEP 8**

EVENT 7 – 3B RHR PUMP FAILS TO AUTO START		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Verify Unit 3 Containment Purge Exhaust And Supply Fans – OFF **STEP 9**
		BOP: Verify Containment Spray and Phase B actuated. **STEP 10**
		BOP: Verify SI – RESET **STEP 11**
		BOP: Verify SI Valve Amber Lights On VPB – ALL BRIGHT **STEP 12**
		BOP: Verify SI Flow: • RCS pressure – LESS THAN 1625 PSIG[1950 PSIG] • High-Head SI Pump flow indicator – CHECK FOR FLOW **STEP 13**
	BOOTH OPERATOR When requested**, trigger LOA – ALIGN U-4 HHSIs TO U3 RWST**	BOP: Realign SI System: • Check Procedure Entry Status – E-0 ENTERED FROM 3-ONOP-047.1, LOSS OF CHARGING FLOW IN MODES 1 THROUGH 4 **(NO)** • Verify Unit 3 High-Head SI Pumps – TWO RUNNING • Stop both Unit 4 High-Head SI • Pumps and place in standby • Direct Unit 4 Reactor Operator to align Unit 4 High-Head SI Pump suction to Unit 3 RWST using Attachment 1. **STEP 14**

EVENT 7 – 3B RHR PUMP FAILS TO AUTO START		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Verify Containment Isolation Phase A – RESET **STEP 15**
		BOP: Reestablish RCP Cooling: • Check RCPs – AT LEAST ONE RUNNING **(NO)** • Go to Step 17 **STEP 16**
		BOP: Verify Control Room Ventilation Isolation: **STEP 17**
	<u>BOOTH OPERATOR</u> When requested**, trigger LOA – PLACE PAHM IN SERVICE,** wait 3 to 5 minutes and then report task complete.	BOP: Place Hydrogen Monitors In Service Using 3-NOP-094, CONTAINMENT POST ACCIDENT MONITORING SYSTEM **STEP 18**
		BOP: Verify All Four EDGs – RUNNING **STEP 19**
		BOP: Verify Power To Emergency 4 KV Buses: **STEP 20**
		BOP: Notify Unit Supervisor Of The Following: • Attachment 3 is complete • Any safeguards equipment that is NOT running is in the required condition • Status of Containment pressure continuous action **STEP 21**

Discussion Points are intentionally NOT included in evaluated scenarios. However, space is available below to document follow-up questions when further information is required to determine an evaluation outcome.

FOLLOW-UP QUESTIONS

QUESTION #1

ANSWER #1

QUESTION #2

ANSWER #2

SIMULATOR POST-SCENARIO RESTORATION:

_______ 1. Restore per Simulator Operator Checklist.

_______ 2. Once exams are complete, restore from SEI-19, Simulator Exam Security.

OPERATIONS SHIFT TURNOVER REPORT

UNIT 3 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B	UNIT 4 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B

ONCOMING CREW ASSIGNMENTS

Shift Mgr:			**Inside SNPO:**	
Field Supv.:			**Outside SNPO:**	
Admin RCO:			**ANPO:**	
Unit 3			**Unit 4**	
Unit Supv.:			**Unit Supv.:**	
RCO:			**RCO:**	
NPO:			**NPO:**	

PLANT STATUS

Unit 3			**Unit 4**	
Mode:	1		**Mode:**	1
Power:	75%		**Power:**	100%
MWe:	608		**MWe:**	842
Gross Leakrate:	.22 gpm		**Gross Leakrate:**	0.03 gpm
RCS Boron Conc:	828 ppm		**RCS Boron Conc:**	642

Operational Concerns:

3A RHR pump taken OOS 4 hours ago for an oil change, expected back by the end of this shift.
3A1 Circ Water pump OOS. Tripped on over current, Electrical Maintenance is investigating.
3A Condensate pump was returned to service last shift following a motor bearing replacement.
Return to full power expected next shift.

U3 Anticipated LCO Actions:

None

U4 Anticipated LCO Actions:

None

Results of Offgoing Focus Area:

UNIT 3 STATUS

REACTOR OPERATOR

UNIT RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B

Mode:	1
Power:	75%
MWe	608
Tavg:	571°F
RCS Pressure:	2235
RCS Boron Conc:	828 ppm

RCS Leakrate	
Gross:	0.22 GPM
Unidentified	0.04 GPM
Charging Pps:	0.00 GPM

Accumulator Ref Levels	
A	6656
B	6608
C	6646

Abnormal Annunciators:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	

Current Tech Spec Action Statements: (Does Not Include "For Tracking Only Items"	
T.S.A.S / Component: Reason: Entry Date:	3A RHR pump, 3.5.2.c – Action g Oil Change 4 hours ago
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	

REACTOR OPERATOR (CONT'D)
UNIT RISK: GREEN (ACCEPTABLE)   PROTECTED TRAIN: B
Changes to Risk Significant Equipment:
No recent changes from last shift. OLRM: GREEN PROTECTED TRAIN: B
Upcoming Reactivity Management Activities:
Maintain current power level ± .5% Xe is stable.
Upcoming Major POD Activities:
NONE
Upcoming ECOs to Hang and /or Release:
• Hang – None • Release – None
Evolutions or Compensatory Actions in Progress:
NONE
General Information, Remarks, and Operator Work Around Status:
• Weather forecast is overcast skies with scattered pockets of severe rain. • U3 supplying Aux Steam • Air In-leakage = 0.0 SCFM

Site:	**Turkey Point Units 3 and 4 (PTN)**
Title:	**L-16-1 AUDIT EXAM SCENARIO 2**
LMS #:	**NRC 22**
LMS Rev Date:	**6/7/16 Rev #: 0.0**
SEG Type:	☐ **Training** ☒ **Evaluation**
Program:	☐ **LOCT** ☒ **LOIT** ☐ **Other**
Duration:	**110 minutes**

Developed by:	**Brian Clark** Instructor/Developer	**6/13/16** Date
Reviewed by:	**Tim Hodge** *Instructor (Instructional Review)*	**6/22/16** *Date*
Validated by :	**Rocky Schoenhals** *SME (Technical Review)*	**6/22/16** *Date*
Approved by:	**Mark Wilson** *Training Supervision*	**6/22/16** *Date*
Approved by:	**Rocky Schoenhals** *Training Program Owner (Line)*	**6/22/16** *Date*

SCENARIO REFERENCES		
DOC NO.	**TITLE**	**REV**
	PTN TECHNICAL SPECIFICATIONS	298
3-EOP-E-0	REACTOR TRIP OR SAFETY INJECTION	12
3-GOP-100	FAST LOAD REDUCTION	12
3-ONOP-028	REACTOR CONTROL SYSTEM MALFUNCTION	4
3-ONOP-046.4	MALFUNCTION OF BORON CONCENTRATION CONTROL SYSTEM	0
3-ONOP-049.1	DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNEL	4
3-ONOP-059.8	POWER RANGE NUCLEAR INSTRUMENTATION MALFUNCTION	0A
3-ONOP-089	TURBINE RUNBACK	1

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
Terminal Objective	Given this simulator scenario and resources normally found in the Control Room, the operating crew will perform Control Room operations IAW approved plant procedures in order to maintain the integrity of the plant and the health and safety of the public.
Enabling Objectives:	Given this simulator scenario and resources normally found in the Control Room, operate in accordance with approved plant procedures, Operations Department Instructions, and management expectations: 1. (ALL) Demonstrate personnel SAFETY awareness in interactions with plant staff and outside agencies. 2. (ALL) Demonstrate ALARA awareness in interactions with plant staff and outside agencies. 3. (ALL) Exchange correct information using 3-point communication/Repeat-backs with Control Room personnel and other plant staff. 4. (ALL) Inform plant personnel and System of plant conditions, as needed. 5. (US) Employ timely and concise crew briefs where appropriate. 6. (ALL) Maintain awareness of plant status and control board indication. 7. (ALL) Correctly diagnose plant situations. 8. (ALL) Solve operational problems as they arise. 9. (RCO/BOP) Manipulate plant controls properly and safely. 10. (ALL) Demonstrate self-checking using STAR and peer checks(when required) 11. (US) Demonstrate command and control of the crew. 12. (US) Coordinate the input of crew members and other plant staff. 13. (US) Utilize the input of crew members and other plant staff. 14. (ALL) Demonstrate conservative decision making. 15. (ALL) Demonstrate teamwork. 16. (ALL) Respond to plant events using procedural guidance (OPs/ONOPs/EOPs) as applicable in accordance with rules of usage. 17. (RCO/BOP) Implement any applicable procedural immediate operator actions without use of references. 18. (SRO) Maintain compliance with Tech Specs. 19. (ALL) Identify/enter applicable Tech Spec action statements. 20. (ALL) Respond to annunciators using ARPs (time permitting). 21. (ALL) Maintain written communication, logs, and documentation as needed to permit post-event reconstruction. Continued on next page

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
	While addressing the following events: 1. LT-3-115 VCT Level Transmitter Fails Low. 2. 3A Heater Drain Pump (Turbine Runback) Rods Fail To Auto Insert 3. N-42 Loss Of Instrument Power 4. 3B S/G Feedwater pump High Vibration (Fast Load Reduction required) 5. FT-3-494 3C S/G Steam Flow Transmitter Fails As Is 6. Common Main Feed Header Break Common Loss Of Suction To All AFW Pumps 7. Main Turbine Fails To Automatically Trip 8. PCV-3-455C Fails To Open (PZR PORV)
Prerequisites:	None
Training Resources:	PTN Unit 3 Plant Simulator
Development References:	• TR-AA-220-1003, Initial NRC and Audit Exam Process • TR-AA-230-1003, SAT Development • TR-AA-230-1007, Conduct of Simulator Training and Evaluation • 0-ADM-232, Time Critical Action Program • OP-AA-100-1000, Conduct Of Operations • OP-AA-103-1000, Reactivity Management • 0-ADM-200, Operations Management Manual • 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage • WCAP-17711-NP, Pressurized Water Reactor Owners Group Westinghouse Emergency Response Guideline Revision 2-Based Critical Tasks
Protected Content:	N/A
Evaluation Method:	Performance Mode
Operating Experience:	None
Risk Significant Operator Actions:	Initiate bleed-and-feed cooling within 36 minutes following a loss of Feed Water with a reactor trip on low SG level.

TASKS ASSOCIATED WITH THIS SIMULATOR EXERCISE GUIDE

SRO TASK #	TASK TITLE
02028033500	Authorize Unit Trip
02046045300	Recover From VCT Level Transmitter Failures
02059026300	Respond To Loss Of Power Range Instrumentation Channel
02074016500	Respond To A Loss Of Heat Sink Following A Reactor Trip
02081006300	Respond To A Loss Of One Heater Drain Pump
02200046500	Respond to Steam Generator Low Level
02200009300	Respond to Unit Runback

RO TASK	TASK TITLE
01046045300	Recover from VCT Level Transmitter Failures
01059026300	Respond to Loss of Power Range Instrumentation Channel
01074006100	Stop Steam Generator Feed Pump
01074011300	Control Steam Generator Level Manually With Main Feed Regulating Valves
01074016500	Respond To A Loss Of Heat Sink Following A Reactor Trip
01081006300	Respond to a Loss of One Heater Drain Pump
01089020100	Trip Turbine Manually
01200001500	Respond To Unit Trip
01200046500	Respond to Steam Generator Low Level
01200009300	Respond to Unit Runback

UPDATE LOG:	
NOTES:	Place this form with the working copies of lesson plans and/or other materials to document changes made between formal revisions. For fleet-wide training materials, keep electronic file of this form in same folder as approved materials. Refer to TR-AA-230-1003 SAT Development for specific directions regarding how and when this form shall be used.

Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / REVIEWER	DATE / DATE
0-0	Initial Revision	Revised for L-16-1 NRC Exam	2108338	Note 5 Note 5	Note 5 Note 5
0-1					
0-2					
0-3					
0-4					
0-5					

1. Individual updating lesson plan or training material shall complete the appropriate blocks on the Update Log.
2. Describe the change to the lesson plan or training materials.
3. State the reason for the change (e.g., reference has changed, typographical error, etc.)
4. Preparer enters name/date on the Update Log and obtains Training Supervisor approval.
5. Initial dates and site approval on cover page.

SCENARIO SUMMARY

Initial Conditions:
The plant is at 100% power (BOL). Online risk is green. B train is protected on both units

Equipment OOS
The 3A RHR pump and 3A1 Circulating Water pump are OOS.

SCENARIO SUMMARY

Event 1
Shortly after the crew takes the shift, LT-3-115, VCT Level Transmitter, Fails Low which causes auto makeup to start. The crew responds using the 3-ONOP-046.4, Malfunction of Boron Concentration Control System. The RCO manually stops auto makeup.

Event 2
Once the crew stabilizes VCT level, the 3A Heater Drain Pump trips, which causes an automatic Turbine Runback. The crew will enter 3-ONOP-089, Turbine Runback. During the runback, Control Rods fail to auto insert. The RCO will manually insert rods to maintain Tave $\pm 3°F$. When the Runback is complete, the RCO will borate as needed to clear Rod Lo Limit and Axial Flux alarms, and the BOP will reset the Steam Dumps. When the plant is stable, the crew will enter 3-ONOP-028, Reactor Control System Malfunction, for the failure of Rods to auto insert.

Event 3
Once the crew resets the Steam Dumps and completes the required actions for the Reactor Control System Malfunction, the N-42 Instrument Power Fuse Blows. The crew will enter 3-ONOP-059.8, Power Range Nuclear Instrumentation Malfunction. The BOP will defeat or bypass the functions of N-42, as directed by the ONOP.

Event 4
After the crew completes the actions of 3-ONOP-059.8, Engineering reports High Vibration on the 3B SGFW pump. The SM directs the crew to start a 3-GOP-100 Fast Load Reduction to secure the 3B SGFW pump.

Event 5
When the crew starts the down power, FT-3-494, 3C S/G Steam Flow Transmitter, will be failed as is. The BOP will take manual control of the 3C S/G level and restore the 3C S/G level to normal. The US will enter 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, and direct the BOP to select an operable channel, then restore 3C S/G level control to automatic.

SCENARIO SUMMARY

Event 6

After the crew reduces power by 5 to 10% and completes the actions for the failed Steam Flow channel, a Main Feed Water Header break occurs. The crew responds to the reactor trip using 3-EOP-E-0, Reactor Trip or Safety Injection. During the loss of Main Feed Water, there's also a loss of the suction piping to all AFW pumps. The crew will transition to 3-EOP-FR-H.1, Response To Loss Of Secondary Heat Sink, and initiate Feed and Bleed.

Event 7

During 3-EOP-E-0, The Main Turbine fails to automatically trip. The BOP will take compensatory action to trip the Turbine manually.

Event 8

When the crew attempts to initiate Feed and Bleed, one of the PZR PORVs fails to open so the crew will open all RCS Vent Valves

The scenario is terminated once the RCS Vent Valves are open or at the Lead Evaluator's discretion once all critical tasks have been evaluated.

Event		CRITICAL TASKS
7	CT1	**Manually Trip the Main Turbine** Manually trip the main turbine before any RCS cold leg temperature decreases by more than 100°F. *Safety Significance* - Failure to trip the main turbine causes an excessive rate of RCS cooldown, well beyond the conditions typically analyzed in the FSAR. The excessive cooldown rate creates large thermal stresses in the reactor pressure vessel and causes rapid insertion of a large amount of positive reactivity. Thus, failure to manually trip the Main Turbine under the postulated conditions can result in challenges to the Integrity and Subcriticality CSFs.
8	CT2	**Initiate Bleed-And-Feed** Initiate bleed-and-feed cooling in accordance with 3-EOP-FR-H.1 within 36 minutes following an automatic or manual Reactor trip due to a loss of Feed Water. (0-ADM-232,Time Critical Operator Actions in the PTN PSA Model) *Safety Significance* - Failure to initiate RCS bleed and feed before the RCS saturates at a pressure above which the high-head ECCS pumps can inject results in significant and sustained core uncovery. If RCS bleed is initiated so that the RCS is depressurized below the shutoff head of the high-head ECCS pumps, then core uncovery is prevented or minimized.

SEQUENCE OF EVENTS	
Event #	**Description**
1.	LT-3-115 VCT Level Transmitter Fails Low
2.	3A Heater Drain Pump (Turbine Runback) Rods Fail To Auto Insert
3.	N-42 Loss Of Instrument Power
4.	3B S/G Feedwater pump High Vibration (Fast Load Reduction required)
5.	FT-3-494 3C S/G Steam Flow Transmitter Fails As Is
6.	Common Main Feed Header Break Common Loss Of Suction To All AFW Pumps
7.	Main Turbine Fails To Automatically Trip
8.	PCV-3-455C Fails To Open (PZR PORV)

SIMULATOR SET UP INSTRUCTIONS	
Check	**Action**
____	Restore IC-11 (100% BOL) or equivalent IC.
____	Place the Simulator in RUN.
____	Stop the 3A1 Circ Water Pump
____	Open & execute lesson file L-16-1 N2
____	Ensure the following lesson steps are triggered: • EVENT 6 SETUP - LOSS OF AFW PUMP SUCTION • EVENT 8 SETUP - PORV 455C FAILED CLOSE • EVENT 7 SETUP - TURBINE FAILS TO TRIP • EVENT 7 SETUP - CV FAIL AS IS • SETUP - 3A RHR PUMP OOS • SETUP - 3A1 CWP OOS
____	• Place 3A RHR pump in PTL and hang an ECO Card • 3A1 CWP in stop and hang an ECO card
____	Verify the trend for 3A1 Screen on the TWS DP Recorder is clear.
____	Ensure Rod Group Step Counters have completed stepping out.
____	Allow the plant to stabilize.
____	Acknowledge any alarms and freeze Simulator.
____	Ensure B train is protected train on VPA.
____	Verify Key 13 for Reactor Head Vent valves is in the key locker (6 keys on one key ring)
____	Perform the SIMULATOR OPERATOR CHECKLIST or equivalent.
____	Place TURNOVER SHEETS on RO's desk or give to the Lead Evaluator.

BRIEFINGS

- Shift turnover information is attached to the back of this guide.
- Ensure all applicants are prior briefed on Appendix E of NUREG 1021, Policies and Guidelines For Taking NRC Examinations.
- Conduct a Crew Pre-brief to cover turnover information. Shift turnover information is attached to the back of this guide.

US: ______________________________

RCO: ______________________________

BOP: ______________________________

SCENARIO NOTE

0-ADM-211 Prudent Operator Actions - If redundant stand-by equipment is available and ready, the operator is permitted to start the redundant equipment for failed or failing operating equipment. Immediate follow up of applicable ARPs and ONOPs (AOPs) shall occur as required.

Critical Tasks are highlighted in red.	**Simulator Operator Actions** are highlighted in blue.	**Operator Verifiable Actions** are Highlighted in green.

EVENT 1 - LT-3-115 VCT LEVEL TRANSMITTER FAILS LOW.		
3-ONOP-046.4, MALFUNCTION OF BORON CONCENTRATION CONTROL SYSTEM		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE Ensure the Simulator is in **RUN** before the crew enters the Simulator.	
		US: Conducts shift turnover.
	BOOTH OPERATOR When directed by the Lead Evaluator, **trigger EVENT 1 - LT-3-115 FAILS LOW**	
		RCO: Responds to Alarms A3/4 VCT AUTO MAKE-UP and A4/6 VCT HI/LOW LEVEL • VERIFY Make-Up Flow. • Reports LT-3-115 failed low.
	BOOTH OPERATOR If dispatched to locally check the VCT level, wait 2-5 minutes and report LT-3-115 failed low. No obvious cause of the failure.	BOP: • Reviews ARP, refer to 3-ONOP-046.4, Malfunction of Boron Concentration Control System. • Checks VCT level on DCS • Dispatches SNPO to locally check LT-3-115 indication in charging pump room.
		US: Directs response using 3-ONOP-046.4, Malfunction of Boron Concentration Control System
		RCO: Check Boric Acid OR Primary Water Makeup Flow Rates – ABNORMAL **(NO Go to Step 28)** **Step 1**

EVENT 1 - LT-3-115 VCT LEVEL TRANSMITTER FAILS LOW.		
3-ONOP-046.4, MALFUNCTION OF BORON CONCENTRATION CONTROL SYSTEM		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Check For VCT Level Transmitter, LT-3-115, Failing Or Failed High. **(NO, Go to Step 31)** **Step 28**
		RCO Check for LI-3-115 Failing Or Failed Low **Step 31**
		US: Review caution and note with the crew. **CAUTION** With no operator action, LT-3-115 failed low with makeup flow greater than charging flow could result in over pressurization of the VCT. **NOTE** Failure of LT-3-115 low will result in the following: • Annunciator Alarm A 4/6 VCT HI/LO LEVEL. • Auto makeup starts, but does not stop automatically. • LCV-3-115A modulating open to attempt to control level at the VCT Level Controller, LC-3-112, setpoint
		RCO: Turn RCS Makeup Control Switch To STOP Go to Step 41. **Step 32**

EVENT 1 - LT-3-115 VCT LEVEL TRANSMITTER FAILS LOW.		
3-ONOP-046.4, MALFUNCTION OF BORON CONCENTRATION CONTROL SYSTEM		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR Respond as WCC/I&C	Notifies WCC/I&C regarding LT-3-115 failure. Directs PWO initiation & troubleshooting. Determines no Tech Specs apply **Steps 41/42**
		Performs a manual MU to the VCT using 0-OP-046, as required **Step 43**
	BOOTH OPERATOR Acknowledge briefing on how to swap charging pump suction to RWST if required.	US: • Conducts crew brief. • Briefs Operator how to manually swap charging pump suction to RWST if required.

EVENT 2 – 3A HEATER DRAIN PUMP TRIP, RODS FAIL TO AUTO INSERT		
3-ONOP-089, TURBINE RUNBACK		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the Lead Evaluator, **trigger EVENT 2 – 3A HDP TRIP**	
		BOP: • Acknowledges Alarms D8/2, HDP A/B MOTOR OVERLOAD TRIP and E2/5, TURBINE RUNBACK. • Reports 3A Heater Drain Pump tripped, Turbine Runback in progress
	NOTE Steps 1 and 2 are Immediate Operator Actions.	BOP: Verifies a SGFP was NOT lost. **IOA-Step 1**
		RCO/BOP: Check for proper operation of the following: • Steam Dumps • Turbine • If Rods are in AUTO, then verify Auto Rod Insertion to match Tavg with Tref. **(NO)** - Manually insert Rods as need to match Tavg with Tref. • Main Feedwater Valves • Pressurizer **IOA-Step 2**
		US: • Enters and directs the actions of 3-ONOP-089, Turbine Runback. • Reviews Notes with the crew.
		BOP: Check Steam Generator levels stabilized and on program. **Step 1**

EVENT 2 – 3A HEATER DRAIN PUMP TRIP, RODS FAIL TO AUTO INSERT		
3-ONOP-089, TURBINE RUNBACK		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Check Tavg is maintained within $\underline{+}3°F$ of Tref. • Place Control Rods in Manual. • Maintain Tavg/Tref ΔT within $\underline{+}3°F$. **Step 2**
		BOP: Check Steam Generator pressures stabilizing. **Step 3**
		RCO: Check Pressurizer Level stabilizing and trending to Program Level. **Step 4**
	<u>NOTE</u> When the plant is stabilized, the US may also enter 3-ONOP-028, Reactor Control System Malfunction due to the failure of rods to auto insert. 3-ONOP-028 starts on page 20.	RCO: Check Pressurizer Pressure stabilizing and trending to 2235 psig. **Step 5**
	<u>BOOTH OPERATOR</u> If dispatched to locally check the 3A Heater Drain pump, wait 5 minutes, and then report its breaker has an over current flag but nothing abnormal at the pump.	BOP: Check following for proper operation: • Steam Gen Feed Pump Recirc • Condensate Pump Recirc • Heater Drain Pumps • Heater Drain Tank Level Controls • Secondary Heater Level Controls **Step 6**

EVENT 2 – 3A HEATER DRAIN PUMP TRIP, RODS FAIL TO AUTO INSERT		
3-ONOP-089, TURBINE RUNBACK		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE The crew should monitor alarms and borate using 50 gallon batches as necessary to withdraw rods until the alarm is clear.	RCO/BOP: • Monitor Annunciator G 5/1, AXIAL FLUX T.S. LIMIT EXCEEDED – CLEAR. • Monitor Annunciator B 9/2, Axial Flux Tilt - CLEAR • Monitor Annunciator B 8/1, ROD BANK LO LIMIT – CLEAR. • Monitor Annunciator B 8/2 ROD BANK A/B/C/D EXTRA LO LIMIT – CLEAR. **Steps 7 - 10**
		BOP: When the turbine runback is complete: • Match the control switch flag for the 3A Heater Drain pump. • Check CV-3-2011, LP HTR BYP CLOSED. **Steps 11.a&b**
	BOOTH OPERATOR Acknowledge reports of plant status and request for support.	BOP: NOTIFY Load Dispatcher of load restrictions. **Step 11c**
		US: Informs SM to notify Plant Management and NRC Resident per 0-ADM-115, Notifications of Plant Events. **Step 11d**
		RCO: If boration used when plant conditions are stable, stop the boration and restore Auto Makeup **Step 12**
	NOTE Since Auto Rod control is not working, the crew should leave rods in manual.	RCO: Check Rod Control in MANUAL. **Step 13**

EVENT 2 – 3A HEATER DRAIN PUMP TRIP, RODS FAIL TO AUTO INSERT		
3-ONOP-089, TURBINE RUNBACK		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR Acknowledge notification to take RCS samples.	BOP: IF change in Reactor Power exceeded 15%, then notify Chemistry that RCS sampling is required within 2 to 6 hours per TS 4.4.8, Table 4.4-4, Item 6b. **Step 14**
	NOTE If the RO promptly inserts rods during the runback, the Steam Dumps may not arm.	BOP: Take Steam Dump To Condenser Mode Switch to Reset, and Release to AUTO. **Step 15**

EVENT 2 – 3A HEATER DRAIN PUMP TRIP, RODS FAIL TO AUTO INSERT		
3-ONOP-028, REACTOR CONTROL SYSTEM MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
		US: Enters and direct the actions of 3-ONOP-028.
		RCO: • Verify Rods in manual. **Step 4.2.1**
		RCO: • Do **NOT** increase reactor power without permission from the Reactor Engineering Supervisor and the Shift Manager. • Manually position the RCC control bank to restore steady state conditions. • Notify Reactor Engineering and I&C • Review Subsection 5.1. • Determine Unit Shut down not required at this time **Steps 5.2.1 - 5.2.5**
	<u>LEAD EVALUATOR NOTE</u> After plant is stabilized or at the Lead Evaluators discretion, proceed to Event 3	

EVENT 3 – N-42 LOSS OF INSTURMENT POWER		
3-ONOP-059.8, POWER RANGE NUCLEAR INSTRUMENTATION MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	<u>BOOTH OPERATOR</u> When directed by the Lead Evaluator **trigger EVENT 3 - N42 BLOWN INST PWR FUSE**	
		RCO: • Respond to multiple alarms associated with the Power Range • Reports N-42 Instrument Power Fuse blown.
		BOP: Reviews ARPs, recommend entering 3-ONOP-059.8, Power Range Nuclear Instrumentation Malfunction.
		US: Enters 3-ONOP-059.8, Power Range NI Malfunction.
	<u>BOOTH OPERATOR</u> If asked, I&C would like be present while tripping bistables. They will report to the control room in about one hour.	BOP: • Place the DROPPED ROD MODE switch for N42 channel in the BYPASS position (ANN B8/4) • Place the N42 ROD STOP BYPASS switch to the failed channel BYPASS position • Transfer the UPPER SECTION comparator defeat switch to the N42. • Transfer the LOWER SECTION comparator defeat switch to the N42. • Transfer POWER MISMATCH BYPASS switch to BYPASS N42. • Transfer the COMPARATOR CHANNEL DEFEAT switch to N42 **Step 5.1.1.1 – 5.1.1.6**

EVENT 3 – N-42 LOSS OF INSTURMENT POWER		
3-ONOP-059.8, POWER RANGE NUCLEAR INSTRUMENTATION MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	<u>**BOOTH OPERATOR**</u> Acknowledge request for assistance. If asked, the STA will monitor QPTR.	US: Notify I&C Monitor the Quadrant Power Tilt Ratio using 3-OSP-059.10, Determination Of Quadrant Power Tilt Ratio. **Step 5.1.1.9 – 5.1.1.10**
		US: Review Tech Specs: • TS 3.3.1, Functional Unit 2, - Action 2, Trip bi-stables in 6 hrs and restrict power to 75% RTP or monitor QPTR
	<u>**Lead Evaluator**</u> Once the crew completes the steps of 3-ONOP-059.8, or at the Lead Evaluators discretion, move on the next event.	

EVENT 4 – 3B S/G FEEDWATER PUMP HIGH VIBRATION		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the Lead Evaluator, call as the SM to report High Vibration on the 3B SG Feedwater pump. Engineering has looked at the pump and recommends it be secured. Direct the crew to start a 3-GOP-100 Fast Load Reduction to secure the 3B SG Feedwater pump.	
	BOOTH OPERATOR Acknowledge request for support.	
		US: • Directs actions to reduce Rx power from 50% per 3-GOP-100. • Completes Attachment 3 • Brief the crew per Attachment 4 **Steps 1-2**
		US: Reviews Foldout page with crew. • 3-EOP-E-0 Transition Criteria • Notify Chemistry Department • Boration Stop Criteria • Restore Blender to AUTO **FOLDOUT PAGE**
	BOOTH OPERATOR Acknowledge notifications.	BOP: Notify The Following Of Fast Load Reduction • System Dispatcher • Plant personnel using the Page Boost • Chemistry to start RCS sampling is required according to Tech Spec Table 4.4-4. **Step 3**

EVENT 4 – 3B S/G FEEDWATER PUMP HIGH VIBRATION		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Begin Boration For Initial Tavg Effect • Set the Boric Acid Totalizer to total boric acid volume value determined on Attachment 3. • Place the Reactor Makeup Selector Switch to BORATE. • Place the RCS Makeup Control Switch to START. • Adjust FC-3-113A, Boric Acid Flow Controller, to achieve 40 gpm boric acid flow as indicated on FR-3-113. • WHEN Tavg begins to lower from the boration, THEN adjust FC-3-113A, Boric Acid Flow Controller, to load reduction value from Attachment 3. **Step 4**
	<u>BOOTH OPERATOR</u> When crew starts reducing power, **trigger EVENT 5 –FT-3-494 FAILED AS IS.**	US: Determine Turbine Load Reduction in MW CNTRL **Step 5**
	<u>NOTE</u> When the crew notices the 3C Steam Flow Chanel, FT-3-494, failed as is, they may place the ramp on hold. The steps for the FT-3-494 failure start on page 27.	BOP: Initiate Turbine Load Reduction in MW CNTRL • Select MW CNTRL • Set TARGET power level – MW VALUE from Attachment 3 • Set RAMP RATE – MW/M VALUE FROM Attachment 3. • Check T_{avg} has lowered 1° to 2°F from the initial value prior to boration. • Depress GO • Ensure FC-3-113A, Boric Acid Flow Controller, has been adjusted to the load reduction boration rate. Go to Step 10 **Step 6**

EVENT 4 – 3B S/G FEEDWATER PUMP HIGH VIBRATION		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Monitor Load Reduction • Adjusts power reduction rate to maintain T_{avg}/T_{ref} within limits of Attachment 3. • Monitors S/G level control to ensure feed reg valves properly maintain level control in automatic. • Refer to Enclosure 1 for expected alarms. **Step 10**
	BOOTH OPERATOR Respond as SNPO. If asked, idle Charging Pump ready for start.	RCO: • Maintain pressurizer level to ensure that automatic pressurizer level control maintains level on program. • If needed, start 2nd Chg Pp and place 2nd orifice in service. • Adjust boration rate to maintain Tavg/Tref within limits of Attachment 3. • Refer to Enclosure 1 for expected alarms. **Step 10**
		RCO: Monitor Boration Rate • Monitor for excessive rod movement by monitoring TR-3-409D, Rod Position Bank D. • Determine if Insertion Limit and Bank D position are converging at a rate that will cause rod insertion limit alarms. • Adjust power reduction rate as needed to control rod insertion • Increase boration rate and/or total amount as necessary to limit control rod insertion **Step 11**

EVENT 4 – 3B S/G FEEDWATER PUMP HIGH VIBRATION		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: • Monitor Annunciator B 8/1, ROD BANK LO LIMIT – CLEAR • Monitor B 8/2 ROD BANK A/B/C/D EXTRA LO LIMIT – CLEAR **Steps 12-13**
		US: Have SM refer to the following procedures: • 0-EPIP-20101, DUTIES OF EMERGENCY COORDINATOR • 0-ADM-115, NOTIFICATION OF PLANT EVENTS **Step 14**
		RCO: Energize Pressurizer Backup Heaters **Step 15**
	NOTE Since the unit is not coming offline, the crew should leave station services on the Auxiliary Transformers.	BOP: • Verify Turbine Load Less Than 675 MWE • Stop one condensate pump • Check Desired Final Power Target – LESS THAN 475 Mwe.
	LEAD EVALUATOR Once power has been reduced by a minimum of 5%, at the Lead Evaluators discretion, proceed to Event 5.	

EVENT 5 – FT-3-494 3C S/G STEAM FLOW TRANSMITTER FAILS AS IS		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR This malfunction was inserted when the crew started the fast load reduction.	BOP: • Recognizes and reports 3C S/G Steam Flow FT-3-494 failure. • Takes Prompt Actions - Take manual control of 3C S/G level control valve FCV-3-498. - Return 3C S/G level to normal.
	NOTE The crew may stop the load reduction.	RCO: Addresses Alarm Response for C6/3, SG C Level Deviation. • CHECK LI-3-496 or LI-3-498, C STM GEN LEVEL controlling channel for SG Level deviation. • CHECK Feedwater Controllers FIC-3-498A or FIC-3-498B for indications of failure, alarm, or input signal failures. • CHECK Feedwater Controller Inputs • IF alarm is due to instrument failure, THEN REFER TO 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
		US: Enters and directs actions of 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, for response.
	NOTE The crew may use the ARP to select an operable channel and restore automatic level control.	BOP: • Verify FT-3-494 failure by channel check comparison. • Verify no off-normal conditions exist on FT-3-495. • Place 3C S/G Steam Flow Control Transfer Switch to FT-3-495 (Yellow) • Place 3C S/G Feed Water Flow Control Transfer Switch to FT-3-496 (Yellow) • Ensure 3C S/G level is returned to auto.

EVENT 5 – FT-3-494 3C S/G STEAM FLOW TRANSMITTER FAILS AS IS		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR WCC/I&C: Acknowledge the report. I&C want to be present to trip bi-stables and should be in the control room in about an hour.	BOP: Notifies WCC to initiate PWO and I&C for troubleshooting.
	BOOTH OPERATOR If asked to reset AMSAC, wait 2- 3 minutes and then **trigger EVENT 5 - RESET AMSAC.**	US: Reviews Tech Specs • Enters Tech Spec Action 3.3.1 Functional Unit 12 - Action 6, within 6 hrs trip bi-stables • Enters Tech Spec Action 3.3.2 Functional Unit 1.f and 4.d - Action 15, within 6 hrs trip bi-stables
	Lead Evaluator If the crew stopped the Rapid Power reduction, they should resume it once the 3C S/G level control is restored to auto and the US completes a review of Tech Specs.	US: Conducts crew brief.

EVENT 6 - COMMON MAIN FEED HEADER BREAK		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**TIME**	**TIME**
	BOOTH OPERATOR When directed by Lead Evaluator or if Reactor trips, **verify EVENT 6 - MAIN FEED HEADER BREAK triggers.**	
		BOP • Reports S/G levels lowing rapidly • Recommends a Manual Rx trip and SI
		US: • Directs RCO to manually trip the Reactor and for operators to perform their IOA's.
CT2 **Start Time**	Note the time the Rx trips for verification of CT2 to Initiate bleed-and-feed cooling within 36 minutes following a loss of Feed Water with a reactor trip on low SG level.	RCO: • Manually trips Reactor.
	NOTE Steps 1 - 4 of 3-EOP-E-0 are Immediate Operator Actions (IOAs). The board operators will call out the high level steps of the IOAs as each step is completed from memory. Once the IOAs are completed, the US will read though Steps 1 – 4 with the crew.	RO/BOP: Perform IOA's.
		RCO: Verifies Reactor Trip **STEP 1**

EVENT - 7 MAIN TURBINE FAILS TO AUTOMATICALLY TRIP		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**TIME**	**TIME**
CT1	**Manually Trip the Main Turbine** Manually trip the main turbine before any RCS cold leg temperature decreases by more than 100°F. **BOOTH OPERATOR** When the BOP trips the turbine **verify EVENT 6 – STEAM LEAK INSIDE CTMT triggers.** When the BOP trips the turbine **verify EVENT 7 - DELETE TURBINE TRIP FAILURE triggers.**	BOP: Verify Turbine Trip **(NO)** • Manually Trips the turbine. **STEP 2**
		BOP: Verifies Power To Emergency 4 KV Buses **STEP 3**
		RCO: Checks If SI Is Actuated **STEP 4**
		US: Directs 3-EOP-E-0 response and reviews the IOAs.

EVENT - 7 MAIN TURBINE FAILS TO AUTOMATICALLY TRIP		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**TIME**	**TIME**
		US: Reviews FOP for 3-EOP-E-0 with the crew • Adverse Containment Conditions • RCP Trip Criteria • Faulted S/G Isolation Criteria • Ruptured S/G Isolation Criteria • AFW System Operation Criteria • CST Makeup Water Criteria • RHR System Operation Criteria(YES, RCO starts timer) • Loss Of Offsite Power Or SI On Other Unit • Loss Of Charging Criteria **FOLDOUT PAGE**
	<u>NOTE</u> Attachment 3 actions start on page 37.	BOP: Continues with ATTACHMENT 3 to complete The Prompt Action Verifications. **STEP 5**
	<u>BOOTH OPERATOR</u> If dispatched to check AFW pumps, wait 3 to 5 minutes and then report there too much steam in the area to check the pumps.	RCO: Check AFW Pumps – AT LEAST TWO RUNNING. **(NO)** **STEP 6**
		RCO: Verify AFW Valve Alignment – PROPER EMERGENCY ALIGNMENT **STEP 7**
		RCO: Verify Proper AFW Flow: **(NO)** • NO feed flow available **STEP 8**

EVENT - 7 MAIN TURBINE FAILS TO AUTOMATICALLY TRIP		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**TIME**	**TIME**
		US: • Monitor Critical Safety Functions using 3-EOP-F-0, Critical Safety Function Status Trees. • Go to 3-EOP-FR-H.1, Response To Loss Of Secondary Heat Sink, Step 1.

EVENT 8 – PCV-3-455C FAILS TO OPEN. (PZR PORV)		
3-EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR If dispatched to start A or B Standby S/G Feedwater Pump, wait 2 to 3 minutes and then report DWDS-3-012, Standby S/G FW Pump Manual Isolation valve, will not open.	US: Enters and directs the actions of 3-EOP-FR-H.1, Response To Loss Of Secondary Heat Sink
		RCO: Check If Secondary Heat Sink Is Required **STEP 1**
	NOTE The transient also cause a small steam leak on the 3B Main Steam Line inside containment.	US: Reviews Foldout Page • Adverse Containment Conditions (MET) **Foldout Page**
	NOTE Step 2 is a continuous action step.	RCO: Check If Bleed And Feed Is Required • Stop all RCPs (Go to Step 13) **STEP 2**
	CAUTION Step 13 through Step 17 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.	US: Reviews Caution with the crew.
	Part of CT2	RCO: Actuate SI And Containment Isolation Phase A **STEP 13**

EVENT 8 – PCV-3-455C FAILS TO OPEN. (PZR PORV)		
3-EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	Part of CT2	RCO: Establish maximum Charging flow • Check power supply to all Charging pumps – Aligned To Offsite Power • Check status of Charging pumps prior to SI actuation in Step 13 – any running **(NO)** • Check CCW flow to RCP(s) Thermal Barrier is lost **(NO)** • Reset SI • Start all available Charging pumps • Adjust Charging pump speed controllers to establish maximum charging flow • Adjust HCV-3-121, Charging Flow To Regen Heat Exchanger, to maintain proper Seal Injection flow • Place RCS Makeup Control Switch in STOP • Check Charging Pump Suction – Aligned To RWST **STEP 14.a**
		RCO: • Check SI pumps status – at least two running • Verify SI valve amber lights on VPB – All Bright **STEP 14.b&c**

EVENT 8 – PCV-3-455C FAILS TO OPEN. (PZR PORV)		
3-EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: Reviews Caution and notes with crew **CAUTION** If Low PRZ Pressure SI signal is NOT blocked prior to PRZ pressure decreasing below 1730 psig, Charging Pumps started in previous step will trip **NOTE** • PRZ pressure must be less than 1987 psig for permissive to block the Low • PRZ Pressure SI signal. • Step 15 should be reviewed in advance to ensure timely performance.
	NOTE PORV 455C was failed closed during the simulator setup. Part of CT2	**RCO:** • Verify power to PRZ PORV Block valves – Available • Verify PRZ PORV Block valves – Both Open • Check Block Low PRZ Press S.I. status light – ON (**NO)** • Open one PRZ PORV • When Block Low PRZ Press. S.I. status light is ON Momentarily place both Safety Injection Block switches to Block and return to Neutral • Verify Low PRZ Press. S.I. Blocked status light – ON • Open remaining PRZ PORV **(NO, Failed Close)** **STEP 15**
		BOP: Verify Instrument Air To Containment **STEP 16**

EVENT 8 – PCV-3-455C FAILS TO OPEN. (PZR PORV)		
3-EOP-FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
CT2 **Stop Time**	**BOOTH OPERATOR** When dispatched to install the fuses for the RCS Vent valves**, trigger EVENT 8 LOA - INSTALL FUSES FOR RX HEAD VENTS.** **Initiate Bleed-And-Feed** Initiate bleed-and-feed cooling in accordance with 3-EOP-FR-H.1 within 36 minutes following an automatic or manual Reactor trip due to a loss of Feed Water.	RCO/BOP: Both PRZ PORVs Open **(NO)** • Install fuses for RCS Vent valves: • WHEN power is restored to RCS vent valves, THEN open all RCS vents: **STEP 17**
The scenario is terminated once the RCS Vent Valves are open or at the Lead Evaluator's discretion once all critical task have been evaluated.		
***** END OF SCENARIO *****		

EVENT 6 – COMMON MAIN FEED HEADER BREAK		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Check Load Centers Associated With Energized 4 KV Buses – ENERGIZED **STEP 1**
		BOP: Verify Feedwater Isolation: • Place Main Feedwater Pump switches in STOP • Feedwater Control Valves – CLOSED • Feedwater Bypass Valves – CLOSED • Feedwater Bypass Isolation Valves – CLOSED • Feedwater Isolation MOVs – CLOSED • Verify Standby Feedwater Pumps – OFF **STEP 2**
		BOP: Check If Main Steam Lines Should Be Isolated **STEP 3**
		BOP: Verify Containment Isolation Phase A Valve White Lights On VPB – ALL BRIGHT **STEP 4**
		BOP: Verify Pump Operation: • At least two High-Head SI Pumps – RUNNING • Only one RHR Pump Available **STEP 5**

EVENT 6 – COMMON MAIN FEED HEADER BREAK		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Verify Proper CCW System Operation: • CCW Heat Exchangers – THREE IN SERVICE • CCW Pumps – ONLY TWO RUNNING • CCW Headers – TIED TOGETHER • MOV-3-626, RCP Thermal Barrier CCW Outlet – OPEN **STEP 6**
		BOP: Verify Proper ICW System Operation: • Verify ICW Pumps – AT LEAST TWO RUNNING • Verify ICW To TPCW Heat Exchanger – ISOLATED • Check ICW Headers – TIED TOGETHER **STEP 7**
		BOP: Check Emergency Containment Coolers – ONLY TWO RUNNING **STEP 8**
		BOP: Verify Unit 3 Containment Purge Exhaust And Supply Fans – OFF **STEP 9**
		BOP: Verify Containment Spray NOT Required: **-STEP 10**
		BOP: Verify SI – RESET **STEP 11**

EVENT 6 – COMMON MAIN FEED HEADER BREAK

3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS

TIME	EVALUATOR ACTIVITIES & NOTES	EXPECTED STUDENT RESPONSE
		BOP: Verify SI Valve Amber Lights On VPB – ALL BRIGHT **STEP 12**
		BOP: Verify SI Flow: • RCS pressure – LESS THAN 1625 PSIG[1950 PSIG] **(NO)** **STEP 13**
	BOOTH OPERATOR When directed by the crew, **trigger LOA – ALIGN U-4 HHSI TO U-3 RWST.** Wait 5 minutes and report local operator steps complete.	BOP: Realign SI System: • Check Procedure Entry Status – E-0 ENTERED FROM 3-ONOP-047.1, LOSS OF CHARGING FLOW IN MODES 1 THROUGH 4 **(NO)** • Verify Unit 3 High-Head SI Pumps – TWO RUNNING • Direct Unit 4 Reactor Operator to align Unit 4 High-Head SI Pump suction to Unit 3 RWST using Attachment 1. **STEP 14**
		BOP: Verify Containment Isolation Phase A – RESET **STEP 15**
		BOP: Reestablish RCP Cooling: • Check RCPs – AT LEAST ONE RUNNING • Open CCW To Normal Containment Cooler Valves: – MOV-3-1417 – MOV-3-1418 • Reset and start Normal Containment Coolers **STEP 16**

EVENT 6 – COMMON MAIN FEED HEADER BREAK		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Verify Control Room Ventilation Isolation: **STEP 17**
	BOOTH OPERATOR When requested by crew, **trigger LOA – PLACE PAHMS IN SERVICE.** Wait 5 minutes and report local operator steps complete.	BOP: Place Hydrogen Monitors In Service Using 3-NOP-094, CONTAINMENT POST ACCIDENT MONITORING SYSTEM • For Each Hydrogen Monitor A/B - ENSURE FUNCTION SELECTOR switch is in SAMPLE. - PLACE control switch in ANALYZE. - PRESS the REMOTE SELECTOR button. - PRESS the ALARM RESET button. • Dispatch an operator to complete local step of 3-NOP-094 **STEP 18**
		BOP: Verify All Four EDGs – RUNNING **STEP 19**
		BOP: Verify Power To Emergency 4 KV Buses: **STEP 20**
		BOP: Notify Unit Supervisor Of The Following: • Attachment 3 is complete • Any safeguards equipment that is NOT In the required condition • Status of Containment pressure continuous action **STEP 21**

Discussion Points are intentionally NOT included in evaluated scenarios. However, space is available below to document follow-up questions when further information is required to determine an evaluation outcome.

FOLLOW-UP QUESTIONS

QUESTION #1

ANSWER #1

QUESTION #2

ANSWER #2

SIMULATOR POST-SCENARIO RESTORATION:

_______ 1. Restore per Simulator Operator Checklist.

_______ 2. Once exams are complete, restore from SEI-19, Simulator Exam Security.

OPERATIONS SHIFT TURNOVER REPORT

UNIT 3 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B	UNIT 4 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B

ONCOMING CREW ASSIGNMENTS

Shift Mgr:			**Inside SNPO:**	
Field Supv.:			**Outside SNPO:**	
Admin RCO:			**ANPO:**	
Unit 3			**Unit 4**	
Unit Supv.:			**Unit Supv.:**	
RCO:			**RCO:**	
NPO:			**NPO:**	

PLANT STATUS

Unit 3			**Unit 4**	
Mode:	1		**Mode:**	1
Power:	100%		**Power:**	100%
MWe:	842		**MWe:**	842
Gross Leakrate:	0.25 gpm		**Gross Leakrate:**	0.03 gpm
RCS Boron Conc:	1145		**RCS Boron Conc:**	642

Operational Concerns:

3A RHR pump taken OOS 4 hours ago for an oil change, expected back by the end of this shift.
3A1 Circ Water pump OOS. Tripped on over current, Electrical Maintenance is investigating.

U3 Anticipated LCO Actions:

None

U4 Anticipated LCO Actions:

None

Results of Offgoing Focus Area:

UNIT 3 STATUS

REACTOR OPERATOR

UNIT RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B

Mode:	1
Power:	100%
MWe	842
Tavg:	580°F
RCS Pressure:	2235
RCS Boron Conc:	1145

RCS Leakrate	
Gross:	0.25 GPM
Unidentified	0.04 GPM
Charging Pps:	0.00 GPM

Accumulator Ref Levels	
A	6656
B	6608
C	6646

Abnormal Annunciators:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	

Current Tech Spec Action Statements: (Does Not Include "For Tracking Only Items"	
T.S.A.S / Component: Reason: Entry Date:	3A RHR pump, 3.5.2.c – Action g Oil Change 4 hours ago
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	

REACTOR OPERATOR (CONT'D)
UNIT RISK: GREEN (ACCEPTABLE)   PROTECTED TRAIN: B
Changes to Risk Significant Equipment:
No recent changes from last shift. OLRM: GREEN PROTECTED TRAIN: B
Upcoming Reactivity Management Activities:
Maintain 99.5% to 100%
Upcoming Major POD Activities:
NONE
Upcoming ECOs to Hang and /or Release:
• Hang – None • Release – None
Evolutions or Compensatory Actions in Progress:
NONE
General Information, Remarks, and Operator Work Around Status:
• Weather forecast is overcast skies with scattered pockets of severe rain. • U3 supplying Aux Steam • Air In-leakage = 0.0 SCFM

Site:	Turkey Point Units 3 and 4 (PTN)
Title:	L-16-1 AUDIT EXAM SCENARIO 3
LMS #:	NRC 23
LMS Rev Date:	5/31/16 Rev #: 0.0
SEG Type:	☐ Training ☒ Evaluation
Program:	☐ LOCT ☒ LOIT ☐ Other
Duration:	120 minutes

Developed by:	Brian Clark Instructor/Developer	6/15/16 Date
Reviewed by:	Tim Hodge *Instructor (Instructional Review)*	6/22/16 *Date*
Validated by :	Rocky Schoenhals *SME (Technical Review)*	6/22/16 *Date*
Approved by:	Mark Wilson *Training Supervision*	6/22/16 *Date*
Approved by:	Rocky Schoenhals *Training Program Owner (Line)*	6/22/16 *Date*

SCENARIO REFERENCES		
DOC NO.	**TITLE**	**REV**
3-GOP-301	HOT STANDBY TO POWER OPERATION	34B
3-ONOP-059.7	INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION MALFUNCTION	0
3-ONOP-041.6	PRESSURIZER LEVEL CONTROL MALFUNCTION	2
0-ADM-211	EMERGENCY AND OFF-NORMAL OPERATING PROCEDURE USAGE	4B
3-ONOP-049.1	DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNEL	4
3-EOP-E-0	REACTOR TRIP OR SAFETY INJECTION	12
3-EOP-ES-0.1	REACTOR TRIP RESPONSE	12
3-ONOP-046.1	EMERGENCY BORATION	4
3-EOP-E-2	FAULTED STEAM GENERATOR ISOLATION	4
3-EOP-FR-P.1	RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION	3
	PTN TECHNICAL SPECIFICATIONS	298

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
Terminal Objective	Given this simulator scenario and resources normally found in the Control Room, the operating crew will perform Control Room operations IAW approved plant procedures in order to maintain the integrity of the plant and the health and safety of the public.
Enabling Objectives:	Given this simulator scenario and resources normally found in the Control Room, operate in accordance with approved plant procedures, Operations Department Instructions, and management expectations: 1. (ALL) Demonstrate personnel SAFETY awareness in interactions with plant staff and outside agencies. 2. (ALL) Demonstrate ALARA awareness in interactions with plant staff and outside agencies. 3. (ALL) Exchange correct information using 3-point communication/Repeat-backs with Control Room personnel and other plant staff. 4. (ALL) Inform plant personnel and System of plant conditions, as needed. 5. (US) Employ timely and concise crew briefs where appropriate. 6. (ALL) Maintain awareness of plant status and control board indication. 7. (ALL) Correctly diagnose plant situations. 8. (ALL) Solve operational problems as they arise. 9. (RCO/BOP) Manipulate plant controls properly and safely. 10. (ALL) Demonstrate self-checking using STAR and peer checks(when required) 11. (US) Demonstrate command and control of the crew. 12. (US) Coordinate the input of crew members and other plant staff. 13. (US) Utilize the input of crew members and other plant staff. 14. (ALL) Demonstrate conservative decision making. 15. (ALL) Demonstrate teamwork. 16. (ALL) Respond to plant events using procedural guidance (OPs/ONOPs/EOPs) as applicable in accordance with rules of usage. 17. (RCO/BOP) Implement any applicable procedural immediate operator actions without use of references. 18. (SRO) Maintain compliance with Tech Specs. 19. (ALL) Identify/enter applicable Tech Spec action statements. 20. (ALL) Respond to annunciators using ARPs (time permitting). 21. (ALL) Maintain written communication, logs, and documentation as needed to permit post-event reconstruction. Continued on the next page:

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
	While addressing the following events: 1. Raise Power to 3% 2. N35 Loss Of Compensating Voltage 3. 3C Charging Pump Speed Controller Air Leak 4. PT-3-1607, 3B S/G Steam Dump To Atmosphere Pressure Transmitter, Drifts High 5. PS-3-2007, Containment Pressure Channel Fails High 6. N36 Fails High, Rx Fails To Automatically Trip 7. 2 Stuck Rods 8. 3B Faulted S/G Inside Containment 9. POV-3-487, Feedwater Bypass Isolation, and FCV-3-489, Feedwater Bypass, Valves Fail To Isolate.
Prerequisites:	None
Training Resources:	PTN Unit 3 Plant Simulator
Development References:	• TR-AA-220-1003, Initial NRC and Audit Exam Process • TR-AA-230-1003, SAT Development • TR-AA-230-1007, Conduct of Simulator Training and Evaluation • 0-ADM-232, Time Critical Action Program • OP-AA-100-1000, Conduct Of Operations • OP-AA-103-1000, Reactivity Management • 0-ADM-200, Operations Management Manual • 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage • WCAP-17711-NP, Pressurized Water Reactor Owners Group Westinghouse Emergency Response Guideline Revision 2-Based Critical Tasks
Protected Content:	N/A
Evaluation Method:	Performance Mode
Operating Experience:	None
Risk Significant Operator Actions:	Trip reactor manually within one minute when automatic trip signal fails. (0-ADM-232 Attachment 2, Time Critical Operator Actions in the PTN PSA Model) During a MSLB inside Containment stop AFW flow to the faulted SG within 10 Minutes. (0-ADM-232 Attachment 1, Time Critical Operator Actions)

TASKS ASSOCIATED WITH SIMULATOR EXERCISE GUIDE

SRO TASK #	TASK TITLE
02028033500	AUTHORIZE UNIT TRIP
02046008300	EMERGENCY BORATE THE R.C.S. (MOV-350)
02047008300	INVESTIGATE CHARGING PUMP MALFUNCTIONS
02059008100	AUTHORIZE REMOVAL OF AN INTERMEDIATE RANGE NIS CHANNEL FROM SERVICE
02059024300	RESPOND TO LOSS OF INTERMEDIATE RANGE INSTRUMENTATION
02059027300	EVALUATE AND DIRECT TECH SPECS REQUIRED ACTIONS DUE TO NIS OUT OF SPEC / SERVICE CONDITIONS
02200002500	EVALUATE CRITICAL SAFETY FUNCTION (CSF) STATUS TREE OUTPUT
02200007500	RESPOND TO A STEAM LINE FAULT
02200022500	DIAGNOSE CAUSE OF SAFEGUARDS ACTUATION
02200023100	COORDINATE UNIT STARTUP
02200050500	RESPOND TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION

RO TASK	TASK TITLE
01200023100	COORDINATE UNIT STARTUP
01059024300	RESPOND TO LOSS OF INTERMEDIATE RANGE INSTRUMENTATION
01046008300	EMERGENCY BORATE THE R.C.S. (MOV-350)
01200007500	RESPOND TO A STEAM LINE FAULT
01200022500	DIAGNOSE CAUSE OF SAFEGUARDS ACTUATION
01200050500	RESPOND TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION
01047013100	START A CHARGING PUMP
01200001500	RESPOND TO UNIT TRIP
01047008300	INVESTIGATE CHARGING PUMP MALFUNCTIONS

UPDATE LOG:

NOTES:	Place this form with the working copies of lesson plans and/or other materials to document changes made between formal revisions. For fleet-wide training materials, keep electronic file of this form in same folder as approved materials. Refer to TR-AA-230-1003 SAT Development for specific directions regarding how and when this form shall be used.

Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / REVIEWER	DATE / DATE
0-0	Initial Revision	New for L-16-1 NRC Exam	2108338	Note 5 / Note 5	Note 5 / Note 5

1. Individual updating lesson plan or training material shall complete the appropriate blocks on the Update Log.
2. Describe the change to the lesson plan or training materials.
3. State the reason for the change (e.g., reference has changed, typographical error, etc.)
4. Preparer enters name/date on the Update Log and obtains Training Supervisor approval.
5. Initial dates and site approval on cover page.

SCENARIO SUMMARY

INITIAL CONDITIONS

The plant is at 10^{-8} Amps power (BOL). Online risk is green. B train is protected on both units. The crew will raise power to 3%.

EQUIPMENT OOS:

None

NOTE

Allow 30 minutes for the crew to brief raising power before entering the control room to brief raising power from 10^{-8} amps to 3%.

Event 1

After the crew takes the shift, the RCO will start raising Rx power by withdrawing Control Rods per 3-GOP-301, Hot Standby to Power Operation. The BOP will manually adjust Feedwater Bypass Flow Control valves, FCV-3-479/489/499, to maintain Steam Generator levels.

Event 2

After the crew stabilizes power at ~ 3%, N35 loses compensating voltage. The US will enter 3-ONOP-059.7, Intermediate Range Nuclear Instrumentation, and direct the BOP to place N35 in bypass.

Event 3

After the actions of 3-ONOP-059.7 are complete, an air leak will develop on the 3C Charging pump speed controller, causing the controller to fail to maximum output. The US will enter 3-ONOP-041.6, Pressurizer Level Control Malfunction, and direct the RCO to start the 3B charging pump and secure the 3C charging pump.

Event 4

After the charging pumps are swapped, PT-3-1607, 3B S/G Steam Dump To Atmosphere Pressure Transmitter, will drift high causing CV-3-1607, 3B Steam Dump To Atmosphere, to slowly open, lowering the 3B S/G pressure. The BOP will place CV-3-1607 in manual and reduce demand to stabilize 3B S/G pressure.

Event 5

After the plant is stabilized, PS-3-2007, Containment Pressure Channel, fails high. The US will enter 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channel, and review Tech Specs for the failed pressure channel.

Event 6

After the US completes the review of Tech Specs for PS-3-2007, Intermediate Range Nuclear Instrumentation Channel, N36, fails high and the Rx fails to auto trip. The RCO will manually trip the Reactor. The US will enter and direct the actions 3-EOP-E-0, Reactor Trip or Safety Injection.

SCENARIO SUMMARY

Event 7

When the Rx is tripped, the RCO will report 2 control rods failed to fully insert. After the crew completes the Immediate Operator Actions they will transition to 3-EOP-ES-0.1, Reactor Trip Response. The RCO will start a boration for the 2 stuck rods.

Event 8

After the boration is started, a steam leak will develop on the 3B S/G inside containment. The crew will return to 3-EOP-E-0, verify SI actuates, and isolate Aux Feed Water to the 3B S/G per the Foldout page of 3-EOP-E-0.

Event 9

When SI actuates, POV-3-487, 3B S/G Feedwater Bypass Isolation valve, is failed as is, and FCV-3-489, 3B FW Bypass, will leak by. The BOP will manually close POV-3-487, 3B FW Bypass Isolation, per Attachment 3 of 3-EOP-E-0. The crew will complete the actions of 3-EOP-E-0. About the time the crew is ready to transition to 3-EOP-E-2, Faulted Steam Generator Isolation, a red path will develop on the RCS Integrity Status Tree. The US will transition to 3-EOP-FR-P.1, Response To Imminent Pressurized Thermal Shock Condition.

The scenario is terminated once the crew transitions to 3-EOP-FR-P.1 or at the Lead Evaluator's discretion once all critical tasks have been evaluated.

CREW CRITICAL TASKS		
Event		**Description**
6	**CT1**	**Manually Trip The Rx** Trip reactor manually within one minute when automatic trip signal fails. (0-ADM-232 Attachment 2, Time Critical Operator Actions in the PTN PSA Model) *Safety Significance:* Failure to manually trip constitutes an incorrect performance that "necessitates the crew taking compensating action that would complicate the event mitigation strategy" and demonstrates the inability of the crew to "recognize a failure or an incorrect automatic actuation of an ESF system or component."
8	**CT2**	**Stop AFW Flow To Faulted SG** During a MSLB inside Containment stop AFW flow to the faulted SG within 10 Minutes. (0-ADM-232 Attachment 1, Time Critical Operator Actions) *Safety Significance:* Failure to isolate a Faulted SG that can be isolated causes challenges to the Critical Safety Functions that may not otherwise occur. Failure to isolate flow could result in an unwarranted Orange or Red Path condition on Integrity and/or Subcriticality (if cooldown is allowed to continue uncontrollably). Additionally, Termination of AFW flow to faulted SG is necessary to limit mass and energy releases into containment to prevent exceeding design pressure.

SEQUENCE OF EVENTS	
Event #	**Description**
1.	Raise Power to 3%
2.	N35 Loss Of Compensating Voltage
3.	3C Charging Pump Speed Controller Air Leak
4.	PT-3-1607, 3B S/G Steam Dump To Atmosphere Pressure Transmitter, Drifts High
5.	PS-3-2007, Containment Pressure Channel Fails High
6.	N36 Fails High, Rx Fails To Automatically Trip
7.	2 Stuck Rods
8.	3B Faulted S/G Inside Containment
9.	POV-3-487, Feedwater Bypass Isolation, and FCV-3-489, Feedwater Bypass, Valves Fail To Isolate.

SIMULATOR SET UP INSTRUCTIONS	
Check	**Action**
____	Restore IC-181 BOL 10^{-8} amps
____	Open & execute lesson file L-16-1 N3.lsn
____	Verify the following step is triggered • SETUP - DEFEAT AUTO RX TRIP
____	Ensure Rod Group Step Counters have completed stepping out.
____	Allow the plant to stabilize.
____	Acknowledge any alarms and freeze Simulator.
____	Ensure B train is protected train on VPB.
____	Perform the SIMULATOR OPERATOR CHECKLIST or equivalent.
____	Place TURNOVER SHEETS on RO's desk or give to the Lead Evaluator.
____	Ensure a copy of the maneuvering guideline is available
____	Ensure a marked copy of 3-GOP-301 is available for power increase.
____	Ensure a copy of 0-ADM-200 is available for briefs.
____	Ensure a copy of ODI-44 is available for briefs.

BRIEFINGS

- Shift turnover information is attached to the back of this guide.
- Ensure all applicants are prior briefed on Appendix E of NUREG 1021, Policies and Guidelines For Taking NRC Examinations.
- Conduct a Crew Pre-brief to cover turnover information. Shift turnover information is attached to the back of this guide

US: ______________________________

RCO: ______________________________

BOP: ______________________________

SCENARIO NOTE

0-ADM-211 Prudent Operator Actions - If redundant stand-by equipment is available and ready, the operator is permitted to start the redundant equipment for failed or failing operating equipment. Immediate follow up of applicable ARPs and ONOPs (AOPs) shall occur as required.

Critical Tasks are highlighted in red.	**Simulator Operator Actions** are highlighted in blue.	**Operator Verifiable Actions** are highlighted in green.

EVENT 1 – RAISE POWER TO 3%		
3-GOP-301, HOT STANDBY TO POWER OPERATION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	NOTE The Reactivity briefing will occur prior to assuming the watch in the Simulator Briefing Room. Allow up to 30 minutes for the briefing before the crew enters the control room.	CREW: Participates in reactivity briefing for raising Rx power to 3% / POAH.
	NOTE Maneuvering Guidelines are attached to the back of this Scenario Guide.	US: Directs the evolution per 3-GOP-301, Hot Standby to Power Operation, in accordance Step 5.24.
		RCO: Pull Rods to establish a startup rate not to exceed 1 dpm while below the POAH.
		BOP: Adjusts Steam Dumps to Atmosphere and Feedwater flow to maintain S/G Level on program when POAH is reached.
		RCO: Once above the POAH withdrawals rods continues to raise power with a startup rate not to exceed .5 dpm.
	LEAD EVALUATOR Once the crew levels power above the POAH or at the lead evaluator's discretion, direct the Booth Operator to trigger the next event.	

EVENT 2 – N35 LOSS OF COMPENSATING VOLTAGE		
3-ONOP-059.7, INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 2 – N-35 FAILS**	
		RCO: Recognizes / reports N-35 IR channel malfunction.
		BOP: Review ARP B5/3 • CHECK LOSS OF COMP VOLT light on N-35 IR channel drawer ON. • Refer to 3-ONOP-059.7
		US: Enters 3-ONOP-059.7, Intermediate Range Nuclear Instrumentation Malfunction.
		RCO: • Check If Reactor Trip Required • Check Annunciator B 5/3 off. **(NO)** • Go to Step 4 **Steps 1-2**
		BOP: • Places N35 level trip switch in BYPASS • Verify N36 Operable RCO: • Verify N-45 recorder selected to N36 **Step 4**

EVENT 2 – N35 LOSS OF COMPENSATING VOLTAGE		
3-ONOP-059.7, INTERMEDIATE RANGE NUCLEAR INSTRUMENTATION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
		RCO: • Check Reactor In Mode 1 Below P-10 **(NO)** • Check Reactor In Mode 1 Below P-10 **(NO)** • Check Reactor In Mode 2 Above P-6 • Maintain power below 5 percent until both IR channels are Operable • Go to Step 10 **Steps 5 - 7**
	<u>BOOTH OPERATOR</u> As WCC / I&C, inform US that ARs & WRs have been written and I&C has been notified to investigate.	US: Initiate a PWO and notify I&C to check affected IR channel.
		US: Reviews Tech Spec applicability. • TS 3.3.1 Functional Unit 3 - Action 3, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% • TS 3.3.1 Functional Unit 17a - Action 7, within 1 hr determine that the interlock is in its required state
	<u>LEAD EVALUATOR</u> Once the US completes the review of Tech Specs, or at your discretion, continue to the next event.	

EVENT 3 - 3C CHARGING PUMP SPEED CONTROLLER AIR LEAK		
3-ONOP-041.6, PRESSURIZER LEVEL CONTROL MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 3 – 3C CHARGING PUMP SPEED CONTROLLER AIR LEAK**	
		RCO: Report Max demand on 3C Charging pump speed controller.
	NOTE The US may use the guidance in the ARP to swap charging pumps prior to entering enter 3-ONOP-041.6.	BOP: Review ARP G1/2, CHARGING PUMP HI SPEED • Check individual charging pump controller and the master charging pump controller. • IF a failure of the individual charging pump controller has occurred in automatic, THEN PLACE the individual controller in manual AND MAINTAIN pressurizer level on program. - GO TO 3-ONOP-041.6, Pressurizer Level Control Malfunction • IF unable to control running charging pump, THEN START a standby charging pump AND SHUTDOWN the affected pump.
		US: Enter 3-ONOP-041.6, Pressurizer Level Control Malfunction

EVENT 3 - 3C CHARGING PUMP SPEED CONTROLLER AIR LEAK		
3-ONOP-041.6, PRESSURIZER LEVEL CONTROL MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR If dispatched to check the 3C Charging Pump, wait 2 to 3 minutes and then report a large air leak on the speed controller. If asked, the 3B Charging pump is ready for a start and once it's started, report back a SAT start.	RCO: • Check pressurizer level indicators LI-3-459A, LI-3-460, and LI-3-461 **(NO)** • Place Master Charging Pump Controller, LC-3-459G in Manual • Place individual Charging Pump Controllers in Manual • Start or stop additional pumps as necessary **Steps 5.1 – 5.3**
		RCO: When desired, place running charge controllers and master controller in auto.
	BOOTH OPERATOR When asked, acknowledge request for addition support. .	US: Initiate a PWO
	LEAD EVALUATOR Continue with next event after the RCO swaps charging pumps.	

EVENT 4 – 3B S/G STEAM DUMP TO ATMOSPHERE PRESSURE TRANSMITTER DRIFTS HIGH		
0-ADM-211, PRUDENT OPERATOR ACTIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator**, trigger EVENT 4 - PT-3-1607, 3B S/G STEAM DUMP TO ATMOSPHERE PRESSURE TRANSMITTER, DRIFTS HIGH**	
		CREW: Hears steam flow.
		BOP: Reports CV-3-1607, 3B Steam Dump to Atmosphere, is open.
		US: Directs BOP to close CV-3-1607.
	BOOTH OPERATOR As WCC, inform US that WCC/AOM has been notified, ARs & WRs were written, and proper work groups have been notified.	BOP: Places CV-3-1607 in manual and reduces demand to stabilize 3B S/G pressure.
	Lead Evaluator After the plant is stabilized, proceed with the next event.	

EVENT 5 – PS-3-2007, CONTAINMENT PRESSURE CHANNEL FAILS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNEL		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 5 - CTMT PRESSURE CHANNEL FAILS HIGH**	
		BOP: Review ARP, H5/1, CNTMT HI-HI/HI PRESS • CHECK Containment pressure indication • CHECK Status lamps on VPB • Refer to 3-ONOP-049.1
		US: Enters 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels
		BOP: Verify instrument loop failure by comparison to adjacent loops and known plant parameters and conditions. **Step 5.1**
		US: Refer to Technical Specifications • Tech Spec 3.3.2, Functional units 1.c, 2.b, 3.b.3, and 4.c - Action 15 – Place the inoperable channel in the tripped condition within 6 hours. **Step 5.6**
		US: Identifies fuses for failed channel using Attachment 7.
	BOOTH OPERATOR WCC/I&C: Acknowledge the report. State that I&C wants to be present before removing the fuses and should be in the control room in about an hour.	BOP: Notifies WCC to initiate PWO and I&C for troubleshooting.

EVENT 5 – PS-3-2007, CONTAINMENT PRESSURE CHANNEL FAILS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNEL		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	LEAD EVALUATOR After the US completes the Tech Spec review, proceed with the next event.	

EVENT 6 - N36 FAILS HIGH		
3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 6 - N36 FAILURE** and **verify EVENT 7 – 2 STUCK RODS triggers..**	
CT1 Start Time	**NOTE** Record the time Annunciator C6/5 INTERM RANGE HI FLUX TRIP Alarms	RCO: Reports conditions met for manual reactor trip. • N-36 has failed high • Associated bistables are lit • Reports manual reactor trip is required
		US: Directs a reactor trip
CT1 Stop Time	**Manually Trip The Rx** Trip reactor manually within one minute when automatic trip signal fails.	RCO: Trips the reactor
	NOTE Steps 1 - 4 of 3-EOP-E-0 are Immediate Operator Actions (IOAs). The board operators will call out the high level steps of the IOAs as each step is completed from memory. Once the IOAs are complete the US read through steps 1 - 4 with the crew.	RCO: Verifies Reactor Trip • Determines 2 rods stuck out • Reports reactor trip **Step 1**
		BOP: Verifies Turbine trip **Step 2**
		BOP: Verifies Power To Emergency 4 KV Buses **Step 3**
		RCO: Checks SI has NOT Actuated and is NOT required **Step 4**

EVENT 6 - N36 FAILS HIGH		
3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: Directs transition to 3-EOP-ES-0.1, Reactor Trip Response **Step 4**

EVENT 7 – 2 STUCK RODS		
3-EOP-ES-0.1, REACTOR TRIP RESPONSE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: Directs 3-EOP-ES-0.1 response
		CREW: Reviews FOP for 3-EOP-ES-0.1 • SI Actuation Criteria • Pressurizer Level Criteria • S/G Level Criteria Using AFW • CST Makeup Water Criteria • Control Room Ventilation Manual Isolation Criteria **Foldout Page**
		BOP: ➔ Checks RCS temperature control • Checks AFW pumps - NONE running • Checks RCPs – all running • Checks RCS Average Temperatures using DCS - Stable between 545°F and 547°F **Step 1**

EVENT 7 – 2 STUCK RODS		
3-EOP-ES-0.1, REACTOR TRIP RESPONSE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Checks Feedwater Status • Checks RCS Average Temperatures - LESS THAN 554°F • Verifies Main Feedwater Flow Control valves – CLOSED AND IN MANUAL • Manually closes Feedwater Isolation valves: - MOV-3-1407 - MOV-3-1408 - MOV-3-1409 • Checks S/G Narrow Range Levels – GREATER THAN 7% IN AT LEAST ONE S/G • Stops all but one Main Feedwater Pump **Step 2**
		RCO: Verifies ALL control rods - fully inserted **(NO)** • Emergency Borate for stuck control rods using 3-ONOP-046.1, EMERGENCY BORATION, while continuing with Step 4. **Step 3**

EVENT 7 – 2 STUCK RODS		
3-EOP-ES-0.1, REACTOR TRIP RESPONSE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE In accordance with step 4 of 3-ONOP-046.1, Emergency borate for 50 minutes for each rod not fully inserted using BAST water at 60 GPM through MOV-3-350. (2 rods x 50 min /rod = 100 min) **LEAD EVALUATOR** When emergency boration is established, continue with next event.	RCO: Commences emergency boration using 3-ONOP-046.1 • Verifies charging pumps - AT LEAST ONE RUNNING • Turns RCS Makeup Control Switch to STOP • Manually starts Boric Acid Pump 3A or 3B • Opens Emergency Boration Valve, MOV-3-350 • Opens Charging Flow to Regen Heat Exchanger, HCV-3-121 • Verify Loop A Charging Isolation,CV-3-310A – OPEN • Establishes emergency boration flow - FI-3-110 > 60 GPM - FI-3-122A > 45 GPM • Informs US emergency boration is established
	NOTE The US and BOP should continue with 3-EOP-ES-0.1.	BOP: • Checks 4KV Power Status To Both Unit 3 And Unit 4 • Checks Pressurizer Level Control • Checks PRZ Pressure Control • Checks S/G Levels ➔ Verifies All 4kV buses energized by OFFSITE POWER • Establish S/G Pressure Control

EVENT 8 – 3B S/G FAULTED INSIDE CONTAINMENT		
3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 8 - 3B S/G STEAM LEAK INSIDE CONTAINMENT** When event 8 is triggered, verify **EVENT 9 – 3B SG FAILS TO ISOLATE triggers**	RCO: • Reports Tavg lowering • Reports containment press rising BOP: • Reports 3B SG pressure lowering uncontrollably
CT2 Start Time	**NOTE** Record the time the Steam Leak is initiated.	US: Transitions to 3-EOP-E-0, Step 1, when Foldout page SI initiation criteria are met.
		RCO: Verifies Reactor Trip **Step 1**
		BOP: Verifies Turbine Trip **Step 2**
		BOP: Verifies Power To Emergency 4 KV Buses **Step 3**
		RCO: Checks If SI Is Actuated **Step 4**
		US: Directs 3-EOP-E-0 response and reviews the IOAs.

EVENT 8 – 3B S/G FAULTED INSIDE CONTAINMENT		
3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
CT2 **Stop Time**	**Stop AFW Flow To Faulted SG** During a MSLB inside Containment, stop AFW flow to the faulted SG within 10 Minutes.	CREW: Reviews FOP for 3-EOP-E-0 • Adverse Cntmt • RCP Trip Criteria (Yes) - Trips RCPs when RCP trip criteria met. • Faulted S/G Isolation - Yes - Place in manual and close or reduce the controller setpoint to 0 o Train 1 AFW Flow to 3B S/G CV-3-2817 o Train 2 AFW Flow to 3B S/G CV-3-2832 • RUPTURED S/G ISOLATION CRITERIA • AFW Sys Operation Criteria - Yes • CST Makeup Water Criteria • RHR System Operation Criteria - Yes • Set timer • Loss of Offsite Power or SI on the Other Unit • Loss of Charging Criteria **Foldout Page**
	NOTE Attachment 3 actions start on page 31.	BOP: Continues with ATTACHMENT 3 to complete the Prompt Action Verifications. **Step 5**
		RCO: Check AFW Pumps – AT LEAST TWO RUNNING - Yes **Step 6**
		RCO: Verify AFW Valve Alignment – PROPER EMERGENCY ALIGNMENT - Yes **Step 7**

EVENT 8 – 3B S/G FAULTED INSIDE CONTAINMENT		
3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: • Verify Proper AFW Flow: • Check Narrow Range Level in at least one S/G – GREATER THAN 7%[27%] • Maintain feed flow to S/G until Narrow Range Levels between 21%[27%] and 50% **Step 8**
		RCO: Check RCP Seal Cooling - satisfied **Step 9**
		RCO: ➔ Check RCS Temperatures • Check RCPs – ANY RUNNING – **(NO)** • Check RCS Cold Leg Temperatures – STABLE BETWEEN 545°F AND 547°F OR TRENDING DOWN TO 547°F - (**NO)** Stops dumping steam **Step 10**
		RCO: Check PRZ PORVs, Spray Valves And Excess Letdown Isolated - Yes **Step 11**
		RCO: Check If RCPs Should Be Stopped – (Not running) **Step 12**
		RCO: • Check If S/Gs Are Faulted - YES • Continue to monitor CSFs **Step 13**

EVENT 8 – 3B S/G FAULTED INSIDE CONTAINMENT

3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION

TIME	EVALUATOR ACTIVITIES & NOTES	EXPECTED STUDENT RESPONSE
		US: • When conditions are met transition to 3-EOP-FR-P.1, Response To Imminent Pressurized Thermal Shock Condition **Step 13**

EVENT 8 – 3B S/G FAULTED INSIDE CONTAINMENT		
3-EOP-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: Enter 3-EOP-FR-P.1, Response To Imminent Pressurized Thermal Shock.
		RCO: Check RCS Pressure – GREATER THAN 275 PSIG [575 PSIG] **(NO)**
The scenario is terminated once the crew transitions to 3-EOP-FR-P.1 or at the Lead Evaluator's discretion once all critical tasks have been evaluated.		

EVENT 9 – 3B S/G FAULTED INSIDE CONTAINMENT		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Check Load Centers - Energized **Step 1**
		BOP: Verify Feedwater Isolation: • Place Main Feedwater Pump switches in STOP • Feedwater Control valves – CLOSED • Feedwater Bypass valves – CLOSED • Feedwater Bypass Isolation valves – NOT CLOSED - Closes POV-3-487 • Feedwater Isolation MOVs – CLOSED • Verify Standby Feedwater Pumps – OFF **Step 2**
		BOP: Check If Main Steam Lines Should Be Isolated (MSIVs Closed) **Step 3**
		BOP: Verify Containment Isolation Phase A Valve White Lights On VPB – ALL BRIGHT **Step 4**
		BOP: Verify Pump Operation: • At least two High-Head SI Pumps – RUNNING • Both RHR Pumps – RUNNING **Step 5**

EVENT 9 – 3B S/G FAULTED INSIDE CONTAINMENT		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Verify Proper CCW System Operation: • CCW Heat Exchangers – THREE IN SERVICE • CCW Pumps – ONLY TWO RUNNING • CCW Headers – TIED TOGETHER • MOV-3-626, RCP Thermal Barrier CCW Outlet – OPEN **Step 3**
		BOP: Verify Containment Isolation Phase A Valve White Lights On VPB – ALL BRIGHT **Step 4**
		BOP: Verify Pump Operation: • At least two High-Head SI Pumps – RUNNING • Both RHR Pumps – RUNNING **Step 5**
		BOP: Verify Proper CCW System Operation: • CCW Heat Exchangers – THREE IN SERVICE • CCW Pumps – ONLY TWO RUNNING • CCW Headers – TIED TOGETHER • MOV-3-626, RCP Thermal Barrier CCW Outlet – OPEN **Step 6**

EVENT 9 – 3B S/G FAULTED INSIDE CONTAINMENT

3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS

TIME	EVALUATOR ACTIVITIES & NOTES	EXPECTED STUDENT RESPONSE
		BOP: Verify Proper ICW System Operation: • Verify ICW Pumps – AT LEAST TWO RUNNING • Verify ICW To TPCW Heat Exchanger – ISOLATED • Check ICW Headers – TIED TOGETHER **Step 7**
		BOP: Check Emergency Containment Coolers – ONLY TWO RUNNING **Step 8**
		BOP: • Verify Unit 3 Containment Purge Exhaust And Supply Fans – OFF **Step 9**
		BOP: ➔ Verify Containment Spray Running **Step 10**
		BOP: Verify SI – RESET **Step 11**
		BOP: Verify SI Valve Amber Lights On VPB – ALL BRIGHT **Step 12**
		BOP: Verify SI Flow: • RCS pressure – LESS THAN 1625 PSIG[1950 PSIG] **(NO)** • Go to Step 14. **Step 13**

EVENT 9 – 3B S/G FAULTED INSIDE CONTAINMENT		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When requested, **trigger LOA – ALIGN U-4 HHSIs TO U3 RWST**	BOP: Realign SI System: • Check Procedure Entry Status – E-0 ENTERED FROM 3-ONOP-047.1, LOSS OF CHARGING FLOW IN MODES 1 THROUGH 4 **(NO)** • Verify Unit 3 High-Head SI Pumps – TWO RUNNING **Step 14**
		BOP: Reset Phase A. **Step 15**
		BOP: Reestablish RCP Cooling: • Check RCPs – AT LEAST ONE RUNNING **(NO Go To Step 17)** **Step 16**
		BOP: Verify Control Room Ventilation Isolation: **Step 17**
	BOOTH OPERATOR When requested, **trigger LOA – ALIGN PAHMS FOR SERVICE**	BOP: Place Hydrogen Monitors In Service Using 3-NOP-094, CONTAINMENT POST ACCIDENT MONITORING SYSTEM **Step 18**
		BOP: Verify All Four EDGs – RUNNING **Step 19**
		BOP: Verify Power To Emergency 4 KV Buses: **Step 20**

EVENT 9 – 3B S/G FAULTED INSIDE CONTAINMENT		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Notify Unit Supervisor Of The Following: • Attachment 3 is complete • Any safeguards equipment that is NOT is in the required condition • Status of Containment pressure continuous action **Step 21**

Discussion Points are intentionally NOT included in evaluated scenarios. However, space is available below to document follow-up questions when further information is required to determine an evaluation outcome.

FOLLOW-UP QUESTIONS

QUESTION #1

ANSWER #1

QUESTION #2

ANSWER #2

SIMULATOR POST-SCENARIO RESTORATION:

______ 1. Restore per Simulator Operator Checklist.

______ 2. Once exams are complete, restore from SEI-19, Simulator Exam Security.

OPERATIONS SHIFT TURNOVER REPORT

UNIT 3 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B	UNIT 4 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B

ONCOMING CREW ASSIGNMENTS

Shift Mgr:			**Inside SNPO:**	
Field Supv.:			**Outside SNPO:**	
Admin RCO:			**ANPO:**	
Unit 3			**Unit 4**	
Unit Supv.:			**Unit Supv.:**	
RCO:			**RCO:**	
NPO:			**NPO:**	

PLANT STATUS

Unit 3			**Unit 4**	
Mode:	2		**Mode:**	1
Power:	10^{-8} amps		**Power:**	100%
MWe:	0		**MWe:**	842
Gross Leakrate:	0.23 gpm		**Gross Leakrate:**	0.03 gpm
RCS Boron Conc:	1670 ppm		**RCS Boron Conc:**	642 ppm

Operational Concerns:

Plant Start Up last shift following a 10 day outage to repair Feed Water Line Leak.
2 ROs and 2 SROs are in JITT for Turbine Roll and power accession. You are to raise power to 3% and hold until they return.

U3 Anticipated LCO Actions:

None

U4 Anticipated LCO Actions:

None

Results of Offgoing Focus Area:

UNIT 3 STATUS

REACTOR OPERATOR

UNIT RISK: GREEN (ACCEPTABLE)   PROTECTED TRAIN: B

Mode:	2
Power:	.2%
MWe	0
Tavg:	547 °F
RCS Pressure:	2235
RCS Boron Conc:	1670 ppm

RCS Leakrate	
Gross:	0.23 GPM
Unidentified	0.02 GPM
Charging Pps:	0.00 GPM

Accumulator Ref Levels	
A	6656
B	6608
C	6646

Abnormal Annunciators:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	

Current Tech Spec Action Statements: (Does Not Include “For Tracking Only Items”	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	

REACTOR OPERATOR (CONT'D)
UNIT RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B
Changes to Risk Significant Equipment:
No recent changes from last shift. OLRM: GREEN PROTECTED TRAIN: B
Upcoming Reactivity Management Activities:
Approval granted to withdraw rods to 125 steps to raise power to Raise Power to 3% to roll the Turbine per 3-GOP-301 Step 5.45. Place unit online and continue power increase to 30%.
Upcoming Major POD Activities:
SM has approved going to Mode 1 Roll the Turbine Place the Unit Online Continue power increase to 30%.
Upcoming ECOs to Hang and /or Release:
• Hang – None
Evolutions or Compensatory Actions in Progress:
NONE
General Information, Remarks, and Operator Work Around Status:
• Weather forecast is overcast skies with scattered pockets of severe rain. • U4 supplying Aux Steam • Air In-leakage = 0.0 SCFM

FOR TRAINING USE ONLY

TURKEY POINT | Reactivity Manipulation Table (USE ONLY AS GUIDELINE) | REACTOR ENGINEERING

Unit 3 Cycle 28 BOC Power Ascension Rev. 0

ELAPSED TIME	POWER (%)	CBD (Steps)	AFD (%)	RAOC (Limit)	BORON (ppm)	CHANGE (ppm)	DILUTE (gal)	BORATE (gal)
00:00	0.1	107	-0.00	N/A	1675	*	*	*
00:02	0.1	109	-0.00	N/A	1675	*	*	*
00:04	0.5	111	-0.00	N/A	1675	*	*	*
00:06	0.5	113	-0.00	N/A	1675	*	*	*
00:08	1.0	115	-0.1	N/A	1675	*	*	*
00:10	1.5	117	-0.1	N/A	1675	*	*	*
00:12	2.0	117	-0.1	N/A	1675	*	*	*
00:14	2.5	119	-0.2	N/A	1675	*	*	*
00:16	3.0	119	-0.2	N/A	1675	*	*	*
00:18	3.0	119	-0.2	N/A	1675	*	*	*
00:20	3.0	119	-0.2	N/A	1675	*	Note 3	*

NOTEs

1. Withdraw rods to establish a SUR of 1 dpm to raise power from 10^{-8} amps to the Point Of Adding Heat
2. The SUR should be limited to .5 dpm above the Point Of Adding Heat
3. Once power is leveled at ~ 3% dilute and operate control rods as required to maintain RCS temperature for current power plateau.

Site:	**Turkey Point Units 3 and 4 (PTN)**
Title:	**L-16-1 NRC EXAM SCENARIO 4**
LMS #:	**NRC 24**
LMS Rev Date:	**6/8/16** **Rev #: 0**
SEG Type:	☐ **Training** ☒ **Evaluation**
Program:	☐ **LOCT** ☒ **LOIT** ☐ **Other**
Duration:	**120 minutes**

Developed by:	**Brian Clark** Instructor/Developer	**6/163/16** Date
Reviewed by:	**Tim Hodge** *Instructor (Instructional Review)*	**6/22/16** *Date*
Validated by :	**Rocky Schoenhals** *SME (Technical Review)*	**6/22/16** *Date*
Approved by:	**Mark Wilson** *Training Supervision*	**6/22/16** *Date*
Approved by:	**Rocky Schoenhals** *Training Program Owner (Line)*	**6/22/16** *Date*

SCENARIO REFERENCES		
DOC NO.	**TITLE**	**REV**
3-EOP-E-0	REACTOR TRIP OR SAFETY INJECTION	12
3-EOP-ECA-0.0	LOSS OF ALL AC POWER	10
3-EOP-FR-P.1	RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION	3
3-GOP-100	FAST LOAD REDUCTION	12
3-ONOP-003.8	LOSS OF 120V VITAL INSTRUMENT PANEL 3P08	6
3-ONOP-041.1	REACTOR COOLANT PUMP OFF-NORMAL	11
3-ONOP-049.1	DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNEL	4
	PTN TECHNICAL SPECIFICATIONS	298

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
Terminal Objective	Given this simulator scenario and resources normally found in the Control Room, the operating crew will perform Control Room operations IAW approved plant procedures in order to maintain the integrity of the plant and the health and safety of the public.
Enabling Objectives:	Given this simulator scenario and resources normally found in the Control Room, operate in accordance with approved plant procedures, Operations Department Instructions, and management expectations: 1. (ALL) Demonstrate personnel SAFETY awareness in interactions with plant staff and outside agencies. 2. (ALL) Demonstrate ALARA awareness in interactions with plant staff and outside agencies. 3. (ALL) Exchange correct information using 3-point communication/Repeat-backs with Control Room personnel and other plant staff. 4. (ALL) Inform plant personnel and System of plant conditions, as needed. 5. (US) Employ timely and concise crew briefs where appropriate. 6. (ALL) Maintain awareness of plant status and control board indication. 7. (ALL) Correctly diagnose plant situations. 8. (ALL) Solve operational problems as they arise. 9. (RCO/BOP) Manipulate plant controls properly and safely. 10. (ALL) Demonstrate self-checking using STAR and peer checks(when required) 11. (US) Demonstrate command and control of the crew. 12. (US) Coordinate the input of crew members and other plant staff. 13. (US) Utilize the input of crew members and other plant staff. 14. (ALL) Demonstrate conservative decision making. 15. (ALL) Demonstrate teamwork. 16. (ALL) Respond to plant events using procedural guidance (OPs/ONOPs/EOPs) as applicable in accordance with rules of usage. 17. (RCO/BOP) Implement any applicable procedural immediate operator actions without use of references. 18. (SRO) Maintain compliance with Tech Specs. 19. (ALL) Identify/enter applicable Tech Spec action statements. 20. (ALL) Respond to annunciators using ARPs (time permitting). 21. (ALL) Maintain written communication, logs, and documentation as needed to permit post-event reconstruction. Continued on next page

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
	While addressing the following events: 1. PT-3-495 3C S/G Pressure Transmitter Fails High 2. 3C RCP Degraded Seals 3. 3P08 Loss Of Power 4. 3C RCP Seal Failure 5. Loss of All AC 6. 3B 4Kv Bus Stripping Relay Failure 7. Small Break LOCA 8. MOV-3-843B HHSI Discharge to Cold Leg Fails To Auto Open
Prerequisites:	None
Training Resources:	PTN Unit 3 Plant Simulator
Development References:	• TR-AA-220-1003, Initial NRC and Audit Exam Process • TR-AA-230-1003, SAT Development • TR-AA-230-1007, Conduct of Simulator Training and Evaluation • 0-ADM-232, Time Critical Action Program • OP-AA-100-1000, Conduct Of Operations • OP-AA-103-1000, Reactivity Management • 0-ADM-200, Operations Management Manual • 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage • WCAP-17711-NP, Pressurized Water Reactor Owners Group Westinghouse Emergency Response Guideline Revision 2-Based Critical Tasks
Protected Content:	N/A
Evaluation Method:	Performance Mode
Operating Experience:	None
Risk Significant Operator Actions:	Following a Loss Of All AC, complete bus stripping and restore power to the 3B 4KV bus prior to actuating SI and within 30 minutes of the loss of power. During a SBLOCA, establish at least one train of HHSI flow prior to completing 3-EOP-E-0, Attachment 3, and within 30 minutes the HHSI pump starting.

TASKS ASSOCIATED WITH THIS SIMULATOR EXERCISE GUIDE

SRO TASK #	TASK TITLE
02005015500	RESPOND TO A LOSS OF ALL A.C. POWER
02028033500	AUTHORIZE UNIT TRIP
02041029300	EVALUATE AND RESPOND TO A LOW PRESSURIZER PRESSURE
02041044300	EVALUATE AND RESPOND TO HIGH RCP NUMBER ONE SEAL LEAKOFF
02089026300	AUTHORIZE FAST LOAD REDUCTION
02200001500	RESPOND TO UNIT TRIP
02200021500	RESPOND TO LOSS OF COOLANT ACCIDENT
02200022500	DIAGNOSE CAUSE OF SAFEGUARDS ACTUATION
02200044500	RESPOND TO HIGH STEAM GENERATOR LEVEL
02200050500	RESPOND TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION

RO TASK	TASK TITLE
01005015500	RESPOND TO A LOSS OF ALL A.C. POWER
01041044300	EVALUATE AND RESPOND TO HIGH RCP NUMBER ONE SEAL LEAKOFF
01089026300	RESPOND TO / ADJUST TURBINE DURING FAST LOAD REDUCTION
01200001500	RESPOND TO UNIT TRIP
01200021500	RESPOND TO LOSS OF COOLANT ACCIDENT
01200022500	DIAGNOSE CAUSE OF SAFEGUARDS ACTUATION
02200044500	RESPOND TO HIGH STEAM GENERATOR LEVEL
02200050500	RESPOND TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION

UPDATE LOG:	
NOTES:	Place this form with the working copies of lesson plans and/or other materials to document changes made between formal revisions. For fleet-wide training materials, keep electronic file of this form in same folder as approved materials. Refer to TR-AA-230-1003 SAT Development for specific directions regarding how and when this form shall be used.

Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / REVIEWER	DATE / DATE
0-0	Initial Revision	Revised for L-16-1 NRC Exam	2108338	Note 5 / Note 5	Note 5 / Note 5
0-1					
0-2					
0-3					
0-4					
0-5					

1. Individual updating lesson plan or training material shall complete the appropriate blocks on the Update Log.
2. Describe the change to the lesson plan or training materials.
3. State the reason for the change (e.g., reference has changed, typographical error, etc.)
4. Preparer enters name/date on the Update Log and obtains Training Supervisor approval.
5. Initial dates and site approval on cover page.

SCENARIO SUMMARY

Initial Conditions:
The plant is at 100% power (MOL). Online risk is green. B train is protected on both units.

Equipment OOS

- The 3A RHR pump and 3A1 Circulating Water pump are OOS.

Event 1
After the crew takes the shift, PT-3-495, 3C S/G Pressure, slowly fails high. The BOP will take manual control of 3C S/G level and restore level to normal. The crew will use the ARP or 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, to select operable channels and restore 3C S/G level control to automatic. The US will enter 3-ONOP-049.1 to verify all required actions are complete and to determine which bistables need to be tripped.

Event 2
Once equipment is restored to automatic, the 3C RCP seals will start degrading. The US will enter 3-ONOP-041.1, Reactor Coolant Pump Off-Normal, and commence a unit shutdown using 3-GOP-100, Fast Load Reduction.

Event 3
After the crew starts the Load reduction, Vital AC Bus 3P08 loses power. The Crew will enter 3-ONOP-003.8, Loss of Vital AC Bus 3P08. The crew will dispatch an operator to 3P08 to attempt to restore power. The 3A and 3B S/G level controllers shift to manual. The 3C S/G level controller shifts to manual on the Backup Controller. The BOP will select operable control channels for the 3A and 3B S/G and then restore automatic control. The operator dispatched to restore power will report the Main Breaker for 3P08 will not close. Electrical Maintenance estimates it will take 2 hours to replace the breaker.

Event 4
After the US completes the Tech Spec review for the loss of 3P08, the 3C RCP Seal degrades to the point that requires a Reactor trip and stopping the 3C RCP. The RCO will trip the Reactor, verify the Reactor is tripped, stop 3C RCP, close 3C RCP CBO Isolation Valve, CV-3-303C, and close PCV-3-455A, PZR Spray Valve Loop C. The crew will enter 3-EOP-E-0, Reactor Trip or Safety Injection, and complete the Immediate Operator Actions.

Event 5
After the RCO completes tripping the 3C RCP, a Loss of AC Power will occur. The 3A and 3B Emergency Diesel Generators will start, but neither will energize its respective 4KV bus. The crew will enter 3-EOP-ECA-0.0, Loss of All AC Power.

SCENARIO SUMMARY

Event 6

The 3A 4KV Bus is locked out, so the US will direct the BOP perform Attachment 2, 3B 4KV Bus Stripping. The BOP will open the 3B ICW pump, 3B CCW pump, 3C CCW pump, 3B Load Center, and 3D Load Center breakers to complete bus stripping. Once Bus Stripping is complete, the 3B EDG will automatically energize the 3B 4KV Bus.

Event 7

Once the 3B 4KV Bus is energized, the crew will transition back to 3-EOP-E-0. Shortly after the transition, a Small Break LOCA will occur.

Event 8

When SI actuates, the slave relay which opens MOV-3-843B fails to actuate. The RCO may open MOV-3-843B any time after SI actuates. If the RCO doesn't open MOV-3-843B, the BOP will open it during the performance of 3-EOP-E-0 Attachment 3, Prompt Action Verifications.

The scenario is terminated after the crew transitions to 3-EOP-FR-P.1 and determines a soak is required, or at the Lead Evaluator's discretion once all critical tasks have been evaluated.

Event	CRITICAL TASKS	
6	CT1	**Re-energize 3B 4KV Bus** Following a Loss Of All AC, complete bus stripping and restore power to the 3B 4KV bus prior to actuating SI and within 30 minutes of the loss of power. *Safety Significance*: The failure to energize an AC emergency bus in a timely manner constitutes a misoperation or incorrect crew performance in which the crew does not prevent a degraded emergency power capacity. The 30 minute time limit is based minimizing DC bus battery depletion and the requirement to manually load a de-energized DC bus battery charger onto the operating EDG. (0-ADM-232, Attachment 1, Time Critical Operator Actions)
7	CT2	**Open MOV-3-843B** During a SBLOCA, establish at least one train of HHSI flow prior to completing 3-EOP-E-0, Attachment 3, and within 30 minutes the HHSI pump starting. *Safety Significance*: Failure to establish at least one train HHSI flow constitutes a misoperation or incorrect crew performance in which the crew does not prevent "degraded emergency core cooling system (ECCS) capacity." The 30 minute time limit is based on the requirement to limit the time the pump is operating at shutoff head to less than 30 minutes. (0-ADM-231 Attachment 1, Time Critical Operator Actions)

SEQUENCE OF EVENTS	
Event #	**Description**
1.	PT-3-495 3C S/G Pressure Transmitter Fails High
2.	3C RCP Degraded Seals
3.	3P08 Loss Of Power
4.	3C RCP Seal Failure
5.	Loss of All AC
6.	3B 4Kv Bus Stripping Relay Failure
7.	Small Break LOCA
8.	MOV-3-843B HHSI Discharge to Cold Leg Fails To Auto Open

SIMULATOR SET UP INSTRUCTIONS	
Check	**Action**
_____	Restore IC-1 (100% MOL) or equivalent IC.
_____	Place the Simulator in RUN.
_____	Stop the 3A1 Circ Water Pump
_____	Open & execute lesson file L-16-1 N4
_____	Ensure the following lesson steps are triggered: • SETUP - 3A RHR PUMP OOS • SETUP - 3A1 CWP OOS • SETUP EVENT 6 - 3B BUS STRIPPING FAILURE • SETUP EVENT 8 - B TRAIN SLAVE RELAY FAILURE
_____	• Place an ECO tag on the 3A RHR pump place it in PTL: • Place an ECO tag on the 3A1 CWP place it in Stop.
_____	Verify the trend for 3A1 Screen on the TWS DP Recorder is clear.
_____	Ensure Rod Group Step Counters have completed stepping out.
_____	Allow the plant to stabilize.
_____	Acknowledge any alarms and freeze Simulator.
_____	Ensure B train is protected train on VPA.
_____	Perform the SIMULATOR OPERATOR CHECKLIST or equivalent.
_____	Place TURNOVER SHEETS on RO's desk or give to the Lead Evaluator.

BRIEFINGS

- Shift turnover information is attached to the back of this guide.
- Ensure all applicants are prior briefed on Appendix E of NUREG 1021, Policies and Guidelines For Taking NRC Examinations.
- Conduct a Crew Pre-brief to cover turnover information. Shift turnover information is attached to the back of this guide.

US: ______________________________

RCO: ______________________________

BOP: ______________________________

SCENARIO NOTE

0-ADM-211 Prudent Operator Actions - If redundant stand-by equipment is available and ready, the operator is permitted to start the redundant equipment for failed or failing operating equipment. Immediate follow up of applicable ARPs and ONOPs (AOPs) shall occur as required.

Critical Tasks are highlighted in red.	**Simulator Operator Actions** are highlighted in blue.	**Operator Verifiable Actions** are Highlighted in green.

EVENT 1 - PT-3-495 3C S/G PRESSURE TRANSMITTER FAILS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 1 – PT-3-495 FAILS HIGH**	
		BOP: • Recognizes and reports PT-3-495 failure. **PROMPT ACTIONS** • Takes manual control of 3C S/G level control valve FCV-3-498. • Restores 3C S/G level to normal.
		RCO: Addresses Alarm Response for C5/3, 6/3, SG C Level Deviation. • CHECK LI-3-496 or LI-3-498, B\C STM GEN LEVEL controlling channel for SG Level deviation. • CHECK Feedwater Controllers FIC-3-498A or FIC-3-498B for indications of failure, alarm, or input signal failures. • CHECK Feedwater Controller Inputs • IF alarm is due to instrument failure, THEN REFER TO 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
		US: Enters and directs actions of 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, for response.

EVENT 1 - PT-3-495 3C S/G PRESSURE TRANSMITTER FAILS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE The crew may use the ARP to select an operable channel and restore automatic level control.	BOP: • Verify PT-3-495 failure by channel check comparison. • Verify no off-normal conditions exist on PT-3-496. • Place 3C S/G Steam Flow Control Transfer Switch to FT-3-495 (Yellow) • Place 3C S/G Feed Water Flow Control Transfer Switch to FT-3-496 (Yellow) • Ensure 3C S/G level is returned to auto.
	BOOTH OPERATOR WCC/I&C: Acknowledge the report. If asked, I&C would like to be present before tripping bi-stables. They will report to the control room within an hour.	BOP: Notifies WCC to initiate PWO and I&C for troubleshooting.
		US: Reviews Tech Specs • LCO 3.3.1 Functional Unit 12 - Action 6 within 6 hours trip bi-stables • LCO 3.3.2 Functional Unit 1e, 1f, and 4d. - Action 15 within 6 hours trip bi-stables
	LEAD EVALUATOR After S/G level control is restored to auto and the US completes the review of Tech Specs, at the Lead Evaluators discretion, direct the Booth Operator to trigger the next event.	US: Conducts crew brief.

EVENT 2 – 3C RCP DEGRADED SEALS		
3-ONOP-041.1, REACTOR COOLANT PUMP OFF-NORMAL		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the lead evaluator, **trigger EVENT 2 - 3C RCP DEGRADED SEAL**	
		BOP: Reviews ARPs A6/4 and A7/4. • Refer to 3-ONOP-041.1, Reactor Coolant Pump Off Normal.
		RCO: Responds to RCP Alarms • Check P2 pressure equal to or less than 1741 psig on VPA **(NO)** • Check P3 pressure greater than 975 psig on VPA **(YES)**
		US: Enter and direct the actions of 3-ONOP-041.1, Reactor Coolant Pump Off Normal.
		US: Reviews Foldout Page with the crew. • RCP Stopping Criteria • RCP Seal Criteria For Stopping RCP • Fast Load Reduction Criteria **(YES, 3C RCP Seal Stage - greater than 1700 and / or CBO exceeds 3.7 gpm)** • Exceeding Vibration Or Stator Temperature Limits • RCP Vibration Assessment Criteria
		US: Enter and direct the actions of 3-GOP-100, Fast Load Reduction

EVENT 2 – 3C RCP DEGRADED SEALS		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: • Directs actions to reduce Rx power per 3-GOP-100 • Completes Attachment 3 • Brief the crew per Attachment 4 **Steps 1-2**
		US: Reviews Foldout page with crew. • 3-EOP-E-0 Transition Criteria • Notify Chemistry Department • Boration Stop Criteria • Restore Blender to AUTO **FOLDOUT PAGE**
	BOOTH OPERATOR Acknowledge notifications.	BOP: Notify The Following Of Fast Load Reduction • System Dispatcher • Plant personnel using the Page Boost • Chemistry to start RCS sampling is required according to Tech Spec Table 4.4-4. **Step 3**

EVENT 2 – 3C RCP DEGRADED SEALS		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Begin Boration For Initial Tavg Effect • Set the Boric Acid Totalizer to total boric acid volume value determined on Attachment 3. • Place the Reactor Makeup Selector Switch to BORATE. • Place the RCS Makeup Control Switch to START. • Adjust FC-3-113A, Boric Acid Flow Controller, to achieve 40 gpm boric acid flow as indicated on FR-3-113. • WHEN Tavg begins to lower from the boration, adjust FC-3-113A, Boric Acid Flow Controller, to load reduction value from Attachment 3. **Step 4**
		US: Determine Turbine Load Reduction in MW CNTRL **Step 5**
		BOP: Initiate Turbine Load Reduction in MW CNTRL • Select MW CNTRL • Set TARGET power level – MW VALUE from Attachment 3 • Set RAMP RATE – MW/M VALUE FROM Attachment 3. • Check T_{avg} has lowered 1° to 2°F from the initial value prior to boration. • Depress GO • Ensure FC-3-113A, Boric Acid Flow Controller, has been adjusted to the load reduction boration rate. Go to Step 10 **Step 6**

EVENT 2 – 3C RCP DEGRADED SEALS		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Monitor Load Reduction • Adjusts power reduction rate to Maintain Tavg/Tref within the expected ΔT identified in Attachment 3. • Monitors S/G level control to ensure feed reg valves properly maintain level control in automatic. • Refer to Enclosure 1 for expected alarms. **Step 10**
	BOOTH OPERATOR Respond as SNPO. If asked, report idle Charging Pump is ready for start.	RCO: • Maintain pressurizer level to ensure that automatic pressurizer level control maintains level on program. • If needed, start 2nd Chg Pp and place 2nd orifice in service. • Adjusts boration rate to Maintain Tavg/Tref within the expected ΔT identified in Attachment 3. • Refer to Enclosure 1 for expected alarms. **Step 10**
		RCO: Monitor Boration Rate • Monitor for excessive rod movement by monitoring TR-3-409D, Rod Position Bank D. • Determine if Insertion Limit and Bank D position are converging at a rate that will cause rod insertion limit alarms. • Adjust power reduction rate as needed to control rod insertion • Increase boration rate and/or total amount as necessary to limit control rod insertion **Step 11**

EVENT 2 – 3C RCP DEGRADED SEALS		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: • Monitor Annunciator B 8/1, ROD BANK LO LIMIT – CLEAR • Monitor B 8/2 ROD BANK A/B/C/D EXTRA LO LIMIT – CLEAR **Steps 12-13**
		US: Have SM refer to the following procedures: • 0-EPIP-20101, DUTIES OF EMERGENCY COORDINATOR • 0-ADM-115, NOTIFICATION OF PLANT EVENTS **Step 14**
	<u>LEAD EVALUATOR</u> Once power has been reduced by a minimum of 5%, at the Lead Evaluators discretion, proceed to the next Event.	RCO: Energize Pressurizer Backup Heaters **Step 15**

EVENT 3 – 3P08 LOSS OF POWER		
3-ONOP-003.8, LOSS OF 120V VITAL INSTRUMENT PANEL 3P08		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
	NOTE Ensure the Simulator is in **RUN** before the crew enters the Simulator.	
	BOOTH OPERATOR When directed by the Lead Evaluator, **trigger EVENT 3 – LOSS OF 3P08**	
		RCO/BOP • Respond to various alarms. • Report a loss of power to 3P08 - Power Range N-43 Failure (NIS Racks Channel III Lights Out) - Loss of Channel III Vital Instrumentation/Indications
		BOP: Review ARP F 1/2, VITAL AC BUS INVERTER TROUBLE. • Ensure Auto transfer to CVT **(NO)** • Determine which inverter has trouble. (3P08 Status Light) • Place spare inverter in service using 3-ONOP-003.8, Loss of Vital Instrument Panel. • Refer to TS 3.8.3 for additional actions.
		US: Enter and direct the actions of 3-ONOP-003.8, Loss Of 120V Vital Instrument Panel 3P08.
	NOTE Step 1 is an immediate action step.	RCO: • Check If A Reactor Trip has occurred **(NO)** • Check If A Reactor Trip is required. **(NO)** **STEP 1**

EVENT 3 – 3P08 LOSS OF POWER		
3-ONOP-003.8, LOSS OF 120V VITAL INSTRUMENT PANEL 3P08		
	BOOTH OPERATOR When dispatched to restore power 3P08 per Attachment 1, wait 3 minutes and then report nothing obviously wrong. Inform the control room you are going to attempt to re-energize the bus, then **trigger EVENT 3 - 3P08/3P23 ALL BREAKERS OFF**.	US: Reviews Foldout Page with the crew. • Verify Turbine Inlet Pressure Control selected to PT-3-447 (CH.4 Yellow) • Dispatch operator to restore power to 3P08 using Attachment 1. • If a Reactor Trip has occurred **(NO)** • If power available to 3P09, then perform the following: - Place CS-3-1608, Power Selector switch for H/A-3-1608, to 3P09. - Restore H/A-3-1608 to Auto. **FOLDOUT PAGE**
		RCO/BOP: Check Unit Operating In Modes 1 Through 3 Prior To Loss Of 3P08. **STEP 2**
		RCO: • Verify Pressurizer PORVs – CLOSED • Check Pressurizer Level control switch in Position 1 (CH 1 & 2) • Control charging flow using the 3A or 3B charging pumps in AUTO speed control - Starts 3A or 3B Charging pump **STEP 3**
		BOP: Control 3C Steam Generator Water Level by using MANUAL control on the backup Controller. **Step 4a**

EVENT 3 – 3P08 LOSS OF POWER		
3-ONOP-003.8, LOSS OF 120V VITAL INSTRUMENT PANEL 3P08		
		BOP: Check 3A Steam Generator Feedwater Primary Controller in AUTOMATIC Mode **(NO)** • Select 3A Steam Generator Feedwater Flow Control Transfer switch to FI-3-476 (Yellow) • Select 3A Steam Generator Steam Flow Control Transfer switch to FI-3-475 (Yellow) • Select 3A Steam Generator Level Control Transfer switch to LI-3-478 (Red) • On 3A Primary Controller, press "A" button for 2 seconds until backlit to return controller to AUTOMATIC Mode **STEP 4b**
		BOP: Check 3B Steam Generator Feedwater Primary Controller in AUTOMATIC Mode **(NO)** • Select 3B Steam Generator Feedwater Flow Control Transfer switch to FI-3-486 (Yellow) • Select 3B Steam Generator Steam Flow Control Transfer switch to FI-3-485 (Yellow) • Select 3B Steam Generator Level Control Transfer switch to LI-3-488 (White) • On 3B Primary Controller press "A" button for 2 seconds until backlit to return controller to AUTOMATIC Mode **STEP 4c**
		RCO/BOP Maintain Plant Parameters - STABLE **STEP 5**

EVENT 3 – 3P08 LOSS OF POWER		
3-ONOP-003.8, LOSS OF 120V VITAL INSTRUMENT PANEL 3P08		
	BOOTH OPERATOR Report the Main breaker for 3P08 will not close. Electrical Maintenance will have to replace it. Estimated repair time is 2 hours.	**RCO:** Check Power Restored To 3P08 **(NO)** IF power can NOT be restored to 3P08 within 1 hour, THEN perform the actions required by Technical Specifications as directed by the NPS, Return to Step 1. **STEP 6**
	LEAD EVALUATOR This Tech Spec review can be quite extensive. The Lead evaluator may choose to have the US identify Tech Specs with actions statements of 2 hours or less as a follow up question. When the US starts the Tech Spec review at the Lead Evaluators discretion proceed to the next event.	**US:** Review Tech Specs • LCO 3.3.1 Functional Unit 17.b due to the loss of PT-3-446 - Action 7 - within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3. • LCO 3.4.4 due to the loss of PORV PCV-3-456 - Action a - within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. • LCO 3.8.3.1.h due to the loss of 3P08 - Action c - Reenergize the A.C. vital panel within 2 hours or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours.

EVENT 4 - 3C RCP SEAL FAILURE		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator**, trigger EVENT 4 - 3C RCP SEAL FAILURE**	
		RCO: • Reports rising CBO Flow and #3 seal differential pressure. • RCP CBO flow exceeds 4.1 gpm or any Seal Stage differential pressure exceeds 2000 psid, recommends a Manual Rx trip.
		US: • Directs 3-EOP-E-0 response after auto Reactor trip.
		RCO: • Manually trips Reactor.
	LEAD EVALUATOR When the RCO completes the required actions for the RCP seal package failure, proceed to the next event.	RCO: • Verifies Reactor Trip • After verifying Rx Trip - Trips 3C RCP - Closes CV-3-303C, CBO Isolation valve CV-303C. - Closes PCV-3-455A, PZR Spray Valve Loop C **STEP 1**
		BOP: Verify Turbine Trip **STEP 2**
		BOP: Verify Power To Emergency 4 KV Buses **STEP 3**
		RCO: Checks If SI Is Actuated **STEP 4**

EVENT 5 - LOSS OF ALL AC		
3-EOP-ECA-0.0, LOSS OF ALL AC POWER		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the Lead Evaluator, **trigger EVENT 5 LOSS OF ALL AC**.	
CT1 **Start Time**	**NOTE** Record the time offsite power is lost for Time Critical Task verification.	
	NOTE Step 1 and Step 2 are IMMEDIATE ACTION steps	RCO: Verify Reactor Trip **STEP 1**
		BOP: Verify Turbine Trip **STEP 2**
		RCO: Check If RCS Is Isolated **(NO)** • Close Letdown Isolation valves CV-3-200A/B/C **STEP 3**
		BOP: Verify Proper AFW Flow **STEP 4**
		US: The Unit Supervisor shall evaluate plant conditions and establish EDG Priority. Since the 3A Bus is locked out the US determines the 3B EDG is the priority. **STEP 5**

EVENT 5 - LOSS OF ALL AC		
3-EOP-ECA-0.0, LOSS OF ALL AC POWER		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Try To Restore Power To 3A OR 3B 4KV Bus • Check EDG Priority – 3A **(NO)** • Go to Step 5.o **STEP 5.a**
		BOP: • Check 3B Bus Lockout Relay – RESET • Check 3B EDG Lockout – RESET • Check 3B EDG – RUNNING • Check 3B 4KV Bus – ENERGIZED (NO) • Go to step 5.t **STEP 5.o – 5.r**
		BOP: Verify 3B 4KV bus stripping using Attachment 2. **STEP 5.t**

EVENT 6 - 3B 4KV BUS STRIPPING RELAY FAILURE		
3-EOP-ECA-0.0, LOSS OF ALL AC POWER ATTACHMENT 2		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Step 1 is N/A **STEP 1**
CT1 **Stop Time**	**NOTE** The BOP will have to manually strip the loads highlighted in red. When all loads are stripped, the 3B EDG will energize the 3B 4KV Bus. **Re-energize 3B 4KV Bus** Following a Loss Of All AC, complete bus stripping and restore power to the 3B 4KV bus prior to actuating SI and within 30 minutes of the loss of power.	BOP: Verify the following breakers OPEN • 3AB22, 3B 4KV BUS TIE TO 3A OR 3C 4KV BUS • 3AB05, STARTUP TRANSFORMER 3B 4KV BUS SUPPLY • 3AB02, AUXILIARY TRANSFORMER 3B BUS SUPPLY • 3AB10, HEATER DRAIN PUMP 3B • 3AB21, CONDENSATE PUMP 3B • 3AB12, SAFETY INJECTION PUMP 3B • 3AB15, RESIDUAL HEAT REMOVAL PUMP 3B • 3AB13, COMPONENT COOLING WATER PUMP 3B • 3AB01, REACTOR COOLANT PUMP 3B • 3AB06, REACTOR COOLANT PUMP 3C • 3AB17, INTAKE COOLING WATER PUMP 3B • 3AB11, TURBINE PLANT COOLING WATER PUMP 3B • 3AB16, CIRCULATING WATER PUMP 3B1 • 3AB18, CIRCULATING WATER PUMP 3B2 • 3AB09, 3B LOAD CENTER • 3AB14, 3D LOAD CENTER **STEP 2**
		BOP: Step 3 is N/A **STEP 3**

EVENT 6 - 3B 4KV BUS STRIPPING RELAY FAILURE		
3-EOP-ECA-0.0, LOSS OF ALL AC POWER ATTACHMENT 2		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: • Verify 3AD05, INTAKE COOLING WATER PUMP 3C BREAKER, is open. • Verify 3AD04, COMPONENT COOLING WATER PUMP 3C BREAKER, is open **STEP 4**
		BOP: Notify Unit Supervisor that 3B 4KV Bus stripping is complete. **STEP 5**

EVENT 6 - 3B 4KV BUS STRIPPING RELAY FAILURE		
3-EOP-ECA-0.0, LOSS OF ALL AC POWER		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: • Verify SI – RESET • Check 3B 4KV Bus – ENERGIZED • Observe CAUTION and NOTE prior to Step 5.f, and return to Step 5.f **STEPs 5.u – 5.w**
		Crew Verify required Safeguards equipment – OPERATING **STEP 5.f**
	LEAD EVALUATOR When the crew returns to 3-EOP-E-0 proceed to the next event.	US: Check status of 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES, prior to entering this procedure – MONITORED FOR INFORMATION ONLY. **(NO)** • Implement FRPs as required • Return to procedure and step in effect. **STEP 5.g**

EVENT 7 – SMALL BREAK LOCA		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the Lead Evaluator, **trigger EVENT 7 - SBLOCA**	
		RCO: Verify Reactor Trip **STEP 1**
		BOP: Verify Turbine **STEP 2**
		BOP: Verify Power To Emergency 4 KV Buses **STEP 3**
CT2 **Start Time** __________	**NOTE** Record the time SI actuates for Time Critical task verification	RCO: Checks If SI Is Actuated **STEP 4**
		US: Reviews Steps 1 - 4 of 3-EOP-E-0 with the crew.

EVENT 7 – SMALL BREAK LOCA		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: Reviews FOP for 3-EOP-E-0 • Adverse Cntmt **(Will Be Met)** • RCP Trip Criteria **(Not Running)** • Faulted S/G Isolation • Ruptured S/G Isolation • AFW Sys Operation Criteria • CST Makeup Water Criteria • RHR System Operation Criteria **(Starts a Timer)** • Loss of Offsite Power or SI on the Other Unit • Loss of Charging Criteria **FOLDOUT PAGE**
	NOTE The actions of Attachment 3 are listed beginning on page 34.	BOP: • Continues with ATTACHMENT 3 to complete The Prompt Action Verifications. **STEP 5**
		RCO: • Check AFW Pumps – AT LEAST TWO RUNNING **STEP 6**
	BOOTH OPERATOR If asked to manually align AFW valves, acknowledge request but take no action. If later asked status update, report still working on it.	RCO: • Verify AFW Valve Alignment – PROPER EMERGENCY ALIGNMENT **(NO)** • Manually align Valves **STEP 7**

EVENT 7 – SMALL BREAK LOCA		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Verify Proper AFW Flow: • Check Narrow Range Level in at least one S/G – GREATER THAN 7%[27%] • Maintain feed flow to S/G until Narrow Range Levels between 21%[27%] and 50% **STEP 8**
		RCO: All RCP Thermal Barrier Alarms – CLEAR **STEP 9**
		RCO: • Check RCS Temperatures: – Check RCPs – ANY RUNNING **(NO)** – Check RCS Cold Leg temperatures stable between 545°F and 547°F or trending down to 547°F **(NO)** • IF T_{COLD} is decreasing, THEN perform the following: – Stop dumping steam. – If cooldown continues, and is due to excessive feed flow, then reduce total feed flow to 400 gpm until Narrow Range Level greater than 7%[27%] in at least one S/G. – IF cooldown continues AND is due to excessive steam flow, THEN close Main Steamline isolation and Bypass valves. **STEP 10**

EVENT 7 – SMALL BREAK LOCA		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Check PRZ PORVs, Spray Valves, And Excess Letdown Isolated: **STEP 11**
		RCO: Check If RCPs Should Be Stopped: • RCPs – ANY RUNNING **(NO)** **STEP 12**
		RCO: • Check If S/Gs Are Faulted: **(NO)** **STEP 13**
		RCO: • Check If S/G Tubes Are Ruptured: **(NO)** **STEP 14**
		RCO: If RCS Is Intact **(NO)** US: • Perform the following: – Monitor Critical Safety Functions using 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES. – INTEGRITY Critical Safety Function Status Tree is RED – Go to 3-EOP-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION **STEP 15**

EVENT 7 – SMALL BREAK LOCA		
3-EOP-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: • Conducts EOP transition brief. • Directs 3-EOP-FR-P.1 response.
		US: Reviews FOP for 3-EOP-FR-P.1 with the crew. • Containment Adverse **(YES)** **FOLDOUT PAGE**
		RCO: Check RCS Pressure – GREATER THAN 275 PSIG [575 PSIG] **Step 1**
		RCO: Check RCS Cold Leg Temperatures decreasing **(NO)** • Go to Step 3 **Step 2**
		RCO: Check PRZ PORV Block Valves **Step 3**
		RCO: Check If PRZ PORVs Should Be Closed **Step 4**
		RCO: Check High-Head SI Pumps – ANY RUNNING **Step 5**
		RCO: Check If SI Can Be Terminated **(NO)** **Go to Step 23** **Step 6**

EVENT 7 – SMALL BREAK LOCA		
3-EOP-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Determine If RCS Temperature Soak Is Required **(YES)** **Step 23**
The scenario is terminated after the crew transitions to 3-EOP-FR-P.1 and determines a soak is required, or at the Lead Evaluator's discretion once all critical tasks have been evaluated.		
***** END OF SCENARIO *****		

EVENT 8 – MOV-3-843B HHSI DISCHARGE TO COLD LEG FAILS TO AUTO OPEN		
3-EOP-E-0 Attachment 3 – Prompt Action Verifications		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Check Load Centers Associated With Energized 4 KV Buses – ENERGIZED **(NO** 3A & 3C no power) **STEP 1**
		BOP: Verify Feedwater Isolation: **STEP 2**
		BOP: Check If Main Steam Lines Should Be Isolated **STEP 3**
	BOOTH OPERATOR If dispatched to verify MOV-3-1426 and MOV-3-1427 closed, wait 3 to 5 minutes and then report the valves are closed.	BOP: Verify Containment Isolation Phase A Valve White Lights On VPB – ALL BRIGHT **(NO)** • Manually actuate Containment Isolation Phase A. • Dispatch Operator to verify closed MOV-3-1426, and MOV-3-1427, • MOV-3-6386 (MOV-3-281 closed) • Containment Purge Valves (fuses pulled) **STEP 4**
		BOP: Verify Pump Operation: • At least two High-Head SI Pumps – RUNNING • Both RHR Pumps – RUNNING **(NO)** - 3A RHR pump OOS **STEP 5**

EVENT 8 – MOV-3-843B HHSI DISCHARGE TO COLD LEG FAILS TO AUTO OPEN		
3-EOP-E-0 Attachment 3 – Prompt Action Verifications		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Verify Proper CCW System Operation: • CCW Heat Exchangers – THREE IN SERVICE • CCW Pumps – ONLY TWO RUNNING **(NO)** - Starts 3C CCW pump • CCW Headers – TIED TOGETHER • MOV-3-626, RCP Thermal Barrier CCW Outlet – OPEN **STEP 6**
		BOP: Verify Proper ICW System Operation: • Verify ICW Pumps – AT LEAST TWO RUNNING **(NO)** - Starts the 3C ICW pump • Verify ICW To TPCW Heat Exchanger – ISOLATED: • Check ICW Headers – TIED TOGETHER **STEP 7**
		BOP: Check Emergency Containment Coolers – ONLY TWO RUNNING **STEP 8**
		BOP: Verify Unit 3 Containment Purge Exhaust And Supply Fans – OFF **STEP 9**
		BOP: → Verify -Containment Spray NOT Required **STEP 10**
		BOP: Verify SI – RESET **STEP 11**

EVENT 8 – MOV-3-843B HHSI DISCHARGE TO COLD LEG FAILS TO AUTO OPEN		
3-EOP-E-0 Attachment 3 – Prompt Action Verifications		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
CT1 **Stop Time**	**Open MOV-3-843B** During a SBLOCA, establish at least one train of HHSI flow prior to completing 3-EOP-E-0 Attachment 3 and within 30 minutes the HHSI pump starting.	BOP: Verify SI Valve Amber Lights On VPB – ALL BRIGHT **(NO)** • Opens MOV-3-843B • Opens MOV-3-744B • No power to equipment powered by the 3A Bus **STEP 12**
		BOP: Verify SI Flow: • RCS pressure – LESS THAN 1625 PSIG[1950 PSIG] • High-Head SI Pump flow indicator – CHECK FOR FLOW **STEP 13**
	BOOTH OPERATOR When requested, **trigger LOA – ALIGN U-4 HHSIs TO U3 RWST**	BOP: Realign SI System: • Check Procedure Entry Status – E-0 ENTERED FROM 3-ONOP-047.1, LOSS OF CHARGING FLOW IN MODES 1THROUGH 4 **(NO)** • Verify Unit 3 High-Head SI Pumps – TWO RUNNING **(NO)** - Stop one Unit 4 High-Head SI Pump and place in standby • Direct Unit 4 Reactor Operator to align Unit 4 High-Head SI Pump suction to Unit 3 RWST using Attachment 1. **STEP 14**
		BOP: Verify Containment Isolation Phase A – RESET **STEP 15**

EVENT 8 – MOV-3-843B HHSI DISCHARGE TO COLD LEG FAILS TO AUTO OPEN		
3-EOP-E-0 Attachment 3 – Prompt Action Verifications		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Reestablish RCP Cooling: • Check RCPs – AT LEAST ONE RUNNING **(NO)** • Go to Step 17 **STEP 16**
		BOP: Verify Control Room Ventilation Isolation: **STEP 17**
	BOOTH OPERATOR When requested, **trigger LOA – PLACE PAHM IN SERVICE,** wait 3 to 5 minutes and then report task complete.	BOP: Place Hydrogen Monitors In Service Using 3-NOP-094, CONTAINMENT POST ACCIDENT MONITORING SYSTEM **STEP 18**
		BOP: Verify All Four EDGs – RUNNING **STEP 19**
	BOOTH OPERATOR If asked to start one train of chilled water, acknowledge request. No action required	BOP: • Verify Power To 3A and 3B Emergency 4KV Buses **(NO)** - Inform US • Check 3A and 3B 4KV buses energized from offsite power. **(NO)** - Start one train of Chilled Water **STEP 20**
		BOP: Notify Unit Supervisor Of The Following: • Attachment 3 is complete • Any safeguards equipment that is NOT In the required condition • Status of Containment pressure continuous action **STEP 21**

Discussion Points are intentionally NOT included in evaluated scenarios. However, space is available below to document follow-up questions when further information is required to determine an evaluation outcome.

FOLLOW-UP QUESTIONS

QUESTION #1

ANSWER #1

QUESTION #2

ANSWER #2

SIMULATOR POST-SCENARIO RESTORATION:

_______ 1. Restore per Simulator Operator Checklist.

_______ 2. Once exams are complete, restore from SEI-19, Simulator Exam Security.

OPERATIONS SHIFT TURNOVER REPORT

UNIT 3 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B	UNIT 4 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B

ONCOMING CREW ASSIGNMENTS

Shift Mgr:			**Inside SNPO:**	
Field Supv.:			**Outside SNPO:**	
Admin RCO:			**ANPO:**	
Unit 3			**Unit 4**	
Unit Supv.:			**Unit Supv.:**	
RCO:			**RCO:**	
NPO:			**NPO:**	

PLANT STATUS

Unit 3			**Unit 4**	
Mode:	1		**Mode:**	1
Power:	100%		**Power:**	100%
MWe:	842		**MWe:**	842
Gross Leakrate:	.22 gpm		**Gross Leakrate:**	0.03 gpm
RCS Boron Conc:	745 ppm		**RCS Boron Conc:**	642

Operational Concerns:

3A RHR pump taken OOS 4 hours ago for an oil change, expected back by the end of this shift.
3A1 Circ Water pump OOS. Tripped on over current, Electrical Maintenance is investigating.

U3 Anticipated LCO Actions:

None

U4 Anticipated LCO Actions:

None

Results of Offgoing Focus Area:

UNIT 3 STATUS

REACTOR OPERATOR

UNIT RISK: GREEN (ACCEPTABLE)   PROTECTED TRAIN: B

Mode:	1
Power:	100%
MWe	842
Tavg:	580°F
RCS Pressure:	2235
RCS Boron Conc:	745 ppm

RCS Leakrate	
Gross:	0.22 GPM
Unidentified	0.04 GPM
Charging Pps:	0.00 GPM

Accumulator Ref Levels	
A	6656
B	6608
C	6646

Abnormal Annunciators:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	

Current Tech Spec Action Statements: (Does Not Include "For Tracking Only Items"	
T.S.A.S / Component: Reason: Entry Date:	3A RHR pump, 3.5.2.c – Action g Oil Change 4 hours ago
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	

REACTOR OPERATOR (CONT'D)
UNIT RISK: GREEN (ACCEPTABLE)   PROTECTED TRAIN: B
Changes to Risk Significant Equipment:
No recent changes from last shift. OLRM: GREEN PROTECTED TRAIN: B
Upcoming Reactivity Management Activities:
Maintain current power level 99.5% -100% Xe is stable.
Upcoming Major POD Activities:
NONE
Upcoming ECOs to Hang and /or Release:
• Hang – None • Release – None
Evolutions or Compensatory Actions in Progress:
NONE
General Information, Remarks, and Operator Work Around Status:
• Weather forecast is overcast skies with scattered pockets of severe rain. • U3 supplying Aux Steam • Air In-leakage = 0.0 SCFM

Site:	Turkey Point Units 3 and 4 (PTN)
Title:	L-16-1 NRC EXAM SCENARIO 5
LMS #:	NRC 25
LMS Rev Date:	6/9/16 Rev #: 0
SEG Type:	☐ Training ☒ Evaluation
Program:	☐ LOCT ☒ LOIT ☐ Other
Duration:	120 minutes

Developed by:	**Brian Clark** Instructor/Developer	**6/13/16** Date
Reviewed by:	**Tim Hodge** *Instructor (Instructional Review)*	**6/22/16** *Date*
Validated by :	**Rocky Schoenhals** *SME (Technical Review)*	**6/22/16** *Date*
Approved by:	**Mark Wilson** *Training Supervision*	**6/22/16** *Date*
Approved by:	**Rocky Schoenhals** *Training Program Owner (Line)*	**6/22/16** *Date*

SCENARIO REFERENCES		
DOC NO.	TITLE	REV
3-EOP-E-0	REACTOR TRIP OR SAFETY INJECTION	12
3-EOP-E-3	STEAM GENERATOR TUBE RUPTURE	9
3-ONOP-028	REACTOR CONTROL SYSTEM MALFUNCTION	4
3-ONOP-041.5	PRESSURIZER PRESSURE CONTROL MALFUNCTION	0A
3-ONOP-049.1	DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR	4
3-ONOP-059.8	POWER RANGE NUCLEAR INSTRUMENTATION MALFUNCTION	0A
3-ONOP-071.2	STEAM GENERATOR TUBE LEAKAGE	11
3-OSP-059.10	DETERMINATION OF QUADRANT POWER TILT RATIO	2
	PTN TECHNICAL SPECIFICATIONS	298

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
Terminal Objective	Given this simulator scenario and resources normally found in the Control Room, the operating crew will perform Control Room operations IAW approved plant procedures in order to maintain the integrity of the plant and the health and safety of the public.
Enabling Objectives:	Given this simulator scenario and resources normally found in the Control Room, operate in accordance with approved plant procedures, Operations Department Instructions, and management expectations: 1. (ALL) Demonstrate personnel SAFETY awareness in interactions with plant staff and outside agencies. 2. (ALL) Demonstrate ALARA awareness in interactions with plant staff and outside agencies. 3. (ALL) Exchange correct information using 3-point communication/Repeat-backs with Control Room personnel and other plant staff. 4. (ALL) Inform plant personnel and System of plant conditions, as needed. 5. (US) Employ timely and concise crew briefs where appropriate. 6. (ALL) Maintain awareness of plant status and control board indication. 7. (ALL) Correctly diagnose plant situations. 8. (ALL) Solve operational problems as they arise. 9. (RCO/BOP) Manipulate plant controls properly and safely. 10. (ALL) Demonstrate self-checking using STAR and peer checks(when required) 11. (US) Demonstrate command and control of the crew. 12. (US) Coordinate the input of crew members and other plant staff. 13. (US) Utilize the input of crew members and other plant staff. 14. (ALL) Demonstrate conservative decision making. 15. (ALL) Demonstrate teamwork. 16. (ALL) Respond to plant events using procedural guidance (OPs/ONOPs/EOPs) as applicable in accordance with rules of usage. 17. (RCO/BOP) Implement any applicable procedural immediate operator actions without use of references. 18. (SRO) Maintain compliance with Tech Specs. 19. (ALL) Identify/enter applicable Tech Spec action statements. 20. (ALL) Respond to annunciators using ARPs (time permitting). 21. (ALL) Maintain written communication, logs, and documentation as needed to permit post-event reconstruction. Continued on next page

SIMULATOR EXERCISE GUIDE REQUIREMENTS	
	While addressing the following events: 1. FT-3-487 3B S/G Feed Water Flow Transmitter Drifts High 2. R-3-17B CCW Hx Radiation Monitor Fails High 3. 3A TPCW Pump Cavitation 4. LT-3-460 Pressurizer Level Fails Low 5. 3A & 3B CRDM Fans Trip 6. 3A SGTR with LOOP 7. Control Room HVAC Fails To Align on SI 8. PCV-3-445C PZR PORV, Fails To Close During E-3 Depressurization
Prerequisites:	None
Training Resources:	PTN Unit 3 Plant Simulator
Development References:	• TR-AA-220-1003, Initial NRC and Audit Exam Process • TR-AA-230-1003, SAT Development • TR-AA-230-1007, Conduct of Simulator Training and Evaluation • 0-ADM-232, Time Critical Action Program • OP-AA-100-1000, Conduct Of Operations • OP-AA-103-1000, Reactivity Management • 0-ADM-200, Operations Management Manual • 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage • WCAP-17711-NP, Pressurized Water Reactor Owners Group Westinghouse Emergency Response Guideline Revision 2-Based Critical Tasks
Protected Content:	N/A
Evaluation Method:	Performance Mode
Operating Experience:	None
Risk Significant Operator Actions:	**Limit RHR Time On Recirculation** When a RHR Pump starts and is operating at shutoff head, limit the operating time at shutoff head with minimum flow recirculation to no more than 44 minutes.

TASKS ASSOCIATED WITH THIS SIMULATOR EXERCISE GUIDE

SRO TASK #	TASK TITLE
02008001300	RESPOND TO TURBINE PLANT COOLING WATER (TPCW) MALFUNCTIONS
02028033500	AUTHORIZE UNIT TRIP
02041029300	EVALUATE AND RESPOND TO A LOW PRESSURIZER PRESSURE
02041057300	RESPOND TO PRESSURIZER LEVEL CONTROL CHANNEL MALFUNCTION
02067009300	RESPOND TO PROCESS RADIATION MONITOR ALARM(S)
02089026300	AUTHORIZE FAST LOAD REDUCTION
02200001500	RESPOND TO UNIT TRIP
02200006300	INVESTIGATE AND CONTROL STEAM GENERATOR TUBE LEAK
02200008500	RESPOND TO A STEAM GENERATOR TUBE RUPTURE
02200022500	DIAGNOSE CAUSE OF SAFEGUARDS ACTUATION
02200046500	RESPOND TO STEAM GENERATOR LOW LEVEL

RO TASK	TASK TITLE
01008001300	RESPOND TO TURBINE PLANT COOLING WATER SYSTEM MALFUNCTION
01041029300	EVALUATE AND RESPOND TO A LOW PRESSURIZER PRESSURE
01041057300	RESPOND TO PRESSURIZER LEVEL CONTROL CHANNEL MALFUNCTION
01067009300	RESPOND TO PROCESS RADIATION MONITOR ALARM(S)
01089026300	RESPOND TO / ADJUST TURBINE DURING FAST LOAD REDUCTION
01200001500	RESPOND TO UNIT TRIP
01200006300	INVESTIGATE AND CONTROL STEAM GENERATOR TUBE LEAK
01200008500	RESPOND TO A STEAM GENERATOR TUBE RUPTURE
01200022500	DIAGNOSE CAUSE OF SAFEGUARDS ACTUATION
01200046500	RESPOND TO STEAM GENERATOR LOW LEVEL

UPDATE LOG:					
NOTES:	Place this form with the working copies of lesson plans and/or other materials to document changes made between formal revisions. For fleet-wide training materials, keep electronic file of this form in same folder as approved materials. Refer to TR-AA-230-1003 SAT Development for specific directions regarding how and when this form shall be used.				
Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.					
#	**DESCRIPTION OF CHANGE**	**REASON FOR CHANGE**	**AR/TWR#**	**PREPARER** **REVIEWER**	**DATE** **DATE**
0-0	Initial Revision	Revised for L-16-1 NRC Exam	2108338	Note 5 Note 5	Note 5 Note 5

1. Individual updating lesson plan or training material shall complete the appropriate blocks on the Update Log.
2. Describe the change to the lesson plan or training materials.
3. State the reason for the change (e.g., reference has changed, typographical error, etc.)
4. Preparer enters name/date on the Update Log and obtains Training Supervisor approval.
5. Initial dates and site approval on cover page.

Initial Conditions:

The plant is at 60% power (MOL). Online risk is green. B train is protected on both units.

Equipment OOS

- The 3A RHR pump and 3A1 Circulating Water pump are OOS.

SCENARIO SUMMARY

Event 1

Shortly after the crew takes the shift, FT-3-487, 3B S/G Feed Water Flow transmitter, drifts high. The BOP will take manual control of 3B S/G level and restore level to normal. The crew may use the ARP or 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection, to select operable channels and restore 3B S/G level control to automatic. The US will enter 3-ONOP-049.1 to verify all required actions are complete and to determine which bistables need to be tripped.

Event 2

After the actions of Event 1 are complete, CCW Surge Tank Radiation Monitor, R-17B, fails high. CCW surge tank vent valve, RCV-3-609, fails to close on the high radiation signal. The US will enter 3-ONOP-067, Radioactive Effluent Release, to verify the failure and direct the RCO to manually close the valve.

Event 3

After the actions of Event 2 are complete, the crew will respond to a TPCW low pressure alarm. The BOP will report signs of cavitation and swap TPCW pumps. The US may enter 3-ONOP-008, Turbine Plant Cooling Water Malfunction, to verify all required actions are complete.

Event 4

After the crew swaps TPCW pumps, Pressurizer Level transmitter, LT-3-460, will fail low. The PZR Heaters will trip and letdown will isolate. The US will enter 3-ONOP-041.6, Pressurizer Level Control Malfunction. The RCO will select an operable channel, re-establish normal letdown flow, and restore PZR heaters to automatic. The US will also enter 3-ONOP-049.1 to verify all required actions are complete and to determine which bistables need to be tripped.

Event 5

After the actions of Event 4 are complete, the 3A CRDM Fan Trips. A few minutes later the 3B CRDM Fan Trips. The crew will commence a shutdown using 3-GOP-100, Fast Load Reduction.

Event 6

After a 5 to 10% downpower, a SGTR develops over a 5 minute period on the 3A S/G. The crew will take actions to maximize Charging and to isolate Letdown. When the leakage exceeds the CVCS capacity, the US will order the RCO to trip the Reactor and enter to 3-EOP-E-0, Reactor Trip Or Safety Injection. When the Generator trips a Loss Of Offsite Power occurs. Both Emergency Diesel Generators will start and energize their respective 4KV buses. When the Ruptured S/G Isolation Criteria are met, the BOP or RCO will isolate Aux Feed Water flow to the 3A S/G.

Event 7

When SI actuates, Control Room Ventilation fails to align for recirc. The BOP will manually open Emergency Inlet Dampers D-2 and D-3 per 3-EOP-E-0 Attachment 3, Prompt Action Verifications.

Event 8

The crew will transition from 3-EOP-E-0 to 3-EOP-E-3, Steam Generator Tube Rupture. The crew will isolate the 3A S/G, cooldown the RCS, Establish Charging Flow, and stop the RHR pumps. When the cooldown is complete the RCO will open PZR PORV, PCV-3-455C, to depressurize the RCS (PCV-3-456 is failed close). When the depressurization is complete, PCV-3-455C will fail to close so the RCO will close block valve MOV-3-536 to stop the depressurization.

The scenario is terminated after the crew completes the depressurization per 3-EOP-E-3, or at the Lead Evaluator's discretion, once all critical tasks have been evaluated.

Event		CRITICAL TASKS
6	CT1	**Isolate the Ruptured S/G** During a Steam Generator Tube Rupture, isolate the ruptured S/G before a the ruptured Steam Generator pressure drops below 450 psig to prevent transition to 3-EOP-ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired. *Safety Significance:* Isolation of the ruptured steam generator minimizes release of radioactivity from this generator. In addition, isolation is necessary to establish a pressure differential between the ruptured and non-ruptured steam generators in order to cool the RCS and stop primary-to secondary leakage. If any ruptured S/G cannot be isolated from at least one intact S/G, the operator is directed to go to 3-ECA-3.1, SGTR With Loss Of Reactor Coolant -Subcooled Recovery Desired.

Event		CRITICAL TASKS
6	CT2	**Control Initial RCS Cooldown** During a Steam Generator Tube Rupture, dump steam from intact S/Gs at maximum rate to achieve Core Exit TCs less than required temperatures based on the lowest ruptured S/G pressure without causing a transition to 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, or 3-EOP-ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired. *Safety Significance*: A SGTR mitigation strategy leading to a transition from 3-EOP-E-3 to a contingency procedure constitutes an incorrect performance requiring the crew to take additional compensatory actions that complicate the event mitigation strategy. With a SGTR, there exists a breach of the RCS fission-product and Containment barriers which allows radioactive RCS inventory to leak into the SG and associated piping. Without controlling the cooldown, the primary-to-secondary leakage is not stopped. This continued leakage results in a larger release of radioactivity to the environment affecting the safety of the public.
6	CT3	**Limit RHR Time On Recirculation** When a RHR Pump starts and is operating at shutoff head, limit the operating time at shutoff head with minimum flow recirculation to no more than 44 minutes. (0-ADM-232, Time Critical Operator Action Program–Attachment 1) *Safety Significance:* Failure to secure the RHR Pumps operating at shutoff head leads to pump overheating and adverse vibration which would constitutes incorrect crew performance in which the crew does not prevent a degradation of the emergency core cooling system (ECCS) capacity.
8	CT4	**Control Initial RCS Depressurization** During a Steam Generator Tube Rupture, depressurize the RCS to the ruptured S/G pressure without causing a transition to 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, or 3-EOP-ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired. *Safety Significance:* A SGTR mitigation strategy leading to a transition from 3-EOP-E-3 to a contingency procedure constitutes an incorrect performance requiring the crew to take additional compensatory actions that complicate the event mitigation strategy. With a SGTR, there exists a breach of the RCS fission-product and Containment barriers which allows radioactive RCS inventory to leak into the SG and associated piping. Without controlling the cooldown, the primary-to-secondary leakage is not stopped. This continued leakage results in a larger release of radioactivity to the environment affecting the safety of the public.

SEQUENCE OF EVENTS	
Event #	**Description**
1.	FT-3-487 3B S/G Feed Water Flow Transmitter Drifts High
2.	R-3-17B CCW Hx Radiation Monitor Fails High
3.	3A TPCW Pump Cavitation
4.	LT-3-460 Pressurizer Level Fails Low
5.	3A & 3B CRDM Fans Trip
6.	3A SGTR with LOOP
7.	Control Room HVAC Fails To Align on SI
8.	PCV-3-445C PZR PORV, Fails To Close During E-3 Depressurization

SIMULATOR SET UP INSTRUCTIONS	
Check	**Action**
_____	Restore IC-24 (60% MOL) or equivalent IC.
_____	Place the Simulator in RUN.
_____	Stop the 3A1 Circ Water Pump
_____	Open & execute lesson file L-16-1 N5
_____	Ensure the following lesson steps are triggered: • SETUP - 3A RHR PUMP OOS • SETUP - 3A1 CWP OOS • SETUP EVENT 2 - RCV-609 FAILED OPEN • SETUP EVENT 7 - CONTROL ROOM VENTILATION FANS FAIL TO START • SETUP EVENT 8 - PORV 356 FAILED CLOSE
_____	• Place the 3A RHR pump in PTL and hang an ECO tag. • Place the 3A1 CWP in STOP and hang an ECO tag.
_____	Verify the trend for 3A1 Screen on the TWS DP Recorder is clear.
_____	Ensure Rod Group Step Counters have completed stepping out.
_____	Allow the plant to stabilize.
_____	Acknowledge any alarms and freeze Simulator.
_____	Ensure B train is protected train on VPA.
_____	Perform the SIMULATOR OPERATOR CHECKLIST or equivalent.
_____	Place TURNOVER SHEETS on RO’s desk or give to the Lead Evaluator.

BRIEFINGS

- Shift turnover information is attached to the back of this guide.
- Ensure all applicants are prior briefed on Appendix E of NUREG 1021, Policies and Guidelines For Taking NRC Examinations.
- Conduct a Crew Pre-brief to cover turnover information. Shift turnover information is attached to the back of this guide.

US: ______________________________

RCO: ______________________________

BOP: ______________________________

SCENARIO NOTE

0-ADM-211 Prudent Operator Actions - If redundant stand-by equipment is available and ready, the operator is permitted to start the redundant equipment for failed or failing operating equipment. Immediate follow up of applicable ARPs and ONOPs (AOPs) shall occur as required.

Critical Tasks are highlighted in red.	**Simulator Operator Actions** are highlighted in blue.	**Operator Verifiable Actions** are Highlighted in green.

EVENT 1 - FT-3-487 3B S/G FEED WATER FLOW TRANSMITTER DRIFTS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE Ensure the Simulator is in **RUN** before the crew enters the Simulator.	
		US: Conducts shift turnover.
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 1 - FT-3-487 DRIFTS HIGH**	BOP: • Recognizes and reports FT-3-487 failure. **PROMPT ACTIONS** • Takes manual control of 3B S/G level control valve, FCV-3-488. • Restores 3B S/G level to normal.
		RCO: Addresses Alarm Response for C4/2 & C6/2 • CHECK LI-3-486 or LI-3-488, B STM GEN LEVEL, controlling channel for SG Level deviation. • CHECK Feedwater Controllers, FIC-3-488A or FIC-3-488B, for indications of failure, alarm, or input signal failures. • CHECK Feedwater Controller Inputs • IF alarm is due to instrument failure, THEN REFER TO 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels.
		US: Enters and directs actions of 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, for response.

EVENT 1 - FT-3-487 3B S/G FEED WATER FLOW TRANSMITTER DRIFTS HIGH		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE The crew may use the ARP to select an operable channel and restore automatic level control.	BOP: • Verify FT-3-487 failure by channel check comparison. • Verify no off-normal conditions exist on FT-3-486. • Place 3B S/G Feed Water Flow Control Transfer Switch to FT-3-486 (Yellow) • Place 3B S/G Steam Flow Control Transfer Switch to FT-3-485 (Yellow) • Ensure 3B S/G level is returned to auto. **Steps 5.1 - 5.4**
	BOOTH OPERATOR WCC/I&C: Acknowledge the report. I&C would like to be present when bistables are tripped. They will be in the control in one hour. If asked to locally check FT-3-487, wait 2 to 3 minutes and then report nothing visibility wrong.	BOP: Notifies WCC to initiate PWO and I&C for troubleshooting.
	BOOTH OPEREATOR If dispatched to reset AMSAC, wait 2 to 3 minutes and then **trigger EVENT 1 - RESET AMSAC**	US: Reviews Tech Specs • LCO 3.3.1 Functional Unit 12 - Action 6 within 6 hours trip bi-stables **Step 5.5 - 5.6**
	Lead Evaluator After S/G level control is restored to auto and the US has reviewed Tech Specs, at the Lead Evaluators discretion, direct the Booth Operator to trigger the next event.	US: Conducts crew brief.

EVENT 2 - R-3-17B CCW HX RADIATION MONITOR FAILS HIGH		
3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 2 - R-17B FAILS HIGH**	
	BOOTH OPERATOR When the RCO closes RCV-3-609, **verify EVENT 2 - ALLOW RCV-609 TO CLOSE MANUALLY triggers.**	BOP: Reviews ARP for H1/4 • IF alarm is on R-17A/B, then refer to 3-ONOP-067, Radioactive Effluent Release, for expected automatic actions. • CHECK alarm valid as follows: - CHECK FAIL/TEST light NOT LIT. - PUSH FAIL/TEST light (meter reading of 288 or 289K) - PUSH SOURCE CHECK light (should get meter increase). - PUSH HIGH ALARM light to determine if meter level is above high alarm setpoint
		US: Enter and direct the actions of 3-ONOP-067, Radioactive Effluent Release
		US: Review the Foldout Page • Notify plant personnel • IF a Reactor Trip occurs AND any following PRMS alarms Actuate, THEN within 30 minutes of the alarm, manually align Control Room ventilation in the Emergency Recirculation Mode R-15/19/20 • **IF** any PRMS high alarm occurs **AND** automatic actions are required, **THEN** verify the applicable automatic actions for the occurring PRMS HIGH ALARMS: - R-17A/B HIGH ALARM, RCV-3-609, CCW Head Tank Vent Valve – CLOSED

EVENT 2 - R-3-17B CCW HX RADIATION MONITOR FAILS HIGH		
3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Closes RCV-3-609, CCW Head Tank Vent Valve, per fold out page.
		BOP: Check High Alarm On R-17B **STEP 1**
	NOTE Parts of this step may have been completed using the ARP.	BOP: • Check R17B readout GREATER THAN OR EQUAL TO ALARM SETPOINT • Check channel operability as follows - Depress and hold FAIL/TEST pushbutton on affected PRMS Channel - Check readout - EQUAL TO 288K OR 289K - Release FAIL/TEST pushbutton • Check affected PRMS drawer responds to source check • Check for PRMS channel failure - Check Fail indicator – OFF - Display and recorder reading – **NOT** FAILED LOW **STEP 2**
	BOOTH OPERATOR Acknowledge reports to SM, RP and Chemistry. If asked to survey or sample CCW for elevated radiation, wait 3 to 5 minutes and then report all radiation levels are normal.	US: • Notify the Shift Manager of problem with R-17B. • Direct Radiation Protection Shift Supervisor to conduct radiological surveys to confirm validity of alarm. • Direct Chemistry to perform sampling to confirm validity of alarm. **STEP 2 RNO**
	NOTE The US may discontinue use of 3-ONOP-067 once it's determined that an actual high radiation condition does not exist.	BOP: Check R-17A and R-17B High Alarms – OFF **(NO)** • Go to Step 29 **STEP 3**

EVENT 2 - R-3-17B CCW HX RADIATION MONITOR FAILS HIGH		
3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Check CCW System For High Activity • Announce the high radiation alarm on page system and warn personnel to remain clear of all CCW piping • Verify RCV-3-609, CCW Head Tank vent Valve - CLOSED • Direct Chemistry Department to sample CCW System to determine its activity level • Route any known CCW system leakage to the WHUT floor drain **STEP 29**
		Crew: • Check Normal CCW Temperatures And Flows Out Of RCP Thermal Barriers • Check Normal CCW Temperature And Flow Out Of NRHX • Check Normal CCW Temperature And Flow Out Of Seal Water Heat Exchanger • Check Normal CCW Temperature And Flow Out Of In-Service Spent Fuel Pit HXs • Check Normal CCW Temperature And Flow Out Of Excess Letdown HX • Check 3A RHR Pump AND 3A RHR Heat Exchanger - IN SERVICE **(NO)** - Go to step 37 **STEP 30 - 35**
	<u>LEAD EVALUATOR</u> Once the crew has verified the operability of the R-17B, go to the next event at your discretion.	RCO: Check 3B RHR Pump AND 3B RHR Heat Exchanger - IN SERVICE **(NO)** • Go to step 39 **STEP 37**

EVENT 3 - 3A TPCW PUMP CAVITATION		
3-ONOP-008, TURBINE PLANT COOLING WATER MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by Lead Evaluator, **trigger EVENT 3 – 3A TPCW PUMP CAVITATION.**	
		BOP: Reports 3A TPCW amps and TPCW pressure fluctuating.
	BOOTH OPERATOR If dispatched to check the 3A TPCW while it's running, wait 2 to 3 minutes and then report it sounds like its cavitating. If the 3A TPCW pump is stopped when you arrive, report it looks okay. If asked to check out the 3B TPCW for a start, report it is ready to start. Once the pump is running, report SAT start. If WCC or maintenance is contacted acknowledge request for additional support.	RCO: Reviews ARP I5/4 • Use DCS to check TPCW Temperature • IF TPCW header low pressure condition exists: - START standby TPCW pump - MONITOR pump amp indication on 3C04. - Locally CHECK for system leakage, including TPCW Supplemental Cooling Chiller(s) • REFER TO 3-ONOP-008, Turbine Plant Cooling Water Malfunction.
		BOP: • Starts 3B TPCW pump • Stops 3A TPCW pump
	NOTE The US may choose not to enter 3-ONOP-008 if the TPCW pumps are swapped per the ARP.	US: Enter and direct the actions of 3-ONOP-008.

EVENT 3 - 3A TPCW PUMP CAVITATION		
3-ONOP-008, TURBINE PLANT COOLING WATER MALFUNCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	<u>BOOTH OPERATOR</u> If dispatched to check TPCW equipment, report all system parameters to be normal. If asked for specific values, use the simulator drawing or DCS to report actual value.	BOP: • Check All Turbine Plant Cooling Water Pump Alarms – OFF • Verify Turbine Plant Cooling Water Pumps - AT LEAST ONE RUNNING • Check Turbine Plant Cooling Water Header Pressure I 5/4, TPCW HI TEMP/LO PRESS NOT LIT • Check Proper Intake Cooling Water Lineup To Turbine Plant Cooling Water Heat Exchangers • Check For Abnormal Surge Tank Level **(NO)** - Go to Step 12 **STEPs 1-5**
	<u>LEAD EVALUTOR</u> Once the TPCW pumps are swapped, move to the next event at your discretion.	BOP: • Check Cooling To Turbine Plant Cooling Water Heat Exchangers • Locally Verify Turbine Plant Cooling Water Basket Strainer ΔP - LESS THAN 1.5 PSID • Check GEN RTD HI-HI TEMP – OFF • Check Generator Alarms – OFF • Check Pump Alarms – OFF • Check Proper Turbine Plant Cooling Water System Operation • Check Temperature Of Components Supplied By Turbine Plant Cooling Water - STABLE OR DECREASING • Go To Appropriate Plant Procedure As Determined By Shift Manager **STEPs 12-17**

EVENT 4 - LT-3-460 PRESSURIZER LEVEL FAILS LOW		
3-ONOP-041.6, PRESSURIZER LEVEL CONTROL MALFUNCTION.		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the Lead evaluator, **trigger EVENT 4 – LT-3-460 FAILS LOW**	RCO: Reports LT-3-460 failed low
	NOTE Failure of LT-3-460 will cause Letdown to isolate and PZR Heaters to de-energize.	BOP: • Acknowledges A8/4, A9/4 • CHECK LI-459A/460/461 less than or equal to 6%. • Check LCV-3-460, and CV-3-200A/B/C closed. • CHECK Control and Backup heaters OFF • Recommends entry into 3-ONOP-041.6, Pressurizer Level Control Malfunction.
		US: Directs 3-ONOP-041.6 response.
		RCO: • Check Pressurizer level indicators LI-3-459A, LI-3-460 AND LI-3-461 • Selects ch 1 & 3 PZR level control (Position 2) • Maintains PZR level on program per 3-ONOP-041.6, Enclosure 1 • May place Master Charging Pump Controller, LC-3-459G, in manual • May Start or Stop one charging pump as required. • Place LR-3-459 Channel Select Pressurizer Level Recorder to position 1 or 3. **Steps 5.1 - 5.4**

EVENT 4 - LT-3-460 PRESSURIZER LEVEL FAILS LOW		
3-ONOP-041.6, PRESSURIZER LEVEL CONTROL MALFUNCTION.		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE This failure may cause VCT Auto Makeup.	RCO: Restore Letdown Flow • Place LC-3-459G, Pzr Lvl Inst Man/Auto Station, in Manual AND adjust charging flow as required for increased letdown flow. • Throttle Low Pressure LTDN Controller, PCV-3-145, as necessary to prevent LTDN relief valve from lifting. • Manually control Low Pressure Letdown Control Valve, PCV-3-145, to limit pressure spike. • Open High Pressure L/D Isol Vlv from Loop B Cold Leg, LCV-3-460. • Open L/D Isolation Valves, CV-3-200 A, B, or C as required to restore pressurizer level to programmed level. • Return Lower Pressure Letdown Control Valve, PCV-3-145, to Automatic. **STEP 5.5**
		RCO: N/A **STEP 5.6**
		RCO: • Restore PRZ heaters to automatic operation or take manual control. • Maintain pressurizer level to be consistent with programmed level as indicated in Enclosure 1. • WHEN desired, THEN place LC-3-459G in Automatic. **STEPs 5.7 – 5.8**

EVENT 4 - LT-3-460 PRESSURIZER LEVEL FAILS LOW		
3-ONOP-041.6, PRESSURIZER LEVEL CONTROL MALFUNCTION.		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US: Perform actions required by 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels. **Step 5.9**

EVENT 4 - LT-3-460 PRESSURIZER LEVEL FAILS LOW		
3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNELS		
TIME	**EVALUATOR ACTIVITIES & NOTEs**	**EXPECTED STUDENT RESPONSE**
		US: Enters and directs actions of 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels, for response
		BOP: • Verify LT-3-460 failure by comparison with LT-3-459/461 and known plant parameters and conditions • Verify no off-normal conditions exist on LT-3-459/461 • Verify ch 1 & 3 PZR level control in (Position 2) • Verify LR-3-459 Channel Selected to Pressurizer Level Recorder to position 1 or 3 • Verify PZR level control function is returned to automatic. **Steps 5.1 – 5.5**
	BOOTH OPERATOR If asked, I&C would like to be present when bi-stables are tripped. They will be in the control room in one hour.	US Reviews TECH Specs • Tech Spec 3.3.-1 Functional Unit 9 not met. - Action 13, inoperable channel must be placed in the tripped condition within 6 hours.
	BOOTH OPERATOR WCC/I&C: Acknowledge the report. If asked, I&C would like to be present when bi-stables are tripped. They will be in the control room in one hour.	US • Notifies WCC to initiate PWO and I&C for troubleshooting. • Notifies Plant Management IAW 0-ADM-115.
	LEAD EVALUATOR After the PZR level control is restored to auto and the US completes a review of Tech Specs, proceed to the next event at the Lead Evaluators discretion	US: • Conducts crew brief.

EVENT 5 – 3A & 3B CRDM FANS TRIP		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When directed by the Lead evaluator, **trigger EVENT 5 - CRDM FANS TRIP**	
	NOTE The 2nd fan trips 60 seconds after the first fan. **BOOTH OPERTOR** If dispatched to check CRMD fan breakers, wait 2 to 3 minutes and report breakers tripped. If asked to reset them, report they will not reset.	BOP: Reviews ARP I8/5 • CHECK indicating lights to determine affected CRDM Cooler on VPB. • TAKE affected CRDM Cooler control switch to OFF. • ENSURE remaining CRDM Cooler in service. • IF neither fan will start, THEN PERFORM the following commence shutdown using 3-GOP-100, Fast Load Reduction.
		US: • Directs actions to reduce Rx power per 3-GOP-100. • Completes Attachment 3 • Brief the crew per Attachment 4 **Steps 1-2**
		US: Reviews Foldout page with crew. • 3-EOP-E-0 Transition Criteria • Notify Chemistry Department • Boration Stop Criteria • Restore Blender to AUTO **FOLDOUT PAGE**

EVENT 5 – 3A & 3B CRDM FANS TRIP		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR Acknowledge notifications.	BOP: Notify The Following Of Fast Load Reduction • System Dispatcher • Plant personnel using the Page Boost • Chemistry to start RCS sampling is required according to Tech Spec Table 4.4-4. **Step 3**
		RCO: Begin Boration For Initial Tavg Effect • Set the Boric Acid Totalizer to total boric acid volume value determined on Attachment 3. • Place the Reactor Makeup Selector Switch to BORATE. • Place the RCS Makeup Control Switch to START. • Adjust FC-3-113A, Boric Acid Flow Controller, to achieve 40 gpm boric acid flow as indicated on FR-3-113. • WHEN Tavg begins to lower from the boration, THEN adjust FC-3-113A, Boric Acid Flow Controller, to load reduction value from Attachment 3. **Step 4**
		US: Determine Turbine Load Reduction in MW CNTRL **Step 5**

EVENT 5 – 3A & 3B CRDM FANS TRIP		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Initiate Turbine Load Reduction in MW CNTRL • Select MW CNTRL • Set TARGET power level – MW VALUE from Attachment 3 • Set RAMP RATE – MW/M VALUE FROM Attachment 3. • Check T_{avg} has lowered 1° to 2°F from the initial value prior to boration. • Depress GO • Ensure FC-3-113A, Boric Acid Flow Controller, has been adjusted to the load reduction boration rate. Go to Step 10 **Step 6**
		BOP: Monitor Load Reduction • Adjusts power reduction rate to maintain T_{avg}/T_{ref} within limits of Attachment 3. • Monitors S/G level control to ensure feed reg valves properly maintain level control in automatic. • Refer to Enclosure 1 for expected alarms. **Step 10**
	<u>BOOTH OPERATOR</u> Respond as SNPO. If asked, idle Charging Pump ready for start.	RCO: • Maintain pressurizer level to ensure that automatic pressurizer level control maintains level on program. • If needed, starts 2nd Chg Pp and places 2nd orifice in service. • Adjusts boration rate to maintain Tavg/Tref within ±4°F ΔT. • Refer to Enclosure 1 for expected alarms. **Step 10**

EVENT 5 – 3A & 3B CRDM FANS TRIP		
3-GOP-100, FAST LOAD REDUCTION		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Monitor Boration Rate • Monitor for excessive rod movement by monitoring TR-3-409D, Rod Position Bank D. • Determine if Insertion Limit and Bank D position are converging at a rate that will cause rod insertion limit alarms. • Adjust power reduction rate as needed to control rod insertion • Increase boration rate and/or total amount as necessary to limit control rod insertion **Step 11**
		RCO: • Monitor Annunciator B 8/1, ROD BANK LO LIMIT – CLEAR • Monitor B 8/2 ROD BANK A/B/C/D EXTRA LO LIMIT – CLEAR **Steps 12-13**
	<u>BOOTH OPERATOR</u> Acknowledge notification to refer to E-Plan and ADM-115.	US: Have SM refer to the following procedures: • 0-EPIP-20101, DUTIES OF EMERGENCY COORDINATOR • 0-ADM-115, NOTIFICATION OF PLANT EVENTS **Step 14**
	<u>LEAD EVALUATOR</u> Once power has been reduced by a minimum of 5%, at the Lead Evaluators discretion, proceed to the next Event.	RCO: Energize Pressurizer Backup Heaters **Step 15**

EVENT 6 – 3A SGTR WITH LOOP		
3-ONOP-071.2, STEAM GENERATOR TUBE LEAKAGE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	NOTE The tube rupture ramps in over 5 minutes. The crew will meet conditions to trip in ~ 2 minutes. The US may direct actions to maximize charging, isolate letdown, and trip the Reactor without enter 3-ONOP-071.2. **BOOTH OPERATOR** When directed by the Lead Evaluator, **trigger EVENT 6 - 3A SGTR**	
		BOP: Reviews ARP for H1/4, PRMS HI RADIATION • Checks alarm on R-15 • REFER TO 3-ONOP-071.2, Steam Generator Tube Leakage. • Check S/G Feedwater flows and levels for indication of a Ruptures S/G.
		RCO: Checks PZR pressure and level for indication of a S/G Tube Rupture.
		US: Enters and directs the actions of 3-ONOP-071.2, Steam Generator Tube Leakage.
		US: Reviews the Foldout page with the crew. • 3-EOP-E-0 Transition Criteria • Control Room Ventilation Manual Isolation Criteria • Turbine Load Within 10% Of Target Power Level • Blowdown Release Path Isolation • AFW Steam Supply Release Path Isolation

EVENT 6 – 3A SGTR WITH LOOP		
3-ONOP-071.2, STEAM GENERATOR TUBE LEAKAGE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Check PRZ level – STABLE OR INCREASING • Start additional charging pumps as required. • Reduce letdown flow as necessary. • IF PRZ level can NOT be maintained, THEN manually trip the reactor AND go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION. **Step 1**
		BOP: • Check R-15 High Alarm light – ON • Check PRMS Channel R-15 Alarm Valid As Follows • Check PRMS Channels R-19 For Proper Operation **Step 2 - 4**

EVENT 6 - 3A SGTR WITH LOOP		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**TIME**	**TIME**
	NOTE 1. When the Generator trips, a Loss Of Offsite Power occurs and both Emergency Diesel Generators start and energize their respective 4KV buses. 2. When SI actuates, Control Room Ventilation fails to align for recirc.	US: • Directs 3-EOP-E-0 response after auto Reactor trip. OR • Directs RCO to manually trip the Reactor, then for operators to perform their IOA's.
		RCO: • Manually trips Reactor. • Manually actuates SI
	NOTE Steps 1 - 4 of 3-EOP-E-0 are Immediate Operator Actions (IOAs). The board operators will call out the high level steps of the IOAs as each step is completed from memory. Once the IOAs are complete, the US will read through Steps 1 – 4 with the crew.	RO/BOP: Perform IOA's.
CT3 **Start Time**	**NOTE** 3A and 3B RHR pumps will start when SI actuates. Record the time for verification of CT3 to secure RHR pumps within 44 minutes.	RCO: Verifies Reactor Trip **STEP 1**
		BOP: Verify Turbine Trip **STEP 2**
	BOOTH OPERATOR When the GEN MID BKR OPENS, **verify EVENT 6 LOOP triggers**.	BOP: Verifies Power To Emergency 4 KV Buses **STEP 3**
		RCO: Checks If SI Is Actuated **STEP 4**

EVENT 6 - 3A SGTR WITH LOOP		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**TIME**	**TIME**
		RCO: Checks if SI is required: • Manually actuate SI. • Manually actuate Containment Isolation Phase A. **STEP 4 RNO**
		US: Directs 3-EOP-E-0 response and reviews the IOAs.
CT1	**<u>Isolate the Ruptured S/G</u>** Closing CV-3-2816 & CV-3-2831 is part of CT1.	US: Reviews FOP for 3-EOP-E-0 with the crew 1. Adverse Containment Conditions 2. RCP Trip Criteria 3. Faulted S/G Isolation Criteria 4. Ruptured S/G Isolation Criteria **(YES)** – When 3A S/G narrow range level is greater than 7%[27%], close CV-3-2816 & CV-3-2831 5. AFW System Operation Criteria 6. CST Makeup Water Criteria **7.** RHR System Operation Criteria **(YES, RCO starts timer)** 8. Loss Of Offsite Power Or SI On Other Unit 9. Loss Of Charging Criteria **FOLDOUT PAGE**
	<u>NOTE</u> Attachment 3 actions start on page 42.	BOP: • Continues with ATTACHMENT 3 to complete The Prompt Action Verifications. **STEP 5**
	.	RCO: • Check AFW Pumps – AT LEAST TWO RUNNING. **STEP 6**

EVENT 6 - 3A SGTR WITH LOOP		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**TIME**	**TIME**
		RCO: Verify AFW Valve Alignment – PROPER EMERGENCY ALIGNMENT **STEP 7**
		RCO: Verify Proper AFW Flow: **STEP 8**
		RCO: Check RCP Seal Cooling: **STEP 9**
		RCO: Check RCS Temperatures between 545°F and 547°F or trending down to 547°F. **(NO)** • Stop dumping steam. • Reduce AFW flow • Close MSIVs **STEP 10**
		RCO: • Check Charging Pumps running **(NO)** • Verify CCW Flow Alarms to RCPs Thermal Barriers clear. • Check Offsite Power Available **(NO check diesel capacity)** • Check SI Reset • Start one charging pump. • Set Boric Acid Totalizer To Highest Volume Of Boric Acid Possible • Set FC-3-113A, Boric Acid Flow Controller, To A Pot Setting Of 4.0 • Place RMCS switch to Borate • Place RCS Makeup Control Switch To START **Attachment 7**

EVENT 6 - 3A SGTR WITH LOOP		
3-EOP-E-0, RX TRIP OR SAFETY INJECTION		
TIME	**TIME**	**TIME**
		RCO: Check PRZ PORVs, Spray Valves And Excess Letdown Isolated: **STEP 11**
		RCO: Check If RCPs Should Be Stopped: (Not Running) **STEP 12**
		RCO: Check If S/Gs Are Faulted: **(NO Go to Step 14)** **STEP 13**
	<u>BOOTH OPERATOR</u> If Chemistry or RP is called, report local secondary radiation readings and samples are highest on 3A SG.	RCO: Check If S/G Tubes Are Ruptured: **(YES for 3A SG)** **STEP 14**
		US: • Monitor Critical Safety Functions using 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES • Go to 3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE, Step 1

EVENT 6 - 3A SGTR with LOOP		
3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
CT3 **Stop Time**	**NOTE** The crew may wait until 3-EOP-E-3 step 10 to Stop RHR Pumps. **Limit RHR Time On recirculation** When a RHR Pump starts and is operating at shutoff head, limit the operating time at shutoff head with minimum flow recirculation to no more than 44 minutes. [0-ADM-232, Time Critical Operator Action Program–Attachment 1]	US: Directs 3-EOP-E-3 response. Reviews Foldout Page • Adverse Containment Setpoints • RCP Trip Criteria • SI Re-Initiation Criteria • Secondary Integrity Criteria • Cold Leg Recirculation Switchover Criteria • CST Makeup Water Criteria • Multiple Tube Rupture Criteria • Loss Of Offsite Power Or SI On Other Unit. • If RHR flow is less than 1100 gpm, then the RHR Pumps shall be shut down within 44 minutes of the initial start signal. **FOP**
		RCO: • Checks If RCPs Should Be Stopped **STEP 1**
	BOOTH OPERATOR If called as RP, report the radiation readings on 3A SG lines are higher than normal.	RCO: Identify Ruptured S/G: • Identify 3A as the Ruptured S/G • Directs RP to take rad readings on Main Steam and Blowdown Lines • Evaluates DAM1 on DCS • Determines ruptured SG by level increase or radiation **STEP 2**

EVENT 6 - 3A SGTR with LOOP

3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE

TIME	EVALUATOR ACTIVITIES & NOTES	EXPECTED STUDENT RESPONSE
CT1	**Isolate the Ruptured S/G** Adjusting 3A S/G Steam Dump To Atmosphere controller setpoint to 1060 psig and verifying it goes closed is part of CT1.	RCO: Isolate Flow From Ruptured S/G • Adjusts 3A S/G Steam Dump To Atmosphere controller setpoint to 1060 psig • Checks 3A S/G Steam Dump To Atmosphere Closed. • WHEN ruptured S/G pressure is less than 1060 psig, THEN Verify S/G Steam Dump to Atmosphere is closed. **STEP 3.a/b**
CT1	**BOOTH OPERATOR** When directed to de-energize MOV-3-1403 BKR 4D01-28, wait 3 minutes and then **trigger LOA – DEENERGIZE MOV-3-1403**. Report when action is complete. **Isolate the Ruptured S/G** Closing and de-energizing MOV-3-1403 is part of CT1.	BOP: Close steam supply valves from ruptured 3A S/G to AFW Pumps using: Attachment 17 • Check SI reset • Check AMSAC reset • Check Both AFW Auto Start White Lights – OFF (3QR50 AND 3QR51) • Close 3A Steam Generator AFW Steam Supply, MOV-3-1403. • Dispatch an Operator to de-energize BKR 4D01-28 for MOV-3-1403. • Verify MOV-3-1403 – CLOSED • Notify Unit Supervisor That 3A S/G AFW Steam Supply Is Isolated and Attachment 17 is complete. **STEP 3.c**
CT1	**Isolate the Ruptured S/G** Closing the 3A MSIV is part of CT1.	BOP: • Verify 3A S/G Blowdown Isolation Valve, FCV-3-6275A, is closed on 3A S/G. • Closes 3A MSIV. **STEP 3.d-e**

EVENT 6 - 3A SGTR with LOOP		
3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: • Check Circulating Water Pumps – ANY RUNNING **(NO)** - Close 3B and 3C MSIVs **STEP 3.f**
	BOOTH OPERATOR When dispatched to Align Main Steam auxiliaries using Attachment 5, **trigger LOA – ALIGN AUX STEAM SUPPLY FROM U4.** After 5 minutes, report complete.	BOP: • Dispatch Operator to align main steam auxiliaries using Attachment 5 **STEP 3.f**
	BOOTH OPERATOR When dispatched to close 3-10-321 and 3-10-896 for S/G 3C per Attachment 16, wait 5 minutes and report steps complete.	BOP: Isolate miscellaneous flow paths from 3A S/G using Attachment 16. • Verify 3A S/G Blowdown Sample MOV MOV-3-1427 – CLOSED • Dispatches Operator to locally isolate - 3A S/G Main Steamline Steam Trap 3-10-121A - Steam Sample Valves 3-10-891 for S/G 3A • Inform Unit Supervisor That Attachment 16 is Complete. **STEP 3.f**

EVENT 6 - 3A SGTR with LOOP

3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE

TIME	EVALUATOR ACTIVITIES & NOTES	EXPECTED STUDENT RESPONSE
CT1	**Isolate the Ruptured S/G** During a Steam Generator Tube Rupture, isolate the ruptured S/G before a the ruptured Steam Generator pressure drops below 450 psig to prevent transition to 3-EOP-ECA-3.1, SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired. Verify 3A S/G has been Isolated by the crew • CV-3-2818, TRN 1 AFW controller, in manual and closed or auto with the setpoint set to zero. • CV-3-2831, TRN 2 AFW controller, in manual and closed or auto with the setpoint set to zero. • MOV-3-1403, 3A Steam Supply to AFW pumps, closed and de-energized • MOV-3-1407, 3A S/G FW Isolation, closed • CV-3-1606, 3A S/G Stm Dump to Atmosphere, closed in Auto and set to 1060# • 3A MSIV CLOSED	BOP: • Check 3A S/G Level Narrow Range level – GREATER THAN 7%[27%] • Verify feed flow stopped to the 3A S/G. **STEP 4**
		US: Checks 3C S/G pressure greater than 450 psig. **STEP 5**

EVENT 6 - 3A SGTR with LOOP		
3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		US/BOP: Initiate RCS Cooldown • Determines required CET Temp for Cooldown. • Check feed sources to intact S/Gs – CAPABLE OF PROVIDING 400 GPM • Check Condenser AVAILABLE **(NO)** • Manually dump steam to atmosphere from 3B & 3C S/G(s) at maximum rate using Steam Dump to Atmosphere Valves. • Continue with step 7 - When Core Exit TCs - LESS THAN REQUIRED TEMPERATURE, then stops cooldown. - Maintains core exit TCs – LESS THAN REQUIRED TEMPERATURE. **STEP 6**
		BOP: Check Intact S/G Level: • Any Narrow Range Level – GREATER THAN 7%[27%]. • Maintain 3A and 3B S/G narrow range level between 21%[27%] and 50%. • Narrow Level – LESS THAN 50%. **STEP 7**
		RCO: Verify SI – RESET **STEP 8**

EVENT 6 - 3A SGTR with LOOP		
3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		RCO: Establish Charging Flow: • Charging Pumps – AT LEAST ONE RUNNING • Check offsite power – AVAILABLE **(NO)** - Check if diesel capacity is adequate to run three Charging Pumps • Start all available Charging Pumps. • Adjust speed controller as necessary to establish maximum Charging flow from the running Charging Pump(s). • Place RCS Makeup Control in STOP. • Adjust HCV-3-121, Charging Flow To Regen Heat Exchanger, to maintain proper Seal Injection flow. • Verify Charging Pump Suction auto transfers to RWST. **STEP 9**
CT3 **Stop Time**	**Limit RHR Time On recirculation** When a RHR Pump starts and is operating at shutoff head, limit the operating time at shutoff head with minimum flow recirculation to no more than 44 minutes. [0-ADM-232, Time Critical Operator Action Program – Attachment 1]	RCO: → Check If RHR Pumps should Be Stopped: • Check RCS pressure – GREATER THAN 275 PSIG[575 PSIG] • Check RHR flow – LESS THAN 1100 GPM • Stop RHR Pumps and place in standby. **STEP 10**

EVENT 6 - 3A SGTR with LOOP

3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE

TIME	EVALUATOR ACTIVITIES & NOTES	EXPECTED STUDENT RESPONSE
		RCO: Check PRZ PORVs And Block Valves: • Check Power to block valves – AVAILABLE • Check PORVs – CLOSED • Check Block valves - AT LEAST ONE OPEN **STEP 11**
		RCO: Reset Containment Isolation Phase A & Phase B. **STEP 12**
		RCO: Verify Instrument Air To Containment: • Verify CV-3-2803, Instrument Air Containment Isolation – OPEN • Verify Instrument Air pressure, as indicated on PI-3-1444 – GREATER THAN 95 PSIG **STEP 13**
CT2	**Control Initial RCS Cooldown** When 3-EOP-E-3, Steam Generator Tube Rupture, is entered, dump steam from 3B and 3C SGs at maximum rate using the Condenser Steam Dump Valves or Steam Dump To Atmosphere Valves to achieve Core Exit TCs less than required temperatures based on the lowest ruptured S/G pressure without causing a required transition to 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition.	BOP: Check If RCS Cooldown Should be Stopped: • Check CETs < REQUIRED • WHEN core exit TCs are less than required temperature • Stop RCS cooldown • Maintain CETs < REQUIRED **STEP 14**
		BOP: Checks Ruptured S/G(s) Pressure STABLE or INCREASING **STEP 15**

EVENT 6 - 3A SGTR with LOOP		
3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Check RCS Subcooling Based On Core Exit TCs – GREATER THAN 39°F[93°F] **STEP 16**

EVENT 8 - PCV-3-445C PZR PORV, FAILS TO CLOSE DURING E-3 DEPRESSURIZATION		
3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
	BOOTH OPERATOR When PCV-3-455C is opened, **verify EVENT 8 - PORV 455C FAILS TO CLOSE triggers.**	
	NOTE PORV PCV-3-456 was failed close in the scenario setup.	RCO: Check Normal PRZ Spray available **(NO)** Go to Step 18 **STEP 17**
CT4	**Control Initial RCS Depressurization** During a Steam Generator Tube Rupture, depressurize the RCS to the ruptured S/G pressure without causing a transition to 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, or 3-EOP-ECA-3.1, SGTR With Loss Of Reactor Coolant - Subcooled Recovery Desired.	RCO: • Open one PRZ PORV until any of the conditions satisfied using Attachment 6. • When the conditions of Attachment 6 are satisfied close PZR PORV • Close Block Valve MOV-3-536 to stop the depressurization. **STEP 18**
The scenario may be terminated at the discretion of the Lead Evaluator after the RCS cooldown is complete and the crew has had the opportunity to complete all critical steps.		
***** END OF SCENARIO *****		

EVENT 7 – CONTROL ROOM HVAC FAILS TO ALIGN ON SI		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Check Load Centers Associated With Energized 4 KV Buses – ENERGIZED **STEP 1**
		BOP: Verify Feedwater Isolation: • Place Main Feedwater Pump switches in STOP **STEP 2**
		BOP: Check If Main Steam Lines Should Be Isolated **STEP 3**
		BOP: Verify Containment Isolation Phase A Valve White Lights On VPB – ALL BRIGHT **STEP 4**
		BOP: Verify Pump Operation: **STEP 5**
		BOP: Verify Proper CCW System Operation: **STEP 6**
		BOP: Verify Proper ICW System Operation: **STEP 7**
		BOP: Check Emergency Containment Coolers – ONLY TWO RUNNING **STEP 8**

EVENT 7 – CONTROL ROOM HVAC FAILS TO ALIGN ON SI		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Verify Unit 3 Containment Purge Exhaust And Supply Fans – OFF **STEP 9**
		BOP: Verify Containment Spray NOT Required: **-STEP 10**
		BOP: Verify SI – RESET **STEP 11**
		BOP: Verify SI Valve Amber Lights On VPB – ALL BRIGHT **STEP 12**
		BOP: Verify SI Flow: **(NO)** Go to Step 14 **STEP 13**
	BOOTH OPERATOR When directed by the crew, **trigger LOA – ALIGN U4 HHSI TO U3 RWST.** Wait 5 minutes and report local operator steps complete.	BOP: Realign SI System: • Verify Unit 3 High-Head SI Pumps – TWO RUNNING • Stop both Unit 4 HHSI pump. • Direct Unit 4 Reactor Operator to align Unit 4 High-Head SI Pump suction to Unit 3 RWST using Attachment 1. • Stop Unit 4 HHSI pumps **STEP 14**
		BOP: Verify Containment Isolation Phase A – RESET **STEP 15**

EVENT 7 – CONTROL ROOM HVAC FAILS TO ALIGN ON SI		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Check RCPs – AT LEAST ONE RUNNING **(NO Go to Step 17)** **STEP 16**
	BOOTH OPERATOR When the BOP opens damper D3, **verify EVENT 7 - ALLOW OPENING D3 triggers** When the BOP opens damper D2, **verify EVENT 7 - ALLOW OPENING D2 triggers**	BOP: • Verify Emergency Air Supply Fans – at least one running • Control Room Ventilation dampers – aligned for recirc **(NO)** - Open Emergency Inlet Damper D-3 - Open Emergency Inlet Damper D-2 • Verify Normal Flow green indicating light (4QR82) – ON • TS-0002, TSC Emergency Vent Auto Initiate Key Switch – IN ENABLE **STEP 17**
	BOOTH OPERATOR When requested by crew, **trigger LOA – PLACE PAHMS IN SERVICE.** Wait 5 minutes and report local operator steps complete.	BOP: Place Hydrogen Monitors In Service Using 3-NOP-094, CONTAINMENT POST ACCIDENT MONITORING SYSTEM • For Each Hydrogen Monitor A/B - ENSURE FUNCTION SELECTOR switch is in SAMPLE. - PLACE control switch in ANALYZE. - PRESS the REMOTE SELECTOR button. - PRESS the ALARM RESET button. • Dispatch an operator to complete local step of 3-NOP-094 **(STEP 18)**

EVENT 7 – CONTROL ROOM HVAC FAILS TO ALIGN ON SI		
3-EOP-E-0 ATTACHMENT 3 – PROMPT ACTION VERIFICATIONS		
TIME	**EVALUATOR ACTIVITIES & NOTES**	**EXPECTED STUDENT RESPONSE**
		BOP: Verify All Four EDGs – RUNNING **STEP 19**
		BOP: Check 3A AND 3B 4 KV Buses – All Energized From Offsite power. **(NO, check computer chiller running)** **STEP 20**
		BOP: Notify Unit Supervisor that Attachment 3 is complete. **STEP 21**

Discussion Points are intentionally NOT included in evaluated scenarios. However, space is available below to document follow-up questions when further information is required to determine an evaluation outcome.

FOLLOW-UP QUESTIONS

QUESTION #1

ANSWER #1

QUESTION #2

ANSWER #2

SIMULATOR POST-SCENARIO RESTORATION:

_______ 1. Restore per Simulator Operator Checklist.

_______ 2. Once exams are complete, restore from SEI-19, Simulator Exam Security.

OPERATIONS SHIFT TURNOVER REPORT

UNIT 3 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B	UNIT 4 RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B

ONCOMING CREW ASSIGNMENTS

Shift Mgr:			**Inside SNPO:**	
Field Supv.:			**Outside SNPO:**	
Admin RCO:			**ANPO:**	
Unit 3			**Unit 4**	
Unit Supv.:			**Unit Supv.:**	
RCO:			**RCO:**	
NPO:			**NPO:**	

PLANT STATUS

Unit 3			**Unit 4**	
Mode:	1		**Mode:**	1
Power:	60%		**Power:**	100%
MWe:	464		**MWe:**	842
Gross Leakrate:	.22 gpm		**Gross Leakrate:**	0.03 gpm
RCS Boron Conc:	885 ppm		**RCS Boron Conc:**	642 ppm

Operational Concerns:

3A RHR pump taken OOS 4 hours ago for an oil change, expected back by the end of this shift.
3A1 Circ Water pump OOS. Tripped on over current, Electrical Maintenance is investigating.
3A Condensate pump was returned to service last shift following a motor bearing replacement and PMT run. Return to full power expected next shift.

U3 Anticipated LCO Actions:

None

U4 Anticipated LCO Actions:

None

Results of Offgoing Focus Area:

UNIT 3 STATUS

REACTOR OPERATOR

UNIT RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B

Mode:	1
Power:	60%
MWe	464
Tavg:	565°F
RCS Pressure:	2235
RCS Boron Conc:	885ppm

RCS Leakrate	
Gross:	0.22 GPM
Unidentified	0.04 GPM
Charging Pps:	0.00 GPM

Accumulator Ref Levels	
A	6656
B	6608
C	6646

Abnormal Annunciators:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	
Annunciator: Comp Actions:	

Current Tech Spec Action Statements: (Does Not Include "For Tracking Only Items"	
T.S.A.S / Component: Reason: Entry Date:	3A RHR pump, 3.5.2.c – Action g Oil Change 4 hours ago
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	
T.S.A.S / Component: Reason: Entry Date:	

REACTOR OPERATOR (CONT'D)
UNIT RISK: GREEN (ACCEPTABLE) PROTECTED TRAIN: B
Changes to Risk Significant Equipment:
No recent changes from last shift. OLRM: GREEN PROTECTED TRAIN: B
Upcoming Reactivity Management Activities:
Maintain current power level ± .5% Xe is stable.
Upcoming Major POD Activities:
NONE
Upcoming ECOs to Hang and /or Release:
• Hang – None • Release – None
Evolutions or Compensatory Actions in Progress:
• NONE
General Information, Remarks, and Operator Work Around Status:
• Weather forecast is overcast skies with scattered pockets of severe rain. • U3 supplying Aux Steam • Air In-leakage = 0.0 SCFM

Site:	Turkey Point Units 3 and 4 (PTN)
Title:	L-16-1 NRC EXAM SCENARIO 1
LMS #:	NRC 21
LMS Rev Date:	6/6/16 Rev #: 0
SEG Type:	☐ Training ☒ Evaluation
Program:	☐ LOCT ☒ LOIT ☐ Other
Duration:	120 minutes

Developed by:	Brian Clark Instructor/Developer	6/13/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by :	SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	Training Program Owner (Line)	06/22/16 Date

NEXTera ENERGY NUCLEAR FLEET	**L-16-1 NRC EXAM SCENARIO 2** DRAFT L-16-1 NRC EXAM SECURE INFOMATION	**SEG**

Site:	**Turkey Point Units 3 and 4 (PTN)**
Title:	**L-16-1 AUDIT EXAM SCENARIO 2**
LMS #:	**NRC 22**
LMS Rev Date:	**6/7/16** **Rev #: 0.0**
SEG Type:	☐ **Training** ☒ **Evaluation**
Program:	☐ **LOCT** ☒ **LOIT** ☐ **Other**
Duration:	**110 minutes**

Developed by:	Brian Clark *Instructor/Developer*	6/13/16 *Date*
Reviewed by:	[signature] *Instructor (Instructional Review)*	6/22/16 *Date*
Validated by :	[signature] *SME (Technical Review)*	06/22/16 *Date*
Approved by:	[signature] *Training Supervision*	6/22/16 *Date*
Approved by:	[signature] *Training Program Owner (Line)*	06/22/16 *Date*

Site:	Turkey Point Units 3 and 4 (PTN)
Title:	L-16-1 AUDIT EXAM SCENARIO 3
LMS #:	NRC 23
LMS Rev Date:	5/31/16 Rev #: 0.0
SEG Type:	☐ Training ☒ Evaluation
Program:	☐ LOCT ☒ LOIT ☐ Other
Duration:	120 minutes

Developed by: Brian Clark — Instructor/Developer — 6/15/16 Date

Reviewed by: [signature] — Instructor (Instructional Review) — 6/22/16 Date

Validated by : [signature] — SME (Technical Review) — 06/22/16 Date

Approved by: [signature] — Training Supervision — 6/22/16 Date

Approved by: [signature] — Training Program Owner (Line) — 06/22/16 Date

Site:	Turkey Point Units 3 and 4 (PTN)
Title:	L-16-1 NRC EXAM SCENARIO 4
LMS #:	NRC 24
LMS Rev Date:	6/8/16 Rev #: 0
SEG Type:	☐ Training ☒ Evaluation
Program:	☐ LOCT ☒ LOIT ☐ Other
Duration:	120 minutes

Developed by:	Brian Clark Instructor/Developer	6/16/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by :	SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	Training Program Owner (Line)	06/22/16 Date

Site:	Turkey Point Units 3 and 4 (PTN)
Title:	L-16-1 NRC EXAM SCENARIO 5
LMS #:	NRC 25
LMS Rev Date:	6/9/16 Rev #: 0
SEG Type:	☐ Training ☒ Evaluation
Program:	☐ LOCT ☒ LOIT ☐ Other
Duration:	120 minutes

Developed by:	Brian Clark Instructor/Developer	6/17/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by :	SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	Training Program Owner (Line)	06/22/16 Date

L-16-1 NRC Exam

Control Room - JPM A

JPM TITLE: Respond to Control Bank D Demanded Past 230 Steps

JPM NUMBER: 01028916302 **REV.** 1-0

TASK NUMBER(S) / TASK TITLE(S): 01028916300 / Respond to Control Bank D Demanded Past 230 Steps

K/A NUMBERS: 001 A4.14 **K/A VALUE:** RO 3.0 / SRO 3.4

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [X] Other: [ ]
Lab: [ ]

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: Yes
Alternate Path [INPO]: Yes

Developed by:	**Brian Clark** Instructor/Developer	**6/21/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/21/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
0-0	New JPM			N/A	N/A
				N/A	N/A
1-0	Formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A	N/A
				N/A	N/A

SIMULATOR SET-UP:

SIMULATOR SETUP INSTRUCTIONS:

______	1.	Reset to **IC 1** or saved IC.
______	2.	Place simulator in RUN.
______	3.	Ensure applicable portions of Simulator Operator Checklist are complete.
______	4.	*N/A if using saved IC* Perform following setup steps: a. Place Rod Motion Control Selector Switch in MANUAL b. Withdraw Control Bank D to 234 steps
______	5.	Open and execute L-16-1 NRC JPM A.lsn
______	6.	Allow plant to stabilize.
______	7.	Acknowledge alarms and place simulator in FREEZE.
______	8.	Save as temporary IC, if JPM will be repeated.
______	9.	When ready to begin, then place Simulator in RUN.

SIMULATOR MALFUNCTIONS:

- TFL10201: Continuous rod insertion in auto

SIMULATOR OVERRIDES:

- TCL1CS9: Reset Bank Overlap Step Counters

SIMULATOR REMOTE FUNCTIONS:

- N/A

Required Materials:	• Handout 3-ONOP-028 • Handout Unit 3 COLR
General References:	• 3-ONOP-028, Reactor Control System Malfunction • Plant Curve Book, Unit 3
Task Standards:	• Restore the Rod Control System to normal configuration with Bank D at 228 steps withdrawn • Respond to a continuous rod insertion event

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 3 is at 100% power.
- Control Bank D has been inadvertently withdrawn past 230 steps.
- The Control Bank D step counters currently indicate 234 steps.
- The crew entered 3-ONOP-028, Reactor Control System Malfunction, and completed the immediate actions.

INITIATING CUES:

- The Unit Supervisor directs you to restore the Rod Control System to its normal configuration by performing Section 5.5 of 3-ONOP-028, Reactor Control System Malfunction.
- The Shift Manager has authorized the return of control rods to automatic, on completion of the procedure.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 3-ONOP-028, Reactor Control System Malfunction.
Evaluator Note:	If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue.
Evaluator Cue:	Provide examinee with copies of handout 3-ONOP-028 and handout Unit 3 COLR.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 2 **Critical: No**	**3-ONOP-028, prior to Step 5.5.1:** **C A U T I O N** ***Demanding RCCs to step past the ARO position may cause failure of the stationary grippers and result in a misaligned, partially inserted or dropped rod.***
Standard:	Read CAUTION and determine it is satisfactory to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: No**	**3-ONOP-028, Step 5.5.1:** **Check for indications of misaligned or dropped RCCs**
Standard:	Observe Control Bank D's rod position indicators and determines that no rods are misaligned or dropped.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: No**	**3-ONOP-028, Step 5.5.2:** **IF a RCC is determined to be misaligned, THEN …**
Standard:	Read step, compare to Initial Conditions, and determine it does not apply.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: No**	**3-ONOP-028, Step 5.5.3:** **IF a RCC is determined to be dropped, THEN …**
Standard:	Read step, compare to Initial Conditions, and determine it does not apply.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: No**	**3-ONOP-028, Step 5.5.4:** **IF Control Bank D control rods have not been withdrawn more than 230 demanded steps, THEN …**
Standard:	Read step, compare to Initial Conditions, and determine it does not apply.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: No**	**3-ONOP-028, prior to Step 5.5.5:** **NOTE** ***Completion of the stepping sequence for each group in the bank is essential before resetting the Demand Counter, otherwise a stepping sequence or group misalignment may occur.***
Standard:	Read NOTE and determine it is satisfactory to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical: No**	**3-ONOP-028, Step 5.5.5:** **IF Control Bank D control rods have been withdrawn more than 230 demanded steps, THEN perform the following:** 1. Ensure group 2 step counter matches group 1 by completing the stepping sequence.
Standard:	Observe the step counters for groups 1 and 2 (Control Bank D) and ensure that both indicate 234 steps.
Evaluator Note:	The step counters for groups 1 and 2 of Control Bank D should both indicate 234 steps, per the Initial Conditions.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical: Yes**	**3-ONOP-028, Step 5.5.5:** **IF Control Bank D control rods have been withdrawn more than 230 demanded steps, THEN perform the following:** 2. Manually set both Control Bank D group demand step counters to 230 steps.
Standard:	Depress the LOWER pushbutton for group 1's step counter four times and observe that the counter indicates 230 steps; repeat for the group 2 step counter.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 10 **Critical: Yes**	**3-ONOP-028, Step 5.5.5:** **IF Control Bank D control rods have been withdrawn more than 230 demanded steps, THEN perform the following:** 3. Reset the bank demand step counters from the DCS as follows: a. Navigate to the RPI BANK DEMAND ALIGNMENT screen. b. Select INITIATE ALIGNMENT for Control Bank D. c. Select NEW VALUE in the overlay. d. Type in the bank demand step value of 230, using the keyboard. e. Press Enter. f. Select INITIATE RE ALIGNMENT in the overlay. g. Select YES in the save changes overlay. h. Select CLOSE OVERLAY.
Standard:	• Select: Main Menu → RPI Rod Position Summary → Related Displays → Bank Demand Alignment → Initiate Alignment for Control Rod Bank D. • Click on the number in the New Value box and observe that the box turns blue; enter 230, using the keyboard; press ENTER and observe that the New Value box turns green. • Select Initiate Realignment; press YES to save; select Close Overlay.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 11 **Critical: No**	**3-ONOP-028, prior to Step 5.5.5.4:** **N O T E** • ***The Bank Overlap Counter is located in 3B MCC Room in the Rod Control Logic Cabinet.*** • ***Manual stepping of the bank overlap counter logic can only be increased. The decrease button decreases the display, but does not properly decrease the internal logic. If the display value is too large, the reset button must be used and the correct value entered with the increase button.***
Standard:	Read NOTE and determine it is satisfactory to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 12 **Critical: No**	**3-ONOP-028, Step 5.5.5:** **IF Control Bank D control rods have been withdrawn more than 230 demanded steps, THEN perform the following:** 4. IF desired, reset the bank overlap counter. 5. Reset the Bank Overlap Counter by depressing the plus pushbutton to obtain a reading of 614, which corresponds to 230 steps.
Standard:	Dispatch a field operator to the Rod Control Logic Cabinet in the 3B MCC Room to perform Step 5.5.5.5 of 3-ONOP-028, Reactor Control System Malfunction.
Booth Operator Cue:	• When directed, trigger **RESET BANK OVERLAP COUNTER.** After complete (3 seconds), inform the examinee that the Bank Overlap Counter is currently reading 614. See Booth Operator Cue in Performance Step 14 below. • Verify **CONTINUOUS ROD INSERTION IN AUTO triggers**.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 13 **Critical: Yes**	**3-ONOP-028, Step 5.5.5:** **IF Control Bank D control rods have been withdrawn more than 230 demanded steps, THEN perform the following:** 6. IF the cycle ARO position is NOT 230 steps, THEN perform the following: a. Verify the Rod Motion Control Selector Switch is in MAN. b. Insert Control Bank D to the ARO position as defined in the Core Operating Limits Report.
Standard:	• Observe that the unit's current ARO value is 228 steps. • Verify that the Rod Motion Selector Switch is in MAN. • Take the Rod Motion IN-HOLD-OUT Switch to the IN position for two steps and observe that Control Bank D's group 1 and 2 step counters indicate 228 steps.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

THIS BEGINS THE ALTERNATE-PATH PORTION OF THE JPM

Performance Step: 14 **Critical: Yes**	**3-ONOP-028, Step 5.5.5:** **IF Control Bank D control rods have been withdrawn more than 230 demanded steps, THEN perform the following:** 6. IF the cycle ARO position is NOT 230 steps, THEN perform the following: c. IF directed by the Shift Manager, THEN Place Rod Control Selector Switch to AUTOMATIC position.
Standard:	• Place the Rod Motion Selector Switch in the AUTO position. • Recognize that Control Bank D begins inserting at fast speed. • Place the Rod Motion Selector Switch in the MAN position.
Evaluator Note:	Examinee should place the Rod Motion Selector Switch in the MAN position, prior to T_{avg} being 6°F less than T_{ref} (i.e., 2°F lower than transient band per 0-ADM-211, Emergency And Off-Normal Operating Procedure Usage).
Evaluator Cue:	When approached as the Shift Manager, indicate that the Rod Control System should be returned to automatic.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **Once the Rod Control Selector Switch is placed in manual, state "Another operator will complete the remaining steps of this procedure. This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ____________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 3 is at 100% power.
- Control Bank D has been inadvertently withdrawn past 230 steps.
- The Control Bank D step counters currently indicate 234 steps.
- The crew entered 3-ONOP-028, Reactor Control System Malfunction, and completed the immediate actions.

INITIATING CUES:

- The Unit Supervisor directs you to restore the Rod Control System to its normal configuration by performing Section 5.5 of 3-ONOP-028, Reactor Control System Malfunction.
- The Shift Manager has authorized the return of control rods to automatic, on completion of the procedure.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-028

CAUTION
Performance of this procedure may affect core reactivity.

Title:

Reactor Control System Malfunction

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Number	4
Revision Approval Date:	4/25/16

PCRs *10-0928, 1605435, 1975461, 2032570*
PC/Ms *92-031, 03-048, 07-019, 09-006*
ECs *246849*

LIST OF EFFECTIVE PAGES
(Rev. 4)

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1	04/25/16	12	11/11/04
2	04/25/16	13	07/14/10
3	09/28/07	14	04/12/02
4	09/28/07	15	04/25/16
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1.0 PURPOSE

1.1 This procedure provides instructions to be followed because of a Reactor Control System Malfunction due to:

1.1.1 Failure of an RCC to move due to being untrippable, CRDM failure, or rod control power supply failure

1.1.2 Failure of an RCC control bank to insert following a change in Turbine load or in boron concentration with reactor control in automatic

1.1.3 Continuous insertion of an RCC control bank

1.1.4 Continuous withdrawal of an RCC control bank

1.2 This procedure provides instructions to be followed when Control Bank D step demands greater than the All Rods Out (ARO) position (228, 229, or 230 steps as defined in Plant Curve Book, Section 7, COLR) while the Rod Motion Control Selector is positioned in MAN.

2.0 SYMPTOMS

2.1 Immovable RCC

2.1.1 The RCC is determined to be immovable during routine operation or performance of 3-OSP-028.6, RCCA PERIODIC EXERCISE.

2.2 Failure of an RCC Control Bank to Insert with Reactor Control in Automatic

2.2.1 Failure of the control banks to insert when Tavg exceeds Tref by greater than 1.5 degrees F

2.2.2 Annunciators

1. B 4/4, TAVG/TAVG - TREF DEVIATION
2. B 4/5, RCS HI/LO TAVG
3. B 9/4, ROD CONTROL URGENT FAILURE

2.3 Continuous Insertion of an RCC Control Bank

2.3.1 RCCs stepping in with Tavg and Tref matched, or Tavg less than Tref

2.3.2 Tavg decreases more than 1.5 degrees F below Tref

2.3.3 Decreasing reactor power

2.3.4 Failure of PT-3-446 (if selected)

2.3.5 Failure of PT-3-447 (if selected)

2.3.6 Failure of Power Range Channel 4

2.3.7 Failure of TM-408, Medium Signal Selector

2.3.8 Annunciators

1. B 4/4, TAVG/TAVG - TREF DEVIATION
2. B 8/1, ROD BANK A/B/C/D LO LIMIT
3. B 8/2, ROD BANK A/B/C/D EXTRA LO LIMIT

2.4 Continuous Withdrawal of an RCC Control Bank

2.4.1 RCCs stepping out as indicated on the RPIs or group demand step counters, and not manually initiated by the operator.

2.4.2 Tavg increases more than 1.5 degrees F above Tref

2.4.3 Annunciators

1. B 4/4, TAVG/TAVG - TREF DEVIATION
2. B 4/5, RCS HI/LO TAVG
3. B 6/3, POWER RANGE OVERPOWER ROD STOP

2.5 Control Bank D Demanded Past ARO Position

2.5.1 Control Bank D group demand step counters indicate greater than the ARO position (228, 229, or 230 steps as defined in Plant Curve Book, Section 7, COLR).

3.0 AUTOMATIC ACTIONS

3.1 Immovable RCC

3.1.1 None

3.2 Failure of an RCC Control Bank to Insert with Reactor Control in Automatic

3.2.1 Charging pump flow decrease in response to increasing pressurizer level.

3.3 Continuous Insertion of an RCC Control Bank

3.3.1 Charging pump flow increase

3.3.2 Actuation of the pressurizer heaters

3.3.3 Pressurizer low pressure reactor trip

3.4 Continuous Withdrawal of an RCC Control Bank

3.4.1 OPΔT reactor trip

3.4.2 OTΔT reactor trip

3.4.3 Power range high flux level reactor trip

3.4.4 Charging pump flow decrease

3.4.5 Pressurizer spray valves open on increasing pressurizer pressure.

3.5 Control Bank D Demanded Past ARO Position

3.5.1 None

C A U T I O N S

If the Rod Control System is inoperable due to Urgent Failure or other cause, the Shift Manager shall be notified immediately.

If a transient occurs and the Reactor cannot be stabilized by boration/dilution or changes in turbine load, the Reactor shall be tripped and a transition made to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

NOTES

Boration/dilution or changes in turbine load will effect shutdown margin and axial offset. If plant conditions permit, the Shift Manager shall be consulted for methods used to achieve and maintain stable plant conditions.

Failure of RCC(s) to move when demanded, (e.g., ROD CONTROL URGENT FAILURE), constitutes inoperability of the associated RCC(s). The requirements of T.S. 3.1.3.1 apply.

4.0 IMMEDIATE ACTIONS

4.1 Immovable RCC

4.1.1 **IF** the Rod Motion Control Selector is in Auto, **THEN** place in the MAN position.

4.1.2 **DO NOT** withdraw any control banks until the RCC(s) have been aligned.

4.2 Failure of an RCC Control Bank to Insert with Reactor Control in Automatic

4.2.1 Place the Rod Motion Control Selector switch to the MAN position.

4.3 Continuous Insertion of an RCC Control Bank

4.3.1 Place the Rod Motion Control Selector switch to the MAN position.

4.3.2 **IF** RCC control cannot be maintained manually, **THEN** trip the Reactor and Turbine and go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

4.4 Continuous Withdrawal of an RCC Control Bank

4.4.1 Place the Rod Motion Control Selector switch to the MAN position.

4.4.2 **IF** RCC control cannot be maintained manually, **THEN** trip the Reactor and Turbine and go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION.

4.5 Control Bank D Demanded Past ARO Position

4.5.1 None

5.0 **SUBSEQUENT ACTIONS**

5.1 Immovable RCC

5.1.1 **DO NOT** increase reactor power without permission from the Reactor Engineering Supervisor and the Shift Manager.

5.1.2 Maintain steady state condition as follows:

1. Maintain Tavg equal to Tref.

   a. Borate/dilute as necessary

   OR

   b. Change turbine load as necessary.

2. **IF** possible, **THEN** avoid insertion of the control rods.

5.1.3 Notify the following:

1. Reactor Engineering Supervisor or designee.
2. I&C Supervisor to verify RPI indication and to investigate CRDM System for possible failure.

5.1.4 **IF** one or more RCC is inoperable due to being immovable because of excessive friction or mechanical interference **OR** known to be untrippable, **THEN** proceed as follows:

1. Determine that the shutdown requirement of Technical Specification 3.1.1.1 is satisfied within 1 hour **AND**;
2. Be in Hot Standby within 6 hours, in accordance with 3-GOP-103, Power Operation to Hot Standby.

5.1.5 **WHEN** more than one full length rod is inoperable **OR** misaligned from the group step counter demand position by more than plus or minus 12 steps, **AND** RTP is greater than 90 percent, **THEN** within 1 hour, perform the following:

1. Restore all indicated rod positions to within the Allowed Rod Misalignment (plus or minus 12 steps),

   OR

2. Reduce RTP to less than 90 percent **AND** confirm that all indicated rod positions are within the Allowed Rod Misalignment (plus or minus 18 steps),

   OR

3. Be in Hot Standby within 6 hours.

5.1.6 **WHEN** more than one full length rod is inoperable **OR** misaligned from the group step counter demand position by more than plus or minus 18 steps **AND** RTP is equal to or less than 90 percent, **THEN** within 1 hour perform the following:

1. Restore all indicated rod positions to within the Allowed Rod Misalignment (plus or minus 18 steps),

OR

2. Be in Hot Standby within 6 hours.

5.1.7 **WHEN** one RCC is trippable but inoperable due to causes other than addressed in Step 5.1.4, **THEN** power operation may continue provided that within 1 hour:

1. The RCC is restored to an operable status

OR

2. The RCC is declared inoperable and the remainder of the RCCs in the bank with the inoperable RCC are aligned to within 12 steps with power greater than 90 percent RTP, **OR** within 18 steps when power is less than or equal to 90 percent RTP of the inoperable rod while not exceeding the rod sequence **AND** insertion limits using the Plant Curve Book, Section VII Figure 3 **AND**,

   a. The thermal power level shall be restricted in accordance with Technical Specification 3.1.3.6 during subsequent operation,

OR

3. The RCC is declared inoperable and the shutdown margin requirement of Technical Specification 3.1.1.1 is satisfied. Power operation may then continue provided that:

   a. The thermal power level is reduced to less than or equal to 75 percent within 1 hour, and within the next 4 hours the power range high neutron flux trip setpoint is reduced to less than or equal to 85 percent of rated thermal power, **AND**;

   b. The shutdown margin requirement of Technical Specification 3.1.1.1 is determined at least once per 12 hours, **AND**;

   c. A power distribution map is obtained from the incore movable detectors and $F_Q(Z)$ and $F_{\Delta H}N$ are verified to be within the limits within 72 hours, **AND**;

   d. A re-evaluation of each accident analysis listed in Enclosure 1 is performed within 5 days to confirm the previously analyzed results of these accidents remain valid under these conditions.

5.1.8 Mode 3 or 4 - Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable the shutdown margin shall be determined to be greater than or equal to 1 percent Δk/k.

1. **IF** the inoperable rod is immovable or untrippable, **THEN** the shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the immovable or untrippable control rod.

5.1.9 Mode 5 - Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the RCC(s) is inoperable the shutdown margin shall be determined to be greater than or equal to 1 percent Δk/k.

1. **IF** the inoperable control rod is immovable or untrippable, **THEN** the shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the inoperable RCC.

5.1.10 At the discretion of the Shift Manager, attempt to move one RCC at a time as follows:

1. Obtain permission from the Reactor Engineering Supervisor or designee.
2. Place all the lift coil disconnect switches for the misaligned rod bank to the disconnect position (toggle switch down) **EXCEPT** the immovable rod switch which is left in the connect position (toggle switch up).

NOTE

The ROD CONTROL URGENT FAILURE, B 9/4 and the RCC power cabinet URGENT FAILURE will alarm for the group with the lift coils disconnected. The RCC(s) in the cabinet with the Urgent Failure shall be considered inoperable until the system is restored to normal and the Shift Manager determines the RCC(s) to be operable.

3. Position the Rod Motion Control Selector switch to the RCC bank which has the immovable RCC.

5.1.10 (Cont'd)

4. Record the step position for the immovable RCC group step counter in the Unit Narrative Log as follows:

   a. Bank ____________

   b. Group ____________

   c. Position ____________ steps

5. Attempt to move the RCC by placing the Rod Motion Lever to the IN/OUT position, as applicable:

   a. **IF** the RCC moves, **THEN**:

      (1) **IF** the RCC is to be inserted, **THEN** adjust Turbine load to maintain Tref equal to Tavg.

      (2) **IF** the RCC is to be withdrawn, **THEN** borate as necessary to maintain steady state Reactor power.

      (3) Align the RCC with the rest of the bank.

6. Place all lift coil disconnect switches to the connect position (toggle switch up).

7. Manually set the associated group step counter to the position recorded in Substep 5.1.10.4.

8. **IF** the immovable rod is in a control bank, **THEN** reset the bank demand step counters from the DCS as follows:

   a. Navigate to the RPI BANK DEMAND ALIGNMENT screen.

   b. Select INITIATE ALIGNMENT for the affected control bank.

   c. Select NEW VALUE in the overlay.

   d. Type in the desired bank demand step value, using the keyboard.

   e. Press Enter.

   f. Select INITIATE RE ALIGNMENT in the overlay.

   g. Select YES in the save changes overlay.

   h. Select CLOSE OVERLAY.

5.1.11 **IF** another RCC is to be moved, **THEN** repeat Step 5.1.10.

5.1.12 After the malfunction has been corrected and prior to increasing power, monitor the following parameters to ensure the flux distribution is normal:

1. Power range nuclear instrumentation - less than 3 percent difference between any two detectors at the same elevation

2. Core exit thermocouples - less than 10°F difference between any two channels at like symmetric locations

NOTE

Performing an incore flux map is optional, at the discretion of the Reactor Engineering Supervisor.

3. No significant axial power shape difference from symmetric assemblies as determined by the Reactor Engineering Supervisor.

4. Axial flux indicators - less than 3 percent difference between any two channels

5.1.13 **IF** unit shutdown is required, **THEN** Reactor power should be reduced by decreasing Turbine Load and Boration to maintain programmed Tavg, in accordance with 3-GOP-103, Power Operation to Hot Standby.

5.1.14 **WHEN** the malfunction has been corrected, **THEN** place Rod Control Selector Switch to MANUAL or AUTOMATIC position.

5.2 Failure of an RCC Control Bank to Insert with Reactor Control in Automatic

CAUTION

For URGENT FAILURE condition rod motion is blocked. The cause must be corrected before moving rods. Resetting the Urgent Failure prior to correcting problem could result in racheting the mechanisms when the RESET pushbutton is depressed.

5.2.1 **DO NOT** increase reactor power without permission from the Reactor Engineering Supervisor and the Shift Manager.

5.2.2 Manually position the RCC control bank to restore steady state conditions.

1. **IF** the RCC control bank will still not move, **THEN** maintain steady state conditions with Tavg equal to Tref by:

   a. Boration/dilution.

   OR

   b. Changing turbine load.

5.2.3 Notify the following:

1. Reactor Engineering Supervisor or designee.
2. I&C Supervisor to verify RPI indication and to investigate CRDM System for possible failure.

5.2.4 Take actions required by Subsection 5.1, Immovable RCC.

5.2.5 **IF** unit shutdown is required, **THEN** Reactor power should be reduced by decreasing Turbine load and boration to maintain programmed Tavg, in accordance with 3-GOP-103, Power Operation to Hot Standby.

5.3 Continuous Insertion of an RCC Control Bank

NOTE

The rod stop bypass will need to be bypassed per 3-ONOP-059.8, POWER RANGE NUCLEAR INSTRUMENTATION MALFUNCTION, prior to withdrawing control rods if Power Range Channel 4 has failed .

5.3.1 Adjust rods or reduce turbine load as determined by the Shift manager to restore Tavg equal to Tref.

5.3.2 **IF** PT-3-446 or PT-3-447 has failed, **THEN** place Channel Select Turbine Inlet Control to the operable channel.

5.3.3 Compare rod position to control rod insertion limits using the Rod Position Bank Recorders (VPA) or using the Plant Curve Book, Section VII, Figure 3.

5.3.4 **IF** the control banks insertion limits are exceeded, **THEN** perform the following:

1. Borate at equal to or greater than 16 gpm using 0-OP-046, CVCS – BORON CONCENTRATION CONTROL, until control rods are above the Low Limit.

2. Ensure compliance with Technical Specifications by performing one of the following:

   a. Restore the control banks to within the limits within 2 hours.

   OR

   b. Reduce thermal power within 2 hours to less than or equal to the fraction of rated thermal power that is allowed by the bank position Plant Curve Book Section VII, Figure 3.

   OR

3. Be in Hot Standby within 6 hours.

5.3.5 **IF** Power Range Channel 4 has failed, **THEN** perform to 3-ONOP-059.8, POWER RANGE NUCLEAR INSTRUMENTATION MALFUNCTION.

5.3.6 **IF** PT-3-446 or PT-3-447 has failed, **THEN** perform the following:

NOTE

A few minutes needs to elapse between the time Turbine Inlet Pressure is transferred and Rod Control is returned to Automatic. This will preclude the possibility of the power mismatch circuitry causing undesired rod motion.

1. Verify Channel Select Turbine Inlet Press Control has been placed to an operable channel **AND** place the Rod Motion Control Selector switch in AUTO.

2. Perform 3-ONOP-049.1, DEVIATION OR FAILURE OF SAFETY RELATED OR REACTOR PROTECTION CHANNELS.

5.3.7 Check TI-3-412D, 422D, 432D for possible indication of a failure of TM-408, Medium Signal Selector.

1. **IF** a failure of a protection channel is indicated, **THEN** notify I&C and refer to 3-ONOP-049.1, Deviation or Failure of Safety Related Reactor Protection Channels.

2. **IF** TM-3-408 input to SDTC is affected, **THEN** place the Steam Dump to Condenser Selector Switch to MAN.

   a. Place a caution tag on the SDTC selector switch stating that placing back to AUTO may result in undesired operation or disable all SDTC function.

5.3.8 Place a caution tag on the Rod Control Selector switch stating that placing rods in auto may result in undesired rod motion until the system is restored to normal. (N/A if rods were restored to AUTO.)

5.3.9 Notify I&C of the problem with the Rod Control System.

5.4 Continuous Withdrawal of an RCC Control Bank

5.4.1 Adjust rods to maintain Tavg equal to Tref.

5.4.2 Notify I&C Department to investigate failure of the rod control system.

5.4.3 Operate rods in manual until cause of rod control system failure has been found and corrected.

5.4.4 Place a caution tag on the Rod Control Selector switch stating that placing rods in auto may result in undesired rod motion until the system is restored to normal.

5.5 Control Bank D Demanded Past ARO Position

CAUTION

Demanding RCCs to step past the ARO position may cause failure of the stationary grippers and result in a misaligned, partially inserted or dropped rod.

5.5.1 Check for indications of misaligned or dropped RCCs.

5.5.2 **IF** a RCC is determined to be misaligned, **THEN** go to 3-ONOP-028.1, RCC MISALIGNMENT.

5.5.3 **IF** a RCC is determined to be dropped, **THEN** go to 3-ONOP-028.3, DROPPED RCC.

5.5.4 **IF** Control Bank D control rods have not been withdrawn more than 230 demanded steps, **THEN** perform the following:

1. Verify the Rod Motion Control Selector switch is in MAN.
2. Insert Control Bank D to All Rods Out as defined in the Core Operating Limits Report.
3. **IF** directed by the Shift Manager, **THEN** Place Rod Control Selector Switch to AUTOMATIC position.

NOTE

Completion of the stepping sequence for each group in the bank is essential before resetting the Demand Counter, otherwise a stepping sequence or group misalignment may occur.

5.5.5 **IF** Control Bank D control rods have been withdrawn more than 230 demanded steps, **THEN** perform the following:

1. Ensure group 2 step counter matches group 1 by completing the stepping sequence.
2. Manually set both Control Bank D group demand step counters to 230 steps.
3. Reset the bank demand step counters from the DCS as follows:
   a. Navigate to the RPI BANK DEMAND ALIGNMENT screen.
   b. Select INITIATE ALIGNMENNT for Control Bank D.
   c. Select NEW VALUE in the overlay.
   d. Type in the bank demand step value of 230, using the keyboard.
   e. Press Enter.
   f. Select INITIATE RE ALIGNMENT in the overlay.
   g. Select YES in the save changes overlay.
   h. Select CLOSE OVERLAY.

NOTE

- *The Bank Overlap Counter is located in 3B MCC Room in the Rod Control Logic Cabinet.*
- *Manual stepping of the bank overlap counter logic can only be increased. The decrease button decreases the display, but does not properly decrease the internal logic. If the display value is too large, the reset button must be used and the correct value entered with the increase button.*

4. **IF** desired, reset the bank overlap counter.
5. Reset the Bank Overlap Counter by depressing the plus pushbutton to obtain a reading of 614, which corresponds to 230 steps.
6. **IF** the cycle ARO position is NOT 230 steps, **THEN** perform the following:
   a. Verify the Rod Motion Control Selector switch is in MAN.
   b. Insert Control Bank D to the ARO position as defined in the Core Operating Limits Report.
   c. **IF** directed by the Shift Manager, **THEN** Place Rod Control Selector Switch to AUTOMATIC position.

6.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

6.1 References

6.1.1 Technical Specifications

1. Section 3/4.1.1, Reactivity Control System - Shutdown Margin
2. Section 3/4.1.3, Reactivity Control Systems - Movable Control Assemblies
3. Section 3/4.2.1, Power Distribution Limits - Axial Flux Difference
4. Section 3/4.2.3, Power Distribution Limits - Nuclear Enthalpy Rise Hot Channel Factor
5. Section 3/4.2.4, Power Distribution Limits - Quadrant Power Tilt Ratio

6.1.2 FSAR

1. Section 14.1.2, Uncontrolled RCCA Withdrawal at Power
2. Section 14.1.4, Rod Cluster Assembly (RCCA) Drop

6.1.3 Plant Procedures

1. 0-ADM-555, Reactivity Management
2. 3-ARP-097.CR, Control Room Annunciator Response - Panel B
3. 3-GOP-103, Power Operation to Hot Standby
4. 3-ONOP-049.1, Deviation or Failure of Safety Related or Reactor Protection Channels
5. 3-ONOP-059.4, Excessive Axial Flux Difference
6. 3-ONOP-059.8, Power Range Nuclear Instrumentation Malfunction
7. 3-OSP-028.6, RCCA Periodic Exercise
8. 0-OSP-040.5, Nuclear Design Verification

6.1.4 Plant Curve Book, Unit 3

1. Section VII, Figure 3, Control Rod Insertion Limits (graph)

6.1.5 INPO

1. SOER 84-2 (1,2, and 8), Control Rod Mispositioning

2. SOER 84-2, Control Rod Mispositioning - Addendum

6.1.6 Miscellaneous Documents (i.e., PC/Ms, Correspondence)

1. PC/M 09-006, Rod Position Indication System Replacement

6.2 Records Required

6.2.1 Completed copies of the below listed items constitute Quality Assurance records and shall be transmitted to QA Records for retention in accordance with Quality Assurance Records Program requirements:

1. None

6.3 Commitment Documents

6.3.1 None

END OF TEXT

ENCLOSURE 1
(Page 1 of 1)

ACCIDENT ANALYSES REQUIRING RE-EVALUATION IN THE EVENT OF AN INOPERABLE RCC

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

FINAL PAGE

1.0 INTRODUCTION

This Core Operating Limits Report for Turkey Point Unit 3 Cycle 28 has been prepared in accordance with the requirements of Technical Specification 6.9.1.7.

The Technical Specifications (TS) affected by this report are listed below with the section and page for each one of the TS addressed in this COLR document.

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in the Introduction are presented below and listed sequentially by Technical Specification (TS). These limits have been developed using the NRC-approved methodologies specified in TS 6.9.1.7.

2.1 Reactor Core Safety Limits – Three Loops in Operation (TS 2.1.1)

- **Figure A1**(page 14A-A7) In Modes 1 and 2, the combination of Thermal Power, reactor coolant system highest loop average temperature and pressurizer pressure shall not exceed the limits in Figure A1.

2.2 Reactor Trip System Instrumentation Setpoints (TS 2.2.1)

NOTE 1 on TS Table 2.2-1 Overtemperature ΔT

- **τ_1 = 0s, τ_2 = 0s** Lead/Lag compensator on measured ΔT
- **τ_3 = 2s** Lag compensator on measured ΔT
- **K_1 = 1.31**
- **K_2 = 0.023/°F**
- **τ_4 = 25s, τ_5 = 3s** Time constants utilized in the lead-lag compensator for T_{avg}
- **τ_6 = 2s** Lag compensator on measured T_{avg}
- **$T' \leq$ 583.0 °F** Indicated Loop T_{avg} at RATED THERMAL POWER
- **K_3 = 0.00116/psi**
- **$P' \geq$ 2235 psig** Nominal RCS operating pressure
- **$f_1(\Delta I)$ = 0 for $q_t - q_b$ between – 18% and + 7%.**

  For each percent that the magnitude of $q_t - q_b$ exceeds – 18%, the ΔT Trip Setpoint shall be automatically reduced by 3.51% of its value at RATED THERMAL POWER; and

  For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 2.37% of its value at RATED THERMAL POWER.

  Where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

NOTE 2 on TS Table 2.2-1 Overtemperature ΔT

The Overtemperature ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.2% ΔT span for the Pressurizer Pressure channel, and 0.4% ΔTspan for the f(ΔI) channel. No separate Allowable Value is provided for Tavg because this function is part of the ΔT value.

NOTE 3 on TS Table 2.2-1 Overpower ΔT

- **K_4 = 1.10**	
- **$K_5 \geq$ 0.0/°F**	For increasing average temperature
- **K_5 = 0.0/°F**	For decreasing average temperature
- $\tau_7 \geq$ **0 s**	Time constants utilized in the lead-lag compensator for T_{avg}
- **K_6 = 0.0016/°F**	For T > T”
- **K_6 = 0.0**	For T $\leq$ T”
- **T” $\leq$ 583.0°F**	Indicated Loop T_{avg} at RATED THERMAL POWER
- **f_2 (ΔI) = 0**	For all ΔI

NOTE 4 on TS Table 2.2-1 Overpower ΔT

The Overpower ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel. No separate Allowable Value is provided for Tavg because this function is part of the ΔT value.

2.3 Shutdown Margin Limit for MODES 1, 2, 3 and 4 (TS 3.1.1.1)

- **Figure A2** (page 14A-A8)

2.4 Shutdown Margin Limit for MODE 5 (TS 3.1.1.2)

- **$\geq$ 1.77 % Δk/k**

2.5 Moderator temperature coefficient (MTC) (TS 3.1.1.3)

- **$\leq$ + 5.0 x 10^{-5} Δk/k/°F** BOL, HZP, ARO and, from HZP to 70% Rated Thermal Power (RTP)

- **From 70% RTP to 100% RTP the MTC decreasing linearly from $\leq$ + 5.0 x 10^{-5} Δk/k/°F to $\leq$ 0.0 x 10^{-5} Δk/k/°F**

- **Less negative than - 41.0 x 10^{-5} Δk/k/°F** EOL, RTP, ARO

2.6 Moderator temperature coefficient (MTC) Surveillance at 300 ppm (TS 4.1.1.3)

- **Less negative than - 35.0 x 10^{-5} Δk/k/°F** Within 7 EFPD of reaching equilibrium boron concentration of 300 ppm.

2.7 Analog Rod Position Indication System (TS 3.1.3.2)

- **Figure A3** (page 14A-A9) The All Rods Out (ARO) position for all shutdown Banks and Control Banks is defined to be 228 steps withdrawn.

2.8 Control Rod Insertion Limits (TS 3.1.3.6)

- **Figure A3** (page 14A-A9) The control rod banks shall be limited in physical insertion as specified in Figure A3 for ARO = 228 steps withdrawn.

2.9 Axial Flux Difference (TS 3.2.1)

- **Figure A4** (page 14A-A10)

2.10 Heat Flux Hot Channel Factor $F_Q(Z)$ (TS 3.2.2)

- **$[F_Q]^L$ = 2.30**

- **K(z) = 1.0** For 0' $\leq$ z $\leq$ 12' where z is core height in ft

2.11 Nuclear Enthalpy Rise Hot Channel Factor (TS 3.2.3)

- **$F_{\Delta H}^{RTP}$ = 1.600 $PF_{\Delta H}$ = 0.3**

2.12 DNB Parameters (TS 3.2.5)

- **RCS Tavg $\leq$ 585.0 °F**
- **Pressurizer Pressure $\geq$ 2204 psig**

Figure A1

Reactor Core Safety Limit – Three Loops in Operation

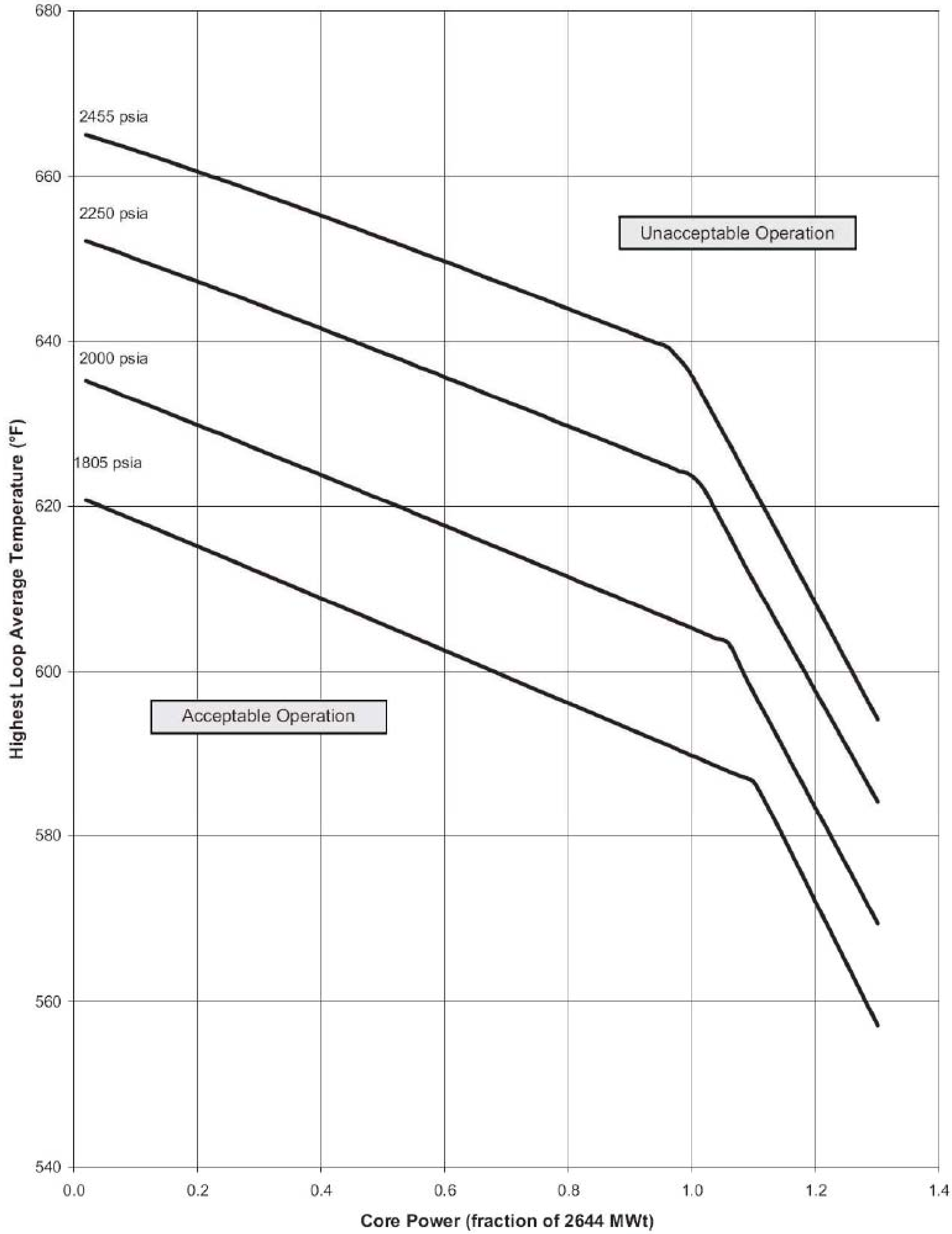


Figure A2

Required Shutdown Margin vs Reactor Coolant Boron Concentration

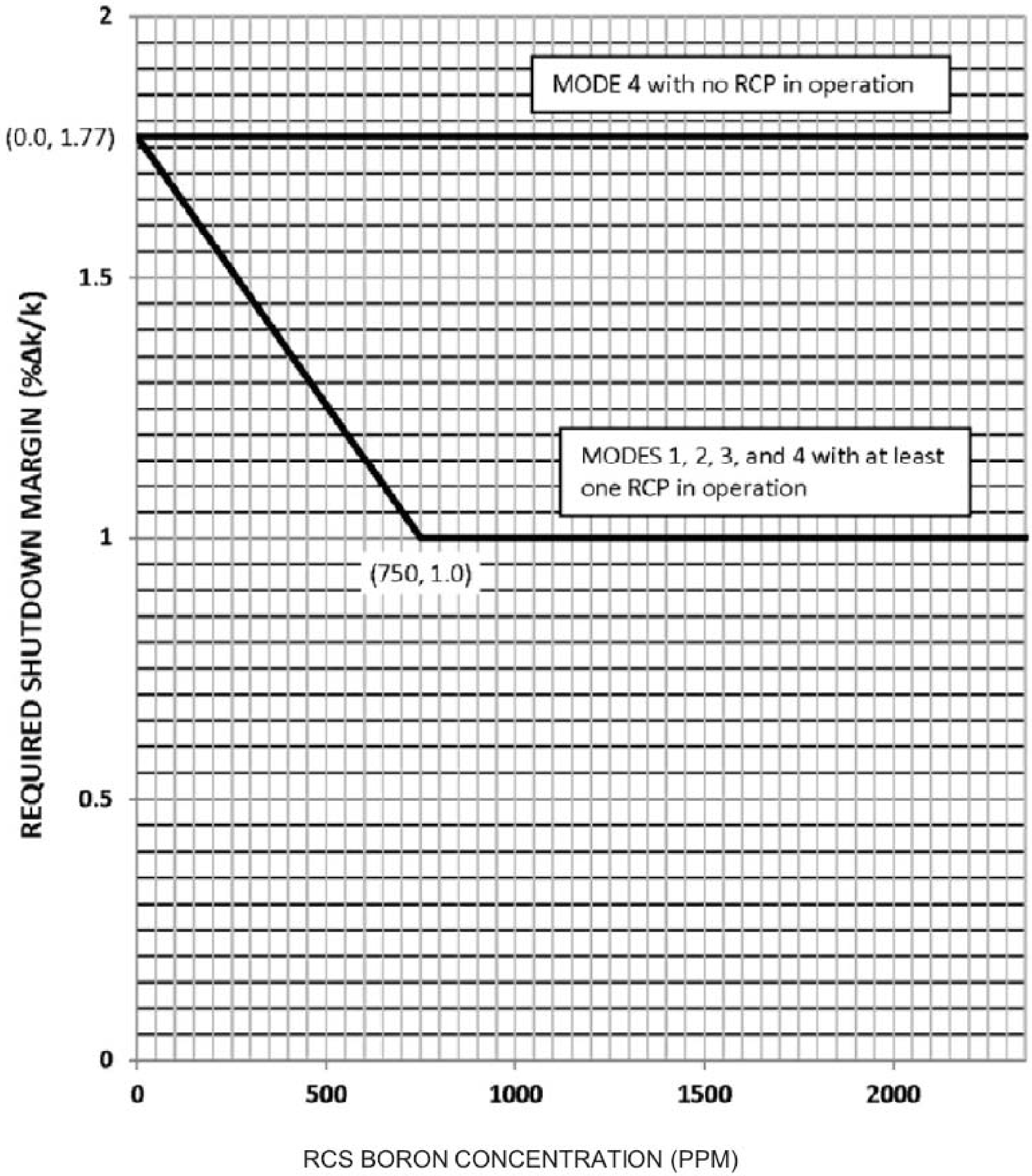


FIGURE A3

Turkey Point Unit 3 Cycle 28 Rod Insertion Limits vs Thermal Power
ARO = 228 Steps Withdrawn, Overlap = 100 Steps

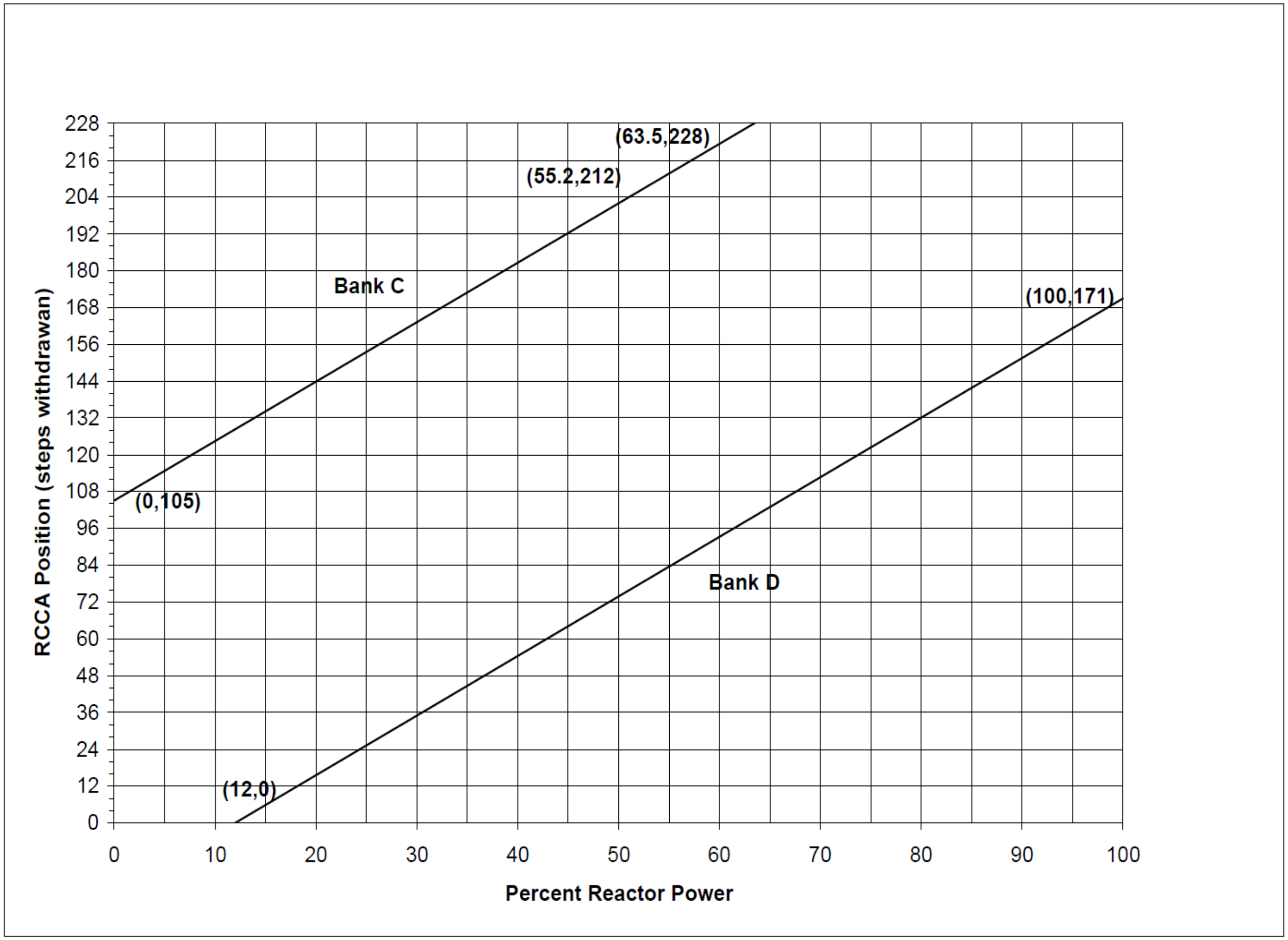
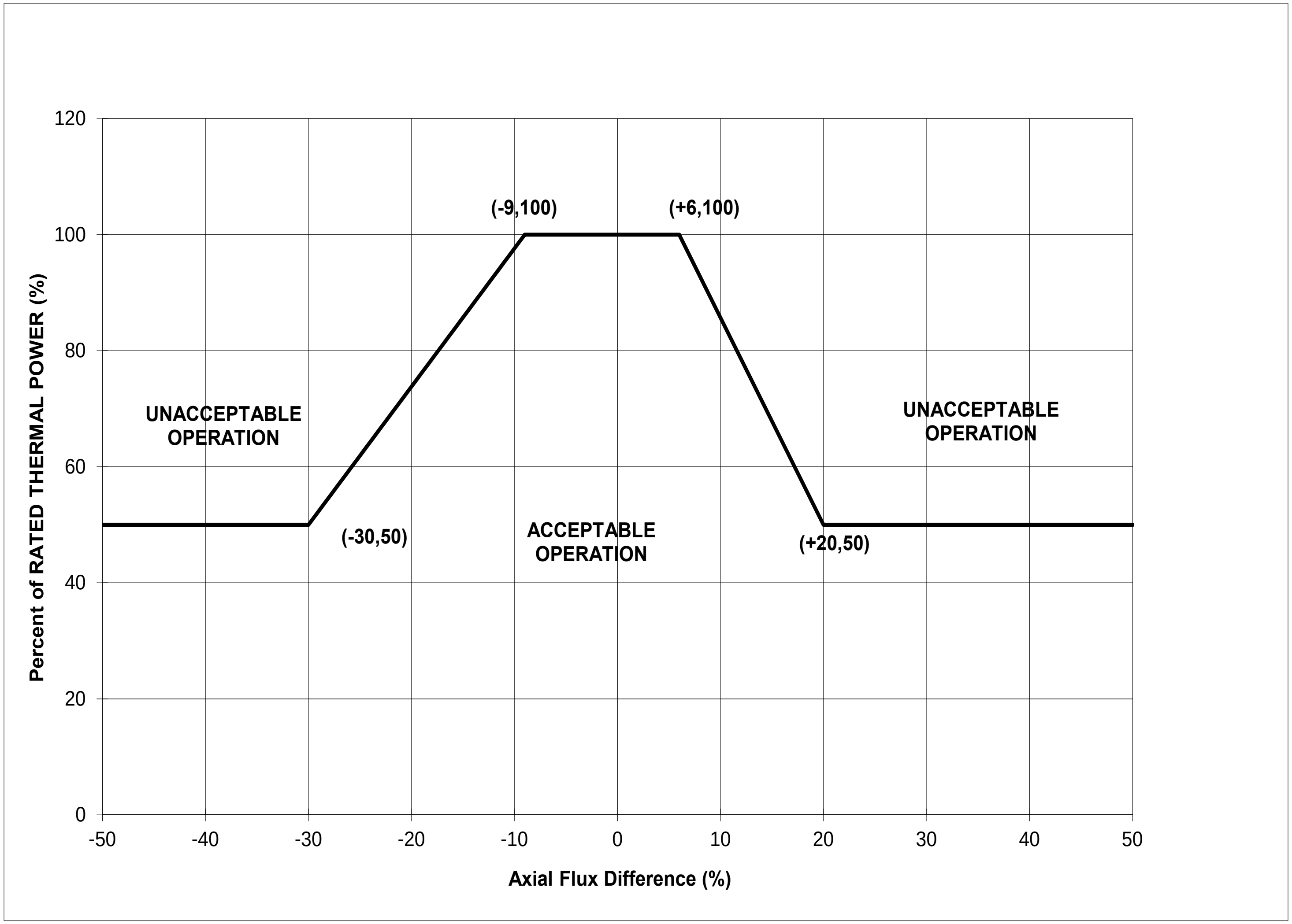


FIGURE A4

Axial Flux Difference as a Function of Rated Thermal Power
Turkey Point Unit 3 Cycle 28



L-16-1 NRC Exam

Control Room - JPM B

JPM TITLE: Place Excess Letdown in Service

JPM NUMBER: 01047016102 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 01047016100 / Initiate Excess Letdown

K/A NUMBERS: 004 A4.06 **K/A VALUE:** RO 3.6 / SRO 3.1

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION:

In-Plant:	[ ]	Control Room:	[ ]
Simulator:	[X]	Other:	[ ]
Lab:	[ ]		

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: Yes

Alternate Path [INPO]: Yes

Developed by:	**Brian Clark** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER SUPERVISOR	DATE DATE
1-0	Updated to fleet template; text/grammar changes	2015 LOCT Annual Exam	N/A	N/A N/A	N/A N/A
2-0	Formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A N/A	N/A N/A

SIMULATOR SET-UP:

SIMULATOR SETUP INSTRUCTIONS:

______	1.	Reset to **IC 1** or equivalent IC.
______	2.	Place simulator in RUN.
______	3.	Ensure applicable portions of Simulator Operator Checklist are complete.
______	4.	Open and execute L-16-1 NRC JPM B.lsn: a. Verify trigger **RV-3-304 Fails Open** is in CONDITION state
______	5.	Allow plant to stabilize.
______	6.	Acknowledge alarms and place simulator in FREEZE.
______	7.	Save as temporary IC, if JPM will be repeated.
______	8.	When ready to begin, then place Simulator in RUN.

SIMULATOR MALFUNCTIONS:

- TFBVO304 – RV-304 Fails Open
- TFBVO10 – 387 Fails Open

SIMULATOR OVERRIDES:

- N/A

SIMULATOR REMOTE FUNCTIONS:

- N/A

Required Materials:	• Handout 3-OP-047
General References:	• 3-OP-047, CVCS Charging and Letdown • 3-ONOP-041.3, Excessive Reactor Coolant System Leakage
Task Standards:	• Place excess letdown in service • Recognize that RV-3-304 has failed open • Start a charging pump, per 3-ONOP-041.3, to maintain pressurizer level

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 3 is in Mode 1.
- Personnel are currently being briefed to perform maintenance and calibration on LCV-3-460, Letdown Isolation Valve.

INITIATING CUES:

- In preparation for the isolation of normal letdown to support maintenance, the Unit Supervisor directs you to place excess letdown in service in accordance with Section 7.12 of 3-OP-047, CVCS Charging and Letdown.
- All applicable prerequisites in Section 3.0 are satisfied.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical : No**	**Obtain required reference materials.**
Standard:	Obtain 3-OP-047, CVCS Charging and Letdown, Section 7.12.
Evaluator Note:	If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue.
Evaluator Cue:	Provide examinee with a copy of handout 3-OP-047.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical : No**	**3-OP-047, Step 7.12.2.1:** **Verify Excess Ltdn Hx CCW Outlet, CV-3-739, is Open**
Standard:	Recognize that CV-3-739 is open.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical : No**	**3-OP-047, Step 7.12.2.2:** **Verify greater than 200 gpm and less than or equal to 238 gpm CCW flow on flow indicator FI-3-624 (located in the Pipe and Valve Room)**
Standard:	Contact field operator and verify that flow on FI-3-624 is satisfactory.
Booth Operator Cue:	Monitor flow rate on FI-3-624 as follows: *Schema → Common Services → Component Cooling → CCW to RCP & XS LTDWN Hxs (bottom left) → Flow displayed on right side of screen*
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical : No**	**3-OP-047, Step 7.12.2.3:** **Verify Excess Ltdn Iso Valve, CV-3-387, is Closed**
Standard:	Recognize that CV-3-387 is closed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical : No**	**3-OP-047, Step 7.12.2.4:** **Verify Excess Ltdn Divert to WDS, CV-3-389, is aligned to the VCT (Switch to Normal)**
Standard:	Recognize that CV-3-389 is aligned to the VCT.
Evaluator Note:	Switch will be in the VCT-NORM position.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical : No**	**3-OP-047, Step 7.12.2.5:** **Slowly Open Excess Letdown Flow Controller, HCV-3-137, to allow excess letdown lines to backfill**
Standard:	Slowly open HCV-3-137 by turning the controller's potentiometer clockwise.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical : No**	**3-OP-047, Step 7.12.2.6:** **<u>WHEN</u> a minimum of 5 minutes have elapsed, <u>THEN</u> Close Excess Letdown Flow Controller, HCV-3-137**
Standard:	When 5 minutes have elapsed, close HCV-3-137 by turning the controller's potentiometer counterclockwise.
Evaluator Note:	When the requirement for the 5 minute wait is identified, state that 5 minutes have elapsed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical : Yes**	**3-OP-047, Step 7.12.2.7:** **Open Excess Ltdn Isol Valve, CV-3-387, AND observe Containment Sump level for indication that RV-3-304 may have lifted**
Standard:	• Open CV-3-387 (critical) • Monitor containment sump level and recognize that it is rising (NOT critical)
Evaluator Note:	The examinee may continue with the procedure before noticing that the sump level is rising; this is addressed below. Annunciators G5/3 (CNTMT LEVEL INCREASING > 1 GPM) and G1/2 (CHARGING PUMP HI SPEED) will eventually actuate.
Evaluator Cue:	Acknowledge any communication to the Unit Supervisor.
Booth Operator Cue:	Verify that **RV-3-304 Fails Open** triggers when the control switch for CV-3-387 is taken to the OPEN position.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical : No**	**3-OP-047, Steps 7.12.2.8 and 7.12.2.9:** **Slowly Open Excess Letdown Flow Controller, HCV-3-137, allowing the heat exchanger to warm up** **Monitor heat exchanger outlet temperature at Excess Ltdn Hx Temp Indicator, TI-3-139**
Standard:	Slowly open HCV-3-137, by turning the controller's potentiometer clockwise, and monitor TI-3-139.
Evaluator Note:	This step may not be performed if the sump level is recognized as rising; if so, mark this step N/A.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 10 **Critical : No**	**Review ARP for G5/3, CNTMT LEVEL INCREASING > 1 GPM**
Standard:	• Check Cntmt Sump Recorders: R-1418 (VPA), R-6308A/B (behind RCO desk) • Monitor RCS parameters for indications of a RCS leak • Monitor Component Cooling Water parameters for indication of a CCW System Leak • Perform 3-OSP-041.1, Reactor Coolant System Leak Rate Calculation, to determine RCS leak rate • Go to 3-ONOP-041.3, Excessive Reactor Coolant System Leakage, and take actions as directed • Refer To Tech Spec 3.4.6.2
Evaluator Note:	• This step may be marked N/A if not used • If applicable, state that the STA will perform the OSP leak rate • Examinee may use 0-ADM-211 guidance to close CV-3-387 and attempt to isolate the leak; CV-3-387 is failed open
Evaluator Cue:	Acknowledge any communication to the Unit Supervisor.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 11 **Critical : No**	**Review ARP for G1/2, CHARGING PUMP HI SPEED**
Standard:	• Check individual charging pump controller and the master charging pump controller • IF leakage is confirmed, THEN go to 3-ONOP-041.3, Excessive Reactor Coolant System Leakage
Evaluator Note:	• If requested, provide examinee with a copy of handout 3-ONOP-041.3 • This step may be marked N/A if not used • Examinee may use 0-ADM-211 guidance to close CV-3-387 and attempt to isolate the leak; CV-3-387 is failed open
Evaluator Cue:	Acknowledge any communication to the Unit Supervisor.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

THIS BEGINS THE ALTERNATE-PATH PORTION OF THE JPM

Performance Step: 12 **Critical : No**	**Close CV-3-387 to isolate RCS leakage**
Standard:	Take the control switch for CV-3-387 to CLOSE.
Evaluator Note:	• Once the leakage is recognized the examinee may attempt to isolate the leak by closing CV-3-387; CV-3-387 is failed open • If no attempt is made here to close CV-3-387 then mark the step N/A
Evaluator Cue:	Acknowledge any communication to the Unit Supervisor.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 13 **Critical : No**	**Enter 3-ONOP-041.3, Excessive Reactor Coolant System Leakage**
Standard:	Locate and enter 3-ONOP-041.3.
Evaluator Note:	• Provide the examinee with a copy of HANDOUT 3-ONOP-041.3 • If the ONOP was entered previously, mark this step as satisfactory
Evaluator Cue:	Acknowledge any communication to the Unit Supervisor.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 14 **Critical : Yes**	**3-ONOP-041.3, Step 1:** **Maintain RCS Inventory**
Standard:	Start additional charging pumps as necessary to maintain RCS Inventory.
Evaluator Note:	• The examinee may first attempt to stabile pressurizer level by taking charging to manual and maximizing output • Satisfactory completion of this step requires that pressurizer level is stabilized or trending to program
Evaluator Cue:	Acknowledge any communication to the Unit Supervisor.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 15 **Critical : No**	**3-ONOP-041.3, Step 2:** **Check RCS Inventory Decreasing**
Standard:	Recognize that RCS inventory is NOT lowering and proceed to Step 10 (per RNO).
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When it is recognized that RCS inventory is no longer decreasing, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ____________

Examinee: ______________________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ____

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 3 is in Mode 1.
- Personnel are currently being briefed to perform maintenance and calibration on LCV-3-460, Letdown Isolation Valve.

INITIATING CUES:

- In preparation for the isolation of normal letdown to support maintenance, the Unit Supervisor directs you to place excess letdown in service in accordance with Section 7.12 of 3-OP-047, CVCS Charging and Letdown.
- All applicable prerequisites in Section 3.0 are satisfied.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-041.3

Title:

Excessive Reactor Coolant System Leakage

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Number:	0B
Revision Approval Date:	10/19/15

PCRs *2026527, 2075588*
ECs *96-092*

W2010:DH/ev/fm/cls

LIST OF EFFECTIVE PAGES
(Rev. 0B)

Page	Revision Date
1	10/19/15
2	10/19/15
3	08/16/04
4	08/16/04
5	08/16/04
6	08/16/04
7	08/16/04
8	08/16/04
9	08/16/04
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11	08/16/04
12	08/16/04
13	08/16/04
14	08/16/04
Foldout	10/19/15

TABLE OF CONTENTS

1.0 PURPOSE

1.1 This procedure provides instructions to be followed in the event of an excessive Reactor Coolant System (RCS) leak that does NOT involve the following:

1.1.1 An RCS leak which results in the initiation of safety injection.

1.1.2 A steam generator tube leak.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 Annunciators

2.1.1 G 1/2, CHARGING PUMP HI SPEED

2.1.2 A 9/3, PZR CONTROL HI/LO LEVEL

2.1.3 I 4/6, CNTMT SUMP HI LEVEL

2.1.4 G 9/5, CNTMT SUMP HI LEVEL

2.1.5 G 5/3, CNTMT LEVEL INCREASING > 1 GPM

2.1.6 H 8/6, CCW HEAD TANK HI/LOW LEVEL

2.1.7 H 1/4, PRMS HI RADIATION

2.1.8 H 3/6, PRMS R-11/R-12 BYPASSED/WARNING ACTUATED

2.2 Indications

2.2.1 Increased charging flow

2.2.2 Possible decrease in pressurizer level

2.2.3 Increased containment sump level

2.2.4 Excessive RCS leakage as calculated by 3-OSP-041.1, REACTOR COOLANT SYSTEM LEAK RATE CALCULATION

2.2.5 Increasing component cooling water head tank level **AND** increasing component cooling water activity indicated on R-3-17A OR R-3-17B

2.2.6 Increase in the frequency of RCS makeup water addition

2.2.7 Increased frequency of operation of containment sump pumps

2.2.8 Possible decrease in pressurizer pressure

2.2.9 Abnormal activity as indicated on R-3-11 OR R-3-12

2.2.10 Abnormal activity as indicated on R-14

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 Technical Specification

1. 3.4.6.2, Reactor Coolant System Operational Leakage

3.1.2 Final Safety Analysis Report

1. Section 6.5, Leakage Detection

3.1.3 Plant Drawings

None

3.1.4 Procedures

1. 3-GOP-103, Power Operation To Hot Standby
2. 3-GOP-305, Hot Standby To Cold Shutdown
3. 3-OSP-041.1, RCS Leak Rate Calculation
4. 3-ONOP-041.7, Shutdown LOCA [Mode 3 (Less than 1000 psig) OR Mode 4].
5. 3-ONOP-041.8, Shutdown LOCA [Mode 5 OR 6]
6. 3-ONOP-067, Radioactive Effluent Release
7. 3-ONOP-071.2, Steam Generator Tube Leakage
8. 0-EPIP-20101, Duties And Responsibilities Of Emergency Coordinator

3.1.5 Plant Change/Modifications

1. PC/M 96-092, Addition Of U/3 CCW Head Tank

3.1.6 Miscellaneous Documents

None

3.2 Records Required

None

3.3 Commitment Documents

None

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Maintain RCS Inventory** a. Maintain RCS Inventory as directed by the Unit Supervisor * Maintain program level **<u>OR</u>** * Maintain ordered band for operational mode **<u>OR</u>** * Maintain unit water solid (if unit water solid prior to event) b. Start additional charging pumps as necessary to maintain RCS Inventory c. **<u>IF</u>** charging flow is maximum, **<u>THEN</u>** isolate letdown flow	
2	**Check RCS Inventory Decreasing**	Go to Step 10.
3	**Check The Following** a. Charging flow - MAXIMUM b. Letdown flow - ISOLATED	Return to Step 1.
4	**Check Unit In Mode 1 Through 3 Greater Than 1000 psig With Safety Injection System Aligned For Injection**	Go to Step 6.
5	**Manually Trip The Reactor <u>AND</u> Go To 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION**	
6	**Check Unit Operating Mode 3 Less Than 1000 psig With Safety Injection Blocked Or Mode 4**	Go to Step 8.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	**Go To 3-ONOP-041.7, SHUTDOWN LOCA [MODE 3 (LESS THAN 1000 PSIG) OR MODE 4]**	
8	**Check Unit Operating Mode 5 or 6 With Refueling Cavity NOT FILLED**	Go to 3-ONOP-033.2, REFUELING CAVITY SEAL FAILURE.
9	**Go To 3-ONOP-041.8, SHUTDOWN LOCA [MODE 5 OR 6]**	
10	**Monitor RCS Leakage**	
	a. Perform The Following	
	1) Determine RCS leak rate using the appropriate leak rate procedure	
	* 3-OSP-041.1, REACTOR COOLANT SYSTEM LEAKRATE CALCULATION	
	OR	
	* 3-OSP-041.2, REACTOR COOLANT SYSTEM VISUAL LEAK INSPECTION AND LEAK EVALUATION	
	2) Attempt to identify the source of the leak	
	3) Check if the leak is isolable	3) Go to Step 11.
	4) Isolate the leak as follows	
	* **IF** leakage is from the RHR System, **THEN** perform ATTACHMENT 1	
	OR	
	* Plant Clearance	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	**Check For Additional Indications Of RCS Leakage**	
	a. Verify R-11 - STABLE **OR** DECREASING	a. Perform the following: 1) Close Containment Instrument Air Bleed Valves, CV-3-2819 And CV–3–2826. 2) Close Containment Sump Pump Discharge Valves, CV-3-2821 And CV-3-2822. 3) Perform 3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE, while continuing with this procedure.
	b. Verify R-12 - STABLE **OR** DECREASING	b. Perform the following: 1) Close Containment Instrument Air Bleed Valves, CV-3-2819 and CV–3–2826. 2) Close Containment Sump Pump Discharge Valves, CV-3-2821 and CV-3-2822. 3) Perform 3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE, while continuing with this procedure.
	c. Verify R-14 - STABLE **OR** DECREASING	c. Perform 3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE, while continuing with this procedure.
	d. Verify SG tubes - INTACT • R-15 - STABLE **OR** DECREASING • R-19 - STABLE **OR** DECREASING • SECONDARY SAMPLE RESULTS	d. Perform 3-ONOP-071.2, STEAM GENERATOR TUBE LEAKAGE, while continuing with this procedure.
	e. Verify RCS to Component Cooling Water boundary - INTACT • R-17A STABLE **OR** DECREASING • R-17B STABLE **OR** DECREASING	e. Perform 3-ONOP-067, RADIOACTIVE EFFLUENT RELEASE, while continuing with this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12	**Determine If RCS Leakage Within Limits Of Technical Specifications**	
	a. Check Technical Specifications	
	b. Verify RCS leakage - LESS THAN LIMIT	b. Perform the following: 1) Perform actions required by Technical Specifications. 2) **<u>IF</u>** unit shutdown is required, **<u>THEN</u>** perform 3-GOP-103, POWER OPERATION TO HOT STANDBY, or 3-GOP-100, FAST LOAD REDUCTION, while continuing with this procedure. 3) **<u>WHEN</u>** the unit is in MODE 3, **<u>THEN</u>** initiate cooldown to cold shutdown using 3–GOP-305, Hot Standby To Cold Shutdown.
13	**Notify The Shift Manager To Refer To 0-EPIP-20101, Duties Of Emergency Coordinator**	
14	**Request The Health Physics Shift Supervisor Perform The Following** • Conduct local area radiation surveys • Post radiation areas as required	
15	**Check If Leakage Isolated**	Return to Step 1.
16	**Verify Containment Instrument Air Bleed Valves, CV-3-2819 And CV-3-2826, <u>AND</u> Containment Sump Pump Discharge Valves, CV-3-2821 And CV-3-2822, Are Aligned As Determined By Unit Supervisor**	
17	**Go To Appropriate Plant Procedure As Determined By The Unit Supervisor**	

END OF TEXT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 1
(Page 1 of 4)

RHR LEAK ISOLATION

C A U T I O N

If High Temperature or High Velocity break hampers leak identification, stop the running RHR pumps to minimize break flow.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Check RHR System Components Between RCS (MOV-3-750/MOV-3-751 AND 3-752A, 3–752B) And 3A/3B RHR Pump Inlet Isolation Valves- NOT LEAKING	Perform the following: a. Stop both RHR pumps. b. Close the following valves: • MOV-3-750, LOOP 3C RHR Pump Suction Stop Vlv • MOV-3-751, LOOP 3C RHR Pump Suction Stop Vlv • 3-752A, 3A RHR Pump Inlet Isolation • 3-752B, 3B RHR Pump Inlet Isolation • MOV-3-862A, RHR Suction from RWST • MOV-3-862B, RHR Suction from RWST • 3-741A, RHR Cross Conn VLV for Hot & Cold Leg c. Go to 3-ONOP-041.8, SHUTDOWN LOCA (MODE 5 OR 6).
2	Check RHR System Components Bounded By The Following RHR Valves - NOT LEAKING • 3-752A, 3A RHR Pump Inlet Isolation • 3-759A, RHR HX A Outlet Valve • 3-757D, RHR HX A Bypass Hdr Isol	Perform the following: a. Verify RHR pumps STOPPED. b. Close the following valves: • 3-752A, 3A RHR Pump Inlet Isolation • 3-757D, RHR HX A Bypass HDR ISOL • 3-759A, RHR HX A Outlet Valve c. Go to Step 8 of this ATTACHMENT.
3	Check RHR System Components Bounded By The Following RHR Valves - NOT LEAKING • 3-752B, 3B RHR Pump Inlet Isolation • 3-759B, RHR HX B Outlet • 3-757C, RHR HX B Bypass Hdr Isol	Perform the following: a. Verify RHR pumps STOPPED. b. Close the following valves: • 3-752B, 3B RHR Pump Inlet Isolation • 3-757C, RHR HX B Bypass Hdr Isol • 3-759B, RHR HX B Outlet c. Go to Step 8 of this ATTACHMENT.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 1
(Page 2 of 4)

RHR LEAK ISOLATION

4	Check RHR System Components Between RCS And RHR DISCH To Cold Leg Isol Valves - NOT LEAKING • MOV-3-744A RHR DISCH To Cold Leg Isol Valve • MOV-3-744B RHR DISCH To Cold Leg Isol Valve	Perform the following: a. Stop both RHR pumps. b. Close the following valves: • MOV-3-744A • MOV-3-744B c. Go to Step 7 of this ATTACHMENT.
5	Check RHR System Components Bounded By The Following RHR Valves - NOT LEAKING • 3-757D, RHR HX A Bypass Hdr Isol • 3-759A, RHR HX A Outlet Vlv • 3-759B, RHR HX B Outlet Vlv • 3-757C, RHR HX B Bypass Hdr Isol • MOV-3-744A RHR Disch to Cold Leg Isol Valve • MOV-3-744B RHR Disch to Cold Leg Isol Valve	Perform the following: a. Stop both RHR pumps. b. Close RHR Disch To Cold Leg Isol valves: • MOV-3-744A • MOV-3-744B c. Close RHR HX A/B Outlet Vlvs: • 3-759A • 3-759B d. Close RHR HX A/B Bypass HDR Isol Valves: • 3-757D • 3-757C e. Go to Step 7 of this ATTACHMENT.
6	Return To Procedural Step In Effect	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 1
(Page 3 of 4)

RHR LEAK ISOLATION

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	Align RHR System For Alternate Cooldown Lineup As Follows	
	a. Close 3-887, RHR To RWST Supply Hdr Isol Vlv	
	b. Check both RHR pump discharge pressures - LESS THAN 210 PSIG • PI-3-600 • PI-3-601	b. Perform the following: 1) Verify RHR Suction From RWST valve - CLOSED • MOV-3-862A • MOV-3-862B 2) Open MOV-3-863A and MOV–3–863B locally. 3) Go to Step 7e of this ATTACHMENT.
	c. Close the following breakers • 30726 for MOV-3-863A • 30626 for MOV-3-863B	
	d. Open an RHR Alternate Disch Isol VLV * MOV-3-863A * MOV-3-863B	
	e. Open MOV-3-872, Alternate Low HEAD SI	
	f. Check RX Vessel Drain Down Level - GREATER THAN OR EQUAL TO 23% • LIS-3-6421 • LIS-3-6423	f. Perform the following: 1) **WHEN** reactor vessel level greater than **OR** equal to 23%, **THEN** return to Step 7g of this ATTACHMENT. 2) Continue with procedure Step 11.
	g. Start One RHR Pump * 3A RHR pump with MOV-3-863A - OPEN * 3B RHR pump with MOV-3-863B - OPEN	
	h. Maintain running RHR pump differential pressure - BETWEEN 105 AND 115 PSIG * Difference between PI-3-601 and PI-3-1595A for 3A RHR pump * Difference between PI-3-600 and PI-3-1595B for 3B RHR pump	h. Throttle RHR Pump A/B Isol valve to running RHR pump to obtain proper differential pressure: * 3-754A for 3A RHR pump * 3-754B for 3B RHR pump
	i. Control RCS cooldown rate locally by throttling CCW from operating RHR heat exchanger * 3-748A for 3A HX * 3-748B for 3B HX	
	j. Go to plant procedure as determined by Unit Supervisor	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 1
(Page 4 of 4)

RHR LEAK ISOLATION

8 Start Unisolated RHR Pump

a. Check Reactor Vessel Drain Down Level - GREATER THAN OR EQUAL TO 23%

- LIS-3-6421
- LIS-3-6423

RESPONSE NOT OBTAINED:

a. Perform the following:

1) **WHEN** reactor vessel level greater than **OR** equal to 23%, **THEN** return to Step 8b of this ATTACHMENT.

2) Continue with procedural step in effect.

b. Start unisolated RHR pump
* 3A RHR pump
* 3B RHR pump

c. Return FCV-3-605, RHR Heat Exchanger Bypass Flow, to automatic at desired flow

d. Open HCV-3-758, RHR Heat Exchanger Outlet Flow, as necessary to maintain desired RCS temperature

e. Return to procedural step in effect

FINAL PAGE

FOLDOUT FOR PROCEDURE 3-ONOP-041.3

1. **3-EOP-E-0 TRANSITION CRITERIA**

   IF Unit 3 is in Modes 1 through 3 greater than 1000 psig with the Safety Injection System aligned for injection **AND** either of the following occurs, **THEN** verify the Reactor Tripped **AND** go to 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION:

   a. RCS leakage greater than charging pump capacity and letdown isolated.

   b. PRZ level – CAN **NOT** BE MAINTAINED GREATER THAN 7%[48%].

Florida Power & Light Company

Turkey Point Nuclear Plant

This procedure may be affected by a T.C. (Temporary Change) Verify information prior to use.
Date verified today *Initials*

Unit 3

3-OP-047

CAUTION
Performance of this procedure may affect core reactivity.

Title:

CVCS – Charging and Letdown

(Continuous Use)

Safety Related Procedure	
Responsible Department:	Operations
Revision Number:	15B
Revision Approval Date:	2/25/16

PCRs *08-4488, 09-0553, 09-2331, 08-4426, 09-3329, 09-3140, 09-4014, 1602148, 1614029, 1691391, 1644831, 1768933, 1781767, 1736925, 1832392, 1821744, 1855113, 1801385, 1873103, 1933915, 1942209, 1934064, 1940472, 2007051, 2015832, 2023064, 2025918 2030924, 2067359, 1983171, 1956291, 2090289, 2112643*
ECs *85-137, 87-258, 88-527, 88-605 , 89-209, 89-224, 89-581, 90-055, 90-440, 91-037, 92-031, 94-006, 94-012, 94-138, 97-039, 02-053, 02-057, 03-001, 03-132, 04-009, 04-047, 06-103, 270716, 249693, 280399*

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26	03/07/14	61	02/04/15	96	02/04/15	130	02/04/15
27	03/07/14	62	02/04/15	97	02/04/15	131	02/04/15
28	03/07/14	63	02/04/15	98	02/04/15	132	02/04/15
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1.0 PURPOSE

1.1 This procedure provides instructional guidance for the operation of the Chemical and Volume Control System (CVCS) for Charging and Letdown of the RCS.

2.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

2.1 References

2.1.1 Technical Specifications

1. Section 3/4.1.2, Boration Systems
2. Section 3.6, Containment Integrity (PC/M 89-581)

2.1.2 FSAR

1. Section 9.2, Chemical and Volume Control

2.1.3 Regulatory Guide 1.33, Quality Assurance Program Requirements

2.1.4 Operating Diagrams

1. 5613-M-3033, Sh 1, Spent Fuel Pit Cooling System
2. 5613-M-3036, Sh 1, Sample System-NSSS, Sample Room
3. 5613-M-3041, Sh 1, Reactor Coolant System-Loops
4. 5613-M-3041, Sh 3, Reactor Coolant System-Reactor Coolant Pumps
5. 5610-M-3046, Sh 1, CVCS-Boric Acid System
6. 5613-M-3047, Sh 1, CVCS Charging and Letdown
7. 5613-M-3047, Sh 2, CVCS Charging and Letdown
8. 5613-M-3047, Sh 3, CVCS Seal Water Injection to RCPs
9. 5610-M-3061, Sh 13, Waste Disposal System-Gas, Waste Decay
10. 5610-M-3094, Sh 1, Containment Post Accident Evaluation Systems
11. 5610-M-3094, Sh 2, Containment Post Accident Evaluation Systems
12. 5613-M-3061, Sh 14, Waste Disposal System Gas-Waste Analyzer
13. 5613-M-3061, Sh 1, Waste Disposal System, Liquid Waste Holdup and Transfer
14. 5613-M-3020, Sh 1, Primary Water Makeup System
15. 5613-M-3020, Sh 2, Primary Water Makeup System
16. 5610-E-855, Breaker List

2.1.5 Operating Procedures

1. 0-ADM-215, Plant Surveillance Tracking Program
2. 3-GOP-503, Cold Shutdown to Hot Standby
3. 3-ONOP-047.1, Loss of Charging Flow in Modes 1 Through 4
4. 3-NOP-041.01A, 3A, Reactor Coolant Pump Normal Operations
5. 3-NOP-041.01B, 3B, Reactor Coolant Pump Normal Operations
6. 3-NOP-041.01C, 3C, Reactor Coolant Pump Normal Operations
7. 3-NOP-041.2, Pressurizer Operation
8. 3-NOP-300, Alternate Shutdown Panel
9. 3-OP-041.8, Filling and Venting the RCS
10. 0-OP-046, CVCS, Boron Concentration Control
11. 3-OP-047.1, VCT Gas Space Concentration Control
12. 3-OP-047.3, CVCS - Demineralizer Operations
13. 0-OSP-200.1, Schedule of Plant Checks and Surveillances
14. 3-OSP-300.2, Alternate Shutdown Panel 3C264 Switch and Instrument Alignment Check

2.1.6 Vendor/Technical Manuals

1. Union Pump Company Instruction Manual, No. 5610-M-420-170-1

2.1.7 Miscellaneous Documents (i.e., PC/M, ECs, Correspondence)

1. MOS Daily Report of 5/13/88-D, Item D-2 (CTRAC 88-1049)
2. JQT-89-319, Observations - Review of Chemical and Volume Control System (CVCS)
3. PC/M 85-137, Appendix R Valve Hand Operator Addition - Unit 3
4. PC/M 87-258, Load Center 3H and Repowering of MCCD (3D)
5. PC/M 88-527, Resolution of Drawing Changes Associated with 5613-M-3041, Sheet 3, Reactor Coolant System, Reactor Coolant Pump
6. PC/M 88-605, Drawing 5613-M-3047, Sheet 1, CVCS - Charging and Letdown System
7. PC/M 89-209, Quick Connects Installed on Stabilizer Vents for Charging Pumps of the Chemical and Volume Control System
8. PC/M 89-224, Drawing Update - Chemical and Volume Control System
9. PC/M 89-581, Containment Isolation Features Design Basis Implementation
10. PC/M 90-055, MOD, Charging Pumps Discharge Drains and Relief Vlv Piping Supports
11. PC/M 91-037, Modification to LT-3-112 and LT-3-115 Instrument Loops
12. PC/M 92-031, Units 3/4 Annunciator Window Upgrading
13. PC/M 94-006, Charging Pump Suction Stabilizer Vent Modification
14. PC/M 94-138, Charging Pump Task Team Modification
15. PC/M 97-039, Plant Reliability Improvement Modification (C-Bus)
16. JPN-PTN-SEMS-96-014, Rev. 3, A Test of the Use of Sub-Micron Ultrafine Filters in the CVCS and SFP
17. PC/M 03-001, Installation of Maintenance Valve, 3-1341
18. CR 03-4281, Lifting of RV-4-304 When Placing Excess Letdown in Service
19. PC/M 04-047, Unit 3 Charging Pump Permanent Drains
20. PC/M 03-132, Installation of New Unit 3 CVCS Letdown Relief Valve
21. PC/M 04-009, Replacement of RV-3-311 Relief Valve with CV-3-310A Bypass Line
22. PC/M 06-103, Component Cooling Water, TCV-3-144, Replacement
23. CR 2009-23599, Charging Pump Post Maintenance Venting

2.1.7 (Cont'd)

24. EC 270716, Addition of High Point Vents for the Unit 3 CVCS Boration Headers
25. CR 01929273, Inconsistencies with TP-10-001 for RCP Low Leak Off Flow
26. CR 1926106, Charging PP Tripped When Speed Was Taken to 0% On HIC
27. EC 280399, U3 - RCP Seals Upgrade Project

2.2 Records Required

2.2.1 The date, time, and section completed shall be entered in the Unit Narrative Log. Also, problems encountered while performing the procedure should be entered; i.e., malfunctioning equipment, delays due to changes in plant conditions, etc.

2.2.2 Completed copies of the QA Record Pages for the below listed items constitute Quality Assurance Records and shall be transmitted to QA Records for retention in accordance with Quality Assurance Records Program requirements:

1. Subsections 7.1, 7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9, 7.10, 7.14, 7.16, 7.17, 7.18, and 7.20
2. Attachment 1
3. Attachment 2
4. Attachment 5

2.2.3 Completed copies of the below listed items shall be retained in the Shift Manager's file until the next performance of that section, enclosure, or attachment:

1. Subsections 7.6 and 7.14
2. Attachment 1
3. Attachment 2
4. Attachment 5

2.2.4 Completed attachments listed below, that have the TAG column checked (√), shall be copied and transmitted to the Labeling Coordinator:

1. Attachment 1

2.3 Commitment Documents

2.3.1 JPN-PTN-SENP-95-026, Safety Evaluation for CCW Flow Balance and Post Accident Alignment Requirements to Support Current and Uprated Conditions (LER 250/95-006)

3.0 PREREQUISITES

3.1 The following systems are operable or in operation as required to support the CVCS - Charging and Letdown System operation:

3.1.1 3-NOP-013, Instrument Air System

3.1.2 3-NOP-020, Primary Water System

3.1.3 3-NOP-030, Component Cooling Water System

3.1.4 0-NOP-065.01, Hydrogen Gas Supply System

3.1.5 0-NOP-065.03, Nitrogen Gas System

3.1.6 0-OP-046, Boric Acid System

3.1.7 0-OP-061.15A, Waste Gas Compressors

3.1.8 0-OP-061.15B, Waste Gas Decay Tanks

3.2 All plant electrical systems are operable to supply power and control functions to support CVCS operation.

3.3 All instruments and control devices are in service for the CVCS - Charging and Letdown System operation with no surveillances required and no outstanding PWOs, clearances, or Temporary System Alterations that affect system operability as per the following:

3.3.1 0-ADM-215, Plant Surveillance Tracking Program

3.3.2 0-OSP-200.1, Schedule of Plant Checks and Surveillances (No surveillances have exceeded the date required on the missed surveillance sheet.)

3.3.3 Missed Surveillance Sheet

3.3.4 Temporary System Alteration (TSA) Log

3.3.5 Clearance Log

3.3.6 Out-of-Service Log

3.4 The CVCS - Charging and Letdown valve and breaker alignments have been verified by the completion of the following attachments:

3.4.1 Attachment 1

3.4.2 Attachment 2

3.5 The Alternate Shutdown Panel Alignment has been verified by satisfactory completion of 3-OSP-300.2, Alternate Shutdown Panel 3C264 Switch and Instrument Alignment Check, for equipment listed in this procedure.

4.0 PRECAUTIONS/LIMITATIONS

4.1 Before changing system status, Technical Specifications should be consulted for system requirements for that plant mode.

4.2 Design restrictions on demineralizer operation require the letdown flow rate to be maintained below 120 gpm and the temperature of the water entering the inlet header to be less than 140 °F.

4.3 Explosive mixtures of hydrogen and oxygen concentration shall be avoided at all times. The oxygen concentration in the VCT shall be maintained less than or equal to 2 percent by volume when hydrogen is greater than 4 percent.

4.4 The CVCS Demineralizers are required to be bypassed prior to adding hydrazine to the CVCS **EXCEPT** a demineralizer with PRC-01.

4.5 All work performed in the Radiation Controlled Area shall be performed in accordance with the requirements of the Radiation Work Permit and ALARA program.

4.6 When aligning remotely operated valves (i.e., chain operated, reach rods, etc.), the position shall be verified by local valve position. This requirement may be waived by the Shift Manager in cases of significant radiation exposure, which occur in areas designated as high radiation areas or in areas deemed inaccessible by the Shift Manager.

4.7 Letdown flow should be maintained through the CVCS Demineralizers to maintain system cleanliness. Securing letdown during plant cooldown may result in high dose rates in the RHR System. The RP Supervisor and the Radiochemist shall be notified if letdown is to be secured.

4.7.1 Letdown orifices should not be changed during delithiation operations. If letdown flow has to be changed, then Chemistry should be notified so that the delithiation bed run time can be recalculated.

4.8 If a charging pump exhibits primary packing leakage symptoms as described below, then issue a PWO to Mechanical Maintenance Department to repack the pump.

4.8.1 Primary packing leakage of greater than 0.05 gpm: place on Plant Status Sheet and repack within 4 weeks.

4.8.2 Primary packing leakage of greater than 0.08 gpm: place on Plant Status Sheet and repack within 2 weeks.

4.8.3 Abnormally high airborne gas concentration in the Charging Pump Room.

4.9 If a charging pump exhibits secondary packing leakage symptoms as described below, then issue a PWO to Mechanical Maintenance Department to repack the pump.

4.9.1 Decreasing seal pot level that requires shiftly seal pot fills.

4.9.2 A steady stream of water leaking out any one of the plungers in the charging pump plunger well.

4.10 Temperature changes of the CVCS letdown will affect the ability of the in-service resin bed to retain boric acid. A temperature increase will cause a minor boration and a temperature decrease will cause a minor dilution. Reactor power should be closely monitored when changing letdown temperatures or changing resin beds.

4.11 The differential pressure across the RCS filters should be monitored for approximately 8 hours after placing a new demineralizer in service due to the possibility of demineralizer resin fine (resin dust present in new resin) carry over.

4.12 Ultrafine filter design limit is 75 psid. Installed monitoring instrumentation limit is 50 psid. Seal Water Injection Filter alarm setpoint is 20 psid. Filter DPs will increase at elevated rates as a filter clogs. To allow sufficient design margin, CVCS filters should be replaced at less than 30 psid or prior to the 20 psid SWI alarm actuation or should be replaced based on projected filter does rates.

4.13 To prevent exceeding filter design limits, filter DP changes should be anticipated prior to changing letdown flowrates. Filter DPs will change as a flowrate ratio square function (i.e., $\Delta P_2 = \Delta P_1 x(\text{flowrate}_2/\text{flowrate}_1)^2$). RCP seal water injection flowrates are expected to more than double when letdown flowrates are increased greater than 100 gpm.

4.14 Increasing reactor power above 100 percent caused by a reduction in boron concentration (for example, due to placing a demineralizer in service) shall be turned and reduced below 100 percent by control rod insertion.

4.15 CCW flow to the Excess Letdown Heat Exchanger must be maintained below 238 gpm as indicated on FI-3-624 (located in the pipe and valve room) to prevent excessive vibration in the heat exchanger.

4.16 A second individual shall be assigned as communicator during venting or draining of CVCS filters. This ensures continuous monitoring of radioactive systems to prevent spills and subsequent contamination.

4.17 Just prior to and during refueling outages, filter sizes of the RCS, Seal Water Injection and Seal Water Return filters should be increased to 2 or 6 microns.

4.18 The Letdown Orifice Valves, CV-3-200A, B and C, include a seal-in valve stroke circuit. When opening a valve, the stem must travel in the open direction in order to release the closed limit switch. At this point, the seal-in is made-up, the valve exhibits dual indication, and the interlock relay changes state causing a click noise. Due to the high dP across the valve, full flow can be established prior to the seal-in circuit being made-up and prior to the click noise. Therefore, the operator should observe letdown flow and pressure to determine when flow is established.

4.19 The Letdown Pressure Control Valve PCV-3-145 Manual/Auto station controls upstream letdown pressure.

4.19.1 An increase in demand increases the pressure by closing the valve. A decrease in demand decreases the pressure by opening the valve.

4.19.2 When increasing letdown pressure using PCV-3-145, the operator should use the **bump-and-wait** technique to prevent overshooting the desired pressure.

4.20 The expected pressure change as a result of opening or closing an orifice stop valve is proportional to the square of the change in flow. Operator action is required to mitigate the expected transient.

4.21 CVCS Demin inlet pressure should be maintained less than 150 psig by placing a clean standby RCS filter inservice OR reducing the letdown flowrate OR by reducing VCT pressure during RCS crud burst activities.

4.22 With HCV-3-121, Charging Flow to Regen Hx Open, a flowpath to RCS Loop A exists through 3-385, CV-310A Bypass Isolation Valve even with Loop Charging Isolation valves CV-3-310A and CV-3-310B Closed. This bypass isolation valve is normally Locked Open to provided thermal relief protection for the Regenerative Heat Exchanger and associated piping.

4.23 The following precautions should be observed regarding operation of HCV-3-121:

4.23.1 HCV-3-121 should only be throttled if acceptable RCP seal injection flows can not be maintained.

4.23.2 When throttling HCV-3-121, charging discharge pressure should be monitored to prevent lifting a charging pump discharge relief valve.

4.23.3 To prevent potentially lifting a charging pump discharge relief valve, HCV-3-121 should be Open prior to starting additional charging pumps for increased charging flowrates.

4.23.4 Care must be exercised when throttling HCV-3-121 in the Closed direction. Throttling this valve completely Closed can cause the Charging Pump discharge relief valve to lift resulting in a possible loss of charging if the relief valve fails to reseat.

4.23.5 If the charging pump discharge valve is lifted while throttling HCV-3-121, Engineering should be contacted for evaluation.

4.24 Motor starting duty limits are:

4.24.1 With motor at ambient temperature, two successive starts are allowed (the motor must coast to rest between starts).

4.24.2 With motor at operating temperature, one start is allowed. Subsequent starts require that the motor is allowed to cool by standing idle for one hour or by running for one-half hour.

4.25 Boration headers to the charging pump suction header (i.e., MOV-3-350, FCV-3-113B, and 3-356 headers) require dynamic venting following maintenance activities that drain the charging pump suction header.

4.26 Letdown Temperature Controller, TC-3-144A, Manual Auto station controls letdown temperature out of the Non-Regenerative Heat Exchanger by adjusting the position of TCV-3-144, Temp Control Vlv for CCW from Non-Regen Hx Outlet.

4.26.1 In Auto, which is the normal mode of operation, the position of TCV-3-144 is automatically adjusted to maintain the letdown temperature at 116°F to 125°F (nominal value is 118°F based on a potentiometer setting of 4.53).

4.26.2 In Manual, an increase in demand increases letdown temperature by closing TCV-3-144. A decrease in demand results in a decrease in letdown temperature by opening TCV-3-144.

4.27 If a charging pump has primary packing leakage greater than 0.05 gpm, charging pump primary packing leakage shall be measured during all RCS leak rate determinations.

INIT Date/Time Started: ____________ / ____________

7.12 Guidance for Placing Excess Letdown in Service

7.12.1 Initial Conditions

____ 1. All applicable prerequisites listed in Section 3.0 are satisfied.

7.12.2 Procedure Steps

____ 1. Verify Excess Ltdn Hx CCW Outlet, CV-3-739, is Open.

____ 2. Verify greater than 200 gpm and less than or equal to 238 gpm CCW flow on flow indicator FI-3-624 (located in the Pipe and Valve Room).

____ 3. Verify Excess Ltdn Iso Valve, CV-3-387, is Closed.

____ 4. Verify Excess Ltdn Divert to WDS, CV-3-389, is aligned to the VCT (Switch to Normal).

____ 5. Slowly Open Excess Letdown Flow Controller, HCV-3-137, to allow excess letdown lines to backfill.

____ 6. **WHEN** a minimum of 5 minutes have elapsed, **THEN** Close Excess Letdown Flow Controller, HCV-3-137.

____ 7. Open Excess Ltdn Isol Valve, CV-3-387 **AND** observe Containment Sump level for indication that RV-3-304 may have lifted.

CAUTION

If Excess Letdown Heat Exchanger outlet temperature exceeds 195°F, then VCT may have an excessive heatup rate.

____ 8. Slowly Open Excess Letdown Flow Controller, HCV-3-137, allowing the heat exchanger to warmup.

____ 9. Monitor heat exchanger outlet temperature at Excess Ltdn Hx Temp Indicator, TI-3-139.

____ 10. **IF** VCT Divert to Hold-up Tk, LCV-3-115A, reaches the 100 percent Divert position (red light On, green light Off) **OR** if desired to direct water to the RCDT, **THEN** align Excess Ltdn Divert to WDS, CV-3-389, to the RCDT (switch to Divert).

____ 11. Enter completion of this procedure subsection in the Unit Narrative Log.

L-16-1 NRC Exam

Control Room - JPM C

JPM TITLE: Establish Auxiliary Pressurizer Spray

JPM NUMBER: 01041052100 **REV.** 0-0

TASK NUMBER(S) / TASK TITLE(S): 01041052100 / Initiate Pressurizer Auxiliary Spray

K/A NUMBERS: EPE 038 EA1.04 **K/A VALUE:** RO 4.3 / SRO 4.1

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]

Simulator: [X] Other: [ ]

Lab: [ ]

Time for Completion: 20 Minutes Time Critical: No

Alternate Path [NRC]: Yes

Alternate Path [INPO]: Yes

Developed by:	**Brian Clark** Instructor/Developer	**6/20/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/21/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER SUPERVISOR	DATE DATE
0-0	New JPM	L-16-1 NRC Exam	N/A	N/A N/A	N/A N/A

SIMULATOR SET-UP:

SIMULATOR SETUP INSTRUCTIONS:

______	1.	Reset to **IC 180** or saved IC.
______	2.	Place simulator in RUN.
______	3.	Ensure applicable portions of Simulator Operator Checklist are complete.
______	4.	*N/A if using saved IC* Open and execute L-16-1 NRC JPM C.lsn • Verify **PORVs FAILED CLOSE** auto triggers • Verify both **WINDUP RESET** triggers are in CONDITION state
______	5.	Allow plant to stabilize.
______	6.	Acknowledge alarms and place simulator in FREEZE.
______	7.	Save as temporary IC, if JPM will be repeated.
______	8.	When ready to begin, then place Simulator in RUN.

SIMULATOR MALFUNCTIONS:

- TFHV456C: PORVs failed closed
- TFHV55CC: PORVs failed closed

SIMULATOR OVERRIDES:

- N/A

SIMULATOR REMOTE FUNCTIONS:

- N/A

Required Materials:	• Handout 3-EOP-E-3
General References:	• 3-EOP-E-3, Steam Generator Tube Rupture
Task Standards:	• During RCS depressurization, recognize that the PORVs are NOT functional and, alternatively, establish auxiliary spray

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 3 tripped from full power, due to a tube failure in the 3A Steam Generator.
- After the trip, a LOOP occurred and both EDGs started and loaded onto their respective bus.
- The crew is performing 3-EOP-E-3, Steam Generator Tube Rupture.
- The 3A Steam Generator is isolated.
- The RCS cooldown has been completed.

INITIATING CUE:

- The Unit Supervisor directs you to perform Step 18 of 3-EOP-E-3 to depressurize the RCS.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical : No**	**Obtain required reference materials.**
Standard:	Obtain 3-EOP-E-3, Steam Generator Tube Rupture.
Evaluator Note:	If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue.
Evaluator Cue:	Provide examinee with a copy of handout 3-EOP-E-3.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 2 **Critical : No**	**3-EOP-E-3, prior to Step 18:** **CAUTION** • ***If a PRZ PORV is used to depressurize the RCS, the PRT rupture disk may rupture. This may result in abnormal Containment conditions.*** • ***Cycling of the PRZ PORV shall be minimized.*** **NOTE** ***If RCPs are NOT running, the upper head region may void during RCS depressurization. This will result in a rapidly increasing PRZ level.***
Standard:	Read CAUTION/NOTE and determine it is satisfactory to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical : No**	**3-EOP-E-3, Step 18:** **Depressurize RCS Using PRZ PORV To Minimize Break Flow And Refill PRZ** a. Check PRZ PORV – AT LEAST ONE AVAILABLE
Standard:	Check the pressurizer PORV light indications and recognize that both are available.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

THIS BEGINS THE ALTERNATE-PATH PORTION OF THE JPM

Performance Step: 4 **Critical : No**	**3-EOP-E-3, Step 18:** **Depressurize RCS Using PRZ PORV To Minimize Break Flow And Refill PRZ** b. Open one PRZ PORV until any of the following conditions satisfied…**(NO)** → (RNO) IF **NO** PORV can be opened, THEN establish Auxiliary Spray using Attachment 4 and return to Step 17.b.
Standard:	Recognize that neither PORV will open and transition to Attachment 4.
Evaluator Cue:	If examinee asks which PORV to manipulate, respond as the Unit Supervisor and state, "Use your own judgment."
Evaluator Note:	When examinee takes either PORV's handswitch to OPEN, the valve will NOT respond (i.e., it will remain closed).
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical : Yes**	**3-EOP-E-3, Attachment 4, Step 1:** **Verify Pressurizer Spray valves – OPEN** • PCV-3-455A, Loop C • PCV-3-455B, Loop B
Standard:	Open PCV-3-455B and PCV-3-455A, Pressurizer Spray Control Valves,
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical : No**	**3-EOP-E-3, Attachment 4, Step 2:** **Verify Aux Spray TI-3-123 And PRZ Temperature TI-3-454 Temperature Difference – LESS THAN 320°F**
Standard:	Verify that the difference between the indications on TI-3-123, RHX Outlet Temperature Indicator, and TI-3-454, Pressurizer Steam Space Temperature Indicator, are within 320°F.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical : Yes**	**3-EOP-E-3, Attachment 4, Step 3:** **Open CV-3-311, Aux Spray Isolation**
Standard:	Take the handswitch for CV-3-311, Auxiliary Spray Control Valve, to OPEN and verify that the red indicating light is lit and the green indicating light is extinguished.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical : Yes**	**3-EOP-E-3, Attachment 4, Step 4:** **Close CV-3-310A, Loop A Charging Isolation**
Standard:	Take the handswitch for CV-3-310A, Charging to RCS Loop A Control Valve, to CLOSE and verify that the green indicating light is lit and the red indicating light is extinguished.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical : No**	**3-EOP-E-3, Attachment 4, Step 5:** **Verify CV-3-310B, Loop C Charging Isolation, is CLOSED**
Standard:	Recognize that CV-3-310B, Charging to RCS Loop B Control Valve, is closed (i.e., the green indicating light is lit and the red indicating light is NOT lit).
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 10 **Critical : Yes**	**3-EOP-E-3, Attachment 4, Step 6:** **Control Aux Spray as Follows:** * Increase Auxiliary Spray flow by closing PCV-3-455A, Pressurizer Loop C, and/or PCV-3-455B, Pressurizer Loop B * Reduce Auxiliary Spray flow by opening PCV-3-455A, Pressurizer Loop C, and/or PCV-3-455B, Pressurizer Spray Loop B
Standard:	Close PCV-3-455A/B as needed to increase spray flow and reduce pressurizer pressure.
Evaluator Note:	Only reducing pressurizer pressure is critical.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When RCS pressure is observed to be lowering, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ____________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 3 tripped from full power, due to a tube failure in the 3A Steam Generator.
- After the trip, a LOOP occurred and both EDGs started and loaded onto their respective bus.
- The crew is performing 3-EOP-E-3, Steam Generator Tube Rupture.
- The 3A Steam Generator is isolated.
- The RCS cooldown has been completed.

INITIATING CUE:

- The Unit Supervisor directs you to perform Step 18 of 3-EOP-E-3 to depressurize the RCS.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

FPL

TURKEY POINT UNIT 3

EMERGENCY OPERATING PROCEDURE

SAFETY RELATED

CONTINUOUS USE

Procedure No.

3-EOP-E-3

Revision No.

9

Title:

STEAM GENERATOR TUBE RUPTURE

Responsible Department: OPERATIONS

Special Considerations:

Last page of this procedure contains fold out page

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED today INITIAL ga

Revision	Approved By	Approval Date
7	Mike Murphy	07/30/14
9	Rich Tucker	10/26/15

UNIT #	UNIT 3
DATE	
DOCT	PROCEDURE
DOCN	3-EOP-E-3
SYS	
STATUS	COMPLETED
REV	9
# OF PGS	

REVISION SUMMARY	
Rev. No.	**Description**
9	**PCR 2006542, 10/26/15, Terry White** Revised to address RCP Seal replacement per **EC 280399**. Specific changes include: ● #1 Seal ΔP replaced by RCS pressure ● #1 Seal Leak-Off flow replaced with Control Bleed Off (CBO) ● Revised RCP Start criteria based on manufacturer recommendations Corrected formatting errors from previous revision to Attachment 5.
8	**PCR 2026972, 03/05/15, Luis H. Jimenez** Revised Attachment 5 to address **EC 282865** Changes. EC 282865 removed valve 3-10-060. Attachment 5 was revised to remove instructions for aligning Main Steam Auxiliaries to Unit 1 and 2.

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ATTACHMENTS (continued)

FOLDOUT

1.0 PURPOSE

1. This procedure provides actions to terminate leakage of reactor coolant into the secondary system following a Steam Generator Tube Rupture.

2.0 SYMPTOMS OR ENTRY CONDITIONS

1. This procedure is entered from:

a. E-0, REACTOR TRIP OR SAFETY INJECTION diagnostic steps, Step 14, and there are indications of a Steam Generator Tube Rupture.

b. E-0, REACTOR TRIP OR SAFETY INJECTION, Step 18,
E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 4,
E-2, FAULTED STEAM GENERATOR ISOLATION, Step 7,
ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, Step 5, and
FR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL, Step 8,
when Secondary Radiation is abnormal.

c. E-0, REACTOR TRIP OR SAFETY INJECTION, Step 19,
E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 3,
ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 4,
ES-3.1, POST-SGTR COOLDOWN USING BACKFILL, Step 5,
ES-3.2, POST-SGTR COOLDOWN USING BLOWDOWN, Step 5,
ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP, Step 5,
ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED, Step 12,
ECA-3.2, SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED, Step 5, and
ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL, Step 6,
when a S/G Narrow Range Level increases in an uncontrolled manner.

d. ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL, Steps 2, 3, 4, 6, and 22 when Pressurizer Pressure Control is restored.

e. E-1, LOSS OF REACTOR OR SECONDARY COOLANT, foldout item 5,
ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, foldout item 5, and
ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, foldout item 4, whenever any S/G Level increases in an uncontrolled manner OR any S/G has abnormal radiation.

2.0 SYMPTOMS OR ENTRY CONDITIONS (continued)

1. (continued)

f. ES-3.1, POST-SGTR COOLDOWN USING BACKFILL, foldout item 7,
ES-3.2, POST-SGTR COOLDOWN USING BLOWDOWN, foldout item 7,
ES 3.3, POST SGTR COOLDOWN USING STEAM DUMP, foldout item 7,
when after identification of a ruptured S/G, any intact S/G Level increases in an uncontrolled manner OR any intact S/G has abnormal radiation.

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

CAUTION

- If a PRZ PORV is used to depressurize the RCS, the PRT rupture disk may rupture. This may result in abnormal Containment conditions.
- Cycling of the PRZ PORV shall be minimized.

NOTE

If RCPs are **NOT** running, the upper head region may void during RCS depressurization. This will result in a rapidly increasing PRZ level.

18. Depressurize RCS Using PRZ PORV To Minimize Break Flow And Refill PRZ

a. Check PRZ PORV – AT LEAST ONE AVAILABLE

Response Not Obtained:

a. Establish Auxiliary Spray using Attachment 4 and return to Step 17.b.

1) IF Auxiliary Spray can **NOT** be established, THEN continue to disregard any false Integrity Status Tree indication caused by ruptured loop T-cold, and go to 3-EOP-ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL, Step 1.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18.	**(continued)**	
	b. Open <u>one</u> PRZ PORV <u>until</u> any of the following conditions satisfied using Attachment 6, as reference: * <u>Both</u> of the following • RCS pressure – LESS THAN RUPTURED S/G(s) PRESSURE • PRZ level – GREATER THAN 7%[48%] <u>OR</u> * PRZ level – GREATER THAN 73%[60%] <u>OR</u> * RCS Subcooling based on Core Exit TCs – LESS THAN 19°F[73°F]	**b.** <u>IF</u> **NO** PORV can be opened, <u>THEN</u> establish Auxiliary Spray using Attachment 4 and return to Step 17.b. **1)** <u>IF</u> Auxiliary Spray can **NOT** be established, <u>THEN</u> continue to disregard any false Integrity Status Tree indication caused by ruptured loop T-cold, and go to 3-EOP-ECA-3.3, SGTR WITHOUT PRESSURIZER PRESSURE CONTROL, Step 1.
	c. Stop depressurization by closing PRZ PORV	**c.** Close PORV Block Valve.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.	**Check RCS Pressure – INCREASING**	Close PRZ PORV Block Valve.

IF pressure continues to decrease, THEN perform the following:

a. Monitor following conditions for indication of leakage from PRZ PORV:

1) PRZ Relief Line temperature, TI-3-463

2) PRZ Relief Tank level, LI-3-470

3) PRZ Relief Tank temperature, TI-3-471

4) PRZ Relief Tank pressure, PI-3-472

b. Go to 3-EOP-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED, Step 1.

ATTACHMENT 1
Natural Circulation Indications
(Page 1 of 1)

The following conditions support or indicate Natural Circulation flow:

- RCS Subcooling based on Core Exit TCs – GREATER THAN 19°F[73°F]
- S/G pressures – STABLE OR DECREASING
- RCS Hot Leg temperatures – STABLE OR DECREASING
- Core Exit TCs – STABLE OR DECREASING
- RCS Cold Leg temperatures – WITHIN 30°F OF SATURATION TEMPERATURE FOR INTACT S/G PRESSURE

End of Attachment 1

ATTACHMENT 2

Control of RCS Pressure and Charging Flow to Minimize RCS-to-Secondary Leakage

(Page 1 of 1)

NOTE

When RCS depressurization is required, Normal Spray should be used whenever possible.
If Normal Spray is **NOT** available AND Letdown is in service, Auxiliary Spray should be used (Refer to Attachment 4).
If Normal Spray AND Auxiliary Spray are **NOT** available, one PRZ PORV should be used.

PRZ LEVEL	RUPTURED S/G(S) NARROW RANGE LEVEL		
	INCREASING	**DECREASING**	**OFF SCALE HIGH**
LESS THAN 26%[50%]	• Increase Charging Flow • Depressurize RCS Refer to note above	Increase Charging Flow	• Increase Charging Flow • Maintain RCS And Ruptured S/G(s) Pressures Equal
BETWEEN 26%[50%] and 50%[55%]	• Depressurize RCS Refer to note above	Turn On PRZ Heaters	Maintain RCS And Ruptured S/G(s) Pressures Equal
BETWEEN 50%[55%] and 73%[60%]	• Depressurize RCS Refer to note above • Decrease Charging Flow	Turn On PRZ Heaters	• Maintain RCS And Ruptured S/G(s) Pressures Equal • Decrease Charging Flow
GREATER THAN 73%[60%]	Decrease Charging Flow	Turn On PRZ Heaters	• Maintain RCS And Ruptured S/G(s) Pressures Equal • Decrease Charging Flow

End of Attachment 2

ATTACHMENT 3
Unit 3 Component KW Load Rating Chart
(Page 1 of 2)

CAUTION

Steady state loading on each Unit 3 Emergency Diesel Generator shall **NOT** exceed 2500 KW. When starting additional equipment, diesel load is required to be monitored to ensure the transient limit of 2750 KW is **NOT** exceeded.

NOTE

- One Computer Room Chiller is required to be restarted within 60 minutes of Loss Of Offsite Power to maintain operability of DCS and QSPDS.
- Battery Charger load is dependent on the status of its parallel charger (i.e., in service or de-energized).

ESSENTIAL LOADS

COMPONENT	KW	COMPONENT	KW
CCW PUMP	380	BATTERY CHARGER 3B1	20/39
HIGH-HEAD SI PUMP	302	BATTERY CHARGER 4A2	20/39
INTAKE COOLING WATER PUMP	265	EMERGENCY LIGHTING	18
RHR PUMP	222	INSTRUMENT AIR DRYER	18
CONTAINMENT SPRAY PUMP	212	DG AUXILIARY EQUIPMENT	17
ED FIRE PUMP (P39)	203	SWITCHGEAR/LC 3A A/C AHU	17
NORMAL CONTAINMENT COOLER	77	SWITCHGEAR/LC 3B A/C AHU	17
CRDM COOLER FAN	48	DG AIR COMPRESSOR	13
COMPUTER ROOM CHILLER	43	EDG RM LIGHTING PANEL 3X87	11
AUXILIARY BLDG EXHAUST FAN	33	AUXILIARY BLDG SUPPLY FAN	9
BATTERY ROOM A/C	30	H2 ANALYZER HEAT TRACE	8
BATTERY CHARGER 3A1	29/56	CABLE SPREADING ROOM A/C	5
BATTERY CHARGER 4B2	29/56	DG VENT FAN	5
CONTROL ROOM A/C COMPR	27	PAGE SYSTEM	5
SWITCHGEAR/LC 3A A/C CHILLER	26	CONTROL ROOM FILTER FAN	3
SWITCHGEAR/LC 3B A/C CHILLER	26	COMPUTER ROOM AIR UNIT	3
ELECTRICAL EQUIP RM A/C	25	SWITCHGEAR 3D SUPPLY FAN	2
EMERGENCY CNTMT COOLER	23	DG FUEL OIL TRANSFER PUMP	1
		H2 ANALYZER PUMP	1

ATTACHMENT 3

Unit 3 Component KW Load Rating Chart

(Page 2 of 2)

NON-ESSENTIAL LOADS

COMPONENT	**KW**	**COMPONENT**	**KW**
TPCW PUMP	299	TURNING GEAR LUBE OIL PUMP	33
CHARGING PUMP	114	BEARING LIFT OIL PUMP	28
SPENT FUEL PIT PUMP	82	AIR SIDE SEAL OIL PUMP	21
PRESSURIZER HEATER (EACH)	50	BORIC ACID TRANSFER PUMP	13
TURNING GEAR DRIVE	41	HYDROGEN SIDE SEAL OIL PUMP	3

End of Attachment 3

ATTACHMENT 4
Establish Auxiliary Pressurizer Spray
(Page 1 of 2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Verify Pressurizer Spray valves – OPEN** • PCV-3-455A, Loop C • PCV-3-455B, Loop B	
2.	**Verify Aux Spray TI-3-123 And PRZ Temperature TI-3-454 Temperature Difference – LESS THAN 320°F**	Perform the following: **a.** Record total time duration that Aux Spray is in service with temperature difference greater than or equal 320°F. **b.** Notify Engineering to perform engineering evaluation required by Technical Specifications
3.	**Open CV-3-311, Aux Spray Isolation**	
4.	**Close CV-3-310A, Loop A Charging Isolation**	
5.	**Verify CV-3-310B, Loop C Charging Isolation, is CLOSED**	
6.	**Control Aux Spray As Follows:** * Increase Auxiliary Spray flow by closing PCV-3-455A, Pressurizer Spray Loop C, and/or PCV-3-455B, Pressurizer Spray Loop B * Reduce Auxiliary Spray flow by opening PCV-3-455A, Pressurizer Spray Loop C, and/or PCV-3-455B, Pressurizer Spray Loop B	

ATTACHMENT 4
Establish Auxiliary Pressurizer Spray
(Page 2 of 2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**Check This Attachment – DIRECTED BY Section 3.0, Step 18.a RNO OR Section 3.0, Step 18.b RNO**	Go to Attachment 4, Step 9.
8.	**Return to Section 3.0, Step 17.b**	
9.	**WHEN Aux Spray Alignment NO Longer Required, THEN Perform The Following:** **a.** Open one of the following: * CV-3-310A, Loop A Charging Isolation * CV-3-310B, Loop B Charging Isolation **b.** Close CV-3-311, Auxiliary Spray Valve	Perform the following: **1.** Reduce Charging Pump speed to minimum. **2.** Close HCV-3-121, Charging Flow To Regen Heat Exchanger. **3.** Adjust Charging Pump speed to maintain Seal Injection flow.

End of Attachment 4

ATTACHMENT 5
Aligning Main Steam Auxiliaries
(Page 1 of 2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Check Current Auxiliary Steam Alignment – FROM UNIT 3**	Go to Attachment 5, Step 5 to align Auxiliary Steam to Unit 3 Main Steam auxiliaries.
2.	**Check That It Is Desired To Align Auxiliary Steam From Unit 4**	Go to Attachment 5, Step 7.
3.	**At U-4 250 psig Auxiliary Steam Reducing Station, Perform the Following:**	
	a. Set pressure dial inside PC-1601 to minimum setting	
	b. Verify Inlet Isolation Valves – OPEN: • 4-10-075, Auxiliary steam 250 psig Reducer Inlet • 4-10-1236, Auxiliary Steam Reducer CV-1601A Inlet Isolation Valve	
	c. Raise Pressure Set dial inside PC-1601, 15 psig at a time until pressure is set between 245 and 255 psig on PI-4-1717	
	d. Lock knob on Pressure Set dial	
4.	**Verify Unit 3 250 psig Reducing Station Inlet Isolation Valves – CLOSED** • 3-10-075, Auxiliary Steam 250 psig Reducer Inlet • 3-10-1236, Auxiliary Steam Reducer CV-1601A Inlet Isolation Valve	

ATTACHMENT 5
<u>Aligning Main Steam Auxiliaries</u>
(Page 2 of 2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**Perform The Following To Align Auxiliary Steam To The Unit 3 Main Steam Auxiliaries** **a.** Open SLWU-3-001, Main Steam Line Warm-Up Isolation Valve **b.** Close 3-10-007, Main Steam To Auxiliary Steam Header Isolation Valve	
6.	**Inform Unit Supervisor That Alignment To Unit 4 Is Complete**	
7.	**Check Condenser Vacuum – GREATER THAN 20" HG**	Place SJAE Hogging Jet in-service as follows: **a.** Open 3-30-043, Steam Supply To Hogging Jet Valve. **b.** Slowly open 3-30-044, Steam Supply To Hogging Jet Valve, <u>until</u> Hogging Jet supply pressure as indicated on 3-PI-1597, is between 250 and 260 psig. **c.** Open 3-30-010, Condenser Air Removal To Hogging Jet.
8.	**Inform Unit Supervisor That Attachment 5 Is Complete**	

End of Attachment 5

ATTACHMENT 6
RCS Depressurization and SI Termination Criteria
(Page 1 of 2)

1.0 RCS DEPRESSURIZATION TERMINATION CRITERIA:

* Both of the following:

- RCS pressure – LESS THAN RUPTURED S/G(s) PRESSURE, AND
- PRZ level – GREATER THAN 7%[48%]

OR

* PRZ level – GREATER THAN 73%[60%]

OR

* RCS Subcooling based on Core Exit TCs – LESS THAN 19°F[73°F]

OR

* Both of the following (**N/A** if using a PRZ PORV):

- RCS pressure – WITHIN 300 PSI OF RUPTURED S/G(s) PRESSURE, AND
- PRZ level – GREATER THAN 37%[50%]

ATTACHMENT 6
RCS Depressurization and SI Termination Criteria
(Page 2 of 2)

2.0 SI TERMINATION CRITERIA:

- RCS Subcooling based on Core Exit TCs – GREATER THAN 19°F[73°F]

AND

- SECONDARY HEAT SINK (One of the following):
  - * Total Feed Flow to S/G(s) – GREATER THAN 400 GPM AVAILABLE

    OR

  - * Narrow Range Level in at least one intact S/G – GREATER THAN 7%[27%]

AND

- RCS Pressure – STABLE OR INCREASING

AND

- PRZ Level – GREATER THAN 7%[48%]

End of Attachment 6

ATTACHMENT 7
Establish Normal Letdown
(Page 1 of 1)

1. Verify B CCW Header flow in NORMAL.

2. Verify Letdown Orifice Isolation valves are CLOSED.

3. Open CV-3-204, Letdown From Regen Heat Exchanger Isolation.

4. Open LCV-3-460, High Pressure Letdown Isolation From Loop B Cold Leg.

5. Manually control PCV-3-145, Low Pressure Letdown Controller, to limit pressure spike when opening Letdown Orifice Isolation Valves.

6. Open Letdown Orifice Isolation Valves to establish desired Letdown flow.

7. Place PCV-3-145, Low Pressure Letdown Controller, in AUTOMATIC.

End of Attachment 7

ATTACHMENT 8
Establish Excess Letdown
(Page 1 of 1)

1. Verify CV-3-739, Excess Letdown Heat Exchanger CCW Outlet, is open.

2. Verify HCV-3-137, Excess Letdown Flow Controller, is closed.

3. Verify CV-3-389, Excess Letdown From Heat Exchanger To VCT Or RCDT, in RCDT-DIVERT position.

4. WHEN Seal Return flow is established, THEN CV-3-389, Excess Letdown From Heat Exchanger To VCT Or RCDT, may be placed in VCT-NORMAL position if desired.

5. Slowly open HCV-3-137, Excess Letdown Flow Control Valve.

6. Close HCV-3-137, Excess Letdown Flow Control Valve.

7. Open CV-3-387, Excess Letdown Isolation Valve From Cold Leg To Excess Letdown Heat Exchanger.

8. Open HCV-3-137, Excess Letdown Flow Controller.

9. Verify Excess Letdown Heat Exchanger Outlet Temperature, TI-3-139 is less than 195°F.

End of Attachment 8

FOLDOUT PAGE
For Procedure 3-EOP-E-3

1. **ADVERSE CONTAINMENT CONDITIONS**
   a. IF either condition listed below occurs, THEN use [Adverse Containment Setpoints]:
      * Containment atmosphere temperature ≥ 180°F

      OR

      * Containment radiation levels ≥ 1.3×10^5 R/hr

   b. WHEN Containment atmosphere temperature returns to less than 180°F, THEN Normal Setpoints can again be used.
   c. WHEN Containment radiation levels return to less than 1.3×10^5 R/hr, THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.
2. **RCP TRIP CRITERIA**
   a. IF all conditions listed below occur, THEN trip all RCPs:
      - High-Head SI pumps – AT LEAST ONE RUNNING AND SI FLOWPATH VERIFIED
      - RCS Subcooling – LESS THAN 19°F[41°F]
      - Controlled RCS cooldown **NOT** initiated

   b. IF Phase B actuated, THEN trip all RCPs.
3. **SI RE-INITIATION CRITERIA**

   IF either condition listed below occurs after Section 3.0, Step 21, THEN manually start SI Pumps as necessary to restore RCS subcooling and PRZ level and go to 3-EOP-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT – SUBCOOLED RECOVERY DESIRED, Step 1:
   * RCS Subcooling based on Core Exit TCs – LESS THAN 19°F[73°F]

   OR

   * PRZ level – CAN **NOT** BE MAINTAINED GREATER THAN 7%[48%]
4. **SECONDARY INTEGRITY CRITERIA**

   IF any S/G pressure is decreasing in an uncontrolled manner OR has completely depressurized, AND that S/G has **NOT** been isolated, AND is **NOT** needed for RCS cooldown, THEN go to 3-EOP-E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.
5. **COLD LEG RECIRCULATION SWITCHOVER CRITERIA**

   IF RWST level decreases to less than 155,000 gallons, THEN go to 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1.
6. **CST MAKEUP WATER CRITERIA**

   IF CST level decreases to less than 12%, THEN add makeup to CST using 3-NOP-018.01, CONDENSATE STORAGE TANK (CST).
7. **MULTIPLE TUBE RUPTURE CRITERIA**

   IF, after identification of a ruptured S/G, any intact S/G level increases in an uncontrolled manner OR any intact S/G has abnormal radiation, THEN stabilize the plant and return to 3 -EOP-E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.
8. **LOSS OF OFFSITE POWER OR SI ON OTHER UNIT**

   IF SI has been reset AND subsequently either offsite power is lost OR SI actuates on the other unit, THEN restore safeguards equipment and at least one Computer Room Chiller to required configuration. Refer to Attachment 3 for essential loads.
9. **RHR SYSTEM OPERATION CRITERIA**

   IF RHR flow is less than 1100 gpm, THEN the RHR Pumps shall be shut down within 44 minutes of the initial start signal.

L-16-1 NRC Exam

Control Room - JPM D

JPM TITLE: Respond to Loss of RHR

JPM NUMBER: 01050004301 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 01050004300 / Respond to Loss of RHR

K/A NUMBERS: APE 025 AA1.03 **K/A VALUE:** RO 3.4 / SRO 3.3

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [X] Other: [ ]
Lab: [ ]

Time for Completion: 20 Minutes Time Critical: No

Alternate Path [NRC]: No
Alternate Path [INPO]: No

Developed by:	**Brian Clark** Instructor/Developer	**6/20/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/21/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☐	☐	☒
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
1-0	Updated to fleet template; text/grammar changes	Updated for 2014 LOIT Annual Exam	1982463	N/A	N/A
				N/A	N/A
1-1	Validation time, Formatting, Enhance cues	NRC Validation	N/A	Hodge	12/19/14
				Wilson	12/22/14
2-0	Formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A	N/A
				N/A	N/A

SIMULATOR SET-UP:

SIMULATOR SETUP INSTRUCTIONS:

______	1.	Reset to **IC 30** or saved IC.
______	2.	NO-OP Cond & Feedwater and Steam Generators.
______	3.	Place simulator in RUN.
______	4.	Verify MODE 4 valve placards are in place.
______	5.	Ensure applicable portions of Simulator Operator Checklist are complete.
______	6.	Open and execute L-16-1 NRC JPM D.lsn.
______	7.	Allow plant to stabilize (allow auto makeup to complete).
______	8.	Acknowledge alarms and place simulator in FREEZE.
______	9.	Save as temporary IC, if JPM will be repeated.
______	10.	When ready to begin, then place Simulator in RUN.

SIMULATOR MALFUNCTIONS:

- TFMUM01S: 3A RHR Pump Shaft Shear
- IMM1S03C: MOV-3-750 Drifts Closed

SIMULATOR OVERRIDES:

- N/A

SIMULATOR REMOTE FUNCTIONS:

- N/A

Required Materials:	• Handout 3-ONOP-050
General References:	• 3-ONOP-050, Loss of RHR • 3-ARP-097.CR, Annunciator Response Procedures
Task Standards:	• Re-open MOV-3-750 • Restart 3B RHR Pump • Open FCV-3-605 to restore RHR flow to the core

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 3 is in Mode 4.
- The 3A RHR Pump is in service and providing core cooling.

INITIATING CUE:

- The Unit Supervisor has directed the crew to maintain current conditions.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

NOTE: 0-ADM-211, Emergency and Off-Normal Operating Procedure Usage (Prudent Operator Actions) – If redundant standby equipment is available and ready, the operator is permitted to start the redundant equipment for failed or failing operating equipment. Immediate follow-up of applicable ARPs or ONOPs shall occur as required.

Performance Step: 1 **Critical: No**	**Recognize closure of MOV-3-750**
Standard:	Recognize closure of MOV-3-750 and enter 3-ONOP-050, Loss of RHR.
Evaluator Note:	• If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue • When ready to begin, have booth operator **trigger MOV-3-750 DRIFTS CLOSED** • Annunciator H 6/2 (RHR HX HI/LO FLOW) will eventually actuate; if the ARP is used, it will direct the use of 3-ONOP-050
Evaluator Cue:	• Provide operator with a copy of Handout 3-ONOP-050 • If auto makeup occurs, state that "Another operator will monitor"
Booth Operator Cue:	After triggering **MOV-3-750 DRIFTS CLOSED**, verify that **ALLOW MOV-3-750 TO REOPEN triggers**.
Performance:	**SATISFACTORY** ______ **UNSATISFACTORY** _____
Comments:	

Performance Step: 2 **Critical: No**	**3-ONOP-050, Step 1:** **Check If RHR Pumps Should Be Stopped** a. RCS level – GREATER THAN 10% PRESSURIZER COLD CAL b. RHR pumps – ANY RUNNING c. RHR pumps – NOT CAVITATING
Standard:	Stop the 3A RHR Pump, if desired.
Evaluator Note:	• The examinee may observe the running pump with little or no flow and elect to stop the pump as a prompt/prudent action; if this occurs, mark Step 4 below as complete (i.e., securing the 3A RHR Pump is the critical step, but <u>when</u> this happens is NOT critical) • Alternatively, the examinee may elect to depress the MOV-3-750 “interrupt” pushbutton at this time (or earlier); if this occurs, a 3A RHR Pump shaft shear will be triggered → the examinee may interpret the results as a non-running pump and proceed to Step 2, per the Step 1.b RNO
Booth Operator Note:	If the “interrupt” pushbutton is depressed, verify **3A RHR PUMP SHAFT SHEAR triggers**.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: No**	**3-ONOP-050, Step 2:** **Check Loop 3C RHR Pump Suction Stop Valves – OPEN** • MOV-3-750 • MOV-3-751
Standard:	Recognize that MOV-3-750 is closed/closing and enter the RNO step.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 4 **Critical: Yes**	**3-ONOP-050, Step 2.a (RNO):** **Stop RHR Pumps**
Standard:	Secure the 3A RHR Pump.
Evaluator Note:	This action may have been performed in Step 2 above; in either case, record the completion here.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 5 **Critical: Yes**	**3-ONOP-050, Step 2.b (RNO):** **IF a momentary pressure spike has caused either or both valves to start closing, THEN perform the following at the Pushbutton Interrupt switches:** 1) Determine affected valve(s) →Yellow light – ON 2) Verify over pressure signal NOT present → Blue light – ON 3) Push Interrupt Pushbutton for affected valve(s) 4) Verify yellow light – DEENERGIZES 5) WHEN blue light DEENERGIZES, THEN verify affected valve(s) – OPEN 6) IF both valves are open, THEN go to Step 3
Standard:	Reopen MOV-3-750.
Evaluator Note:	• The examinee may have reopened the valve previously (prior to entering the ONOP) as a prompt/prudent action • In either case, record the completion here; opening the valve is the critical step, but when this happens is NOT critical
Booth Operator Note:	When the “interrupt” pushbutton is depressed, verify **3A RHR PUMP SHAFT SHEAR triggers**.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: No**	**3-ONOP-050, Step 3:** **Dispatch An Operator To Monitor RHR Pumps**
Standard:	Direct a field operator to locally monitor the RHR pumps.
Booth Operator Cue:	Acknowledge request to monitor RHR pumps and report that both RHR pumps are secured.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: No**	**3-ONOP-050, Step 4:** **Monitor RCS Heatup Rate**
Standard:	Per the NOTE prior to Step 4, direct the STA or available operator to monitor RCS heatup rate.
Evaluator Cue:	Acknowledge request to monitor RCS heatup rate.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical: No**	**3-ONOP-050, Step 5:** **Verify RHR Discharge to Cold Leg Isolation Valves – OPEN** • MOV-3-744A • MOV-3-744B
Standard:	Verify that MOV-3-744A and MOV-3-744B are OPEN.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical: Yes**	**3-ONOP-050, Step 6:** **Establish Conditions For Restarting An RHR Pump** a. RHR Pumps – BOTH STOPPED b. Close RHR Heat Exchanger Outlet Flow valve, HCV-3-758 c. Close RHR Heat Exchanger Bypass Flow valve, FCV-3-605 d. Verify MOV-3-750 and MOV-3-751 – OPEN
Standard:	Close HCV-3-758 and FCV-3-605.
Evaluator Note:	Only the valve closures are critical.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 10 **Critical: No**	**3-ONOP-050, Step 6:** **Establish Conditions For Restarting An RHR Pump** e. Start the previously running RHR pump
Standard:	Start the 3A RHR Pump, recognize that its shaft is sheared, secure the pump, and proceed to the RNO step.
Evaluator Note:	• If shaft shear recognized prior to this step, examinee may choose not to attempt to start the pump, this is acceptable • Motor amps will be low (due to the sheared shaft) and flow will be zero (due to the pump discharge valves being closed); the examinee may not recognize the sheared shaft until Step 6.f below, when MOV-3-605 is opened and no flow is observed • If the 3A RHR Pump is secured, the examinee may also elect to place the pump in PTL
Booth Operator Cue:	If asked about the 3A RHR Pump's status, report the following: • If the examinee has recognized the sheared shaft, then report that the motor is running but no flow noise is heard • Otherwise, report that there is nothing obviously wrong with the pump
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 11 **Critical: Yes**	**3-ONOP-050, Step 6.e (RNO):** **Start the Standby RHR pump**
Standard:	Start the 3B RHR Pump.
Booth Operator Cue:	If contacted, report a satisfactory start of the 3B RHR Pump.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 12 **Critical: Yes**	**3-ONOP-050, Step 6.f:** **Return RHR Heat Exchanger Bypass Flow valve, FCV-3-605, to AUTOMATIC operation increasing flow in increments of 500 gpm until desired flow is established**
Standard:	Reopen FCV-3-605 and raise RHR flow until annunciator H6/2 clears.
Evaluator Note:	• The examinee may not recognize until this point that the 3A RHR Pump's shaft has sheared • The setpoints for annunciator H6/2 are <3000 gpm and >3750 gpm • Opening the valve and clearing the alarm are critical, but NOT the incremental operation
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 13 **Critical: No**	**3-ONOP-050, Step 6.g:** **Open RHR Heat Exchanger Outlet Flow valve, HCV-3-758, as necessary to maintain desired RCS temperature**
Standard:	Maintain RCS temperature.
Evaluator Note:	The examinee may leave HCV-3-758 closed or throttle it open to establish a slight cooldown; either is acceptable.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When examinee has restored core cooling, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 3 is in Mode 4.
- The 3A RHR Pump is in service and providing core cooling.

INITIATING CUE:

- The Unit Supervisor has directed the crew to maintain current conditions.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

3-ONOP-050

Title:

Loss of RHR

(Continuous Use)

Safety Related Procedure

Responsible Department:	Operations
Revision Number:	8
Revision Approval Date:	4/14/16

PCRs *08-4015, 1614056, 1643677, 1929205, 1941507, 1998582, 1983203, 2091543, 1968625, 2119656*
PC/M *89-332, 96-081*
ECs *247008, 247009, 280399, 280301*

LIST OF EFFECTIVE PAGES
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Foldout	04/14/16

1.0 PURPOSE

1.1 This procedure provides the actions necessary for maintaining core cooling in the event that RHR cooling is lost.

2.0 SYMPTOMS OR ENTRY CONDITIONS

2.1 Annunciators

2.1.1 H 6/2, RHR HX HI/LO FLOW

2.1.2 H 6/4, RHR PP A/B TRIP

2.1.3 I 7/6, RHR SUMP PUMP ROOM A HI LEVEL

2.1.4 I 8/6, RHR SUMP PUMP ROOM B HI LEVEL

2.1.5 I 3/6, RHR SUMP HX ROOM HI LEVEL

2.1.6 I 7/3, RX VESSEL DRAINDOWN LO-LO-LEVEL

2.1.7 A 7/1, PRT HI/LO LEVEL HI PRESS/TEMP

2.1.8 A 9/6, RHR MOV-750/751 LETDOWN ISOLATION

2.1.9 A 4/2, QSPDS INADEQUATE CORE COOLING

2.2 Indications

2.2.1 Neither RHR pump is operating when required for decay heat removal

2.2.2 Loop 3C RHR Suction Stop Valve(s), MOV-3-750 or MOV-3-751, indicate closed when RHR is required for decay heat removal

2.2.3 Rapid increase in RCS pressure and OMS actuation when the RCS is solid

2.2.4 Low flow indicated on FI-3-605

2.2.5 Air-binding of the operating RHR pump as indicated by any of the following:

1. Motor current oscillations
2. Erratic flow oscillations
3. Excessive pump noise
4. Pump cavitation

3.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

3.1 References

3.1.1 Technical Specifications for Turkey Point Unit 3 and Unit 4

3.1.2 Turkey Point Unit 3 and Unit 4 Final Safety Analysis Report

3.1.3 Operating Diagrams

1. 5613-M-3050, Residual Heat Removal System
2. 5613-M-3062, Safety Injection System

3.1.4 Procedures

1. 0-ADM-051, Outage Risk Assessment and Control
2. 3-NOP-041.01A, 3A Reactor Coolant Pump Operations
3. 3-NOP-041.01B, 3B Reactor Coolant Pump Operations
4. 3-NOP-041.01C, 3C Reactor Coolant Pump Operations
5. 3-NOP-073, Condensate System
6. 0-NOP-074.01, Standby Steam Generator Feedwater System
7. 3-ONOP-004, Loss of Offsite Power
8. 3-ONOP-004.10, Loss of Offsite Power While on Backfeed
9. 3-ONOP-004.14, Loss of All AC Power While in Mode 5, 6, or Defueled
10. 3-ONOP-004.15, Loss of All AC Power in Mode 3 (Less Than 1000 PSIG) or Mode 4
11. 3-ONOP-030, Loss of Component Cooling Water
12. 3-ONOP-041.3, Excessive Reactor Coolant System Leakage
13. 3-ONOP-041.8, Shutdown LOCA [Mode 5 or 6]
14. 3-OP-050, Residual Heat Removal System

3.1.5 Plant Change/Modifications

1. PC/M 89-332, Generic Letter 88-17, Loss of Decay Heat Removal Programmed Enhancement - RCS Redundant Level Monitors
2. PC/M 96-081, Setpoint Change for RCP Seal Leakoff Low Flow
3. EC 247008, PC/M 09-139 EPU LAR Umbrella Doc Only PC/M
4. EC 247009, PC/M 09-140 EPU LAR Umbrella Doc Only PC/M
5. EC 280399, Unit 3 RCP Seal Upgrade
6. EC 280301, Fukushima FLEX Strategy Implementation Umbrella Modification

3.1.6 Miscellaneous Documents

1. JPN-PTN-SEMJ-89-094, Adequacy of Core Cooling
2. JPN-PTN-SENP-92-009, Substantial Safety Hazards Evaluation Related to Pressurizer Vents at Cold Shutdown
3. Westinghouse Technical Bulletin ESBU-TB-93-01, Revision 1
4. Westinghouse EOP Rev 1C Changes
5. Westinghouse Owners Group Abnormal Response Guideline, ARG-1, Loss of RHR While Operating at Mid-Loop Conditions, dated 6/6/96

3.2 Records Required

3.2.1 None

3.3 Commitment Documents

3.3.1 NRC Inspection Report 89-053, March 14, 1990

3.3.2 NRC IEIN No. 86-101, Loss of Decay Heat Removal Due to Loss of Fluid Levels In Reactor Coolant System

3.3.3 NRC Generic Letter 88-17, Loss of Decay Heat Removal

3.3.4 NRC IN-92-16, Loss of Flow from the Residual Heat Removal Pump During Refueling Cavity Draindown

3.3.5 INPO SOER 85-4, Loss or degradation of Residual Heat Removal Capability in PWRs (CTRAC No. 85-1178-34)

3.3.6 INPO SER 17-86, Loss of Shutdown Cooling Flow (CTRAC No. 87-0823)

3.3.7 INPO SER 23-86, Loss of Decay Heat Removal Flow (CTRAC No. 86-0982)

3.3.8 INPO OE-1744, RHR Gas Binding Due to Erroneous Half Loop Indication (CTRAC No. 85-1178-34)

3.3.9 INPO SER 9-92, Loss of Residual Heat Removal with Reduced Reactor Vessel Water Level

3.3.10 JPN-PTN-SENP-95-026, CCW Flow Balance and Post-Accident Alignment Requirements to Support Thermal Up-Rate (LER 250/95-006)

3.3.11 GL 88-17, Loss of Decay Heat Removal

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C A U T I O N

If leakage from the RHR system is discovered, the leak should be isolated using 3-ONOP-041.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE.

NOTES

- *Oscillations in flow or motor amps may be indicative of RHR pump cavitation.*
- *If loss of RHR is due to a loss of off-site power capability, power and RHR flow should be restored utilizing one of the following:*
- *3-ONOP-004, LOSS OF OFFSITE POWER*

  OR

  - *3-ONOP-004.10, LOSS OF OFFSITE POWER WHILE ON BACKFEED.*

  OR

  - *3-ONOP-004.14, LOSS OF ALL AC POWER WHILE IN MODE 5, 6, OR DEFUELED, Attachment 17, Loss of All AC Recovery On Station Blackout Tie.*

  OR

  - *3-ONOP-004.15, LOSS OF ALL AC POWER IN MODE 3 (LESS THAN 1000 PSIG) OR MODE 4, Attachment 6, Loss of All AC Recovery On Station Blackout Tie.*
- *During an Extended Loss of AC Power (ELAP), this procedure should be used for reference only.*
- *During a Loss of Power (excluding ELAP), this procedure should be used to establish containment closure and alternate cooling if RHR flow remains unavailable.*
- *The foldout page shall be monitored during the performance of this procedure.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check If RHR Pumps Should Be Stopped**	
	a. RCS level - GREATER THAN 10% PRESSURIZER COLD CAL	a. **<u>IF</u>** RCS Draindown Level Instrumentation is not available or RCS draindown level is LESS than 23%, **<u>THEN</u>** stop the running RHR pump **<u>AND</u>** go to 3-ONOP-041.8, Shutdown LOCA (Mode 5 or 6).
	b. RHR pumps - ANY RUNNING	b. Go to Step 2.
	c. RHR pumps - NOT CAVITATING • Amps Stable at normal value • Flow Stable at normal value	c. Stop RHR pumps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Interrupt feature for MOV-3-750 and MOV-3-751 is functional only with OMS in LO PRESS OPS.

2 **Check Loop 3C RHR Pump Suction Stop Valves – OPEN**

- MOV-3-750
- MOV-3-751

Perform the following:

a. Stop RHR pumps.

b. **IF** a momentary pressure spike has caused either or both valves to start closing, **THEN** perform the following at the Pushbutton Interrupt switches:

1) Determine affected valve(s).
   - Yellow light – ON

2) Verify over pressure signal **NOT** present:
   - Blue light – ON

3) Push Interrupt Pushbutton for affected valve(s).

4) Verify yellow light - DE-ENERGIZES.

5) **WHEN** blue light DE-ENERGIZES, **THEN** verify affected valve(s) - OPEN.

6) **IF** both valves are open, **THEN** go to Step 3.

c. **IF** RCS pressure GREATER THAN 525 psig, **THEN** perform the following:

1) Stop the charging pump(s).

2) Reduce RCS pressure to 425 psig.

d. **IF** MOV-3-750 and MOV-3-751 were **NOT** closed to isolate system leakage, **THEN** reopen MOV-3-750 and MOV–3-751. **IF** either valve can **NOT** be opened, **THEN** direct an operator to locally reopen MOV-3-750 and MOV-3-751.

e. **IF** BOTH valves can **NOT** be reopened, **THEN** monitor RCS Heatup Rate using Step 4 **AND** go to Step 11.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	**Dispatch An Operator To Monitor RHR Pumps**	
	a. Monitor RHR pump locally	
	b. Maintain communication with Control Room	

NOTE

RCS heatup rate is required to be monitored by the Shift Technical Advisor or any available operator until RHR cooling has been re-established.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Monitor RCS Heatup Rate**	
	a. Plot core exit temperature every minute for 5 minutes	a. **IF** core exit temperatures are **NOT** available, **THEN** perform the following: 1) Assume a 12°F per minute heatup rate unless the refueling cavity is flooded. **IF** the refueling cavity is flooded, **THEN** use 4°F per minute. 2) Go to Step 5.
	b. Calculate RCS heatup rate	
	c. Determine time required to reach saturation in RCS	
	d. Report results to Unit Reactor Operator and the Shift Manager	
	e. Repeat this step every 15 minutes until RHR cooling is restored	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	**Verify RHR Discharge To Cold Leg Isolation Valves – OPEN** • MOV-3-744A • MOV-3-744B	**IF** RHR Discharge To Cold Leg Isolation valve(s) were **NOT** closed to isolate system leakage, **THEN** perform the following: a. Reopen RHR discharge valve(s). b. **IF** at least one valve can **NOT** be opened, **THEN** perform the following: 1) Stop RHR pump(s). 2) Direct operators to locally reopen RHR Discharge To Cold Leg Isolation Valve(s). 3) Go to Step 11.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C A U T I O N

RCS Cooldown Rate shall be maintained LESS than 90 degrees per hour.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	**Establish Conditions For Restarting An RHR Pump**	
	a. RHR pumps – BOTH STOPPED	a. Go to Step 7.
	b. Close RHR Heat Exchanger Outlet Flow valve, HCV-3-758	
	c. Close RHR Heat exchanger Bypass Flow valve, FCV-3-605	
	d. Verify MOV-3-750 and MOV-3-751 – OPEN	d. Go to Step 11.
	e. Start the previously running RHR pump	e. Start the Standby RHR pump. 1) **IF** neither RHR pump can be restarted, **THEN** perform the following: a) Direct appropriate personnel to restore at least one RHR pump to operable status. b) Go to Step 11.
	f. Return RHR Heat Exchanger Bypass Flow valve, FCV-3-605, to AUTOMATIC operation increasing flow in increments of 500 gpm until desired flow is established	
	g. Open RHR Heat Exchanger Outlet Flow valve, HCV-3-758, as necessary to maintain desired RCS temperature	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7	**Verify RHR Flow Is 3000 GPM To 3750 GPM**	
	a. Verify RHR Heat Exchanger Bypass Flow, FCV-3-605 - MAINTAINING DESIRED FLOW IN AUTOMATIC	a. Manually control RHR Heat Exchanger Bypass flow, FCV-3-605, to establish desired flow. 1) **<u>IF</u>** unable to control RHR Heat Exchanger Bypass Flow, FCV-3-605, **<u>THEN</u>** perform the following at the 10-foot elevation platform in the RHR Heat Exchanger room to locally control RHR flow: a) Remove seal and place Safe Shutdown FCV-3-605 Manual Control Air Isolation Valve, 3-40-1895, in MANUAL. b) Verify Safe Shutdown FCV-3-605 Manual Control Air Vent Valve, 3-40-1896, in NORMAL. c) Adjust Safe Shutdown FCV-3-605 Manual Controller, PCV-3-605, to establish desired flow.
8	**Verify Stable RHR Pump Operation** • Running RHR pump amps – STABLE • RHR flow – STABLE • RHR pump noise level – NORMAL	**<u>IF</u>** stable RHR pump operation can not be verified, **<u>THEN</u>** perform the following: 1) Stop the running RHR pump. 2) Direct appropriate personnel to restore at least one RHR train to operable status. 3) Go to Step 11.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	**Maintain Stable Plant Conditions**	
	a. Verify RCS temperature – STABLE **OR** DECREASING	a. Perform the following: • Adjust HCV-3-758 to obtain desired cooldown rate. • Adjust FCV-3-605 to maintain desired RHR flow rate.
	b. Verify RCS temperature – LESS THAN 200°F **OR** trending to TEMPERATURE DESIRED BY SHIFT MANAGER	b. Go to Step 11.
10	**Go To Step 24**	
11	**Isolate Containment If Required**	
	a. Direct appropriate personnel to close any open containment penetrations • Equipment hatch • Airlocks • Refueling transfer tube • Any other openings	
	b. Direct personnel to stop work on all RCS openings	
	c. Check RCS temperature – * LESS THAN 180 DEGREES AND STABLE OR DECREASING **OR** * STABLE AT PRE-EVENT VALUE	c. **IF** RCS temperature is greater than 180°F **AND** increasing, **THEN** go to Step 12.
	d. Go to Step 13	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12	**Evacuate And Further Isolate Containment**	
	a. Announce over the plant PA system	
	• **Attention all personnel inside Unit 3 Containment, Evacuate Unit 3 Containment**	
	b. Actuate Containment Evacuation Alarm	
	c. Announce over the plant PA system	
	• **Attention all personnel inside Unit 3 Containment, Evacuate Unit 3 Containment**	
	d. Actuate Containment Isolation Phase A	
	1) Manually actuate containment isolation phase A	
	2) Containment isolation phase A valve white lights on VPB - ALL BRIGHT	2) **IF** any containment isolation phase A valve is **NOT** closed, **THEN** manually close valve. **IF** valve(s) can **NOT** be manually closed, **THEN** manually or locally isolate affected containment penetration.
	e. Reset Phase A Containment Isolation	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

13 **Establish Secondary Heat Sink**

a. Verify RCS intact

RESPONSE NOT OBTAINED: a. Perform the following:

1) **IF** the Reactor Vessel Cavity is flooded **AND** level is decreasing, **THEN** go to 3-ONOP-033.2, REFUELING CAVITY SEAL FAILURE.

2) Go to Step 24.

b. Verify at least two S/Gs available

- Secondary side manways – INSTALLED
- S/G hot leg manway – INSTALLED
- LI-3-462 -GREATER THAN 10%
- RCS Loops – FILLED

RESPONSE NOT OBTAINED: b. Perform the following:

1) **IF** RCS temperature is decreasing, **THEN** continue efforts to restore RHR cooling **AND** go to Step 24.

2) **IF** RCS temperature is increasing, **THEN** go to 3-ONOP-041.8, SHUTDOWN LOCA [MODE 5 OR 6].

c. Establish S/G makeup to the available S/Gs using one of the following methods

* Start a standby feedwater pump using 0-NOP-074.01, STANDBY STEAM GENERATOR FEEDWATER SYSTEM

OR

* Start a condensate pump using 3-NOP-073, CONDENSATE SYSTEM

OR

* Start a condensate transfer pump aligned to S/G fill line

RESPONSE NOT OBTAINED: c. Perform the following:

1) **IF** RCS temperature is decreasing, **THEN** continue efforts to restore RHR cooling **AND** go to Step 24.

2) **IF** RCS temperature is increasing, **THEN** go to 3-ONOP-041.8, SHUTDOWN LOCA [MODE 5 OR 6].

d. Open available S/G steam dump to atmosphere valves as necessary to maintain desired RCS temperatures

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

The effectiveness of steaming the available S/Gs may NOT be readily apparent during natural circulation. Plant conditions should be allowed to stabilize prior to performing Step 14.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14	**Determine If Blowdown Should Be Established**	
	a. Core exit temperatures – INCREASING	a. Go to Step 17.
	b. Available S/G steam dump to atmosphere valves - FULL OPEN	b. Open available S/G steam dump to atmosphere valves as necessary to maintain desired RCS temperatures. **IF** RCS temperature can be controlled using steam dump to atmosphere valves, **THEN** go to Step 17.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

15 **Align Blowdown From Available S/G(s)**

a. Verify RE-19, S/G Blowdown Radiation Monitor, - IN SERVICE

Response not obtained: a. Direct Nuclear Chemistry to sample available S/G(s) for activity.

b. Prepare for blowdown

Response not obtained: b. Go to Step 24.

1) Place blowdown keylock switch(s) for available S/G(s) in DRAIN/FILL position

- HS-3-1427X for S/G A
- HS-3-1426X for S/G B
- HS-3-1425X for S/G C

2) Verify S/G Liquid Sample valve(s) on available S/G(s) - OPEN

- MOV-3-1427 for S/G A
- MOV-3-1426 for S/G B
- MOV-3-1425 for S/G C

3) Verify Blowdown Flow valves - CLOSED

- FCV-3-6278A
- FCV-3-6278B
- FCV-3-6278C

4) Locally close S/G blowdown Manual Containment Isolation valve(s) on available S/G(s)

- SGB-3-007 for S/G A
- SGB-3-008 for S/G B
- SGB-3-009 for S/G C

5) Open Blowdown Containment Isolation valve(s) on available S/G(s)

- CV-3-6275A for S/G A
- CV-3-6275B for S/G B
- CV-3-6275C for S/G C

6) Locally open S/G Blowdown Manual Containment Isolation valve(s) on available S/G(s)

- SGB-3-007 for S/G A
- SGB-3-008 for S/G B
- SGB-3-009 for S/G C

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16	**Establish Blowdown From Available S/G(s)**	
	a. Align blowdown to discharge canal 1) Open Blowdown Tank Vent To Atmosphere, CV-3-6267A 2) Close Blowdown Tank Vent To Feedwater Heaters, CV-3-6267B 3) Open Blowdown Tank to Canal, HIS-3-6265B	
	b. Throttle open Blowdown Flow Control Valve on available S/G(s) to obtain maximum flow • FCV-3-6278A for S/G A • FCV-3-6278B for S/G B • FCV-3-6278C for S/G C	
17	**Maintain Level In Available S/G(s)**	
	a. Check narrow range levels - GREATER THAN 7%	a. Increase S/G makeup to available S/G(s).
	b. Continue S/G makeup to maintain narrow range level between 7% and 50%	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18	**Determine If One RCP Should Be Started**	
	a. RCS – LOOPS FILLED	a. Go to Step 24.
	b. Verify RCS Cold Leg Temperature – GREATER THAN 275°F	b. Perform the following: 1) Locally obtain S/G secondary temperature measurements. Refer to 3-NOP-041.01A, 3A Reactor Coolant Pump Operations; 3-NOP-041.01B, 3B Reactor Coolant Pump Operations; or 3-NOP-041.01C, 3C Reactor Coolant Pump Operations, as appropriate, for methods of obtaining S/G temperatures. 2) **IF** all S/G secondary temperatures are less than 10°F above RCS cold leg temperature, **AND** it is desired to start a RCP, **THEN** go to Step 19. 3) **IF** any S/G secondary water temperature is greater than 10°F above any RCS cold leg temperature, **THEN** verify natural circulation using ATTACHMENT 1. **IF** natural circulation can **NOT** be verified, **THEN** increase dumping steam. 4) Go to Step 24.
19	**Check Plant Conditions For Starting Desired RCP**	Perform the following: 1. Verify natural circulation using ATTACHMENT 1. **IF** natural circulation can **NOT** be verified, **THEN** increase dumping steam. 2. Go to Step 24.
	a. A or B 4KV bus - ENERGIZED FROM STARTUP TRANSFORMER	
	b. RCS Pressure – GREATER THAN 325 PSIG	
	c. Thermal barrier ΔP – GREATER THAN 0 INCHES OF WATER	
	d. RCP CBO flow - WITHIN LIMIT of Attachment 2	
	e. RCP CBO temperature - LESS THAN 195°F	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C A U T I O N

CCW System load requirements of 3-NOP-030, COMPONENT COOLING WATER SYSTEM, shall NOT be exceeded.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20	**Maintain Proper CCW System Alignment For RCP Operation**	
	a. CCW Heat Exchangers - THREE IN SERVICE	a. Perform the following: 1) Start or stop CCW pumps as necessary to establish ONLY ONE RUNNING CCW PUMP. 2) **IF** MOV-3-749A and MOV-3-749B are open, **THEN** stop and place in PULL-TO-LOCK all except one running CCW pump. 3) Go to Step 20c.
	b. CCW pumps - ONLY TWO RUNNING	b. Start or stop CCW pumps as necessary to establish ONLY TWO RUNNING CCW PUMPS.
	c. Check CCW from RHR Heat Exchangers - AT LEAST ONE CLOSED • MOV-3-749A • MOV-3-749B	c. Perform the following: 1) Isolate one Emergency Containment Cooler by placing one ECC Control Switch in STOP **AND** go to Step 20d. 2) **IF** unable to isolate one ECC, **THEN** stop all RCPs **AND** verify natural circulation using ATTACHMENT 1. 3) Go to Step 24.
	d. Verify B CCW header flow - NORMAL	d. Perform the following: 1) Verify natural circulation using ATTACHMENT 1. **IF** natural circulation can **NOT** be verified, **THEN** increase dumping steam. 2) Go to Step 24.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	**Establish Proper CCW Valve Alignment For RCP Operation**	
	a. RCP Thermal Barrier CCW Outlet, MOV-3-626 – OPEN	a. **IF** containment isolation phase B **NOT** actuated, CCW radiation levels are normal, and RCP number one seal leak-off temperature is less than 225°F, **THEN** perform the following: 1) Manually open MOV-3-626. 2) **IF** MOV-3-626 can **NOT** be manually opened, **THEN** direct the operator to locally open MOV-3-626. 3) **IF** MOV-3-626 can **NOT** be opened, **THEN** verify natural circulation using Attachment 1 **AND** go to Step 24.
	b. Verify the following valves – OPEN • MOV-3-716A, RCP CCW Inlet • MOV-3-716B, RCP CCW Inlet • MOV-3-730, RCP Bearing CCW Outlet	b. **IF** containment isolation phase B **NOT** actuated, **THEN** manually open MOV(s). 1) **IF** MOV(s) can **NOT** be manually opened, **THEN** direct operator to locally open MOV(s). 2) **IF** any RCP CCW MOV can **NOT** be opened, **THEN** verify natural circulation using Attachment 1 **AND** go to Step 24.
	c. Open CCW To Normal Containment Cooler valves • MOV-3-1417 • MOV-3-1418	
	d. Reset and start normal containment coolers	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

If possible, RCP B should be run to provide normal PZR spray. If RCP B cannot be started, RCP C should be started followed by RCP A.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22	**Try To Start One RCP**	
	a. Start oil lift pump	
	b. Check that the oil lift pump has been running - AT LEAST 2 MINUTES	b. **WHEN** 2 minute oil lift pressure time delay is satisfied, **THEN** verify Permissive To Start light ON **AND** perform Steps 22c, 22d, and 22e. Continue with Step 23.
	c. Start one RCP	c. Perform the following: 1) Verify natural circulation using ATTACHMENT 1. **IF** natural circulation can **NOT** be verified, **THEN** increase dumping steam. 2) Stop oil lift pumps. 3) Go to Step 24.
	d. Check that the RCP has been running – GREATER THAN 1 MINUTE	d. **WHEN** RCP has been running greater than 1 minute, **THEN** stop oil lift pump **AND** continue at Step 23.
	e. Stop the oil lift pump	
23	**Maintain Stable Plant Conditions**	
	a. Maintain PZR pressure – STABLE	
	b. Maintain PZR level – STABLE	b. **IF** PZR level can **NOT** be maintained, **THEN** perform 3-ONOP-041.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE while continuing with this procedure.
	c. Maintain intact S/G narrow range levels – STABLE	
	c. Maintain RCS average temperature – STABLE AT DESIRED TEMPERATURE	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
24	**Verify RHR Flow Restored**	Return to Step 6.
25	**Go To Appropriate Plant Procedure As Determined By The Shift Manager**	

END OF TEXT

FOLDOUT PAGE

1. **<u>CONTAINMENT CLOSURE CRITERIA</u>**

When at reduced inventory operations, containment closure shall be initiated within 5 minutes of the loss of RHR and shall be completed within the time to core boiling, or within 30 minutes of the loss of RHR, whichever is less.

When not in reduced inventory operations, containment closure shall be completed within the time to core boiling, or within 30 minutes of the loss of RHR, whichever is less, unless the containment closure time limit has been extended as allowed by 0-ADM-051, Outage Risk Assessment and Control, Enclosure 13, Containment Closure Time Limits.

FINAL PAGE

L-16-1 NRC Exam

Control Room - JPM E

JPM TITLE: Manually Initiate Containment Spray

JPM NUMBER: 01068007502 **REV.** 1-0

TASK NUMBER(S) / TASK TITLE(S): 01068007500 / Manually Initiate Containment Spray

K/A NUMBERS: 026 A3.01 **K/A VALUE:** RO 4.3 / SRO 4.5

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [X] Other: [ ]
Lab: [ ]

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No
Alternate Path [INPO]: No

Developed by:	**Brian Clark** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
0-0	New JPM	2015 LOCT Annual Exam	N/A	N/A / N/A	N/A / N/A
1-0	Formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A / N/A	N/A / N/A

SIMULATOR SET-UP:

SIMULATOR SETUP INSTRUCTIONS:

______	1.	Reset to **IC 1** or saved IC.
______	2.	Place simulator in RUN.
______	3.	Ensure applicable portions of Simulator Operator Checklist are complete.
______	4.	*N/A if using saved IC* Open and execute L-16-1 NRC JPM E: a. Trigger lesson step **LBLOCA WITH CTMT SPRAY FAILURE** b. Wait 5 minutes for other triggers to auto trigger c. Perform 3-EOP-E-0, Attachment 3, steps 1-7
______	5.	Allow plant to stabilize.
______	6.	Acknowledge alarms and place simulator in FREEZE.
______	7.	Save as temporary IC, if JPM will be repeated.
______	8.	When ready to begin, then place Simulator in RUN.

SIMULATOR MALFUNCTIONS:

- TFL3S1 & TFL3S2: L3-S1 & L3-S2 Fails to Actuate
- TFL3B11 & TFL3B1: L3-CIB11 & L3-CIB1 Fails to Actuate
- TVHHCLB: Large Break LOCA

SIMULATOR OVERRIDES:

- N/A

SIMULATOR REMOTE FUNCTIONS:

- N/A

Required Materials:	• Handout Attachment 3
General References:	• 3-EOP-E-0, Reactor Trip or Safety Injection
Task Standards:	• Manually initiate at least one train of containment spray, by starting at least one CSP and opening its associated discharge isolation valve • Manually close MOV-3-716B, MOV-3-626, and MOV-3-730 • Stop all RCPs • Secure the Unit 4 HHSI pumps

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- The Unit 3 reactor tripped due to a LOCA inside containment.
- Safety injection has actuated.
- Phase A containment isolation has actuated.
- The crew has completed Step 5 of 3-EOP-E-0, Reactor Trip or Safety Injection.

INITIATING CUE:

- You are directed to complete Attachment 3, Prompt Action Verifications, of 3-EOP-E-0 starting with Step 8.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: __________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

NOTE: Annunciators will be continuously coming in during the performance of this JPM. Determine and coordinate with the Booth Operator how this will be handled prior to administering the JPM.

Performance Step: 1 **Critical: No**	**Obtain required materials.**
Standard:	Obtain Attachment 3, Prompt Action Verifications, of 3-EOP-E-0, Reactor Trip or Safety Injection.
Evaluator Note:	• If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue • Examinee may notice at any time that phase B failed to actuate and may choose to manually initiate it; phase B will NOT initiate, but it is acceptable for them to attempt it
Evaluator Cue:	Provide examinee with a copy of HANDOUT Attachment 3.
Performance:	**SATISFACTORY _____ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 8:** **Verify Containment Cooling:** a. Check Emergency Containment Coolers – ONLY TWO RUNNING
Standard:	Recognize that two ECCs are running.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 9:** **Verify Containment Ventilation Isolation:** a. Unit 3 Containment Purge Exhaust And Supply Fans – OFF
Standard:	Recognize that the containment purge supply and exhaust fans are OFF.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 10:** **Verify Containment Spray NOT Required:** a. Containment pressure – HAS REMAINED LESS THAN 20 PSIG • PR-3-6306A • PR-3-6306B
Standard:	Recognize that containment pressure has NOT remained <20 psig and proceed to Step 10.a (RNO).
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: Yes**	**3-EOP-E-0, Attachment 3, Step 10.a (RNO):** **Perform the following:** 1) IF Containment Spray NOT initiated, THEN manually initiate Containment Spray
Standard:	• Start at least one CSP by taking its control switch to START • Open the associated discharge valve, MOV-3-880A/B, by taking its control switch to OPEN
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 10.a (RNO):** **Perform the following:** 2) Verify Containment Isolation Phase B has actuated
Standard:	Recognize that Phase B containment isolation did NOT actuate and depress both Phase B Isolation pushbuttons.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: Yes**	**3-EOP-E-0, Attachment 3, Step 10.a (RNO):** **Perform the following:** 3) Verify Containment Isolation Phase B Valve white lights on VPB are <u>all</u> bright 4) <u>IF</u> <u>any</u> Containment Isolation Phase B Valve did NOT close, <u>THEN</u> manually or locally isolate affected Containment Penetration
Standard:	Recognize that the following valves did NOT close and take their control switches to CLOSE: • MOV-3-716B, CCW to RCP Inlet • MOV-3-626, RCP Seal Cooling Water Outlet • MOV-3-730, RCP Bearing Cooling Water Outlet
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical: Yes**	**3-EOP-E-0, Attachment 3, Step 10.a (RNO):** **Perform the following:** 5) Stop all RCPs
Standard:	Secure all running RCPs.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 11:** **Verify SI – RESET**
Standard:	Recognize that SI is NOT reset and reset SI by depressing BOTH reset pushbuttons.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 10 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 12:** **Verify SI Valve Amber Lights On VPB – ALL BRIGHT**
Standard:	Recognize that all SI status lights are bright.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 11 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 13:** **Verify SI Flow:** a. RCS pressure – LESS THAN 1625 PSIG[1950 PSIG] b. High-Head SI Pump flow indicator – CHECK FOR FLOW c. RCS pressure – LESS THAN 275 PSIG[575 PSIG] d. RHR Pump flow indicator – CHECK FOR FLOW
Standard:	Recognize that all pressure and flow conditions are met.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 12 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 14:** **Realign SI System:** a. Check Procedure Entry Status – E-0 ENTERED FROM 3-ONOP-047.1, LOSS OF CHARGING FLOW IN MODES 1 THROUGH 4
Standard:	Recognize that the condition is NOT met and proceed to the RNO step.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 13 **Critical: No**	**3-EOP-E-0, Attachment 3, Step 14.a (RNO):** **Go to Attachment 3, Step 14.e**
Standard:	Proceed to Step 14.e.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 14 **Critical: No**	**3-EOP-E-0, Attachment 3, Steps 14.e and 14.f:** **Verify Unit 3 High-Head SI Pumps – <u>TWO</u> RUNNING** **Stop <u>both</u> Unit 4 High-Head SI Pumps and place in Standby**
Standard:	Recognize that both Unit 3 HHSI pumps are running and secure both Unit 4 HHSI pumps by taking their control switches to STOP.
Booth Operator Cue:	If contacted as the Unit 4 RCO, state that Unit 4 does NOT require its HHSI pumps to be running.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When the Unit 4 HHSI pumps have been secured, state “This completes the JPM.”**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ____________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- The Unit 3 reactor tripped due to a LOCA inside containment.
- Safety injection has actuated.
- Phase A containment isolation has actuated.
- The crew has completed Step 5 of 3-EOP-E-0, Reactor Trip or Safety Injection.

INITIATING CUE:

- You are directed to complete Attachment 3, Prompt Action Verifications, of 3-EOP-E-0 starting with Step 8.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

ATTACHMENT 3
Prompt Action Verifications
(Page 1 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Check Load Centers Associated With Energized 4 KV Buses – ENERGIZED** • 3A LC • 3B LC • 3C LC • 3D LC • 3H LC	Close the load center supply breakers.
2.	**Verify Feedwater Isolation**	
	a. Place Main Feedwater Pump switches in STOP	
	b. Feedwater Control valves – CLOSED • FCV-3-478 • FCV-3-488 • FCV-3-498	b. Manually close valves.
	c. Feedwater Bypass valves – CLOSED • FCV-3-479 • FCV-3-489 • FCV-3-499	c. Manually close valves.
	d. Feedwater Bypass Isolation valves – CLOSED: • POV-3-477 • POV-3-487 • POV-3-497	a. Locally close valves by turning manual override located below solenoid clockwise to the stop: (3 o'clock) • SV-3-477 • SV-3-487 • SV-3-497

ATTACHMENT 3
Prompt Action Verifications
(Page 2 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**(continued)**	
	e. Feedwater Isolation MOVs – CLOSED • MOV-3-1407 • MOV-3-1408 • MOV-3-1409	**b.** Locally close valves.
	f. Verify Standby Feedwater Pumps – OFF	**c.** IF Standby Feedwater aligned to Unit 3, THEN stop Standby Feedwater pump(s).
3.	**Check If Main Steam Lines Should Be Isolated**	
	a. Check Main Steamline Isolation and Bypass valves – ANY OPEN	**a.** Go to Attachment 3, Step 4.
	b. Check if either Main Steam Isolation Signal has actuated: * High Steam Flow with either Low S/G Pressure 614 psig OR Low T_{AVE} 543°F OR * Hi-Hi Containment Pressure 20 psig	**b.** Go to Attachment 3, Step 4.
	c. Verify Main Steam Isolation and Bypass valves – CLOSED	**c.** Push Main Steamline Isolation pushbuttons on VPB or manually close valves.

ATTACHMENT 3
Prompt Action Verifications
(Page 3 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Verify Containment Isolation Phase A Valve White Lights On VPB – ALL BRIGHT**	Perform the following: **a.** Manually actuate Containment Isolation Phase A. **b.** IF any Containment Isolation Phase A valve is **NOT** closed, THEN manually close valve. IF valve(s) can **NOT** be manually closed, THEN manually or locally isolate affected Containment penetration. **c.** IF any Containment Purge Valve can **NOT** be manually closed, THEN behind VPB, pull fuse for any open valve(s): * XEP for POV-3-2600 * XLAG for POV-3-2601 * XEQ for POV-3-2602 * XLAH for POV-3-2603
5.	**Verify Pump Operation**	
	a. At least two High-Head SI Pumps – RUNNING	**a.** Manually start High-Head Pump(s).
	b. Both RHR Pumps – RUNNING	**b.** Manually start RHR Pump(s).

ATTACHMENT 3
Prompt Action Verifications
(Page 4 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**Verify Proper CCW System Operation**	
a.	CCW Heat Exchangers – THREE IN SERVICE	**a.** Perform the following: **1)** Start or stop CCW Pumps as necessary to establish only one running CCW Pump. **2)** Verify only two running Emergency Containment Coolers. **3)** Go to Attachment 3, Step 6.c.
b.	CCW Pumps – ONLY TWO RUNNING	**b.** Start or stop CCW Pumps as necessary to establish only two running CCW Pumps.
c.	CCW Headers – TIED TOGETHER	**c.** IF both CCW Headers are intact, THEN direct a field operator to tie the headers together.
d.	MOV-3-626, RCP Thermal Barrier CCW Outlet – OPEN	**d.** IF all the following conditions exist: • Containment Isolation Phase B **NOT** actuated • CCW radiation levels are normal • RCP CBO temperature is less than 260°F THEN manually open MOV-3-626. IF MOV-3-626 can **NOT** be manually opened, THEN direct operator to open MOV-3-626 locally.

ATTACHMENT 3
<u>Prompt Action Verifications</u>
(Page 5 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**Verify Proper ICW System Operation**	
	a. Verify ICW Pumps – AT LEAST <u>TWO</u> RUNNING	**a.** Start ICW Pump(s) to establish at least <u>two</u> running.
	b. Verify ICW To TPCW Heat Exchanger – ISOLATED: • POV-3-4882 – CLOSED • POV-3-4883 – CLOSED	**b.** Manually close valve(s). <u>IF</u> valve(s) can **NOT** be closed, <u>THEN</u> locally close the following valves: * 3-50-319 for POV-3-4882 * 3-50-339 for POV-3-4883
	c. Check ICW Headers – TIED TOGETHER	**c.** <u>IF</u> <u>both</u> ICW headers are intact, <u>THEN</u> direct operator to tie headers together.
8.	**Verify Containment Cooling**	
	a. Check Emergency Containment Coolers – <u>ONLY</u> TWO RUNNING	**a.** Manually start or stop Emergency Containment Coolers to establish <u>only</u> two running.
9.	**Verify Containment Ventilation Isolation**	
	a. Unit 3 Containment Purge Exhaust And Supply Fans – OFF	**a.** Manually stop fans.

ATTACHMENT 3
Prompt Action Verifications
(Page 6 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
➔ **10.**	**Verify Containment Spray NOT Required**	
	a. Containment pressure – HAS REMAINED LESS THAN 20 PSIG: • PR-3-6306A • PR-3-6306B	**a.** Perform the following: **1)** IF Containment Spray **NOT** initiated, THEN manually initiate Containment Spray. **2)** Verify Containment Isolation Phase B has actuated. **3)** Verify Containment Isolation Phase B Valve white lights on VPB are all bright. **4)** IF any Containment Isolation Phase B Valve did **NOT** close, THEN manually or locally isolate affected Containment Penetration. **5)** Stop all RCPs.
11.	**Verify SI – RESET**	Reset SI.
12.	**Verify SI Valve Amber Lights On VPB – ALL BRIGHT**	Manually align valves to establish proper SI alignment for an injection flowpath.

ATTACHMENT 3
Prompt Action Verifications
(Page 7 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**Verify SI Flow**	
	a. RCS pressure – LESS THAN 1625 PSIG[1950 PSIG]	**a.** Go to Attachment 3, Step 14.
	b. High-Head SI Pump flow indicator – CHECK FOR FLOW	**b.** Manually start pumps and align valves to establish an injection flowpath.
	c. RCS pressure – LESS THAN 275 PSIG[575 PSIG]	**c.** Go to Attachment 3, Step 14.
	d. RHR Pump flow indicator – CHECK FOR FLOW	**d.** Manually start pumps and align valves to establish an injection flowpath.

ATTACHMENT 3
Prompt Action Verifications
(Page 8 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	**Realign SI System**	
	a. Check Procedure Entry Status – E-0 ENTERED FROM 3-ONOP-047.1, LOSS OF CHARGING FLOW IN MODES 1 THROUGH 4	**a.** Go to Attachment 3, Step 14.e.
	b. Check High-Head SI Pump flow indicator – FLOW **NOT** INDICATED	**b.** Go to Attachment 3, Step 14.e.
	c. Establish only one High-Head SI Pump running	
	d. Go to Attachment 3, Step 14.g	
	e. Verify Unit 3 High-Head SI Pumps – TWO RUNNING	**e.** Perform the following: **1)** Operate Unit 3 and Unit 4 High-Head SI Pumps to establish injection to Unit 3 from two High-Head SI Pumps. **2)** Go to Attachment 3, Step 14.g.
	f. Stop both Unit 4 High-Head SI Pumps and place in standby	
	g. Direct Unit 4 Reactor Operator to align Unit 4 High-Head SI Pump suction to Unit 3 RWST using Attachment 1.	
15.	**Verify Containment Isolation Phase A – RESET**	Reset Phase A.

ATTACHMENT 3
<u>Prompt Action Verifications</u>
(Page 9 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16.	**Reestablish RCP Cooling**	
	a. Check RCPs – AT LEAST <u>ONE</u> RUNNING	**a.** Go to Attachment 3, Step 17.
	b. Open CCW To Normal Containment Cooler Valves • MOV-3-1417 • MOV-3-1418	**b.** Stop <u>all</u> RCPs.
	c. Reset and start Normal Containment Coolers	**c.** Stop <u>all</u> RCPs.
17.	**Verify Control Room Ventilation Isolation**	
	a. Verify Emergency Air Supply Fans – AT LEAST <u>ONE</u> RUNNING * SF-1A * SF-1B	**a.** Manually start one Emergency Air Supply Fan.
	b. Control Room Ventilation dampers – ALIGNED FOR RECIRC	**b.** Manually align dampers for Recirculation.
	c. Verify Normal Flow green indicating light (4QR82) – ON	**c.** Manually start a second Emergency Air Supply Fan.
	d. TS-0002, TSC Emergency Vent Auto Initiate Key Switch – IN ENABLE	**d.** Place switch in ENABLE.
18.	**Place Hydrogen Monitors In Service Using 3-NOP-094, CONTAINMENT POST ACCIDENT MONITORING SYSTEM**	

ATTACHMENT 3
Prompt Action Verifications
(Page 10 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.	**Verify All Four EDGs – RUNNING**	Emergency Start any available EDG **NOT** running.
20.	**Verify Power To Emergency 4 KV Buses**	
	a. Check 3A, 3B AND 3D 4 KV Buses – ALL ENERGIZED	**a.** Inform Unit Supervisor that Attachment 3 is complete with the exception of the de-energized bus or buses. IF Unit Supervisor decides **NOT** to energize de-energized bus or buses, THEN go to Attachment 3, Step 20.b. IF Unit Supervisor decides to energize 3A, 3B or 3D Bus, THEN perform the following: **1)** IF 3A 4 KV Bus de-energized, THEN restore power to bus using 3-ONOP-004.2, LOSS OF 3A 4KV BUS. **2)** IF 3B 4 KV Bus de-energized, THEN restore power to bus using 3-ONOP-004.3, LOSS OF 3B 4KV BUS. **3)** IF 3D 4 KV Bus de-energized, THEN restore power to bus using 3-ONOP-004.5, LOSS OF 3D 4KV BUS.

ATTACHMENT 3
Prompt Action Verifications
(Page 11 of 11)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20.	**(continued)**	
	b. Check 3A AND 3B 4 KV Buses – ALL ENERGIZED FROM OFFSITE POWER	**b.** Check at least one Computer Room Chiller running. IF **neither** Computer Room Chiller is running, THEN perform the following: **1)** Evaluate if diesel capacity adequate to run one train of Chilled Water for Computer Room. IF adequate diesel capacity is **NOT** available, THEN shed non-essential loads. Refer to Attachment 2 for component KW load rating. **2)** Start one train of Chilled Water.
21.	**Notify Unit Supervisor Of The Following** • Attachment 3 is complete • Any safeguards equipment that is **NOT** In the required condition • Status of Containment pressure continuous action	

End of Attachment 3

L-16-1 NRC Exam

Control Room - JPM F

JPM TITLE: Restore Power to the 3A 4KV Bus

JPM NUMBER: 03005032300 **REV.** 0-0

TASK NUMBER(S) / TASK TITLE(S): 03005032300 / Cross-Tie 3D and 4D 4KV Buses

K/A NUMBERS: EPE 055 EA1.07 **K/A VALUE:** RO 4.3 / SRO 4.5

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]

Simulator: [X] Other: [ ]

Lab: [ ]

Time for Completion: 15 Minutes Time Critical: Yes

Alternate Path [NRC]: Yes

Alternate Path [INPO]: Yes

Developed by:	**Brian Clark** Instructor/Developer	**6/20/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/21/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☒	☐	☐
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
0-0	New JPM	L-16-1 NRC Exam	N/A	N/A	N/A
				N/A	N/A

SIMULATOR SET-UP:

SIMULATOR SETUP INSTRUCTIONS:

______	1.	Reset to **IC 1** or saved IC.
______	2.	Place simulator in RUN.
______	3.	Ensure applicable portions of Simulator Operator Checklist are complete.
______	4.	*N/A if using saved IC* Open and execute lesson L-16-1 NRC JPM F: a. Trigger **SETUP – 3A EDG FAILURE & RX TRIP** b. Verify **SETUP – LOOP & BUS LOCKOUTS** auto triggers (60 sec delay) c. Reduce total AFW flow to between 400 and 450 gpm d. Select “Silence All” in Main Menu of Orchid to disable the Annunciators
______	5.	Allow plant to stabilize.
______	6.	Acknowledge alarms and place simulator in FREEZE.
______	7.	Ensure Key 82 is available and functions to operate SBO Tie Breaker.
______	8.	Ensure Key Log is clean and a fresh logout page is available.
______	9.	Save as temporary IC, if JPM will be repeated.
______	10.	When ready to begin, then place Simulator in RUN.

SIMULATOR MALFUNCTIONS:

- TFQ5GAFS: 3A EDG START FAILURE
- TFE2Z51S: 3B BUS LOCKOUT
- TFE2Z53S: 3D BUS LOCKOUT (reset enabled)
- TFP8D6MT & TFP8D6BT: LOSS OF UNIT 3 STARTUP TRANSFORMER
- TFP8D8MT & TFP8D8BT: LOSS OF UNIT 4 STARTUP TRANSFORMER
- TFP8D3MT & TFP8D3BT: LOSS OF 3C TRANSFORMER
- TFH2FTRA/B/C: TRIP RCPS (setup)

SIMULATOR OVERRIDES:

- A302_A1_S14_3: FAILURE OF EDG EMERGENCY CONTROL SWITCH

SIMULATOR REMOTE FUNCTIONS:

- TCE2E33C: CLOSE 4AD07

Required Materials:	• Handout 3-EOP-ECA-0.0 • Key 82
General References:	• 3-EOP- ECA-0.0, Loss of All AC Power • 3-EOP-E-0, Reactor Trip or Safety Injection
Task Standards:	• Reset lockout on 3D 3kV bus. • Realign 3D bus to 3A bus. • Close SBO tie breaker 3AD07. • Restore power to 3A 4kV bus within 10 minutes. (0-ADM-232)

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 3 tripped from full power, due to a loss of offsite power.
- The plant is in a normal electrical alignment.
- While performing the IOAs of 3-EOP-E-0, Reactor Trip or Safety Injection, the crew recognized the following:
    - The 3A EDG did NOT start.
    - The 3B EDG started but did not energize the 3B 4kv bus.
- The crew transitioned to 3-EOP-ECA-0.0, Loss of All AC Power, and have completed through Step 4.

INITIATING CUE:

- The Unit Supervisor directs you to perform Step 5 of 3-EOP-ECA-0.0, with a priority on the 3B EDG.
- Some elements of this JPM are time-critical.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 3-EOP-ECA-0.0, Loss of All AC Power.
Evaluator Note:	If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue.
Evaluator Cue:	Provide examinee with a copy of handout 3-EOP-ECA-0.0.
Performance:	**SATISFACTORY** _____ **UNSATISFACTORY** _____
Comments:	

Performance Step: 2 **Critical: No**	**3-EOP-ECA-0.0, prior to Step 5:** **NOTE** ***The Unit Supervisor shall evaluate plant conditions and establish EDG Priority.***
Standard:	Read NOTE and recognize that it is safe to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: No**	**3-EOP-ECA-0.0, Step 5:** **Try To Restore Power To 3A OR 3B 4KV Bus** a. Check EDG Priority – 3A **(NO)** → (RNO) Go to Step 5.o.
Standard:	Recognize, from the Initial Conditions, that the priority is on the 3B EDG and transition to Step 5.o.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: No**	**3-EOP-ECA-0.0, Step 5:** **Try To Restore Power To 3A <u>OR</u> 3B 4KV Bus** o. Check 3B Bus Lockout Relay – RESET **(NO)** → (RNO) Perform the following: 1) Reset Lockout Relay. 2) <u>IF</u> Lockout relay can **NOT** be reset, <u>AND</u> EDG Priority was 3A, <u>THEN</u> go to Step 5.y. 3) <u>IF</u> Lockout relay can **NOT** be reset, <u>THEN</u> return to Step 5.b.
Standard:	• Recognize that the blue light for 4KV BUS 3B LOCKOUT RELAY is <u>blinking</u> and depress the associated reset pushbutton. • Recognize that the lockout relay for the 3B 4kV Bus will NOT reset, recall that the existing priority is on the 3B EDG, and transition to Step 5.b.
Evaluator Note:	• Under normal conditions (i.e., when the 3B 4kV Bus is NOT locked out), the blue light will be solid and NOT blinking. • When examinee attempts to reset the lockout relay for the 3B 4kV Bus, it will NOT reset (due to a fault on the 3B 4kV Bus).
Booth Operator Cue:	Ensure **LOCKOUT RELAY DOES NOT RESET** is triggered.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: No**	**3-EOP-ECA-0.0, Step 5:** **Try To Restore Power To 3A <u>OR</u> 3B 4KV Bus** b. Check 3A Bus Lockout Relay – RESET
Standard:	Recognize that the blue light for 4KV BUS 3A LOCKOUT RELAY is <u>NOT flashing</u> (i.e., the lockout relay for the 3A 4kV Bus is reset).
Evaluator Note:	If the 3AB 4kV Bus was locked out, the blue light would be <u>flashing</u>.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: No**	**3-EOP-ECA-0.0, Step 5:** **Try To Restore Power To 3A <u>OR</u> 3B 4KV Bus** c. Check 3A EDG Lockout – RESET **(NO)** → (RNO) Perform the following: 1) Locally reset 3A EDG Start Failure relay by pressing Alarm RESET pushbutton. 2) Reset Lockout Relay. 3) <u>IF</u> Lockout relay can NOT be maintained reset, <u>AND</u> EDG Priority was 3B, <u>THEN</u> go to Step 5.y. 4) <u>IF</u> Lockout relay can NOT be maintained reset, <u>THEN</u> go to Step 5.o.
Standard:	• Recognize that the blue light for 3A EDG is <u>flashing</u> and go to RNO. • Dispatch operator to locally reset Start Failure and Lockout Relay. • Recognize EDG will NOT reset and 3B was priority and go to Step 5.y.
Booth Operator Cue:	When directed to reset EDG start failure, state that relay will NOT reset.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: No**	**3-EOP-ECA-0.0, Step 5:** **Try To Restore Power To 3A <u>OR</u> 3B 4KV Bus** y. Check 3A <u>AND</u> 3B 4KV Buses – AT LEAST <u>ONE</u> ENERGIZED **(NO)** → (RNO) Observe CAUTION and NOTE prior to Step 6 and go to Step 6.
Standard:	Recognize that <u>neither</u> of Unit 3's 4kV buses is energized and transition to Step 6.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical: No**	**3-EOP-ECA-0.0, prior to Step 6:** **<u>CAUTION</u>** ***When power is restored to 3A <u>OR</u> 3B 4KV Bus from a <u>non-FLEX</u> source, then recovery actions should continue by performing Step 26.*** **<u>NOTE</u>** ***The following constitutes an <u>available</u> 4KV bus:*** • ***Bus Lockout reset*** • ***If bus stripping verification has been performed, then all loads are stripped***
Standard:	Read CAUTION/NOTE and recognize that it is safe to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical: No**	**3-EOP-ECA-0.0, Step 6:** **Dispatch Personnel To Restore AC Power** a. Check any of the following: * Unit 3 Startup Transformer Potential Light – ON OR * Opposite unit Startup Transformer Potential Light – ON OR * Opposite unit 4KV busses (A and B) – ANY ENERGIZED OR * 3C 4KV bus – ENERGIZED
Standard:	• Verify that the white potential lights for the Unit 3 and Unit 4 Startup Transformers are NOT lit. • Contact the Unit 4 RCO and determine that both of Unit 4's 4kV buses are energized.
Evaluator Note:	If examinee checks the status 3C 4kV Bus, it will be de-energized.
Booth Operator Cue:	When contacted as the Unit 4 RCO, state "The 4A and 4B 4kV Buses are energized from their respective EDGs."
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 10 **Critical: No**	**3-EOP-ECA-0.0, Step 6:** **Dispatch Personnel To Restore AC Power** b. Initiate restoring AC power to available 4KV bus(es) using the following: * Attachment 6, 3A 4KV Bus Restoration OR * Attachment 7, 3B 4KV Bus Restoration
Standard:	Recall that the lockout relay for the 3B 4kV Bus could NOT be reset, recognize that the 3B 4kV Bus is NOT available, and transition to Attachment 6.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 11 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 1:** **Confirm Bus Stripping On 3A 4KV Bus** a. Check if 3A 4KV Bus Stripping was verified in Section 3.0 **(NO)** → (RNO) Verify 3A 4KV Bus Stripping per Attachment 1.
Standard:	Recognize that stripping of the 3A 4kV Bus was NOT previously verified in Section 3 (i.e., Step 5.i) and transition to Attachment 1.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 12 **Critical: No**	**3-EOP-ECA-0.0, Attachment 1, Step 1:** IF 3A 4KV Bus is de-energized AND 3D 4KV Bus is aligned to 3A 4KV Bus…
Standard:	Recognize that the 3D 4kV Bus is NOT aligned to the 3A 4kV Bus (i.e., it is aligned to the 3B 4kV Bus) and this step is NOT applicable.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 13 **Critical: No**	**3-EOP-ECA-0.0, Attachment 1, Step 2:** IF 3A 4KV Bus is de-energized AND 3D 4KV Bus is **NOT** aligned to 3A 4KV Bus OR Station Blackout Tie permissive blue light is OFF, THEN perform the following: a. IF 3AA16 closed, THEN … b. IF 3AA18 closed, THEN …
Standard:	Recognize that the 3A 4kV Bus is de-energized (and the 3D 4kV Bus is NOT aligned to this bus) and the Station Blackout Tie permissive blue light is NOT lit, but breakers 3AA16 and 3AA18 are open (hence this step is NOT applicable).
Evaluator Note:	Breakers 3AA16 and 3AA18 will have automatically opened, due to the LOOP-induced bus stripping process.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 14 **Critical: No**	**3-EOP-ECA-0.0, Attachment 1, Step 2:** IF 3A 4KV Bus is de-energized AND 3D 4KV Bus is **NOT** aligned to 3A 4KV Bus OR Station Blackout Tie permissive blue light is OFF, THEN perform the following: c. Verify the following breakers open: • 3AA22, 3A 4KV BUS EMERGENCY TIE TO UNIT 4 STARTUP TRANSFORMER • 3AA09, 3A 4KV BUS TIE TO 3B OR 3C 4KV BUS • 3AA05, STARTUP TRANSFORMER 3A 4KV BUS SUPPLY • 3AA02, AUXILIARY TRANSFORMER 3A BUS SUPPLY • 3AA03, STEAM GENERATOR FEED PUMP 3A • 3AA07, HEATER DRAIN PUMP 3A • 3AA21, CONDENSATE PUMP 3A • 3AA13, SAFETY INJECTION PUMP 3A • 3AA15, RESIDUAL HEAT REMOVAL PUMP 3A • 3AA12, COMPONENT COOLING WATER PUMP 3A • 3AA01, REACTOR COOLANT PUMP 3A • 3AA19, INTAKE COOLING WATER PUMP 3A • 3AA11, TURBINE PLANT COOLING WATER PUMP 3A • 3AA16, CIRCULATING WATER PUMP 3A1 (90 sec delay if MOV > 5% OPEN) • 3AA18, CIRCULATING WATER PUMP 3A2 (90 sec delay if MOV > 5% OPEN) • 3AA08, 3A LOAD CENTER • 3AA14, 3C LOAD CENTER
Standard:	Recognize that the 3A 4kV Bus is de-energized (and the 3D 4kV Bus is NOT aligned to this bus), the Station Blackout Tie permissive blue light is NOT lit, and the listed breakers are open.
Evaluator Note:	• The listed breakers will have automatically opened, due to the LOOP-induced bus stripping process. • Breakers 3AA16 and 3AA18 were verified open in the previous step.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 15 **Critical: No**	**3-EOP-ECA-0.0, Attachment 1, Step 3:** IF 3AD01, SUPPLY FROM 4KV BUS 3A, is open, THEN verify 3AA17, FEEDER TO 4KV BUS 3D, is open.
Standard:	Recognize that breakers 3AD01 and 3AA17 are open.
Evaluator Note:	These breakers are open, as the 3D 4kV Bus is currently aligned to the 3B 4kV Bus.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 16 **Critical: No**	**3-EOP-ECA-0.0, Attachment 1, Step 4:** IF 3AD01, SUPPLY FROM 4KV BUS 3A, is closed, THEN perform the following…
Standard:	Recognize (as in the previous step) that 3AD01 is open and this step is NOT applicable.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 17 **Critical: No**	**3-EOP-ECA-0.0, Attachment 1, Step 5:** Notify Unit Supervisor that 3A 4KV Bus stripping is complete.
Standard:	Make the appropriate notification and return to Attachment 6.
Evaluator Cue:	If addressed as the Unit Supervisor or Shift Manager, acknowledge the communication.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 18 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 2:** **Verify SI – RESET**
Standard:	Recognize that a safety injection was NOT actuated and this step is NOT applicable.
Evaluator Note:	Some examinees may elect to depress the SAFETY INJECTION RESET pushbuttons to ensure it's reset, despite the absence of an automatic actuation. This is acceptable.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 19 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 3:** **Energize 3A 4KV Bus From Unit 3 Startup Transformer** a. Check Unit 3 Startup Transformer Potential white light is ON **(NO)** → (RNO) Go to Attachment 6, Step 4.
Standard:	Recognize that the white potential light for the Unit 3 Startup Transformer is NOT lit and transition to Step 4.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 20 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 4:** **Energize 3A 4KV Bus From Opposite Unit Startup Transformer** a. Check opposite unit Startup Transformer Potential white light – ON **(NO)** → (RNO) Observe NOTE prior to Attachment 6, Step 5 and go to Attachment 6, Step 5.
Standard:	Recognize that the white potential light for the Unit 4 Startup Transformer is NOT lit and transition to Step 5.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

THIS BEGINS THE TIME-CRITICAL PORTION OF THE JPM

Performance Step: 21 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, prior to Step 5:** **NOTE** ***Power needs to be restored to at least one 4KV bus (3A OR 3B) within 10 minutes to satisfy station blackout requirements.*** ***(Record Current Time: _______)***
Standard:	Read NOTE, record current time, and recognize that it is safe to proceed.
Evaluator Note:	Record current time: ____________
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 22 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 5:** **Check At Least One Of The Following – ENERGIZED** * Opposite Unit A 4KV Bus * Opposite Unit B 4KV Bus
Standard:	Contact the Unit 4 RCO and determine that both of Unit 4's 4kV buses are energized.
Booth Operator Cue:	When contacted as the Unit 4 RCO, state "The 4A and 4B 4kV Buses are energized from their respective EDGs."
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 23 **Critical: Yes**	**3-EOP-ECA-0.0, Attachment 6, Step 6:** **Check 3D 4KV Bus Lockout Relay – RESET (NO)** → (RNO) Reset 3D 4KV Bus Lockout Relay
Standard:	Recognize that the blue light for 4KV BUS 3D LOCKOUT RELAY is blinking and depress the associated reset pushbutton.
Evaluator Note:	Under normal conditions (i.e., when the 3D 4kV Bus is NOT locked out), the blue light will be solid and NOT blinking.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 24 **Critical: Yes**	**3-EOP-ECA-0.0, Attachment 6, Step 7:** **Check 3D 4KV Bus – ALIGNED TO 3A 4KV BUS (NO)** → (RNO) Perform the following: a. Open 3AB19, Feeder To 4KV Bus 3D. b. Open 3AD06, Supply From 4KV Bus 3B. c. Close 3AD01, Supply From 4KV Bus 3A. d. Close 3AA17, Feeder To 4KV Bus 3D.
Standard:	• Recognize, per the Initial Conditions, that the 3D 4kV Bus is aligned to the 3B 4kV Bus. • Recognize breakers 3AB19 and 3AD06 are open. • Close breakers 3AD01 and 3AA17.
Evaluator Note:	Examinee may match flags for breakers 3AB19 and 3AD06.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 25 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 8:** **Check Station Blackout Permissive Blue Light For 3AD07, Station Blackout Breaker – ON**
Standard:	Recognize that the blue PERMISSIVE light for STATION BLACKOUT BREAKER 3AD07 is lit.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 26 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 9:** **Check 4D 4KV Bus – ENERGIZED**
Standard:	Contact the Unit 4 RCO and determine that the 4D 4kV Bus is energized.
Booth Operator Cue:	When contacted as the Unit 4 RCO, state “The 4D 4kV Bus is energized from the 4B 4kV Bus.”
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 27 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 10:** **Check 4KV Bus Supplying Power To 4D 4KV Bus – ENERGIZED BY OFFSITE POWER (NO)** → (RNO) Perform the following: a. IF only one opposite Unit 4KV Bus (4A OR 4B) is energized… b. Direct Unit 4 RO to align **non**-running safeguards equipment switches powered from opposite Unit 4KV bus supplying 4D 4KV Bus as follows: • Unit 4 High Head SI Pumps – PULL-TO-LOCK • Containment Spray Pumps – PULL-TO-LOCK • Emergency Containment Coolers – STOP • RHR Pumps – PULL-TO-LOCK • CCW Pumps – PULL-TO-LOCK
Standard:	• Contact the Unit 4 RCO and determine that the 4D 4kV Bus is NOT energized by offsite power and both the 4A and 4B 4kV Buses are energized (i.e., Step 10.a is NOT applicable). • Direct the Unit 4 RCO to place all non-running Unit 4 HHSI pumps, CSPs, ECCs, RHR pumps, and CCW pumps in pull-to-lock.
Booth Operator Cue:	• When contacted as the Unit 4 RCO, state “The 4A and 4B 4kV Buses remain energized from their respective EDGs and the 4D 4kV Bus is aligned to the 4B 4kV Bus.” • When directed to place all non-running safeguards equipment in pull-to-lock, acknowledge the direction and state “all non-running safeguards equipment are in pull-to-lock or stop.”
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 28 **Critical: Yes**	**3-EOP-ECA-0.0, Attachment 6, Step 11:** **Energize 3A 4KV Bus From Station Blackout Tie Line** a. Close 3AD07, Station Blackout Breaker, using keylock switch (Key 82) b. Direct opposite Unit RO to close 4AD07, Station Blackout Breaker, using keylock switch (Key 82)
Standard:	• Obtain Key 82, insert it into the keylock at STATION BLACKOUT BREAKER 3AD07, and turn the switch to CLOSE. • Contact the Unit 4 RCO and direct that breaker 4AD07 be closed.
Evaluator Note:	• Only Step 11.a is critical. • When 4AD07 is closed, record current time: ___________ (Refer to Performance Step 22; time differential may NOT exceed 10 minutes)
Booth Operator Cue:	When directed to close breaker 4AD07, acknowledge the direction and **trigger CLOSE 4AD07**. Report when complete.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 29 **Critical: No**	**3-EOP-ECA-0.0, Attachment 6, Step 9:** **Check 3A 4KV Bus – ENERGIZED**
Standard:	Observe that the A 4KV BUS KILOVOLTS meter is indicating ~4160 volts.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **After 3A 4kV bus voltage is checked, state “This completes the JPM.”**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 3 tripped from full power, due to a loss of offsite power.
- The plant is in a normal electrical alignment.
- While performing the IOAs of 3-EOP-E-0, Reactor Trip or Safety Injection, the crew recognized the following:
  - The 3A EDG did NOT start.
  - The 3B EDG started but did not energize the 3B 4kv bus.
- The crew transitioned to 3-EOP-ECA-0.0, Loss of All AC Power, and have completed through Step 4.
- The crew transitioned to 3-EOP-ECA-0.0, Loss of All AC Power, and have completed through Step 4.

INITIATING CUE:

- The Unit Supervisor directs you to perform Step 5 of 3-EOP-ECA-0.0, with a priority on the 3B EDG.
- Some elements of this JPM are <u>time-critical</u>.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

FPL	**TURKEY POINT UNIT 3** **EMERGENCY OPERATING PROCEDURE** SAFETY RELATED **CONTINUOUS USE**	Procedure No. **3-EOP-ECA-0.0** Revision No. **10**

Title:

LOSS OF ALL AC POWER

Responsible Department: OPERATIONS

Special Considerations:

Last page of this procedure contains fold out page

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED today INITIAL

Revision	Approved By	Approval Date
6	Bob Pell	08/15/14
10	Rich Tucker	04/21/16

UNIT #	UNIT 3
DATE	
DOCT	PROCEDURE
DOCN	3-EOP-ECA-0.0
SYS	
STATUS	COMPLETED
REV	10
# OF PGS	

REVISION SUMMARY	
Rev. No.	**Description**
10	**PCR 1970576, 04/21/16, Gerry T Slaby** Changes to incorporate revision 3 from Westinghouse Owners Group (WOG) Emergency Response Guideline (ERG). These changes include Diverse and Flexible Coping Strategies (FLEX) Implementation per NEI 12-06. **PCR 2101331** Correct step 6b for CCW cooling of non-SBO HHSI pumps: This revision resolved this issue.
9B	**PCR 2101600, 01/16/16, James Speicher** Correct breaker designation in Attachment 7, Step 6 right (RNO) column Substep d should be"3AB19."
9A	**PCR 2097666, 12/15/15, David Houtz** Editorial correcting Attachment 3, Section 1.A.5 from C22 to C22A.
9	**PCR 2015652, 11/03/15, Terry White** Revised Step 7 and Attachment 4 to incorporate RCP Seal Replacement per **EC 280399**. Updated Implementing References (calculations) **PCR 2074386**, Attachment 14 revised per **EC 284549** for removal of Control Room HVAC Supply Fan SF-1B (V29B) High Flow Trip Interlock.
8	**PCR 2044334, 05/07/15, Gerard T Slaby** Deleted last bullet for NOTE for Step 5 regarding resetting of EDG Lockout Relay
7	**AR 1999426, 11/03/14, Gerry Slaby** Revised Calc 506.3, Turkey Point EOP Setpoints - Steam Generator Pressure, corrects errors in EOP Setpoint O.7, Minimum S/G pressure to prevent accumulator nitrogen injection (from 130 to 140 psig) in CAUTION Step 14. EOP Setpoint O.8, Minimum S/G pressure to prevent accumulator nitrogen injection plus margin (from 230 to 240 psig) in Steps 14, 14d and 14e. Revised Developmental Reference 6.d. Ref CR 01998831. Clarified wording in RNO step 5.y.

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(continued)

1.0 PURPOSE

1) This procedure provides actions to respond to a Loss Of All AC Power.

2) This procedure is applicable for MODES 1, 2, and 3 (greater than 1000 psig).

2.0 SYMPTOMS AND ENTRY CONDITIONS

1) The symptom of a Loss Of All AC Power is the indication that the A AND B 4KV Buses are both de-energized.

2) This procedure is entered from E-0, REACTOR TRIP OR SAFETY INJECTION, Step 3, on the indication that A AND B 4KV Buses are de-energized.

3) If E-0, REACTOR TRIP OR SAFETY INJECTION, Step 3 has been completed, this procedure is entered any time a Loss Of All AC Power occurs while in the EOP Network.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.0 OPERATOR ACTIONS

NOTE

- Step 1 and Step 2 are IMMEDIATE ACTION steps.
- CSF Status Trees are required to be monitored for Information Only. FRPs shall **NOT** be implemented.

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Verify Reactor Trip** • Rod Bottom Lights – ON • Reactor Trip and Bypass Breakers – OPEN • Rod Position Indicators – AT ZERO • Neutron flux – DECREASING	Manually trip Reactor.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**Verify Turbine Trip**	
a.	All Turbine Stop OR associated Control Valves – CLOSED	**a.** Manually trip Turbine. IF unable to verify turbine trip, THEN close Main Steamline Isolation and Bypass valves and go to Step 2.c.
b.	Moisture Separator Reheater Steam valves – CLOSED:	
1)	MSR Main Steam Supply Stop MOVs	**1)** Manually close valves. IF any valve can **NOT** be closed, THEN close Main Steamline Isolation and Bypass valves and go to Step 2.c.
2)	Reheater Timing valves	**2)** Close Main Steamline Isolation and Bypass valves and go to Step 2.c.
3)	MSR Purge Steam valves	**3)** Manually close valves. IF any valve can **NOT** be closed, THEN close Main Steamline Isolation and Bypass valves.
c.	Mid and East GCBs – OPEN (Normally 30-second delay)	**c.** Manually open breakers. IF breakers do **NOT** open, THEN actuate EMERGENCY GEN. BKR. TRIP SWITCH for the affected breaker(s).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**Check If RCS Is Isolated**	
	a. Letdown Isolation valves – CLOSED: • CV-3-200A • CV-3-200B • CV-3-200C	**a.** Manually close valves.
	b. PRZ PORVs – CLOSED	**b.** IF PRZ pressure less than 2335 psig, THEN manually close PORVs.
	c. Excess Letdown Isolation valves – CLOSED: • CV-3-387, Excess Letdown Isolation Valve From Cold Leg To Excess Letdown Heat Exchanger • HCV-3-137, Excess Letdown Flow Controller	**c.** Manually close valves.
	d. RCS Sample Isolation valves – CLOSED: • SV-3-6428, Loop 3A And 3B Sample Isolation • CV-3-956A, Pressurizer Steam Space Sample Isolation • CV-3-956B, Pressurizer Liquid Space Sample Isolation	**d.** Manually close valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Verify Proper AFW Flow**	
	a. Check AFW Steam Supply MOVs – OPEN: * Console indication OR * DCS indication OR * Local indication	**a.** Manually or locally open AFW Steam Supply MOVs.
	b. Check AFW Pumps – ANY RUNNING	**b.** IF **NO** AFW pumps can be started, THEN perform 3-ONOP-075, AUXILIARY FEEDWATER SYSTEM MALFUNCTION, while observing NOTE prior to Step 5, and continuing with Step 5.
	c. Check AFW Pumps – AT LEAST TWO RUNNING	**c.** IF both units require AFW, THEN perform the following: **1)** Establish 340 gpm flow to each unit. **2)** Use a setpoint of 340 gpm to each unit for required AFW flow instead of the 400 (or 450) gpm specified in subsequent steps and procedures.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**(continued)**	
d.	Check total AFW flow – GREATER THAN 400 GPM	**d.** Perform the following: **1)** Verify proper alignment of AFW valves. IF alignment **NOT** proper, THEN manually align valves as necessary to establish proper alignment. **2)** IF AFW flow greater than 400 gpm can **NOT** be established, THEN perform 3-ONOP-075, AUXILIARY FEEDWATER SYSTEM MALFUNCTION, while observing NOTE prior to Step 5, and continuing with Step 5.
e.	Check total AFW flow – LESS THAN 450 GPM →	**e.** Manually reduce total AFW flow to between 400 and 450 gpm.

NOTE

The Unit Supervisor shall evaluate plant conditions and establish EDG Priority.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**Try To Restore Power To 3A OR 3B 4KV Bus**	
a.	Check EDG Priority – 3A	**a.** Go to Step 5.o.
b.	Check 3A Bus Lockout Relay – RESET	**b.** Perform the following: **1)** Reset Lockout Relay. **2)** IF Lockout relay can **NOT** be reset, AND EDG Priority was 3B, THEN go to Step 5.y. **3)** IF Lockout relay can **NOT** be reset, THEN go to Step 5.o.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**(continued)**	
	c. Check 3A EDG Lockout – RESET	**c.** Perform the following: **1)** Locally reset 3A EDG Start Failure relay by pressing Alarm RESET pushbutton. **2)** Reset Lockout Relay. **3)** IF Lockout relay can **NOT** be maintained reset, AND EDG Priority was 3B, THEN go to Step 5.y. **4)** IF Lockout relay can **NOT** be maintained reset, THEN go to Step 5.o.
	d. Check 3A EDG – RUNNING	**d.** Perform the following: **1)** Manually start 3A EDG from Control Room: * Emergency Start OR * Rapid Start OR * Normal Start **2)** IF 3A EDG can **NOT** be started, AND EDG Priority was 3B, THEN go to Step 5.y. **3)** IF 3A EDG can **NOT** be started, THEN go to Step 5.o.
	e. Check 3A 4KV Bus – ENERGIZED	**e.** Go to Step 5.i.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. (continued)

CAUTION

If SI has been reset OR SI Actuation occurs on the other unit, Safeguards Equipment and at least one Computer Room Chiller needs to be restored to the required configuration.

NOTE

Attachment 5 provides a reference for Emergency Diesel Generator loads.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
f. Verify required Safeguards equipment – OPERATING	**f.** Manually start equipment as required.
g. Check status of 3-EOP-F-0, CRITICAL SAFETY FUNCTION STATUS TREES, prior to entering this procedure – MONITORED FOR INFORMATION ONLY	**g.** Implement FRPs as required, unless this procedure was directly entered from outside the EOP network. Return to procedure and step in effect.
h. Continue monitoring 3-EOP-F-0 CRITICAL SAFETY FUNCTION STATUS TREES, for information only, and return to procedure and step in effect	
i. Verify 3A 4KV Bus Stripping using Attachment 1	**i.** Perform the following: **1)** IF any load can **NOT** be disconnected from 3A 4KV Bus, AND EDG Priority was 3B, THEN go to Step 5.y. **2)** IF any load can **NOT** be disconnected from 3A 4KV Bus, THEN go to Step 5.o.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. (continued)

CAUTION

If an SI signal exists OR is actuated during this procedure, it must be reset to ensure restoration of a power source and to ensure controlled loading of equipment on the 4KV bus.

NOTE

If a Sequencer Failure has occurred AND SI has actuated, the associated EDG Output Breaker may **NOT** close unless SI is reset.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
j. Verify SI – RESET	
k. Check 3A 4KV Bus – ENERGIZED	**k.** Go to Step 5.m.
l. Observe CAUTION and NOTE prior to Step 5.f, and return to Step 5.f	
m. Manually energize 3A 4KV Bus **1)** Place EDG A Synch To 3A 4KV Bus, in ON **2)** Close 3AA20, A EDG To 3A 4KV Bus **3)** Place EDG A Synch To 3A 4KV Bus, in OFF	**m.** Perform the following: **a.** Locally energize 3A 4KV Bus using 3-ONOP-023.2, EMERGENCY DIESEL GENERATOR FAILURE, while continuing with this procedure. **b.** IF EDG Priority was 3B, THEN go to Step 5.y. **c.** Go to Step 5.o.
n. Observe CAUTION and NOTE prior to Step 5.f, and return to Step 5.f	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**(continued)**	
o.	Check 3B Bus Lockout Relay – RESET	**o.** Perform the following: **1)** Reset Lockout Relay. **2)** <u>IF</u> Lockout relay can **NOT** be reset, <u>AND</u> EDG Priority was 3A, <u>THEN</u> go to Step 5.y. **3)** <u>IF</u> Lockout relay can **NOT** be reset, <u>THEN</u> return to Step 5.b.
p.	Check 3B EDG Lockout – RESET	**p.** Perform the following: **1)** Locally reset 3B EDG Start Failure relay by pressing alarm RESET pushbutton. **2)** Reset Lockout relay. **3)** <u>IF</u> Lockout relay can **NOT** be maintained reset, <u>AND</u> EDG Priority was 3A, <u>THEN</u> go to Step 5.y. **4)** <u>IF</u> Lockout relay can **NOT** be maintained reset, <u>THEN</u> return to Step 5.b.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**(continued)**	
q.	Check 3B EDG – RUNNING	**q.** Perform the following: **1)** Manually start 3B EDG from Control Room: * Emergency Start <u>OR</u> * Rapid Start <u>OR</u> * Normal Start **2)** <u>IF</u> 3B EDG can **NOT** be started, <u>AND</u> EDG Priority was 3A, <u>THEN</u> go to Step 5.y. **3)** <u>IF</u> 3B EDG can **NOT** be started, <u>THEN</u> return to Step 5.b.
r.	Check 3B 4KV Bus – ENERGIZED	**r.** Go to Step 5.t.
s.	Observe CAUTION and NOTE prior to Step 5.f, and return to Step 5.f	
t.	Verify 3B 4KV bus stripping using Attachment 2	**t.** Perform the following: **1)** <u>IF</u> <u>any</u> load can **NOT** be disconnected from 3B 4KV Bus, <u>AND</u> EDG Priority was 3A, <u>THEN</u> go to Step 5.y. **2)** <u>IF</u> <u>any</u> load can **NOT** be disconnected from 3B 4KV Bus, <u>THEN</u> return to Step 5.b.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**(continued)**	

CAUTION

If an SI signal exists OR is actuated during this procedure, it must be reset to ensure restoration of a power source and to ensure controlled loading of equipment on the 4KV bus.

NOTE

If a Sequencer Failure has occurred AND SI has actuated, the associated EDG Output Breaker may **NOT** close unless SI is reset.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
u.	Verify SI – RESET	
v.	Check 3B 4KV Bus – ENERGIZED	**v.** Go to Step 5.x.
w.	Observe CAUTION and NOTE prior to Step 5.f, and return to Step 5.f	
x.	Manually energize 3B 4KV Bus: **1)** Place EDG B Synch To 3B 4KV Bus, in ON **2)** Close 3AB20, B EDG To 3B 4KV Bus **3)** Place EDG B Synch To 3B 4KV Bus, in OFF	**x.** Perform the following: **a.** Locally energize 3B 4KV Bus using 3-ONOP-023.2, EMERGENCY DIESEL GENERATOR FAILURE, while continuing with this procedure. **b.** IF EDG Priority was 3A, THEN go to Step 5.y. **c.** Return to Step 5.b.
y.	Check 3A AND 3B 4KV Buses – AT LEAST ONE ENERGIZED	**y.** Observe CAUTION and NOTE prior to Step 6 and go to Step 6.
z.	Observe CAUTION and NOTE prior to Step 5.f, and return to Step 5.f	

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

CAUTION

When power is restored to 3A OR 3B 4KV Bus from a **non**-FLEX source, then recovery actions should continue by performing Step 26.

NOTE

The following constitutes an available 4KV bus:

- Bus Lockout reset
- If bus stripping verification has been performed, then all loads are stripped

6. Dispatch Personnel To Restore AC Power

a. Check any of the following:

* Unit 3 Startup Transformer Potential Light – ON

OR

* Opposite unit Startup Transformer Potential Light – ON

OR

* Opposite unit 4KV busses (A and B) – ANY ENERGIZED

OR

* 3C 4KV bus – ENERGIZED

Response Not Obtained:

a. Notify Shift Manager to coordinate with System Dispatcher to restore offsite power while continuing with this procedure.

Go to Step 7.

b. Initiate restoring AC power to available 4KV bus(es) using the following:

* Attachment 6, 3A 4KV Bus Restoration

OR

* Attachment 7, 3B 4KV Bus Restoration

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

7. Align The Following Equipment Switches As Follows:

- Unit 3 High-Head SI Pumps – PULL-TO-LOCK
- Containment Spray Pumps – PULL-TO-LOCK
- Emergency Containment Coolers – STOP
- RHR Pumps – PULL-TO-LOCK
- CCW Pumps – PULL-TO-LOCK

8. Isolate RCP Seals

a. Manually close RCP CBO Control valves:

- CV-3-303A for RCP 3A
- CV-3-303B for RCP 3B
- CV-3-303C for RCP 3C

b. Locally close valves to isolate RCP Seals using Attachment 8, Locally Isolate RCP Seals, while continuing with Step 9

ATTACHMENT 1
<u>3A 4KV Bus Stripping</u>
(Page 1 of 3)

1. <u>IF</u> 3A 4KV Bus is de-energized <u>AND</u> 3D 4KV Bus is aligned to 3A 4KV Bus,
<u>THEN</u> verify the Station Blackout Tie permissive blue light is ON and 4AD07 open.

2. <u>IF</u> 3A 4KV Bus is de-energized <u>AND</u> 3D 4KV Bus is **NOT** aligned to 3A 4KV Bus
<u>OR</u> Station Blackout Tie permissive blue light is OFF,
<u>THEN</u> perform the following:

a. <u>IF</u> 3AA16 closed, <u>THEN</u> momentarily place 3A1 Circulating Pump control switch in STOP.

b. <u>IF</u> 3AA18 closed, <u>THEN</u> momentarily place 3A2 Circulating Pump control switch in STOP.

ATTACHMENT 1
3A 4KV Bus Stripping
(Page 2 of 3)

2. (continued)

c. Verify the following breakers open:

- 3AA22, 3A 4KV BUS EMERGENCY TIE TO UNIT 4 STARTUP TRANSFORMER
- 3AA09, 3A 4KV BUS TIE TO 3B OR 3C 4KV BUS
- 3AA05, STARTUP TRANSFORMER 3A 4KV BUS SUPPLY
- 3AA02, AUXILIARY TRANSFORMER 3A BUS SUPPLY
- 3AA03, STEAM GENERATOR FEED PUMP 3A
- 3AA07, HEATER DRAIN PUMP 3A
- 3AA21, CONDENSATE PUMP 3A
- 3AA13, SAFETY INJECTION PUMP 3A
- 3AA15, RESIDUAL HEAT REMOVAL PUMP 3A
- 3AA12, COMPONENT COOLING WATER PUMP 3A
- 3AA01, REACTOR COOLANT PUMP 3A
- 3AA19, INTAKE COOLING WATER PUMP 3A
- 3AA11, TURBINE PLANT COOLING WATER PUMP 3A
- 3AA16, CIRCULATING WATER PUMP 3A1 (90 sec delay if MOV > 5% OPEN)
- 3AA18, CIRCULATING WATER PUMP 3A2 (90 sec delay if MOV > 5% OPEN)
- 3AA08, 3A LOAD CENTER
- 3AA14, 3C LOAD CENTER

3. IF 3AD01, SUPPLY FROM 4KV BUS 3A, is open, THEN verify 3AA17, FEEDER TO 4KV BUS 3D, is open.

ATTACHMENT 1
3A 4KV Bus Stripping
(Page 3 of 3)

4. IF 3AD01, SUPPLY FROM 4KV BUS 3A, is closed, THEN perform the following:

a. IF 3AD07, STATION BLACKOUT BREAKER, is closed, THEN perform the following:

1) Open 3AD07, STATION BLACKOUT BREAKER.

2) Direct opposite Unit Reactor Operator to open 4AD07, STATION BLACKOUT BREAKER.

b. Verify breaker 3AD05, INTAKE COOLING WATER PUMP 3C, is open.

c. Verify 3AD04, COMPONENT COOLING WATER PUMP 3C Breaker, is open.

d. IF 3AD05, INTAKE COOLING WATER PUMP 3C Breaker, OR 3AD04, COMPONENT COOLING WATER PUMP 3C Breaker, can **NOT** be opened, THEN open 3AA17, FEEDER TO 4KV BUS 3D, and 3AD01, SUPPLY FROM 4KV BUS 3A.

5. Notify Unit Supervisor that 3A 4KV Bus stripping is complete.

End of Attachment 1

ATTACHMENT 2
3B 4KV Bus Stripping
(Page 1 of 3)

1. IF 3B 4KV Bus is de-energized AND 3D 4KV Bus is aligned to 3B 4KV Bus, THEN verify the Station Blackout Tie Permissive blue light is ON AND 4AD07 open.

2. IF 3B 4KV Bus is de-energized AND 3D 4KV Bus is **NOT** aligned to 3B 4KV Bus OR Station Blackout Tie Permissive blue light is OFF, THEN perform the following:

a. IF 3AB16 closed, THEN momentarily place 3B1 Circulating Water Pump control switch in STOP.

b. IF 3AB18 closed, THEN momentarily place 3B2 Circulating Water Pump control switch in STOP.

ATTACHMENT 2
3B 4KV Bus Stripping
(Page 2 of 3)

c. Verify the following breakers OPEN:

- 3AB22, 3B 4KV BUS TIE TO 3A OR 3C 4KV BUS
- 3AB05, STARTUP TRANSFORMER 3B 4KV BUS SUPPLY
- 3AB02, AUXILIARY TRANSFORMER 3B BUS SUPPLY
- 3AB10, HEATER DRAIN PUMP 3B
- 3AB21, CONDENSATE PUMP 3B
- 3AB12, SAFETY INJECTION PUMP 3B
- 3AB15, RESIDUAL HEAT REMOVAL PUMP 3B
- 3AB13, COMPONENT COOLING WATER PUMP 3B
- 3AB01, REACTOR COOLANT PUMP 3B
- 3AB06, REACTOR COOLANT PUMP 3C
- 3AB17, INTAKE COOLING WATER PUMP 3B
- 3AB11, TURBINE PLANT COOLING WATER PUMP 3B
- 3AB16, CIRCULATING WATER PUMP 3B1 (90 sec delay if MOV > 5% OPEN)
- 3AB18, CIRCULATING WATER PUMP 3B2 (90 sec delay if MOV > 5% OPEN)
- 3AB09, 3B LOAD CENTER
- 3AB14, 3D LOAD CENTER

3. IF 3AD06, SUPPLY FROM 4KV BUS 3B, is open,
THEN verify 3AB19, FEEDER TO 4KV BUS 3D, is open.

ATTACHMENT 2
3B 4KV Bus Stripping
(Page 3 of 3)

4. IF 3AD06, SUPPLY FROM 4KV BUS 3B, is closed, THEN perform the following:

a. IF 3AD07, STATION BLACKOUT BREAKER, is closed, THEN perform the following:

1) Open 3AD07, STATION BLACKOUT BREAKER.

2) Direct opposite Unit Reactor Operator to open 4AD07, STATION BLACKOUT BREAKER.

b. Verify 3AD05, INTAKE COOLING WATER PUMP 3C BREAKER, is open.

c. Verify 3AD04, COMPONENT COOLING WATER PUMP 3C BREAKER, is open.

d. IF 3AD05, INTAKE COOLING WATER PUMP 3C breaker, OR 3AD05, COMPONENT COOLING WATER PUMP 3C breaker, can **NOT** be opened, THEN open 3AB19, FEEDER TO 4KV BUS 3D, and 3AD06, SUPPLY FROM 4KV BUS 3B.

5. Notify Unit Supervisor that 3B 4KV Bus stripping is complete.

End of Attachment 2

ATTACHMENT 6
<u>3A 4KV Bus Restoration</u>
(Page 1 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Confirm Bus Stripping On 3A 4KV Bus**	
	a. Check if 3A 4KV Bus Stripping was verified in Section 3.0	**a.** Verify 3A 4KV Bus Stripping per Attachment 1.
2.	**Verify SI – RESET**	
3.	**Energize 3A 4KV Bus From Unit 3 Startup Transformer**	
	a. Check Unit 3 Startup Transformer Potential white light, is ON	**a.** Go to Attachment 6, Step 4.
	b. Place Startup Transformer Sync To 3A 4KV Bus 3AA05, in ON	
	c. Close 3AA05, Startup Transformer 3A 4KV Bus Supply	**c.** Locally close breaker.
	d. Place Startup Transformer Sync To 3A 4KV Bus 3AA05, in OFF <u>and</u> remove handle	
	e. Check 3A 4KV Bus – ENERGIZED	**e.** Go to Attachment 6, Step 4.
	f. Go to Attachment 6, Step 15	

ATTACHMENT 6
3A 4KV Bus Restoration
(Page 2 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Energize 3A 4KV Bus From Opposite Unit Startup Transformer**	
	a. Check opposite unit Startup Transformer Potential white light – ON	**a.** Observe NOTE prior to Attachment 6, Step 5 and go to Attachment 6, Step 5.
	b. Locally unlock and rack in 3AA22, 3A 4KV Bus Emergency Tie To Unit 4 Startup Transformer	**b.** Observe NOTE prior to Attachment 6, Step 5 and go to Attachment 6, Step 5.
	c. Close 3AA22, 3A 4KV Bus Emergency Tie To Unit 4 Startup Transformer	**c.** Locally close breaker.
	d. Check 3A 4KV Bus – ENERGIZED	**d.** Observe NOTE prior to Attachment 6, Step 5 and go to Attachment 6, Step 5.
	e. Maintain loading on the opposite unit Startup Transformer Tie Line less than 600 amps	
	f. Go to Attachment 6, Step 15	

NOTE

Power needs to be restored to at least one 4KV bus (3A OR 3B) within 10 minutes to satisfy station blackout requirements.
(Record Current Time: ________)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**Check At Least One Of The Following – ENERGIZED**	Notify Unit Supervisor of opposite unit 4KV bus status.
	* Opposite Unit A 4KV Bus	Go to Attachment 6, Step 14.
	* Opposite Unit B 4KV Bus	

ATTACHMENT 6
3A 4KV Bus Restoration
(Page 3 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**Check 3D 4KV Bus Lockout Relay – RESET**	Reset 3D 4KV Bus Lockout Relay. IF 3D 4KV Bus Lockout Relay can **NOT** be reset, THEN go to Attachment 6, Step 14.
7.	**Check 3D 4KV Bus – ALIGNED TO 3A 4KV BUS:** • 3AD01, Supply From 4KV Bus 3A – CLOSED • 3AA17, Feeder To 4KV Bus 3D – CLOSED	Perform the following: **a.** Open 3AB19, Feeder To 4KV Bus 3D. **b.** Open 3AD06, Supply From 4KV Bus 3B **c.** Close 3AD01, Supply From 4KV Bus 3A. **d.** Close 3AA17, Feeder To 4KV Bus 3D. **e.** IF 3D 4KV Bus can **NOT** be aligned to 3A 4KV Bus, THEN go to Attachment 6, Step 14.
8.	**Check Station Blackout Permissive Blue Light For 3AD07, Station Blackout Breaker – ON**	Perform the following: **a.** Re-verify 3A 4KV Bus stripping per Attachment 1. **b.** IF Station Blackout Permissive can **NOT** be satisfied, THEN go to Attachment 6, Step 14.
9.	**Check 4D 4KV Bus – ENERGIZED**	Perform the following: **a.** Request opposite Unit RO energize 4D 4KV Bus per 4-ONOP-004.5, LOSS OF 4D 4KV BUS. **b.** IF 4D 4KV Bus can **NOT** be energized, THEN go to Attachment 6, Step 14.

ATTACHMENT 6
<u>3A 4KV Bus Restoration</u>
(Page 4 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

10. Check 4KV Bus Supplying Power To 4D 4KV Bus – ENERGIZED BY OFFSITE POWER

Perform the following:

a. <u>IF</u> only <u>one</u> <u>opposite</u> Unit 4KV Bus (4A OR 4B) is energized, <u>THEN</u> verify opposite unit RO has completed placing **<u>non</u>**<u>-running</u> safeguards equipment in PULL-TO-LOCK or STOP per one of the following applicable procedures:

* Attachment 2 of 4-EOP-ES-0.1, REACTOR TRIP RESPONSE

<u>OR</u>

* Attachment 2 of 4-ONOP-004, LOSS OF OFFSITE POWER

<u>OR</u>

* Attachment 2 of 4-ONOP-004.10, LOSS OF OFFSITE POWER WHILE ON BACKFEED

Go to Attachment 6, Step 11.

b. Direct Unit 4 RO to align **<u>non</u>**<u>-running</u> safeguards equipment switches powered from opposite Unit 4KV bus supplying 4D 4KV Bus as follows:

- Unit 4 High Head SI Pumps – PULL-TO-LOCK
- Containment Spray Pumps – PULL-TO-LOCK
- Emergency Containment Coolers – STOP
- RHR Pumps – PULL-TO-LOCK
- CCW Pumps – PULL-TO-LOCK

ATTACHMENT 6
3A 4KV Bus Restoration
(Page 5 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.	**Energize 3A 4KV Bus From Station Blackout Tie Line**	
	a. Close 3AD07, Station Blackout Breaker, using keylock switch (Key Number 82)	**a.** Go to Attachment 6, Step 14.
	b. Direct opposite Unit RO to close 4AD07, Station Blackout Breaker, using keylock switch (Key Number 82)	**b.** Go to Attachment 6, Step 14.
12.	**Check 3A 4KV Bus – ENERGIZED**	Go to Attachment 6, Step 14.
13.	**Go to Attachment 6, Step 15**	

ATTACHMENT 6
3A 4KV Bus Restoration
(Page 6 of 6)

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

14. Energize 3A 4KV Bus From 3C 4KV Bus:

a. Check 3C 4KV Bus – ENERGIZED

b. Locally check the following breakers local blue Power Available light – ON:

1) 3AC16, 3C Bus Supply Breaker

2) 3AC01, 3C Bus Emergency Supply Breaker

c. Locally unlock and rack in 3AC13, 3C To 3A/3B Bus Tie

d. Close 3AC13, 3C To 3A/3B Bus Tie

e. Locally unlock and rack in 3AA09, 3A 4KV Bus Tie To 3B Or 3C 4KV Bus

f. Locally check 3AA09, 3A 4KV Bus Tie To 3B Or 3C 4KV Bus, breaker local white Closing Spring Charged / BKR Racked In and green Breaker Open lights are ON

g. Close 3AA09, 3A 4KV Bus Tie To 3B Or 3C 4KV Bus

h. Check 3A 4KV Bus – ENERGIZED

RESPONSE NOT OBTAINED:

Perform the following:

1. Notify unit supervisor that initial efforts to energize 3A 4KV Bus have failed.

2. Continue efforts to energize 3A 4KV Bus from any of the following:

* 3A Emergency Diesel
* Unit 3 Startup Transformer
* Unit 4 Startup Transformer
* Station Blackout Tie
* 3C Bus

3. WHEN 3A 4KV Bus is energized, THEN notify unit supervisor 3A 4KV Bus is energized, and of need to proceed to Section 3.0, Step 26.

15. Notify Unit Supervisor 3A 4KV Bus Is Energized, And Of The Need To Proceed To Section 3.0, Step 26

End of Attachment 6

ATTACHMENT 7
3B 4KV Bus Restoration
(Page 1 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Confirm Bus Stripping On 3B 4KV Bus**	
	a. Check if 3B 4KV Bus Stripping was verified in Section 3.0	**a.** Verify 3B 4KV Bus Stripping per Attachment 2, 3B 4KV Bus Stripping.
2.	**Verify SI – RESET**	
3.	**ENERGIZE 3B 4KV Bus From Unit 3 Startup Transformer**	
	a. Check Unit 3 Startup Transformer Potential white light is ON	**a.** Observe NOTE prior to Attachment 7, Step 4 and go to Attachment 7, Step 4.
	b. Place Startup Transformer Sync To 3B 4KV Bus 3AB05, in ON	
	c. Close 3AB05, Startup Transformer 3B 4KV Bus Supply	**c.** Locally close breaker.
	d. Place Startup Transformer Sync To 3B 4KV Bus 3AB05, in OFF <u>and</u> remove handle	
	e. Check 3B 4KV Bus – ENERGIZED	**e.** Observe NOTE prior to Attachment 7, Step 4 and go to Attachment 7, Step 4.
	f. Go to Attachment 7, Step 14	

ATTACHMENT 7
3B 4KV Bus Restoration
(Page 2 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Power needs to be restored to one 4KV bus (3A OR 3B) within 10 minutes to satisfy Station Blackout requirements. (Record Current Time: _______)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Check At Least One Of The Following – ENERGIZED:** * Opposite Unit A 4KV Bus * Opposite Unit B 4KV Bus	Notify unit supervisor of opposite unit 4KV bus status. Go to Attachment 7, Step 13.
5.	**Check 3D 4KV Bus Lockout Relay – RESET**	Perform the following: **a.** Reset 3D 4KV Bus Lockout Relay. **b.** IF 3D 4KV Bus Lockout Relay can **NOT** be reset, THEN go to Attachment 7, Step 13.
6.	**Check 3D 4KV Bus – ALIGNED TO 3B 4KV BUS** • 3AD06, Supply From 4KV Bus 3B – CLOSED • 3AB19, Feeder To 4KV Bus 3D – CLOSED	Perform the following: **a.** Open 3AA17, Feeder To 4KV Bus 3D. **b.** Open 3AD01, Supply From 4KV Bus 3A. **c.** Close 3AD06, Supply From 4KV Bus 3B. **d.** Close 3AB19, Feeder To 4KV Bus 3D. **e.** IF 3D 4KV Bus can **NOT** be aligned to 3B 4KV bus, THEN go to Attachment 7, Step 13.

ATTACHMENT 7
3B 4KV Bus Restoration
(Page 3 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**Check Station Blackout Permissive Blue Light For 3AD07, Station Blackout Breaker – ON**	Perform the following: **a.** Re-verify 3B 4KV Bus stripping per Attachment 2, 3B 4KV Bus Stripping. **b.** IF Station Blackout Permissive can **NOT** be satisfied, THEN go to Attachment 7, Step 13.
8.	**Check 4D 4KV Bus – ENERGIZED**	Perform the following: **a.** Request opposite Unit RO to energize 4D 4KV Bus per 4-ONOP-004.5, LOSS OF 4D 4KV BUS. **b.** IF 4D 4KV Bus can **NOT** be energized, THEN go to Attachment 7, Step 13.

ATTACHMENT 7
3B 4KV Bus Restoration
(Page 4 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9. Check 4KV Bus Supplying Power To 4D 4KV Bus – ENERGIZED BY OFFSITE POWER

Perform the following:

a. IF only one opposite Unit 4KV Bus (4A OR 4B) is energized, THEN verify opposite unit RO has completed placing **non**-running safeguards equipment in PULL-TO-LOCK or STOP per one of the following applicable procedures:

* Attachment 2 of 4-EOP-ES-0.1, REACTOR TRIP RESPONSE

OR

* Attachment 2 of 4-ONOP-004, LOSS OF OFFSITE POWER

OR

* Attachment 2 of 4-ONOP-004.10, LOSS OF OFFSITE POWER WHILE ON BACKFEED

Go to Attachment 7, Step 10.

b. Direct Unit 4 RO to align **non**-running safeguards equipment switches powered from opposite Unit 4KV bus supplying 4D 4KV Bus as follows:

- Unit 4 High Head SI Pumps – PULL-TO-LOCK
- Containment Spray Pumps – PULL-TO-LOCK
- Emergency Containment Coolers - STOP
- RHR Pumps – PULL-TO-LOCK
- CCW Pumps – PULL-TO-LOCK

ATTACHMENT 7
3B 4KV Bus Restoration
(Page 5 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	**Energize 3B 4KV Bus From Station Blackout Tie Line**	
	a. Close 3AD07, Station Blackout Breaker, using keylock switch (Key Number 82)	**a.** Go to Attachment 7, Step 13.
	b. Direct opposite Unit RO to close 4AD07, Station Blackout Breaker, using keylock switch (Key Number 82)	**b.** Go to Attachment 7, Step 13.
11.	**Check 3B 4KV Bus – ENERGIZED**	Go to Attachment 7, Step 13.
12.	**Go to Attachment 7, Step 14**	

ATTACHMENT 7
3B 4KV Bus Restoration
(Page 6 of 6)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**Energize 3B 4KV Bus From 3C 4KV Bus**	Perform the following:
	a. Check 3C 4KV Bus – ENERGIZED	**1.** Notify Unit Supervisor that initial efforts to energize 3B 4KV Bus have failed.
	b. Locally check the following breakers local blue Power Available light – ON: **1)** 3AC16, 3C Bus Supply Breaker **2)** 3AC01, 3C Bus Emergency Supply Breaker	**2.** Continue efforts to energize 3B 4KV Bus from <u>any</u> of the following: * 3B Emergency Diesel * Unit 3 Startup Transformer * Station Blackout Tie * 3C Bus
	c. Locally unlock and rack in 3AC13, 3C To 3A/3B Bus Tie	
	d. Close 3AC13, 3C To 3A/3B Bus Tie	**3.** <u>WHEN</u> 3B 4KV Bus is energized, <u>THEN</u> notify Unit Supervisor 3A 4KV Bus is energized, <u>and</u> of need to proceed to Section 3.0, Step 26.
	e. Locally unlock and rack in 3AB22, 3B 4KV Bus Tie To 3B Or 3C 4KV Bus	
	f. Locally check 3AB22, 3B 4KV Bus Tie To 3A Or 3C 4KV Bus breaker local white Closing Spring Charged / Breaker Racked In <u>AND</u> green Breaker Open lights are ON	
	g. Close 3AB22, 3B 4KV Bus Tie To 3A Or 3C 4KV Bus	
	h. Check 3B 4KV Bus – ENERGIZED	
14.	**Notify Unit Supervisor 3B 4KV Bus Is Energized, And Of The Need To Proceed To Section 3.0, Step 26**	

End of Attachment 7

L-16-1 NRC Exam

Control Room - JPM G

JPM TITLE: Place N-3-42 Power Range Drawer in Service

JPM NUMBER: 01059016200 **REV.** 1-0

TASK NUMBER(S) / TASK TITLE(S): 01059016200 / Place N-42 Power Range Drawer in Service

K/A NUMBERS: 015 A4.02 **K/A VALUE:** RO 3.9 / SRO 3.9

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]

Simulator: [X] Other: [ ]

Lab: [ ]

Time for Completion: 10 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Brian Clark** Instructor/Developer	**6/20/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/21/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER SUPERVISOR	DATE DATE
0-0	New JPM	N/A	N/A	N/A N/A	N/A N/A
1-0	Formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A N/A	N/A N/A

SIMULATOR SET-UP:

SIMULATOR SETUP INSTRUCTIONS:

_____	1.	Reset to **IC 1** or saved IC.
_____	2.	Place simulator in RUN.
_____	3.	Ensure applicable portions of Simulator Operator Checklist are complete.
_____	4.	N/A if using saved IC Perform switch manipulations in accordance with section 7.2.1, steps 1-23.
_____	5.	Allow plant to stabilize.
_____	6.	Acknowledge alarms and place simulator in FREEZE.
_____	7.	Save as temporary IC, if JPM will be repeated.
_____	8.	When ready to begin, then place Simulator in RUN.

SIMULATOR MALFUNCTIONS:

- N/A

SIMULATOR OVERRIDES:

- N/A

SIMULATOR REMOTE FUNCTIONS:

- N/A

Required Materials:	• Handout 3-OSP-059.4
General References:	• 3-OSP-059.4, Power Range Nuclear Instrumentation Analog Channel Operational Test
Task Standards:	• Place the N-3-42 Power Range drawer in service

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 3 is at 100% power.
- Unit 4 is in Mode 1.
- 3-OSP-059.4, Power Range Nuclear Instrumentation Analog Channel Operational Test, is in progress for N-3-42 and is complete through Step 7.2.2.23.

INITIATING CUE:

- You have been directed to place the N-3-42 power range drawer in service using 3-OSP-059.4, beginning at step 7.2.2.24.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 3-OSP-059.4, Power Range Nuclear Instrumentation Analog Channel Operational Test.
Evaluator Note:	If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue.
Evaluator Cue:	Provide examinee with a copy of handout 3-OSP-059.4.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 2 **Critical: No**	**3-OSP-059.4, Step 7.2.2.24:** **At the N-42, Drawer A, perform the following:** a. Verify the DROPPED ROD ROD STOP Light is OFF
Standard:	Recognize that the DROPPED ROD ROD STOP light is NOT lit.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: Yes**	**3-OSP-059.4, Step 7.2.2.24:** **At the N-42, Drawer A, perform the following:** b. Place the DROPPED ROD MODE switch to NORMAL
Standard:	Place the DROPPED ROD MODE switch in the NORMAL position.
Evaluator Cue:	Annunciator B 7/3, NIS CHANNEL IN TEST, will clear.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: No**	**3-OSP-059.4, Step 7.2.2.25:** **Verify the N-42 ROD DROP IN BYPASS status light (VPA) is OFF**
Standard:	Recognize that the ROD DROP IN BYPASS status light is NOT lit.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: No**	**3-OSP-059.4, Step 7.2.2.26:** **Verify Annunciator B 8/4, NIS TRIP BYPASSED, is OFF. Mark N/A if Annunciator B 8/4 is ON due to another NIS channel in BYPASS**
Standard:	Recognize that annunciator B 8/4 is NOT actuated.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: Yes**	**3-OSP-059.4, Step 7.2.2.27:** **At the COMPARATOR AND RATE Drawer, perform the following:** a. Place the COMPARATOR CHANNEL DEFEAT switch to NORMAL b. Verify COMPARATOR DEFEAT light is OFF
Standard:	Place the COMPARATOR CHANNEL DEFEAT switch in the NORMAL position and recognize that the COMPARATOR DEFEAT light is NOT lit.
Evaluator Note:	Only the switch manipulation is critical.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: Yes**	**3-OSP-059.4, Step 7.2.2.28:** **At the MISCELLANEOUS CONTROL AND INDICATION PANEL (NIS panel), perform the following:** a. Place the ROD STOP BYPASS switch associated with PRN42 to OPERATE
Standard:	Place the N-42 ROD STOP BYPASS switch in the OPERATE position.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical: Yes**	**3-OSP-059.4, Step 7.2.2.28:** **At the MISCELLANEOUS CONTROL AND INDICATION PANEL (NIS panel), perform the following:** b. Place the POWER MISMATCH BYPASS switch associated with PRN42 to OPERATE
Standard:	Place the N-42 POWER MISMATCH BYPASS switch in the OPERATE position.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical: Yes**	**3-OSP-059.4, Step 7.2.2.29:** **At the DETECTOR CURRENT COMPARATOR panel, perform the following:** a. Place the UPPER SECTION defeat switch to NORMAL 1) Verify CHANNEL DEFEAT light is OFF
Standard:	Place the UPPER SECTION defeat switch in the NORMAL position and recognize that the CHANNEL DEFEAT light is NOT lit.
Evaluator Note:	Only the defeat switch manipulation is critical.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 10 **Critical: Yes**	**3-OSP-059.4, Step 7.2.2.29:** **At the DETECTOR CURRENT COMPARATOR panel, perform the following:** b. Place the LOWER SECTION defeat switch to NORMAL 1) Verify CHANNEL DEFEAT light is OFF
Standard:	Place the LOWER SECTION defeat switch in the NORMAL position and recognize that the CHANNEL DEFEAT light is NOT lit.
Evaluator Note:	Only the defeat switch manipulation is critical.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 11 **Critical: Yes**	**3-OSP-059.4, Step 7.2.2.30:** **At Protection Channel II, Rack No. 11, place the Protection Channel bistable test switches in the NORMAL (Left) position:** a. BS-3-422B-1, Overpower ΔT Trip b. BS-3-422C-1, Overtemperature ΔT Trip
Standard:	Place the BS-3-422B-1 and BS-3-422C-1 bistable test switches in the NORMAL (left) positions.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When examinee places the bistable test switches in NORMAL, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ____________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 3 is at 100% power.
- Unit 4 is in Mode 1.
- 3-OSP-059.4, Power Range Nuclear Instrumentation Analog Channel Operational Test, is in progress for N-3-42 and is complete through Step 7.2.2.23.

INITIATING CUE:

- You have been directed to place the N-3-42 power range drawer in service using 3-OSP-059.4, beginning at step 7.2.2.24.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

Florida Power & Light Company

Turkey Point Nuclear Plant

Unit 3

This procedure may be affected by a T.C. (Temporary Change) Verify information prior to use. Date verified today *Initials* ____

3-OSP-059.4

Title:

Power Range Nuclear Instrumentation Analog Channel Operational Test

(Continuous Use)

Safety Related Procedure

Responsible Department:	Operations
Revision Number:	9
Revision Approval Date:	1/27/16

PCRs *565252, 1734938, 1810349, 1877915, 1880709, 1941747, 1969388, 1982434, 2000745, 2014939, 2063083*
TCs *08-061*
PC/Ms *84-210, 90-220, 90-508, 92-031, 93-005, 03-048, 04-112*

LIST OF EFFECTIVE PAGES
(Rev. 9)

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33	01/27/16	73	01/27/16		
34	01/27/16	74	01/27/16		
35	01/27/16	75	01/27/16		
36	01/27/16	76	01/27/16		
37	01/27/16	77	01/27/16		
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1.0 PURPOSE

1.1 This procedure provides instructional guidance to perform the monthly surveillance requirements of the Power Range Nuclear Instrumentation. This surveillance satisfies the requirements of References 2.1.1.

1.2 This procedure also provides instructions for I&C to reset the Power Range High Flux Trip setpoint when necessary.

2.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

2.1 References

2.1.1 Technical Specification

1. Section 2.2.1, Table 2.2-1, Items 2, 17b, 17c and 17d, Reactor Trip System Instrumentation Trip Setpoints
2. Section 3.3.1, Table 3.3-1, Item 2, Reactor Trip System Instrumentation
3. Section 4.3.1.1, Table 4.3-1, Items 2, 17b, 17c and 17d Reactor Trip System Instrumentation Surveillance Requirements
4. Section 4.10.3.2, Physics Tests
5. Section 3/4.7.1, Turbine Cycles, Safety Valves

2.1.2 Final Safety Analysis Report

1. Section 7.2, Protective Systems
2. Section 7.4, Nuclear Instrumentation

2.1.3 Plant Procedures

1. 0-ADM-031, Independent Verification
2. 0-ADM-724, Instrumentation Protection Channel Determination of Channel Operability
3. 3-ONOP-059.8, Power Range Nuclear Instrumentation Malfunction
4. 0-OP-003.3, 120V Vital Instrument AC System
5. 0-OSP-200.1, Schedule of Plant Checks and Surveillances
6. EN-AA-203-1001, Operability Determinations/Functionality Assessments

2.1.4 Miscellaneous Documents (i.e., PC/M, Correspondence)

1. PC/M 84-210, Turbine Runback Modifications

2. PC/M 90-220, RTD Bypass Elimination

3. PC/M 90-508, Implementation of Setpoint Methodology

4. PC/M 93-005, Elimination of Turbine Runback from Dropped Rod

5. Safety Evaluation PTN-BFJM-94-007

6. Unit 3 Plant Curve Book, Section 5, Figure 5A

7. PC/M 03-048, OTΔT and OPΔT Turbine Runback Elimination

8. PC/M 04-112, Emergency Response Data Acquisition and Display System (ERDADS) Replacement

2.2 Records Required

2.2.1 The date, time and section started and the date, time and section completed shall be entered in the Unit Narrative Log. Also, problems encountered while performing the procedure should be entered; i.e., malfunctioning equipment, delays due to changes in plant conditions, etc.

2.2.2 Completed copies of the below listed items document compliance with Technical Specification surveillance requirements and shall be sent to QA Records for retention in accordance with Quality Assurance Records Program requirements:

1. Section 7.0

2. Attachments 1 through 9

2.2.3 Completed copies of the below listed section and attachments shall be transmitted to the system engineer for trending whenever bistable adjustments are performed:

1. Section 7.0

2. Attachments 1 through 9

2.3 Commitment Documents

2.3.1 JPN-PTN-SENP-94-014, No Significant Hazards Evaluation for Extension of Technical Specification Surveillance Intervals and Out of Service Times for the Reactor Protection and Engineered Safety Features Instrumentation Systems

3.0 **PREREQUISITES**

3.1 Instrument AC Panels, 3P06, 3P07, 3P08 and 3P09 should be powered from their normal sources per 0-OP-003.3, 120V Vital Instrument AC System, during this test.

3.2 Obtain permission from System Load Dispatcher to perform this test.

4.0 **PRECAUTIONS/LIMITATIONS**

4.1 Tests shall be performed on only one power range channel at a time with the remaining channels operable. (T.S. Table 3.3-1, Item 2, Action 2)

4.2 Any nuclear instrumentation channel should be energized for at least one hour prior to being tested.

4.3 This test should not be performed during changes of plant reactor power.

4.4 Discrepancies noted during this test shall be investigated for their effect on channel operability and actions taken per 3-ONOP-059.8, Power Range Nuclear Instrumentation Malfunction as applicable.

4.5 Do NOT turn the Gain potentiometer while performing a power range channel test.

4.6 Annunciator B 9/2, Axial Flux Tilt, and B-5/5, OTΔT/OPΔT Rod Stop may be actuated intermittently when the test signals are varied on the channel under test. This is more likely to occur at low power levels. Monitor the in service channels to verify actual axial flux and OPΔT/OTΔT are not exceeded.

4.7 The Shift Manager shall be notified immediately if any acceptance criteria is not met or any malfunction or abnormal conditions occur. This information shall also be recorded in the Remarks section.

4.8 Notify Reactor Engineering if this procedure has been successfully used to return a power range channel to service after maintenance, or if the acceptance criteria of Subsection 6.1 is not met.

4.9 Entry into the Control Room may be restricted as the Shift Manager deems necessary during the performance of this procedure.

4.10 Performance of this procedure has resulted in unit Trips.

4.11 All instructions and verifications of each step are conducted at the NIS Panel, unless otherwise noted.

4.12 If the Power Range Hi Flux Trip Setpoint is set at other than 108%, then a Caution Tag and/or in Information Placard shall be placed near the meter face with at least the following:

- Hi Flux Trip Setpoint is (enter setpoint) %.
- Maintain RX Power at or below (setpoint minus 5%) %.

4.13 Acceptance Criteria specified in this procedure are based on Drawer meter indication. Variations between Drawer meter and other indicators do NOT constitute a failure to meet Acceptance Criteria, but may warrant a PWO if the deviation is significant.

The Engineering recommended acceptable deviations for the various reactor power indications from the Drawer meter are summarized in the following table:

Description	**Component Tag / Point**	**Acceptable Deviation from Drawer Meter (tolerance bands)**
Drawer meter	Part of Drawer	Baseline value
Console meter	NI-41, N-42, N-43, N-44	1.5%
Console recorder	NR-*-45A/B	1.0%
VPA recorder	NR-*-46, -47	1.5%
ERDADS	Including ERDADS N41_PWR_A, N42_PWR_A, N43_PWR_A, N44_PWR_A, TLPWRN_V	1.1%

As long as the displayed values are within the limits specified above, no action is required. When a displayed value is outside the limit specified above, a PWO should be generated for maintenance to investigate and correct.

4.14 When bistables have been restored for 6 hours for testing, return to Trip condition. (Tech Spec 3.3.1, Table 3.3-1, Item 2)

5.0 **<u>SPECIAL TOOLS/EQUIPMENT</u>**

5.1 None

6.0 ACCEPTANCE CRITERIA

6.1 The Reactor Trip System Instrumentation and Interlock Setpoints associated with the Power Range NIS shall be as follows:

6.1.1 Interlock P-10 Trip Setpoint 10%.

6.1.2 Interlock P-10 Reset Setpoint 8%.

6.1.3 With power level greater than or equal to P-10 setpoint, verify P-10 is in the required state.

6.1.4 Overpower Trip Low Range setpoint less than or equal to 25% (24.4% to 25.6%).

6.1.5 Overpower Trip Low Range Reset setpoint 22% to 24%.

6.1.6 With power level greater than or equal to Overpower Trip Low Range Setpoint, verify the Trip is in its required state.

6.1.7 Interlock P-8 Trip Setpoint less than or equal to 45% (44% to 45%).

6.1.8 Interlock P-8 Reset Setpoint 42% to 44%.

6.1.9 With power level greater than or equal to P-8 setpoint, verify P-8 is in the required state.

6.1.10 Overpower Rod Stop Setpoint 103% (102% to 104%).

6.1.11 Overpower Rod Stop Reset Setpoint 100% to 102%.

6.1.12 Overpower Trip High Range Setpoint is the value in the Unit 3 Plant Curve Book, Section 5, Figure 5A, plus 0.6% or minus 0.6%.

6.1.13 Overpower Trip High Range Reset Setpoint is the value in the Unit 3 Plant Curve Book, Section 5, Figure 5A, plus or minus 1%.

INIT Date/Time Started: today /

7.2 Operational Test of N-42

7.2.1 Initial Conditions

SHIFT MANAGER
ADMINISTRATIVE VERIFICATION POINT

1. *The execution of this procedure has resulted in unit Trips.*

2. *All personnel participating in the performance of this procedure shall discuss with their Supervisor the applicable procedure subsection(s) and the Precautions and Limitations, and Prerequisites sections.*

Requirements of this Verification Point have been met. Permission granted to proceed beyond this Verification Point.

________________________ *Date:* today
Shift Manager

______ 3. Verify Prerequisites in Section 3.0 have been completed.

INIT 7.2.1 (Cont'd)

NOTES

- *Test signals for the power range channels are superimposed on the actual detector signal being received; therefore, it is NOT possible to check a bistable Trip and reset point if reactor power is greater than the bistable setpoint. This test is written for performance with reactor power less than 10%, but can be performed at any power level. Notes and instructions are provided in the procedure addressing test requirements at power levels greater than the associated setpoints.*

- *Section 7.2 of 3-OSP-059.4, Power Range Nuclear Instrumentation Analog Channel Operational Test are required to be performed monthly per Tech Specs. However, credit for this monthly surveillance can be taken if the following procedures, which have been verified by Engineering to perform the same operability (ACOT) testing, have been performed SAT within the past 31 days:*

  - *3-SMI-059.08A-D Channel Calibration (once per 84 days)*
  - *3-SMI-059.09A-D Channel Standard Calibration (once per 18 months)*

- *If no gains adjustments have been made since the last determination, then credit may be taken for these equivalent I&C tests, and the OSP should be considered complete and SAT at the date/time the I&C procedure was marked as complete and SAT, before gain adjustments are made to the power range channel (whether post- calibration or during daily channel checks).*

n/a 4. **IF** a gain adjustment of the power range channel has **NOT** been performed since the last determination of NIS 100% power detector current has been performed, **THEN** this subsection is **NOT** required to be performed for operability if any of the following procedures, which have been verified by Engineering to perform the same operability (ACOT) testing, have been performed satisfactorily within the past 31 days:

- 3-SMI-059.08B, N-3-42 Channel Calibration (84 Days)
- 3-SMI-059.09B, N-3-42 Channel Standard Calibration (18 Month)

7.2.2 Procedure Steps

OF 1. Record reactor power indicated on N-42, Drawer A ____________ percent.

2. Verify the following channels are **NOT** Tripped:

OF a. The other Power Range channels.

OF b. The other ΔT channels in Protection Racks.

INIT 7.2.2 (Cont'd)

NOTE

Annunciator J 7/4, EAGLE 21 TROUBLE, alarm is expected when the bistables are placed in the TRIPPED position.

3. Place the following Protection Channel bistables in the Tripped condition at Protection Channel II, Rack No. 11, by placing its test switches in the TEST (right) position:
   a. BS-3-422B-1, Overpower ΔT Trip
   b. BS-3-422C-1, Overtemperature ΔT Trip
4. Verify the following reactor protection logic status lights (VPB) are ON:
   a. OPΔT LOOP B TC 422B1
   b. OTΔT LOOP B TC 422C1
5. At the COMPARATOR AND RATE Drawer, perform the following:
   a. Place the COMPARATOR CHANNEL DEFEAT switch to N42.
   b. Verify the COMPARATOR DEFEAT light is ON.
6. At the MISCELLANEOUS CONTROL AND INDICATION PANEL, perform the following:
   a. Place the ROD STOP BYPASS switch associated with PRN42, to BYPASS PRN42.
   b. Place the POWER MISMATCH BYPASS switch associated with PRN42 to BYPASS PRN42.
7. At the DETECTOR CURRENT COMPARATOR, (NIS panel), perform the following:
   a. Place the UPPER SECTION switch to PRN42.
      (1) Verify the CHANNEL DEFEAT light is ON.
   b. Place the LOWER SECTION switch to PRN42.
      (1) Verify the CHANNEL DEFEAT light is ON.
8. At the N-42, Drawer A, perform the following:
   a. Place the DROPPED ROD MODE switch to BYPASS.
      (1) Verify the DROPPED ROD BYPASS light is ON.
9. Verify the following:
   a. N-42 ROD DROP IN BYPASS status light (VPA) is ON.
   b. Annunciator B 8/4, NIS TRIP BYPASSED is ON.

INITIALS CK'D

7.2.2 (Cont'd)

10. At the N-42, Drawer B, perform the following:

OE a. Verify the N-42, DETECTOR A, TEST SIGNAL potentiometer is adjusted fully counterclockwise.

OE b. Verify the N-42, DETECTOR B, TEST SIGNAL potentiometer is adjusted fully counterclockwise.

OE c. Place the DETECTOR A RANGE switch to 1 MILLI-AMPS.

OE d. Place the DETECTOR B RANGE switch to 1 MILLI-AMPS.

OE e. Place the OPERATION SELECTOR switch to DET A & B.

OE f. Verify the CHANNEL ON TEST Light ON.

OE 11. Verify Annunciator B 7/3, NIS CHANNEL IN TEST, is ON.

NOTES

Substep 7.2.2.12 is conditional on reactor power level:

- ***IF*** *reactor power level is less than the P-10 setpoint (10%),* ***THEN*** *perform Substep 7.2.2.12.a and mark Substep 7.2.2.12.b N/A.*
- ***IF*** *reactor power level is greater than or equal to the P-10 setpoint (10%),* ***THEN*** *mark Substep 7.2.2.12.a N/A and perform Substep 7.2.2.12.b.*

12. Perform Interlock P-10 setpoint testing per Substep 7.2.2.12.a **OR** 7.2.2.12.b depending on reactor power level. (Mark the steps **NOT** performed N/A.)

a. For reactor power less than P-10 setpoint, 10%, perform the following:

n/a (1) Adjust N-42, DETECTOR B, TEST SIGNAL potentiometer clockwise until POWER ABOVE PERMISSIVE P10 light turns ON.

n/a (2) Record percent power indicated on N-42, Drawer A ___________ percent.

Acceptance Criteria: 10%

n/a (3) **IF** acceptance criteria is not met, **THEN** direct I&C to perform Attachment 1, Permissive P-10 Bistable Adjustment. (Mark the remaining Substeps of 7.2.2.12.a N/A)

INITIALS CK'D

7.2.2.12 (Cont'd)

n/a (4) Verify the reactor protection logic status light (VPB), HI POW RANGE P-10 NC42M is ON.

n/a (5) Adjust N-42, DETECTOR B, TEST SIGNAL potentiometer counterclockwise until POWER ABOVE PERMISSIVE P10 light turns OFF.

n/a (6) Record percent power indicated on N-42, Drawer A ___________ percent

Acceptance Criteria: 8%

n/a (7) Verify the reactor protection logic status light (VPB), HI POW RANGE P-10 NC42M is OFF.

b. For reactor power greater than or equal to P-10 setpoint, 10%, perform the following:

(1) Verify the POWER ABOVE PERMISSIVE P10 light N-42, Drawer A is ON.

(2) Verify the reactor protection logic status light (VPB), HI POW RANGE P-10 NC42M is ON.

Acceptance Criteria: P-10 is in its required state for power above 10%

NOTES

Substep 7.2.2.13 is conditional on reactor power level:

- ***IF*** *reactor power level is less than the Overpower Trip Low Range setpoint (25%),* ***THEN*** *perform Substep 7.2.2.13.a and mark Substep 7.2.2.13.b N/A.*
- ***IF*** *reactor power level is greater than or equal to the Overpower Trip Low Range setpoint (25%),* ***THEN*** *mark Substep 7.2.2.13.a N/A and perform Substep 7.2.2.13.b.*

13. Perform Overpower Trip Low Range setpoint testing per Substep 7.2.2.13.a **OR** 7.2.2.13.b depending on reactor power level. (Mark the steps **NOT** performed N/A.)

a. For reactor power less than Overpower Trip Low Range setpoint, 25%, perform the following:

n/a (1) Adjust N-42, DETECTOR B, TEST SIGNAL potentiometer (NIS panel N-42B) clockwise until OVERPOWER TRIP LOW RANGE light (NIS panel N42A) turns ON.

INIT — 7.2.2.13 (Cont'd)

n/a (2) Record percent power indicated on N-42, Drawer A ___________ percent

Acceptance Criteria: 25% (24.4% to 25.6%)

n/a (3) **IF** Acceptance Criteria is **NOT** met, **THEN** direct I&C to perform Attachment 2. (Mark the remaining Substeps of 7.2.2.13.a N/A.)

n/a (4) Verify the Reactor Protection Logic status light (VPB), Lo Pow Range Hi Flux NC42P, is On.

n/a (5) Verify annunciator B 6/2, POWER RANGE SINGLE CHNL LO RANGE ALERT, is On.

n/a (6) Adjust N-42, Detector B, Test Signal potentiometer, counterclockwise until Overpower Trip Low Range light turns Off.

n/a (7) Record percent power indicated on N-42, Drawer A ___________ percent

Acceptance Criteria: 22% to 24%

n/a (8) Verify the reactor protection logic status light (VPB), Lo Pow Range Hi Flux NC42P, is Off.

n/a (9) Verify annunciator B 6/2, POWER RANGE SINGLE CHNL LO RANGE ALERT, is Off.

b. For reactor power greater than or equal to Overpower Trip Low Range Setpoint, 25%, perform the following:

DL (1) Verify the Overpower Trip Low Range light is On.

DL (2) Verify the reactor protection logic status light (VPB), Lo Pow Range Hi Flux NC42P, is On.

Acceptance Criteria: Overpower Trip Low Range is in its required state for power greater than 25%

INITIALS
CK'D

7.2.2 (Cont'd)

n/a

14. Adjust N-42, DETECTOR B, TEST SIGNAL potentiometer clockwise to obtain 40% indication. (Mark N/A if reactor power is greater than or equal to 40%.)

NOTES

Substep 7.2.2.15 is conditional on reactor power level:

- ***IF*** *reactor power level is less than the P-8 setpoint (45%),* ***THEN*** *perform Substep 7.2.2.15.a and mark Substep 7.2.2.15.b N/A.*
- ***IF*** *reactor power level is greater than or equal to the P-8 setpoint (45%),* ***THEN*** *mark Substep 7.2.2.15.a N/A and perform Substep 7.2.2.15.b.*

15. Perform Interlock P-8 setpoint testing per Substep 7.2.2.15.a **OR** 7.2.2.15.b depending on reactor power level. (Mark the steps **NOT** performed N/A.)

a. For reactor power less than P-8 setpoint, 45%, perform the following:

n/a (1) Adjust N-42, DETECTOR A, TEST SIGNAL potentiometer clockwise until POWER ABOVE PERMISSIVE P8 light turns ON.

n/a (2) Record percent power indicated on N-42, Drawer A ___________ percent

Acceptance Criteria: 45% (44% to 45%)

n/a (3) **IF** acceptance criteria is not met, **THEN** direct I&C to perform Attachment 3, Permissive P-8 Bistable Adjustment. (Mark the remaining Substeps of 7.2.2.15.a N/A)

n/a (4) Verify the reactor protection logic status light (VPB), LO-POW RANGE P-8 NC42N is ON.

n/a (5) Adjust DETECTOR A, TEST SIGNAL potentiometer counterclockwise until POWER ABOVE PERMISSIVE P8 Light turns OFF.

n/a (6) Record percent power indicated on N-42, Drawer A ___________ percent

Acceptance Criteria: 42% to 44%

n/a (7) Verify the reactor protection logic status light (VPB), LO-POW RANGE P-8 NC42N is OFF.

INITIALS CK'D

7.2.2.15 (Cont'd)

b. For reactor power greater than or equal to P-8 setpoint, 45%, perform the following:

(1) Verify the POWER ABOVE PERMISSIVE P8 light is ON.

(2) Verify the reactor protection logic status light (VPB), LO-POW RANGE P-8 NC42N is ON.

Acceptance Criteria: P-8 is in its required state for power above 45%

16. Perform Overpower Rod Stop setpoint, 103%, testing as follows:

a. Adjust N-42, DETECTOR A, TEST SIGNAL potentiometer clockwise until OVERPOWER ROD STOP light turns ON.

b. Record percent power N-42, Drawer A ____________ percent

Acceptance Criteria: 103% (102% to 104%)

c. Adjust N-42, DETECTOR A, TEST SIGNAL potentiometer counterclockwise until OVERPOWER ROD STOP light turns OFF.

d. Record percent power N-42, Drawer A ____________ percent

Acceptance Criteria: 100% to 102%

n/a

e. **IF** Acceptance Criteria is not met, **THEN** have I&C perform Attachment 5, Overpower Rod Stop Bistable Adjustment.

INIT 7.2.2 (Cont'd)

17. Perform Overpower Trip High Range setpoint, 108%, testing as follows:

NOTE

The Overpower Trip High Range Setpoint may be set at various setpoints when performing a startup. The setpoints will be lowered for Initial Criticality and Core Mapping after refueling based on the directions from the Reactor Engineering Supervisor or based on the number of Main Steam Safety Valves on any S/G that are out of service when operating with a positive MTC.

JE a. Adjust N-42, Detector A, Test Signal potentiometer clockwise until Overpower Trip High Range light turns On.

JE b. Record percent power N-42, Drawer A ____________ percent

Acceptance Criteria: The Unit 3 Plant Curve Book, Section 5, Figure 5A, Setpoint value plus or minus 0.6%

n/a c. **IF** Acceptance Criteria are **NOT** met, **THEN** direct I&C to perform Attachment 4. (Mark the remaining Substeps of 7.2.2.17 N/A)

JE d. Verify the Reactor Protection Logic status light (VPB), Hi Pow Range Hi Flux NC42R, is On.

JE e. Verify Annunciator B 6/1, POWER RANGE SINGLE CHNL HI RANGE ALERT, is On.

JE f. Adjust N-42, Detector A, Test Signal potentiometer counterclockwise until Overpower Trip High Range Light turns Off.

JE g. Record percent power indicated on N-42, Drawer A ____________ percent

Acceptance Criteria: The Unit 3 Plant Curve Book, Section 5, Figure 5A, Reset Point plus or minus 0.6%

JE h. Verify the Reactor Protection Logic status light (VPB), Hi Pow Range Hi Flux NC42R, is Off.

JE i. Verify Annunciator B 6/1, POWER RANGE SINGLE CHNL HI RANGE ALERT, is Off.

n/a j. Place Caution Tag and/or Information Placard near the Power Range NIS meter face with the information specified in Subsection 4.12. (N/A if Hi Flux Trip Setpoint is 108%, plus or minus 0.6%.)

INITIALS
CK'D

7.2.2 (Cont'd)

18. Perform Dropped Rod Rod Stop setpoint testing as follows:

a. Adjust N-42, DETECTOR A, TEST SIGNAL potentiometer fully counterclockwise.

b. Place the OPERATION SELECTOR switch to DET B.

c. Adjust N-42, DETECTOR B, TEST SIGNAL potentiometer to obtain a meter reading of 50% (102% if power level greater than 45%) on N-42, Drawer A.

d. Place the OPERATION SELECTOR switch to DET A & B.

e. Adjust N-42, DETECTOR A, TEST SIGNAL potentiometer to obtain a meter reading of 54% (106% if power level greater than 45%) on N-42, Drawer A.

(1) Wait for 30 seconds before proceeding.

f. Place the OPERATION SELECTOR switch to DET B.

(1) Verify the DROPPED ROD ROD STOP light is OFF.

g. Place the OPERATION SELECTOR switch to DET A & B.

h. Adjust the N-42, DETECTOR A, TEST SIGNAL potentiometer to obtain a meter reading of 56% (108% if power level greater than 45%) on N-42, Drawer A.

(1) Wait for 30 seconds before proceeding.

NOTE

Dropped Rod Rod Stop indicator light will only be on momentarily for verification.

i. Place the OPERATION SELECTOR switch to DET B.

(1) Verify the DROPPED ROD ROD STOP light turns ON momentarily.

j. Adjust N-42, DETECTOR A, TEST SIGNAL potentiometer fully counterclockwise.

k. Adjust N-42, DETECTOR B, TEST SIGNAL potentiometer fully counterclockwise.

INITIALS
CK'D

7.2.2 (Cont'd)

19. Perform testing of Δ FLUX ALARM setpoints as follows:

NOTE

Two Operators should be used in the performance of this subsection, one at the NIS panel and the other observing meter and alarm responses.

CAUTION

Test signals shall be adjusted slowly to ensure accurate readings.

a. Place the OPERATION SELECTOR switch (NIS Panel N-42B) to DET A.

b. Slowly adjust N-42, DETECTOR A, TEST SIGNAL potentiometer clockwise until the following responses are obtained at +10% (+9% to +11%) ΔFLUX:

(1) Annunciator B 9/2, AXIAL FLUX TILT, turns ON.

(2) Percent FLUX DIFFERENCE PR#2, NI-3-42C (console), indicates +10% (+9% to +11%).

c. Adjust N-42, DETECTOR A, TEST SIGNAL potentiometer fully counterclockwise.

(1) Lock N-42, DETECTOR A, TEST SIGNAL potentiometer.

d. Place the OPERATION SELECTOR switch to DET B.

e. Slowly adjust N-42, DETECTOR B, TEST SIGNAL potentiometer clockwise until the following responses are obtained at -10% (-9% to -11%) ΔFLUX:

(1) Annunciator B 9/2, AXIAL FLUX TILT, turns ON.

(2) Percent FLUX DIFFERENCE PR#2, NI-3-42C (console), indicates -10% (-9% to -11%)

f. Adjust N-42, DETECTOR B, TEST SIGNAL potentiometer (NIS panel N-42B) fully counterclockwise.

(1) Lock N-42, DETECTOR B, TEST SIGNAL potentiometer.

INITIALS
CK'D VERIF

7.2.2 (Cont'd)

20. At N-42, Drawer B, perform the following:

    a. Place the DETECTOR A, RANGE switch as required to maintain detector current indication on scale.

    b. Place the DETECTOR B, RANGE switch as required to maintain detector current indication on scale.

21. Verify the percent power indicated on N-42, Drawer A is within 1% of the other channels.

22. At the N-42, Drawer B, perform the following:

    a. Place the OPERATION SELECTOR switch to NORMAL.

    b. Verify the CHANNEL ON TEST light is OFF.

23. Verify Annunciator B 7/3, NIS CHANNEL IN TEST, is OFF.

24. At the N-42, Drawer A, perform the following:

    a. Verify the DROPPED ROD ROD STOP Light is OFF.

    b. Place the DROPPED ROD MODE switch to NORMAL.

25. Verify the N-42 ROD DROP IN BYPASS status light (VPA) is OFF.

26. Verify Annunciator B 8/4, NIS TRIP BYPASSED, is OFF. Mark N/A if Annunciator B8/4 is ON due to another NIS channel in BYPASS.

27. At the COMPARATOR AND RATE Drawer, perform the following:

    a. Place the COMPARATOR CHANNEL DEFEAT switch to NORMAL.

    b. Verify COMPARATOR DEFEAT light is OFF.

28. At the MISCELLANEOUS CONTROL AND INDICATION PANEL (NIS panel), perform the following:

    a. Place the ROD STOP BYPASS switch associated with PRN42 to OPERATE.

    b. Place the POWER MISMATCH BYPASS switch associated with PRN42 to OPERATE.

INITIALS
CK'D VERIF

7.2.2 (Cont'd)

29. At the DETECTOR CURRENT COMPARATOR panel, perform the following:

____ ____ a. Place the UPPER SECTION defeat switch to NORMAL.

____ ____ (1) Verify CHANNEL DEFEAT light is OFF.

____ ____ b. Place the LOWER SECTION defeat switch to NORMAL.

____ ____ (1) Verify CHANNEL DEFEAT light is OFF.

30. At Protection Channel II, Rack No. 11, place the Protection Channel bistable test switches in the NORMAL (Left) position:

____ ____ a. BS-3-422B-1, Overpower ΔT Trip

____ ____ b. BS-3-422C-1, Overtemperature ΔT Trip

31. Verify the following reactor protection logic status lights (VPB) are OFF:

____ ____ a. OPΔT LOOP B TC422B1

____ ____ b. OTΔT LOOP B TC422C1

____ ____ 32. Verify Acceptance Criteria specified in Subsection 6.1 has been satisfied.

____ a. **IF** the Acceptance Criteria has been met **AND** the channel had been out of service prior to performing this procedure, **THEN** notify Reactor Engineering.

b. **IF** Acceptance Criteria is not met, **THEN** perform the following:

____ (1) Declare the channel inoperable.

____ (2) Perform actions required by 3-ONOP-059.8, POWER RANGE NUCLEAR INSTRUMENTATION MALFUNCTION.

____ (3) Notify Reactor Engineering.

____ 33. Perform Attachment 7, N-42 Status Light, Annunciator, and Reactor Protection Logic Status Lights Verification.

INITIALS
CK'D

7.2.2 (Cont'd)

____ 34. Verify all log entries specified in Subsection 2.2 have been recorded.

REMARKS: ____

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Date/Time Completed:__________________

PERFORMED BY (Print)	INITIALS	VERIFIED BY (Print)	INITIALS
______________________	__________	______________________	__________
______________________	__________	______________________	__________
______________________	__________	______________________	__________

REVIEWED BY:____ Date:__________________
Shift Manager or SRO Designee

ATTACHMENT 7
(Page 1 of 2)

N-42 STATUS LIGHT, ANNUNCIATOR AND REACTOR PROTECTION LOGIC STATUS LIGHTS VERIFICATION

Annunciator Panel 3B

Annunciator	Acceptance Criteria	Initials
B 6/1 POWER RANGE SINGLE CHNL HI RANGE ALERT	OFF	
B 6/2 POWER RANGE SINGLE CHNL LO RANGE ALERT	OFF	
B 7/3 NIS CHANNEL IN TEST	OFF	
B 8/4 NIS TRIP BYPASSED	OFF (Note 1)	

Status Lights (VPA)

Status Light	Acceptance Criteria	Initials
N-42 ROD DROP IN BYPASS	OFF	
POWER ABOVE P-6	POWER > P-6 ON	
	POWER < P-6 OFF	
POWER ABOVE P-10	POWER > P-10 ON	
	POWER < P-10 OFF	
POWER BELOW P-8	POWER > P-8 OFF	
	POWER < P-8 ON	
10^5 CPS TRIP BLOCKED	POWER > P-6 ON	
	POWER< P-6 OFF	
25% PWR RNG TRIP BLOCKED	POWER > P-10 ON	
	POWER < P-10 OFF	
25% INTER RNG TRIP BLOCKED	POWER > P-10 ON	
	POWER < P-10 OFF	
AT POWER TRIPS BLOCKED	POWER > P-10 OFF	
	POWER < P-10 ON	

Reactor Protection Logic Status Lights (VPB)

Reactor Protection Logic Status Light	Acceptance Criteria	Initials
OPΔT LOOP B TC 422 B1	OFF	
OTΔT LOOP B TC 422 C1	OFF	
HI POW RANGE P-10 NC 42 M	POWER > P-10 ON	
	POWER < P-10 OFF	
LO POW RANGE HI FLUX NC 42 P	POWER > 25% ON	
	POWER < 25% OFF	
LO POW RANGE P-8 NC 42 N	POWER > P-8 ON	
	POWER < P-8 OFF	
HI POW RANGE HI FLUX NC 42 R	OFF	

Note 1: Annunciator B8/4 may be ON if another NIS channel is in BYPASS.

ATTACHMENT 7
(Page 2 of 2)

N-42 STATUS LIGHT, ANNUNCIATOR AND REACTOR PROTECTION LOGIC STATUS LIGHTS VERIFICATION

Nuclear Instrumentation System Panel

NIS Light	Acceptance Criteria	Initials
CONTROL POWER ON	ON	
LOSS OF DETECTOR VOLTAGE	OFF	
OVERPOWER TRIP HIGH RANGE	OFF	
OVERPOWER ROD STOP	OFF	
OVERPOWER TRIP LOW RANGE	POWER > P-10 ON	
	POWER < P-10 OFF	
POWER ABOVE PERMISSIVE P-10	POWER > P-10 ON	
	POWER < P-10 OFF	
POWER ABOVE PERMISSIVE P-8	POWER > P-8 ON	
	POWER < P-8 OFF	
DROPPED ROD ROD STOP	OFF	
DROPPED ROD BYPASS	OFF	
INSTRUMENT POWER ON	ON	
CHANNEL ON TEST	OFF	

REMARKS: ____

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Performed By: ______________________ ______________________ ____________
(Signature) *(Print)* *(Date)*

Reviewed By: ______________________ ______________________ ____________
Shift Manager or SRO Designee *(Print)* *(Date)*

L-16-1 NRC Exam

Control Room - JPM H

JPM TITLE: Respond To Control Room Evacuation Condition – Unit 3 RO

JPM NUMBER: 01200011301 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 01200011300 / Respond To Control Room Evacuation Condition – Unit 3 RO

K/A NUMBERS: APE 068 AA1.23 **K/A VALUE:** RO 4.3 / SRO 4.4

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]

Simulator: [X] Other: [ ]

Lab: [ ]

Time for Completion: 10 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Brian Clark** Instructor/Developer	**6/20/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/21/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
1-0	New JPM	Update for 2014 Annual Exam	01982473	N/A N/A	N/A N/A
2-0	Formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A N/A	N/A N/A

SIMULATOR SET-UP:

SIMULATOR SETUP INSTRUCTIONS:

______	1.	Reset to **IC 1** or equivalent IC.
______	2.	Place simulator in RUN.
______	3.	Ensure applicable portions of Simulator Operator Checklist are complete.
______	4.	Acknowledge alarms and place simulator in FREEZE.
______	5.	Save as temporary IC, if JPM will be repeated.
______	6.	When ready to begin, then place Simulator in RUN.

SIMULATOR MALFUNCTIONS:

- N/A

SIMULATOR OVERRIDES:

- N/A

SIMULATOR REMOTE FUNCTIONS:

- N/A

Required Materials:	• Handout Attachment 14
General References:	• 0-ONOP-105, Control Room Evacuation • 3-EOP-E-0, Reactor Trip or Safety Injection
Task Standards:	• Trip Unit 3 reactor • Close Unit 3 MSIVs • Trip Unit 3 Main Feedwater Pumps • Place Unit 3 SDTA controllers in manual and close the valves • Close both PORV block valves • Trip the Unit 3 RCPs

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Both units are at 100% power.
- A fire in the North-South Breezeway has compromised Control Room habitability.

INITIATING CUE:

- The Shift Manager/Unit Supervisor directs you to perform the Unit 3 Reactor Operator immediate actions as required by Attachment 14 of 0-ONOP-105, Control Room Evacuation.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain Attachment 14 of 0-ONOP-105, Control Room Evacuation.
Evaluator Note:	Although the following steps are immediate actions and normally performed from memory, the examinee may use the procedure during the performance of this JPM. If the procedure is NOT used, mark this step N/A.
Evaluator Cue:	Provide operator with a copy of handout Attachment 14.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: Yes**	**0-ONOP-105, Attachment 14, Step 1:** **Perform the following:** A. TRIP Unit 3 Reactor
Standard:	Trip the reactor at the operator console or VPB and verify the following: • Rod Bottom Lights – ON • Reactor Trip AND Bypass Breakers – OPEN • Rod Position Indicators – AT ZERO • Neutron flux – DECREASING
Evaluator Note:	Only the switch manipulation is critical.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: No**	**0-ONOP-105, Attachment 14, Step 1:** **Perform the following:** B. TRIP Unit 3 Main Turbine

Standard:	Trip the main turbine at the operator console and verify the following:
	• All Turbine Stop OR associated Control Valves – CLOSED
	• Moisture Separator Reheater Steam Valves – CLOSED:
	  • MSR Main Steam Supply Stop MOVs
	  • Reheater Timing Valves
	  • MSR Purge Steam Valves
	• Mid and East GCBs – OPEN (Normally 30-second delay)

Evaluator Note:	Only the pushbutton manipulation is critical.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: Yes**	**0-ONOP-105, Attachment 14, Step 2:** **CLOSE Unit 3 MSIVs and Bypass Valves**
Standard:	Close the Unit 3 MSIVs and their bypass valves.
Evaluator Note:	Only the MSIV closures are critical.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: Yes**	**0-ONOP-105, Attachment 14, Step 3:** **PERFORM as many of the following Manual Actions as possible <u>prior</u> to leaving the Control Room:** A. TRIP Unit 3 Main Feedwater Pumps • 3A Main Feedwater Pump • 3B Main Feedwater Pump B. TRIP A Standby S/G Feedwater Pump C. PLACE Unit 3 Steam Dump to Atmosphere Controllers in MANUAL and CLOSE the Steam Dump Valves • CV-3-1606 • CV-3-1607 • CV-3-1608 D. ENSURE 3B Charging Pump TRIPPED E. CLOSE <u>both</u> PORV Block Valves • MOV-3-536 • MOV-3-535 F. TRIP Unit 3 Reactor Coolant Pumps • 3A RCP • 3B RCP • 3C RCP G. OBTAIN the following: • Set of prints • Radio
Standard:	Trip the main feedwater pumps, trip the A SSGFP, manually close the SDTAs, verify that the 3B Charging Pump is tripped, close the PORV block valves, trip the RCPs, and obtain a set of prints and a radio.
Evaluator Note:	Only the main feedwater pump trips, SDTA closures, PORV block valve closures, and RCP trips are critical.
Evaluator Cue:	If asked how many steps must be completed, state "There is no immediate danger; complete all steps."
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: No**	**0-ONOP-105, Attachment 14, Step 4:** **EVACUATE Control Room as follows:** A. PROCEED to Turbine Deck Work Station
Standard:	Evacuate the Control Room and report to the Turbine Deck Work Station.
Performance:	**SATISFACTORY _____ UNSATISFACTORY _____**
Comments:	

Terminating Cue: **When the examinee begins to exit the Control Room, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: __________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Both units are at 100% power.
- A fire in the North-South Breezeway has compromised Control Room habitability.

INITIATING CUE:

- The Shift Manager/Unit Supervisor directs you to perform the Unit 3 Reactor Operator immediate actions as required by Attachment 14 of 0-ONOP-105, Control Room Evacuation.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

FPL	**TURKEY POINT PLANT** **OFF NORMAL OPERATING PROCEDURE** SAFETY RELATED **CONTINUOUS USE**	Procedure No. **0-ONOP-105** Revision No. **16**

Title:

CONTROL ROOM EVACUATION

Responsible Department: OPERATIONS

Special Considerations:

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED today INITIAL

Revision	Approved By	Approval Date
3	Randy Flynn	07/29/12
16	Sam Shafer	04/19/16

UNIT #	
DATE	
DOCT	PROCEDURE
DOCN	0-ONOP-105
SYS	
STATUS	COMPLETED
REV	16
# OF PGS	

REVISION SUMMARY	
Rev. No.	**Description**
16	**PCR 1984699, 04/19/16, Christopher Roda** Note added to attachments 14 and 15 to adress loss of instrument air for AFW and Step added to Attachment 20 to secure Unit 4 Seal Injection. **EC 282069**
15	**PCR 2022458, 02/19/16, Terry White** Revised to incorporate changes identified in **EC 280401**, Unit 4 RCP Seal Upgrade Project. **PCR 2105184, 01/29/16, Jim Speicher** Editorial correction to unit designation for valves in Attachment 22 (Unit 3 Cool Down from ASP) Step 11c. MOV-4-843B, MOV-4-869, MOV-4-866A, MOV-4-866B should all be Unit 3 designators.
14	**PCR 2048169, 12/04/15, Michael Hargis** Revised Per **EC 283697**, which is providing a configuration update to support the demolition of the original Water Treatment Plant.
13	**PCR 2068196,10/28/15, Michael Lambert** **EC 280399** removed requirement for 13 minute actions for RCP thermal barrier cooling. Added requirement for RCPs to be secured within one hour after loss of seal cooling. Also changed name "Seal Leakoff" to "Control Bleed Off." **PCR 2051305, 09/18/15, Terry White** Revise Note prior to Step 1 in Attachment 22 and Note prior to Step 15 in Attachment 24 to incorporate nomenclature changes per **EC 280399**, Unit 3 RCP Seals Upgrade Project.
12	**PCR 2033530, Joseph Turek, 03/20/15** **CR 2028300** Added attachment providing redundant field indications in event ASP indications are not working and references to same attachment in body of procedure.
11	**AR 01954959, 09/11/14, Gerard T Slaby** Corrected KW for Bearing Lift Oil Pump in Attachments 5 and 6 to agree with 5613/4-E-6 and deleted Oil Vapor Extractor. Basis Doc not affected.

TABLE OF CONTENTS

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(continued)

1.0 PURPOSE

1. This procedure provides specific operating instructions in the event of a Control Room evacuation for any reason including fire in vital areas.

2. The following assumptions apply to this procedure.

   - A loss of Off-Site Power may occur anytime within the first 72 hours.
   - All alternate shutdown protected components will be OPERABLE throughout the event.
   - The use of non-protected components is **NOT** required to achieve cold shutdown, however they may be used if free of damage.

3. Instructions for the following plant evolutions are provided in this procedure:

   - Achieve and maintain subcritical reactivity conditions in the reactor.
   - Maintain reactor coolant inventory.
   - Achieve and maintain Hot Standby conditions.
   - Achieve Cold Shutdown conditions within 72 hours.
   - Maintain Cold Shutdown conditions thereafter.

4. Entry conditions assume each Unit to be in any MODE from Refueling to Full Power.

2.0 ENTRY CONDITIONS

1. Fire in any of the following areas:
   - Control Room, Fire Zone 106
   - HVAC Equipment Room, Fire Zone 097
   - Control Room Roof, Fire Zone 106R
   - Cable Spreading Room, Fire Zone 098
   - North-South Breezeway, Fire Zone 079A
   - Units 3 and 4 Control Room Electrical Chase, Fire Zone 132
2. A loss of, a potential loss of, or unreliable operation of Control Room controls and indicators exists.
3. Possible spurious actuation of any plant circuitry exposed to a fire.
4. Personnel safety necessitates evacuation of the Control Room for any of the following conditions:
   - Moderate level of smoke (without fire) from any source
   - Toxic gas
   - Confirmable bomb threat
   - Radiation
   - Other life threatening conditions, as determined by the Shift Manager or SRO designee cause the Control Room to be rendered uninhabitable.
5. The following are AUTOMATIC ACTIONS that COULD occur based on the initiating event requiring Control Room evacuation:
   - Activation of any Fire Suppression System in the Cable Spreading Room or N-S Breezeway.
   - Fast transfer from the Auxiliary Transformer to the Startup Transformer.
   - AUTO-START of EDGs and energizing of associated 4KV Buses.

3.0 ACTIONS

NOTE

- It is recommended that non-protected equipment **NOT** be loaded onto the *B EDGs since fire damage to the circuits for such equipment can **NOT** be prevented or protected. However, if non-protected equipment is loaded onto the *B EDGs, the following minimum checks are required to prevent EDG overloading:
  - The plant shall first be stabilized with all required loads for alternate shutdown operations.
  - The circuits for non-protected equipment shall be checked for possible fire damage before loading onto the EDG to ensure the equipment can be unloaded when necessary.
- EOPs are **NOT** applicable during Control Room evacuation. They should be used for information only or as directed by the TSC while performing this procedure.
- This procedure is written for the plant initially in MODE 1, 2, or 3. If the plant is in MODE 4, 5, or 6, only those steps to restore RHR cooling and to stabilize plant systems after evacuation are necessary.
- If installed emergency lighting is **NOT** adequate, portable emergency lights may be used.
- Compliance with 0-ADM-744 Electrical Arc Flash Personal Protective Equipment is required prior to opening a cubicle door on any 4KV BUS unless special permission is obtained from the Shift Manager.

3.1 Immediate Actions

3.1.1 Shift Manager

- **PERFORM** Attachment 12, Step 1 and Attachment 12, Step 2.

3.1.2 Unit Supervisor

- **PERFORM** Attachment 13, Step 1.

3.1.3 Unit 3 Reactor Operator

- **PERFORM** Attachment 14, Step 1 through Attachment 14, Step 4.

3.1.4 Unit 4 Reactor Operator

- **PERFORM** Attachment 15, Step 1.B through Attachment 15, Step 4.

3.1.5 Third Licensed Reactor Operator

- **PERFORM** Attachment 16, Step 1 and Attachment 16, Step 2.

3.1.6 Non-Fire Brigade Nuclear Plant Operator

- **PERFORM** Attachment 17, Step 1 through Attachment 17, Step 6.

3.1.7 Non-Fire Brigade Number 1 Senior Nuclear Plant Operator (Outside SNPO)

- **PERFORM** Attachment 18, Step 1 and Attachment 18, Step 2.

3.1.8 Non-Fire Brigade Number 2 Senior Nuclear Plant Operator (Inside SNPO)

- **PERFORM** Attachment 20, Step 1 and Attachment 20, Step 2.

3.1.9 Fire Brigade Members

- **RESPOND** to nearest fire equipment locker.

3.1.10 Shift Technical Advisor

- **ASSIST** Shift Manager in TSC.

NOTE

- If radio frequency interference is affecting Alternate Shutdown Panel instrumentation, the use of radios shall be discontinued.
- Dedicated Alternate Shutdown Headsets should be disconnected when leaving an area to prevent excessive background noise on the communication circuit.

3.1.11 Unit ROs

- **ESTABLISH** communication using the dedicated Alternate Shutdown Communications Headsets located in the locked Dedicated Alternate Shutdown Communications Boxes.

3.2 Subsequent Actions

1. Shift Manager

A. **COMPLETE** Attachment 12, Shift Manager.

B. **COORDINATE** plant operations from TSC.

2. Unit Supervisor

A. **COMPLETE** Attachment 13, Unit Supervisor.

B. **PERFORM** Attachment 24, Maintaining a Safe, Stable Configuration Following Control Room Evacuation.

3. Unit 3 Reactor Operator

- **COMPLETE** Attachment 14, Unit 3 Reactor Operator.

4. Unit 4 Reactor Operator

- **COMPLETE** Attachment 15, Unit 4 Reactor Operator.

5. Third Licensed Reactor Operator

- **COMPLETE** Attachment 16, Third Licensed Reactor Operator.

6. Non-Fire Brigade Nuclear Plant Operator

- **COMPLETE** Attachment 17, Non-Fire Brigade Nuclear Plant Operator.

7. Non-Fire Brigade Number 1 Senior Nuclear Plant Operator (Outside SNPO)

- **COMPLETE** Attachment 18, Non-Fire Brigade Number 1 Senior Nuclear Plant Operator (Outside SNPO).

8. Non-Fire Brigade Number 2 Senior Nuclear Plant Operator (Inside SNPO)

- **COMPLETE** Attachment 20, Non-Fire Brigade Number 2 Senior Nuclear Plant Operator (Inside SNPO).

9. Shift Technical Advisor

- **ASSIST** Shift Manager in TSC.

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 1 of 28)

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

NOTE

- Attachment 14, Step 1 through Attachment 14, Step 4 are IMMEDIATE ACTION steps.
- Attachment 14, Step 2 should be completed within 5 minutes.
- Attachment 14, Step 1 through Attachment 14, Step 22 should be completed within 15 minutes.
- Attachment 25, Redundant Field Indication for Alternate Shutdown Panel Indication, should be referred to if Alternate Shutdown Panel indication is not working.

1. **PERFORM** the following:

   A. **TRIP** Unit 3 Reactor.

   B. **TRIP** Unit 3 Main Turbine.

2. **CLOSE** Unit 3 MSIVs and Bypass Valves.

3. **PERFORM** as many of the following Manual Actions as possible prior to leaving the Control Room:

   A. **TRIP** Unit 3 Main Feedwater Pumps.

   B. **TRIP** A Standby S/G Feedwater Pump.

   C. **PLACE** Unit 3 Steam Dump to Atmosphere Controllers in MANUAL and **CLOSE** the Steam Dump Valves.

   D. **ENSURE** 3B Charging Pump TRIPPED.

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**(continued)**	
	E. CLOSE both PORV Block Valves: • MOV-3-536 • MOV-3-535	
	F. TRIP Unit 3 Reactor Coolant Pumps.	
	G. OBTAIN the following: • Set of prints • Radio	**G. USE** prints available from TSC.
4.	**EVACUATE** Control Room as follows:	
	A. PROCEED to Turbine Deck Work Station.	**A. PROCEED** to Work Control Center.
	B. OBTAIN the following: • One copy of this procedure • One high voltage kit	
5.	**PROCEED** to Unit 3 ASP.	
6.	**PERFORM** the following:	
	A. OPEN key box on side of ASP.	
	B. Using Alt Comm Key, **OPEN** Alternate Shutdown Communications Headset Box.	

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**(continued)**	
	C. OBTAIN 12 Transfer Switch handles.	
	D. TURN DC lights ON.	

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 4 of 28)

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

NOTE

- AFW Flow and Steam Dump Controllers are set to zero output (closed valves), therefore when placing these transfer switches to LOCAL, AFW and Steam Dump Valves will CLOSE, requiring reopening using the respective controllers (Manual only).
- Transfer switches listed in Attachment 14, Step 7.A are identified with yellow border.
- The Third Licensed RO is available to restart C AFW Pump if necessary.

CAUTION

The AFW Pump may TRIP when placing AFW Pump T&T Valve Transfer Switch to LOCAL.

7. At the ASP, **PERFORM** the following:

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

7. (continued)

A. INSERT Transfer Switch handles into the following 10 ASP yellow-bordered switches and **PLACE** to LOCAL:

- MOV-3-535, PRZ PORV BLOCK VALVE
- PCV-3-455C, PRZ PORV
- U3 AFW FCV'S CONTROL TRANSFER SWITCH
- LCV-3-460, HIGH PRESS L/D ISOL VLV FROM LOOP B COLD LEG
- CV-3-387, EXCESS L/D ISOL FROM COLD LEG TO EXCESS L/D HX
- MOV-3-1403, 3A STM SUPPLY TO AFW PUMPS
- MOV-6459C, C AFW PUMP T&T VALVE
- POV-3-2604, 3A MAIN STEAM ISOLATION VALVE
- POV-3-2605, 3B MAIN STEAM ISOLATION VALVE
- POV-3-2606, 3C MAIN STEAM ISOLATION VALVE

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	**ENSURE** the RCS is ISOLATED as follows:	
	A. **CHECK** MOV-3-535, PRESSURIZER PORV BLOCK VALVE, is CLOSED.	A. **ENSURE** PCV-3-456, PRZ PORV, is CLOSED.
	B. **CHECK** PCV-3-455C, PRESSURIZER PORV, is CLOSED.	B. **ENSURE** MOV-3-536, PRZ PORV BLOCK VALVE, is CLOSED.
	C. **CHECK** LCV-3-460, HIGH PRESS L/D ISOL VLV FROM LOOP B COLD LEG, is CLOSED.	
	D. **CHECK** CV-3-387, EXCESS L/D ISOL VLV FROM COLD LEG TO EXCESS L/D HX, is CLOSED.	
9.	**DON** Alternate Shutdown Communication System Headset and **NOTIFY** Third Licensed Operator at AFW Cage that MOV-6459C, C AFW PUMP T&T VALVE, is in LOCAL.	

NOTE

- If C AFW Pump can **NOT** be started, AFW flow would still be possible if B AFW Pump (Unit 4 ASP) is operating.
- If only <u>one</u> AFW Pump is available for <u>both</u> Units and <u>both</u> Units require AFW flow, flow to Unit 3 should **NOT** exceed 340 gpm.

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 7 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	**ALIGN** C AFW Pump as follows:	
	A. **CHECK** either Unit in MODE 1, 2, or 3.	A. IF S/Gs **NOT** required for heat sink on either Unit, THEN: (1) **ENSURE** MOV-6459C, C AFW PUMP T&T VALVE, is CLOSED. (2) **ENSURE** MOV-3-1403, 3A STM SUPPLY TO AFW PUMPS, is CLOSED. (3) **ENSURE** MOV-3-1404, 3B STM SUPPLY TO AFW PUMPS, is CLOSED. (4) **GO TO** Attachment 14, Step 11.
	B. **CHECK** MOV-6459C, C AFW PUMP T&T VALVE, is OPEN.	B. **OPEN** MOV-6459C, C AFW PUMP T&T VALVE.
	C. **CHECK** MOV-3-1403, 3A STM SUPPLY TO AFW PUMPS, is OPEN.	C. **OPEN** MOV-3-1403, 3A STM SUPPLY TO AFW PUMPS.

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE It is **NOT** necessary to place Spare transfer switches to LOCAL or ISOLATE.	
11.	At the ASP, **PERFORM** the following:	
	A. **PLACE** the remaining 18 Transfer and Isolate switches to LOCAL or ISOLATE.	
	B. **TURN** chart recorders ON.	
12.	**LIMIT** flow from S/Gs:	
	A. **CHECK** all MSIVs CLOSED.	**A.** **CLOSE** all MSIVs.
	B. **CHECK** MSIV BYPASS Valves were CLOSED prior to Control Room evacuation.	**B.** **DIRECT** Third Licensed RO to ENSURE MSIV Bypass Valves CLOSED.
	C. **CHECK** S/G Blowdown Flow Control Valves CLOSED: • FCV-3-6278A, 3A BLOWDOWN FLOW CONTROL VALVE • FCV-3-6278B, 3B BLOWDOWN FLOW CONTROL VALVE • FCV-3-6278C, 3C BLOWDOWN FLOW CONTROL VALVE	**C.** **NOTIFY** TSC to direct any available operator to locally ISOLATE all S/G Blowdown Flow Control Valves.

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Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- IF required and if available, P82B (STANDBY STEAM GENERATOR FEED PUMP B) may be locally operated. A loss of instrument air would require local manual operation of FCV-3-479, FCV-3-489, and FCV-3-499 (FEEDWATER FLOW BYPASS VALVES) to support P82B use.
- For control of AFW valves, use appropriate procedures for aligning nitrogen bottles as necessary if instrument air is lost.

CAUTION

Excessive RCS cool down could result if AFW flow is **NOT** carefully controlled.

13. ESTABLISH Secondary Heat Sink:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A. CHECK Unit 3 in MODE 1, 2, or 3.	**A.** IF S/Gs **NOT** required for heat sink, THEN: **(1) CLOSE** <u>all</u> Manual AFW Flow Controllers. **(2) GO TO** Attachment 14, Step 14.
B. CONTROL AFW flow using Manual AFW Flow Controllers.	**B. PERFORM** the following: **(1) NOTIFY** TSC that local control of AFW flow is required. **(2) DIRECT** local operation of Train 2 AFW Control Valves.
C. MAINTAIN S/G Wide Range levels between 20% and 50% Narrow Range Equivalent using curve on the ASP.	

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**(continued)**	
	D. MAINTAIN S/G pressure at pre-Control Room evacuation values using S/G DUMP TO ATMOSPHERE hand stations.	
14.	**CHECK** Neutron Count Rate LOWERING.	**PERFORM** the following: **A.** IF Reactor was SHUT DOWN prior to Control Room evacuation, THEN: **(1) CHECK** Neutron Count Rate STABLE. **(2) GO TO** Attachment 14, Step 15. **B.** IF Reactor subcriticality can **NOT** be confirmed, THEN **ALLOW** RCS to heat up until boration is completed in subsequent steps. Attachment 14, Step 25
15.	**PROCEED** to 3B 4KV Switchgear.	

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

NOTE

- Auto Sequential Loading on 3B 4KV Bus is inhibited when NORMAL / ISOLATE switches are placed in ISOLATE. Loading will have to be manually performed by the operator.
- Startup Transformer Lockout Protective Relaying for Breaker 3AB05 is DEFEATED when the breaker NORMAL / ISOLATE switch is in ISOLATE.

CAUTION

- To prevent spurious 3B EDG trips, 3B 4KV Bus Sequencer is DISABLED, 3B 4KV Bus is STRIPPED, and 3B 4KV Bus Lockout Relays are RESET prior to transferring 3B EDG to LOCAL.
- A spurious Safety Injection Signal could cause the SI Pump to start. The SI Pump is stopped to prevent damage due to improper valve lineup caused by fire damage.

16. ALIGN 3B 4KV Bus as follows:

A. At 3C23B-1, 3B EDG SEQUENCER, **PLACE** the following Key Switches in OFF:

- XS-1, SEQUENCER ENABLE keylock (Inside Cabinet)
- PLC Power Supply
- I/O Power Supply

B. At 3C23B, 3B EDG SEQUENCER CUBICLE 2, **REMOVE** Fuse FU-1 (to prevent spurious trip of 3B EDG).

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 12 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

16. (continued)

C. **PLACE** all yellow color coded NORMAL/ISOLATE switches to ISOLATE on all breakers.

D. **ENSURE** the following breakers TRIPPED:

- 3AB01, 3B REACTOR COOLANT PUMP
- 3AB02, 4KV BUS 3B FD FROM UNIT 3 AUX XFMR
- 3AB06, 3C REACTOR COOLANT PUMP
- 3AB10, HEATER DRAIN PUMP 3B
- 3AB11, TURBINE PLANT COOLING WTR PUMP 3B
- 3AB12, SAFETY INJECTION PUMP 3B
- 3AB16, CIRC WATER PUMP 3B1
- 3AB18, CIRC WATER PUMP 3B2
- 3AB19, BUS TIE BREAKER FOR 3AD06
- 3AB21, CONDENSATE PUMP 3B

E. **ENSURE** 3AB22, 3B 4KV BUS TIE TO 3A OR 3C 4KV BUS, is RACKED OUT.

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 13 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17.	At 3C23B, 3B EMERGENCY LOAD SEQUENCER CUBICLE NO. 2, **RESET** 3B 4KV Bus Lockout Relays: • **PLACE** XS-186-3B, LOCKOUT RESET TRANSFER SWITCH, to LOCAL. • **ENSURE** both 3B 4KV Bus Lockout Relays RESET (orange handles vertical).	

NOTE

3B Boric Acid Transfer Pump will trip on a loss of power and will require local restart if needed for borating the RCS.

CAUTION

If all power is lost after 3B Sequencer is disabled, EDG failure could result if ASP control switches for 4KV bus loads, 3B NCC Fan, 3B Charging Pump, and 3B Pressurizer Backup Group Heaters, are **NOT** placed in STOP/TRIP prior to closing the associated EDG Breaker(s).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18.	**DETERMINE** if 3B EDG should be placed in LOCAL:	
	A. CHECK the following: * Control Room evacuation due to fire or explosion * Loss of Off-Site Power exists	**A. PERFORM** the following: **(1) NOTIFY** Unit Supervisor to STOP 3B EDG. **(2) GO TO** Attachment 14, Step 19.
	B. TRIP 3AB05, 4KV BUS 3B FD FROM UNIT 3 STARTUP XFMR.	

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 14 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

18. (continued)

C. At the ASP, **PLACE** control switches for load shedding in required position:

- 3AB13, 3B CCW PUMP switch in TRIP
- 3AB15, 3B RHR PUMP switch in TRIP
- 3AB17, 3B ICW PUMP switch in TRIP
- 3B CHARGING PUMP switch in TRIP
- 3B PRESSURIZER BACKUP GROUP HEATER switch in TRIP
- 3B NORMAL CNTMT COOLER switch in STOP
- 3D NORMAL CNTMT COOLER switch in STOP

D. **NOTIFY** Unit Supervisor of the following:

- Sequencer Switches in OFF
- All Transfer Switches in ISOLATE
- 3B 4KV bus STRIPPED, except Load Centers and EDG
- 3B Bus Lockout Relay RESET
- All switch positioning at the ASP required for load shedding is COMPLETE.

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 15 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18.	**(continued)**	
	E. WHEN Unit Supervisor reports successful transfer of 3B EDG to local control, THEN **CHECK** 3B 4KV Bus ENERGIZED.	**E. DIRECT** Unit Supervisor to perform the following: **(1) START** 3B EDG. **(2) ENERGIZE** 3B 4KV Bus.
19.	**CHECK** 3B and 3D 480 Volt Load Centers ENERGIZED: • 3AB09, 3B LOAD CENTER Transfer, is CLOSED • 3AB14, 3D LOAD CENTER Transfer, is CLOSED	Manually **CLOSE** breaker(s) using breaker control switch.
20.	**PROCEED** to Unit 3 ASP.	
21.	IF <u>any</u> of the following pumps are required, THEN **DIRECT** 3B EDG operator to monitor EDG Loading while starting: • 3B ICW Pump • 3B CCW Pump.	

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 16 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22.	**ALIGN** RHR System for cooling:	
	A. **CHECK** RHR System was IN SERVICE prior to Control Room evacuation.	A. **PERFORM** the following: (1) **CONFIRM** Natural Circulation: • RCS Subcooling based on T_{HOT} - GREATER THAN 19°F • RCS T_{HOT} - STABLE or LOWERING • RCS T_{COLD} - WITHIN 10°F OF SATURATION TEMPERATURE FOR S/G PRESSURE • S/G Pressure - STABLE OR SLOWLY LOWERING (2) **GO TO** Attachment 14, Step 23.

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 17 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

22. (continued)

> **NOTE**
>
> Breakers listed in Attachment 14, Step 22.B RNO are physically LOCKED OFF to prevent spurious valve operation. The breakers are UNLOCKED and ENERGIZED only when the Unit(s) are in MODE 4.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B. CHECK RHR Hot Leg Suctions OPEN: • MOV-3-750, LOOP C RHR PUMP SUCTION STOP VLV • MOV-3-751, LOOP C RHR PUMP SUCTION STOP VLV	**B. PERFORM** the following: **(1) OPEN** valves using any of the following: * **REQUEST** TSC to OPEN valve(s) by entering Containment to locally OPEN valve(s). * **REQUEST** Electricians OPEN valve(s) from MOV breakers: • 30615 for MOV-3-750 • 30731 for MOV-3-751 **(2)** WHEN both valves are OPEN, THEN **CONTINUE** with Attachment 14, Step 22.C. **(3) GO TO** Attachment 14, Step 23.
C. CHECK outside SNPO has aligned RHR.	**C.** WHEN RHR is ALIGNED, THEN **CONTINUE** with Attachment 14, Step 22.D.
D. START 3B RHR Pump.	

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 18 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22.	**(continued)**	
	E. MAINTAIN steady RCS temperatures and RHR flow by directing SNPO local operation of the following valves: • HCV-3-758, RHR HX OUTLET FLOW CONTROL VALVE • FCV-3-605, RHR HX BYPASS FLOW CNTL VLV (Refer to Attachment 18, Step 19.L)	**A. PERFORM** the following: **(1) REQUEST** TSC to walkdown RHR System and **CHECK** proper alignment. **(2)** WHEN proper alignment is ESTABLISHED, THEN **RETURN TO** Attachment 14, Step 22.C.
23.	**NOTIFY** TSC that Attachment 14, Step 1 through Attachment 14, Step 22, are COMPLETE.	WHEN TSC communications are AVAILABLE, THEN **REPORT** status.
24.	**ALIGN** CVCS from ASP as follows:	
	A. ENSURE HCV-3-121, CHARGING FLOW TO REGEN HX, is OPEN.	
	B. ENSURE CV-3-310A, LOOP A CHARGING ISOLATION, is OPEN.	
	C. ENSURE CV-3-387, EXCESS L/D ISOLATION VLV, is CLOSED.	
	D. ENSURE CV-3-311, AUX SPRAY ISOLATION, is CLOSED.	

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 19 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
24.	**(continued)**	
E.	**ENSURE** LCV-3-460, HIGH PRESSURE L/D ISOLATION VLV, is CLOSED.	
F.	**ENSURE** 3B CHARGING PUMP is in STOP.	
G.	**OPEN** LCV-3-115B, RWST TO CHARGING PUMP SUCTION.	G. **DIRECT** the outside SNPO to OPEN valve 3-358, RWST EMERG MAKEUP TO CHARGING PUMPS LCV-3-115B BYPASS
H.	**ENSURE** outside SNPO has CLOSED LCV-3-115C, VCT TO CHARGING PUMP SUCTION.	
I.	**ENSURE** outside SNPO has performed the following: • 3B Charging Pump Auto-Manual Speed Controller selected to MANUAL. • Controller output set to 6 psig.	

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- During Boration, Pressurizer water level is allowed to rise above no-load programmed level and remain above normal operating band in subsequent steps until boration requirements are met.
- A 136 minute boration at 75% Charging Pump speed (Auto/Manual Speed Controller setpoint of 12 psig) will add 5576 gallons of boric acid to the RCS. In the event Charging Pump speed is reduced or pump is cycled, a longer boration time will be required.
- After Pressurizer level of 70% is attained, charging flow rate may be adjusted to maintain Pressurizer level to compensate for shrinkage.
- With each Charging Pump change or cycle, a new data entry should be completed on Attachment 10, Unit 3 Boration, until 136 minute boration or equivalency is accomplished.
- After 5576 gallons from Boric Acid Storage Tanks is charged to the RCS, suction of 3B Charging Pump is shifted to RWST by closing MOV-3-350. Boration continues, as necessary, to maintain Pressurizer program level of 23 to 53%.
- Letdown (RHR and RCS) shall be secured except for RCP seals.

CAUTION

When on OMS and the RCS is SOLID, Charging Pump operation shall be carefully controlled as RCS pressure responds rapidly to Charging Pump operation.

25. **BORATE** the RCS as follows:

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A.	**CONFIRM** with TSC that RCS boron Concentration is less than Cold Shutdown Concentration.	A. **OBSERVE** NOTE prior to Attachment 14, Step 26 and **GO TO** Attachment 14, Step 26.

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
25.	**(continued)**	
	B. DIRECT outside SNPO to OPEN MOV-3-350, EMERGENCY BORATION VALVE.	
	C. START 3B Charging Pump and **COMMENCE** logging 3B Charging Pump operations on Attachment 10, Unit 3 Boration.	
	D. DIRECT outside SNPO to gradually, over one minute, **RAISE** Charging Pump Speed Controller setpoint to 12 psig.	
	E. CONTINUE Emergency Boration for 136 minutes at 12 psig Speed Controller setting while performing subsequent steps.	
	F. CHECK 136 minute Emergency Boration COMPLETE.	**F.** WHEN 136 minute Emergency Boration is complete, THEN **CONTINUE** with Attachment 14, Step 25.G.
	G. DIRECT outside SNPO to CLOSE MOV-3-350, EMERGENCY BORATION VALVE, to prevent gas admission to the Charging Pump suction.	

ATTACHMENT 14
Unit 3 Reactor Operator
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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- The Pressurizer Level Correction Curve (Plant Curve Book Section V, Figure 3C or on ASP) should be used to correct LI-3-462 to the Hot Calibration Channel equivalent.
- Pressurizer Heater Low Level Cutout is bypassed when operation of heaters is from Alternate Shutdown Panel.
- RCP seal injection flowpath is **NOT** protected for Alternate Shutdown and may **NOT** be available.

26. ESTABLISH Pressurizer Level Control as follows:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A. CHECK Pressurizer **NOT** DRAINED.	**A. GO TO** Attachment 14, Step 28.
B. CHECK Pressurizer **NOT** SOLID.	**B. PERFORM** the following: **(1) CYCLE** 3B Charging Pump as necessary to maintain RCS pressure at pre-Control Room evacuation value. **(2) GO TO** Attachment 14, Step 28.

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 23 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
26.	**(continued)**	
	C. CHECK 136 minute Emergency Boration COMPLETE.	**C.** WHEN 136 minute Emergency Boration COMPLETE, THEN: **(1) ADJUST** 3B Charging Pump Speed Controller output setting to: • **MAINTAIN** Pressurizer Wide Range level between 22% and 53% Corrected Narrow Range level. • **REDUCE** start/stop cycles. **(2) GO TO** Attachment 14, Step 27.
	D. CYCLE 3B Charging Pump as necessary to maintain PRZ Wide Range level between 22% and 53% Corrected Narrow Range level using curve on ASP.	

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 24 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
27.	**ESTABLISH** Pressurizer Pressure Control as follows:	
	A. CHECK Third RO has placed Pressurizer Backup Heater Key Switch in EMERGENCY per Attachment 16, Step 23.A.	**A.** WHEN PRZ Backup Heater Key Switch is in EMERGENCY, THEN **CONTINUE** with Attachment 14, Step 27.B.
	B. CHECK Corrected NR Pressurizer level greater than 22%.	**B. CHARGE** as necessary to maintain Pressurizer level greater than 22%.
	C. ENERGIZE 3B Pressurizer Backup Group Heaters as necessary to maintain RCS pressure at pre-Control Room evacuation conditions.	
28.	**CHECK** the following CCW valves OPEN:	
	• MOV-3-1418, CCW FROM NORMAL CONTAINMENT COOLERS	• Locally **OPEN** valve.

NOTE

Because all Unit 3 RCPs have been TRIPPED, operation of only one Normal Containment Cooler is necessary.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
29.	**ESTABLISH** Containment Cooling:	
	A. CHECK 3B Normal Containment Cooler Fan RUNNING.	**A. OBSERVE** NOTE prior to Attachment 14, Step 30 and **GO TO** Attachment 14, Step 30.
	B. CHECK 3B Normal Containment Cooler Damper OPEN.	

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 25 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Steady state loading on each Unit 3 EDG shall **NOT** exceed 2.5 MW. Load transients of up to 2.75 MW are acceptable when starting additional equipment.

30. IF power is LOST, THEN **GO TO** Attachment 14, Step 18.

31. **MAINTAIN** adequate power supply:

A. **START** additional loads as directed by TSC per Attachment 5, Unit 3 Component KW Load Rating Chart.

B. **CONTACT** EDG operator to determine 4KV Bus status.

C. **CHECK** 3B 4KV Bus ENERGIZED by Off-Site Power.

RESPONSE NOT OBTAINED: **C.** **COORDINATE** with EDG operator to maintain EDG voltage and KW within limits.

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 26 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- Adverse trends should be reported to the Shift Manager as soon as possible.
- Pressurizer Level Correction Curve (Plant Curve Book Section V, Figure 3C or on ASP) should be used to correct LI-3-462 to the Hot Calibration Channel equivalent.
- S/G Wide Range Correction Curve (Plant Curve Book Section V, Figure 3B or on ASP) should be used to correct S/G level indication to Narrow Range Equivalent.

32. MAINTAIN stable plant conditions:

A. ENSURE RCS Cold Leg Temperatures STABLE near pre-evacuation value.

B. ENSURE RCS Hot Leg Temperatures at least 30°F subcooled.

C. ENSURE RCS pressure STABLE near pre-evacuation value.

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 27 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
32.	**(continued)**	
	D. CHECK RCS inventory NEAR pre-evacuation value: * Pressurizer Corrected Narrow Range Level STABLE between 22% and 53% * RCS Drain Down Level Hose Indication STABLE near pre-evacuation value * Pressurizer SOLID	**D. PERFORM** the following: **(1)** IF boration is **NOT** COMPLETE, THEN **CYCLE** 3B Charging Pump as necessary to maintain RCS inventory. **(2)** IF boration is COMPLETE, THEN **ADJUST** 3B Charging Pump Speed Controller output setting as necessary to maintain RCS inventory. **(3)** IF Pressurizer level less than 22%, THEN **TURN** Pressurizer Heaters OFF.
	E. CHECK RHR System IN SERVICE.	**B. MAINTAIN** S/G Wide Range levels between 20% and 50% Narrow Range Equivalent.
	F. ENSURE S/G pressure STABLE near pre-evacuation values.	
33.	**NOTIFY** TSC that Attachment 14, Unit 3 Reactor Operator, is COMPLETE.	

ATTACHMENT 14
Unit 3 Reactor Operator
(Page 28 of 28)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
34.	**DETERMINE** if RCS Cooldown is REQUIRED:	
	A. **CHECK** Unit 3 in MODE 1, 2, or 3.	A. **PERFORM** the following: (1) **OPERATE** equipment as directed by the TSC. (2) **RETURN TO** Attachment 14, Step 32.
	B. **GO TO** Attachment 22, Unit 3 Cool Down from ASP.	

L-16-1 NRC Exam

In-Plant - JPM I

JPM TITLE: Locally Trip the Reactor and Turbine

JPM NUMBER: 14028009501 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 14028009500 / Respond to an ATWS

K/A NUMBERS: EPE 029 EA1.12 **K/A VALUE:** RO 4.1 / SRO 4.0

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: X Perform:

EVALUATION LOCATION: In-Plant: X Control Room:
Simulator: Other:
Lab:

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No
Alternate Path [INPO]: No

Developed by:	**Brian Clark** Instructor/Developer	**6/20/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/21/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

<u>Protected Content:</u> (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
1-0	Updated to fleet template; text/grammar changes	2015 LOCT Annual Exam	N/A	N/A N/A	N/A N/A
2-0	Added turbine trip; formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A N/A	N/A N/A

SIMULATOR SET-UP:

- N/A

Required Materials:	• Handout 3(4)-EOP-FR-S.1
General References:	• 3(4)-EOP-FR-S.1, Response to Nuclear Power Generation/ATWS
Task Standards:	• Locally open the Reactor Trip Breakers in the 3(4)B MCC Room • Locally open the 3(4)A and 3(4)B Rod Drive Motor Generator Set Input and Output Breakers in the 3(4)B MCC Room • Locally trip the main turbine at the front standard

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- The Unit 3(4) reactor and main turbine could NOT be tripped from the Control Room.
- The crew has entered 3(4)-EOP-FR-S.1, Response to Nuclear Power Generation/ATWS.

INITIATING CUE:

- You are directed to perform RNO Steps 6.a and 6.b of 3(4)-EOP-FR-S.1.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Evaluator Note:	**Prior to administering, determine which Unit the JPM will be performed on and provide the appropriate procedure and initiating cue.**

Performance Step: 1 **Critical: No**	**Obtain required reference materials and proceed to the 3(4)B MCC Room.**
Standard:	Obtain a copy of 3(4)-EOP-FR-S.1, Response to Nuclear Power Generation/ATWS, and proceed to the 3(4)B MCC Room.
Evaluator Note:	If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue.
Evaluator Cue:	Provide examinee with a copy of handout 3(4)-EOP-FR-S.1.
Performance:	**SATISFACTORY _____ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: No**	**3(4)-EOP-FR-S.1, prior to Step 6:** **CAUTION** ***If an SI signal exists or occurs AND the reactor is subcritical, proper safeguards equipment alignment is required to be verified using Attachment 3 of 3(4)-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.***
Standard:	Read CAUTION and recognize that it is safe to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: Yes**	**3(4)-EOP-FR-S.1, Step 6:** **Check If The Following Trips Have Occurred:** a. Reactor Trip **(NO)** → (RNO) In 3(4)B MCC Room, locally trip reactor as follows: • Open 3(4)A and 3(4)B Reactor Trip Breakers.
Standard:	Locally open the reactor trip breakers by pressing the TRIP button on the front of each breaker.
Evaluator Cue:	• If asked initially, inform examinee that a red CLOSED flag is showing at each breaker • When the breaker trips are properly simulated, state that the associated green lights are lit, the red lights are extinguished, and green OPEN flags are showing
Evaluator Note:	Breakers may be tripped in any order.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: No**	**3(4)-EOP-FR-S.1, Step 6:** **Check If The Following Trips Have Occurred:** a. Reactor Trip **(NO)** → (RNO) In 3(4)B MCC Room, locally trip reactor as follows: • Open 3(4)A and 3(4)B Reactor Trip Bypass Breakers.
Standard:	Recognize that both bypass breakers are racked out and the green OPEN flags are showing.
Evaluator Cue:	• When checked, state that the breaker face plates are protruding from their cubicle cover plates • If asked, inform examinee that a green flag is showing on each breaker
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: Yes**	**3(4)-EOP-FR-S.1, Step 6:** **Check If The Following Trips Have Occurred:** a. Reactor Trip **(NO)** → (RNO) In 3(4)B MCC Room, locally trip reactor as follows: • Open A/B MG Set Generator Output Breakers.
Standard:	Open both motor-generator set output breakers by placing their control switches in the TRIP position.
Evaluator Cue:	• If asked initially, inform examinee that the associated red lights are lit and the green lights are NOT lit at each breaker • When the breaker trips are properly simulated, state that the associated green lights are lit and the red lights are extinguished
Evaluator Note:	Breakers may be tripped in any order.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: Yes**	**3(4)-EOP-FR-S.1, Step 6:** **Check If The Following Trips Have Occurred:** a. Reactor Trip **(NO)** → (RNO) In 3(4)B MCC Room, locally trip reactor as follows: • Open A/B MG Set Generator Input Breakers.
Standard:	Open both motor-generator set input breakers by placing their control switches in the TRIP position.
Evaluator Cue:	• If asked initially, inform examinee that the associated red lights are lit and the green lights are NOT lit at each breaker • When the breaker trips are properly simulated, state that the associated green lights are lit and the red lights are extinguished
Evaluator Note:	Breakers may be tripped in any order.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: Yes**	**3(4)-EOP-FR-S.1, Step 6:** **Check If The Following Trips Have Occurred:** b. Turbine Trip **(NO)** → (RNO) Locally trip turbine at Turbine Front Standard.
Standard:	Rotate the RESET/TRIP lever, to the TRIP position.
Evaluator Cue:	• When the Examinee identifies the trip lever box provide photo A and have the Examinee simulate tripping the Turbine. • When properly simulated, and if asked, use the following cues as applicable: ◦ Turbine stop and control valves are closing ◦ Reheat stop and intercept valves are closing ◦ Turbine shaft is slowing down ◦ Turbine rpm indicator is lowering ◦ Bearing oil pressures are lowering • Note any other cues given in the comments section
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When the examinee locally trips the turbine, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ____

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- The Unit 3(4) reactor and main turbine could NOT be tripped from the Control Room.
- The crew has entered 3(4)-EOP-FR-S.1, Response to Nuclear Power Generation/ATWS.

INITIATING CUE:

- You are directed to perform RNO Steps 6.a and 6.b of 3(4)-EOP-FR-S.1.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

FPL	TURKEY POINT UNIT 3 EMERGENCY OPERATING PROCEDURE SAFETY RELATED **CONTINUOUS USE**	Procedure No. **3-EOP-FR-S.1** Revision No. **4**

Title:

RESPONSE TO NUCLEAR POWER GENERATION/ATWS

Responsible Department: OPERATIONS

Special Considerations:

Last page of this procedure contains fold out page

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED today INITIAL

Revision	Approved By	Approval Date
4	Tom Wall	08/06/14

UNIT #	UNIT 3
DATE	
DOCT	PROCEDURE
DOCN	3-EOP-FR-S.1
SYS	
STATUS	COMPLETED
REV	4
# OF PGS	

REVISION SUMMARY	
Rev. No.	**Description**
4	**AR 1926754, 08/06/14, G.T. Slaby** Changes to incorporate revisions 2, 2+ and 3 from Westinghouse Owners Group (WOG) Emergency Response Guideline (ERG). The changes consist of reformatting the procedure to better align with and reduce the number of exceptions to the ERG, and adding enhancements for usability. Specific changes include: Addition of a Foldout page for Adverse Containment Conditions Inserted new symbol ➔to designate Continuous Action steps, and created new Attachment 3 for a summary of Continuous Action Steps Added new Attachment 1 Local AFW Isolation of Faulted S/Gs, and Attachment, 2 Faulted S/G Isolation

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1.0 PURPOSE

This procedure provides actions to add negative reactivity to a core which is observed to be critical when expected to be Shutdown.

2.0 SYMPTOMS AND ENTRY CONDITIONS

This procedure is entered from:

1) E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1, when Reactor Trip is **NOT** verified AND manual trip is **NOT** effective.

2) F-0.1, SUBCRITICALITY, CRITICAL SAFETY FUNCTION STATUS TREE on either a RED or an ORANGE condition.

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

3.0 OPERATOR ACTIONS

CAUTION

RCPs should **NOT** be tripped with reactor power greater than 5%.

NOTE

Step 1 and Step 2 are IMMEDIATE ACTION steps.

1. Verify Reactor Trip:

- Rod Bottom Lights – ON
- Reactor Trip and Bypass Breakers – OPEN
- Rod Position Indicators – AT ZERO
- Neutron flux – DECREASING

Response Not Obtained:

Perform the following:

a. Manually trip reactor.

b. IF reactor will **NOT** trip, THEN ensure Control Rod Insertion in Auto or Manual.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**Verify Turbine Trip:**	
	a. All Turbine Stop OR associated Control Valves – CLOSED	**a.** Manually trip turbine. IF unable to verify Turbine Trip, THEN close Main Steamline Isolation and Bypass Valves. Observe NOTE prior to Step 3 and go to Step 3.
	b. Moisture Separator Reheater Steam Valves – CLOSED:	
	1) MSR Main Steam Supply Stop MOVs	**1)** Manually close valves. IF any valve can **NOT** be closed, THEN close Main Steamline Isolation and Bypass valves. Observe NOTE prior to Step 3 and go to Step 3.
	2) Reheater Timing Valves	**2)** Close Main Steamline Isolation and Bypass valves. Observe NOTE prior to Step 3 and go to Step 3.
	3) MSR Purge Steam Valves	**3)** Manually close valves. IF any valve can **NOT** be closed, THEN close Main Steamline Isolation and Bypass Valves.

NOTE

FOLDOUT Page shall be monitored for the remainder of this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**Check AFW Pumps – ALL RUNNING**	Manually open steam supply valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Initiate Emergency Boration Of RCS:**	
	a. Verify SI – RESET	
	b. Verify Charging Pumps – AT LEAST <u>ONE</u> RUNNING IN MANUAL	
	c. Stop Makeup System	
	d. Manually start Boric Acid Pump 3A or 3B	**d.** Align Charging Pump suction to the RWST as follows: **1)** Hold closed LCV-3-115C Control switch. **2)** Direct an operator to open Breaker 30669 for LCV-3-115C. **3)** <u>WHEN</u> 30669 is open, <u>THEN</u> release LCV-3-115C Control switch. **4)** Go to Step 4.f.
	e. Open MOV-3-350, Emergency Boration Valve	**e.** Perform the following: **1)** Open FCV-3-113A, Boric Acid To Blender. **2)** Open FCV-3-113B, Blender Flow To Charging Pump. **3)** Locally open 3-356, Manual Emergency Boration Valve. **4)** <u>WHEN</u> 3-356, Manual Emergency Boration Valve is open, <u>THEN</u> close FCV-3-113B, Blender To Charging Pump. **5)** Continue with Step 4.f.
	f. Open HCV-3-121, Charging Flow To Regen Heat Exchanger	

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

4. (continued)

g. Verify CV-3-310A, Loop A Charging Isolation – OPEN

g. Open CV-3-310B, Loop C Charging Isolation.

h. Establish Emergency Boration flow:

- FI-3-110 – GREATER THAN 60 GPM
- FI-3-122A – GREATER THAN 45 GPM

h. Perform one or more of the following as necessary to establish Emergency Boration flow:

* Adjust operating Charging Pump(s) speed controller(s).
* Start additional Charging Pumps.
* Manually align valves.

5. Verify Containment Ventilation Isolation:

a. Verify Unit 3 Containment Purge Exhaust AND Supply Fans – OFF

b. Verify Containment Purge Supply AND Exhaust Isolation Valves – CLOSED:

- POV-3-2600
- POV-3-2601
- POV-3-2602
- POV-3-2603

b. IF any Purge Valve can **NOT** be closed, THEN pull fuses for any open Purge Valves from behind VPB:

- XEP for POV-3-2600
- XLAG for POV-3-2601
- XEQ for POV-3-2602
- XLAH for POV-3-2603

c. Verify Containment Instrument Air Bleed Isolation Valves – CLOSED

- CV-3-2819
- CV-3-2826

c. IF **neither** valve can be closed, THEN locally close:

- MPAS-3-005, Containment Air Bleed to Purge Air Return Line Isolation.
- 3-11-018A, Instrument Air Bleed Line Drain Isolation Valve, (reach rod, Aux Bldg Hallway outside P&V Room)

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

CAUTION

If an SI signal exists or occurs AND the reactor is subcritical, proper safeguards equipment alignment is required to be verified using Attachment 3 of 3-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.

6. Check If The Following Trips Have Occurred:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Reactor Trip	**a.** In 3B MCC Room, locally trip reactor as follows: • Open 3A and 3B Reactor Trip Breakers. • Open 3A and 3B Reactor Trip Bypass Breakers. • Open A/B MG Set Generator Output Breakers. • Open A/B MG Set Motor Input Breakers
b. Turbine Trip	**b.** Locally trip turbine at Turbine Front Standard.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**(continued)**	
	c. Mid and East GCBs – OPEN (Normally 30 seconds delay)	**c.** Perform the following: **1)** Manually open breakers. **2)** IF breakers do **NOT** open, THEN actuate Emergency Gen Bkr Trip Switch for the affected breaker(s). **3)** IF breaker position indication **NOT** available AND turbine speed is **NOT** decreasing, THEN direct field operator to perform the following: **a)** Obtain key 17 from the Shift Manager key locker. **b)** Locally trip Mid and East GCBs from the switchyard: • 8W33 • 8W68

NOTE

When Adverse Containment conditions exist, Gamma-Metrics indication needs to be used.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
➔ 7.	**Check If Reactor Is Subcritical:**	
	a. Power Range Channels – LESS THAN 5%	**a.** Observe CAUTION prior to Step 8 and go to Step 8.
	b. Intermediate Range Channels – NEGATIVE STARTUP RATE	**b.** Observe CAUTION prior to Step 8 and go to Step 8.
	c. Observe CAUTION prior to Step 16 and go to Step 16	

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

CAUTION

If CST level decreases to less than 12%, makeup water sources for CST will be necessary to maintain secondary heat sink.

➔ **8. Check S/G Level:**

a. Narrow Range Level in at least one S/G – GREATER THAN 7%[27%]

RESPONSE NOT OBTAINED:

a. Perform the following:

1) Establish total feedwater flow greater than 800 gpm.

2) IF feed flow **NOT** greater than 800 gpm, THEN manually start pumps and align valves to establish greater than 800 gpm.

3) Maintain total feedwater flow greater than 800 gpm until Narrow Range Level greater than 7%[27%] in at least one S/G.

b. Control feed flow to maintain Narrow Range Level between 21%[27%] and 50%

9. Verify All Dilution Paths – ISOLATED:

a. Check FR-3-113 – **NO** PRIMARY WATER FLOW

RESPONSE NOT OBTAINED:

a. Perform the following:

1) Close FCV-3-114A, Demin Water To Blender.

2) Locally close the following valves:

- 3-359A, Primary Water To Chemical Addition Tank
- 3-272, Primary Water From Chemical Addition Tank
- 3-353A, Manual Dilution Valve

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	**Check For Reactivity Insertion From Uncontrolled RCS Cool Down:** * RCS temperatures – DECREASING IN AN UNCONTROLLED MANNER OR * Any S/G pressure – DECREASING IN AN UNCONTROLLED MANNER	Perform the following: **a.** Stop any controlled cool down. **b.** Go to Step 14.
11.	**Check Main Steamline Isolation AND Bypass Valves – CLOSED**	Manually close valves.
12.	**Identify Faulted S/G(s):**	
	a. Check pressures in all S/Gs: * ANY S/G PRESSURE DECREASING IN AN UNCONTROLLED MANNER OR * ANY S/G COMPLETELY DEPRESSURIZED	**a.** Go to Step 14.

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

CAUTION

- At least one S/G must be maintained available for RCS cool down.
- If all S/Gs are faulted, at least 50 gpm feed flow should be maintained to each S/G.
- If the AFW Pumps are the only available source of feed flow, the steam supply to the AFW Pumps must be maintained from at least one S/G.

13. Isolate Faulted S/G(s):

a. Isolate Main Feed Line:

- Close Feedwater Isolation valve(s)
  * MOV-3-1407 OR FCV-3-478 for S/G A
  * MOV-3-1408 OR FCV-3-488 for S/G B
  * MOV-3-1409 OR FCV-3-498 for S/G C
- Close Feedwater Bypass valve(s)
  * POV-3-477 OR FCV-3-479 for S/G A
  * POV-3-487 OR FCV-3-489 for S/G B
  * POV-3-497 OR FCV-3-499 for S/G C

RESPONSE NOT OBTAINED: **a.** Locally isolate Main Feed Line.

b. Isolate AFW flow

RESPONSE NOT OBTAINED: **b.** Locally isolate using Attachment 1.

c. Dispatch operator to isolate AFW steam supply from faulted S/G(s) using Attachment 2

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**(continued)**	
	d. Verify S/G Dump To Atmosphere valve(s) – CLOSED	**d.** Perform the following: **1)** Place Steam Dump To Atmosphere controller in MANUAL and close the Steam Dump To Atmosphere valve. **2)** IF Steam Dump To Atmosphere can **NOT** be closed, THEN locally isolate Steam Dump To Atmosphere valve. * 3-10-001 for S/G A * 3-10-002 for S/G B * 3-10-003 for S/G C
	e. Verify S/G Blowdown Isolation valve(s) – CLOSED	
	f. Verify S/G Sample Line(s) – ISOLATED	
14.	**Check Core Exit TCs – LESS THAN 1200°F**	IF Core Exit temperatures greater than 1200°F and increasing, THEN go to SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE, Step 1.

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

CAUTION

- These CAUTIONS apply to AFW Pump operation throughout all the EOPs.
- If two AFW Pumps are operating on a single train, one of the pumps SHALL be shut down within one hour of the initial start signal using 3-NOP-075, AUXILIARY FEEDWATER SYSTEM, Section for Shutdown of AFW Pump(s) from Emergency Plant Operation.
- If two AFW Trains are operating AND one of the AFW Pumps has been operating at a low flow of less than 80 gpm, the pump SHALL be shut down within one hour of operating at less than 80 gpm using 3-NOP-075, AUXILIARY FEEDWATER SYSTEM.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15.	**Verify Reactor Subcritical:** **a.** Power Range Channels – LESS THAN 5% **b.** Intermediate Range Channels – NEGATIVE STARTUP RATE	Perform the following: **1.** Continue to borate. **2.** IF boration **NOT** available, THEN allow RCS to heat up. **3.** Perform actions of other Function Restoration Procedures in effect which do **NOT** cool down or otherwise add positive reactivity to the core. **4.** Return to Step 4.

CAUTION

Boration should continue during subsequent actions until adequate Shutdown Margin is obtained.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16.	**Return To Procedure And Step In Effect**	

End of Section 3.0

4.0 REFERENCES AND COMMITMENTS

4.1 <u>References</u>

4.1.1 Implementing References

1. SACRG-1, Severe Accident Control Room Guideline Initial Response

2. 3-NOP-075, Auxiliary Feedwater System

4.1.2 Developmental References

1. Technical Specifications for Turkey Point Unit 3 and Unit 4

2. Turkey Point Unit 3 and Unit 4 Final Safety Analysis Report

3. As-Built Plant Drawings

4. BD-EOP-FR-S.1, Response to Nuclear Power Generations/ATWS

5. Plant Change/Modifications/Engineering Changes:

- **a.** PC/M 92-040, Addition of Reverse Power Relays and Main Generator Protection Modifications
- **b.** PC/M 09-139, EPU EC 247008, LAR Umbrella Mod
- **c.** EC 277336, Instrument Air Bleed Line Modification
- **d.** EC 242095, PC/M 05029, Justification of Increased AFW Pump Performance

6. Miscellaneous Documents:

- **a.** Generic Technical Guidelines developed by the Westinghouse Owners Group (WOG), Revision 3. This consists of the following documents:
  - **1)** Low pressure version of the WOG Optimal Recovery Guidelines, Status Trees, and Functional Restoration Guidelines
  - **2)** Background documents for each low pressure version Optimal Recovery Guidelines, Status Trees, and Functional Restoration Guidelines
  - **3)** WOG Emergency Response Guidelines Executive Volume
  - **4)** WOG Emergency Response Guidelines Maintenance Program Summary
- **b.** Calculation 514.2, Turkey Point EOP Setpoints - Miscellaneous (J, K, L, P, Q, X, Y Series)

4.1.2 Developmental References (continued)

6. (continued)

c. Calculation 511.2, Turkey Point EOP Setpoints - Tank Levels (U Series)

d. Calculation 505.3, Turkey Point EOP Setpoints - Steam Generator Level (M, N Series, X.1, X.2)

e. Calculation 509.2, Turkey Point EOP Setpoints - Flows (S Series)

f. Calculation 502.2, Turkey Point EOP Setpoints - RCS Temperature (E Series, F Series, G Series, H Series, I Series)

g. Calculation 510.2, Turkey Point EOP Setpoints-Containment Parameters (T Series)

h. PTN-ENG-SENS-98-047, AFW Pump Low-Flow Operating Restrictions

4.1.3 Management Directives

None

4.2 <u>Commitments</u>

None

End of Section 4.0

ATTACHMENT 1
Local AFW Isolation of Faulted S/Gs
(Page 1 of 1)

IF this valve will NOT close:	**THEN unlock and close:**
CV-3-2816	3-20-141, AFW To S/G 3A Train 1
CV-3-2831	AFPD-3-008, AFW To S/G 3A Train 2 Isol.
CV-3-2817	3-20-241, AFW To S/G 3B Train 1 Isol.
CV-3-2832	AFPD-3-007, AFW To S/G 3B Train 2 Isol.
CV-3-2818	3-20-341, AFW To S/G 3C Train 1 Isol.
CV-3-2833	AFPD-3-006, AFW To S/G 3C Train 2 Isol.

End of Attachment 1

ATTACHMENT 2
<u>Faulted S/G Isolation</u>
(Page 1 of 2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**<u>IF</u> 3A S/G Is Faulted, <u>THEN</u> Perform The Following:**	
	a. Locally place Breaker 4D01-28, 3A Steam Supply To Aux Feedwater Pumps MOV-3-1403, in OFF	
	b. Locally close MOV-3-1403	
	c. Check MOV-3-1404 – ALIGNED TO TRAIN 2	**c.** <u>IF</u> MOV-3-1405 open, <u>THEN</u> perform the following: **1)** Locally unlock and open AFSS-3-006. **2)** Locally unlock and close AFSS-3-007.
	d. Check MOV-3-1404 – OPEN	**d.** Manually or locally open MOV-3-1404.
	e. Check AFW Trains – AT LEAST <u>ONE</u> AVAILABLE	**e.** Manually or locally align valves to restore at least <u>one</u> AFW Train.
2.	**<u>IF</u> 3B S/G Is Faulted, <u>THEN</u> Perform The Following:**	
	a. Locally place Breaker 30833, 3B Steam Supply To Aux Feedwater Pumps MOV-3-1404, in OFF	
	b. Locally close MOV-3-1404	
	c. Check AFW Trains – AT LEAST <u>ONE</u> AVAILABLE	**c.** Manually or locally align valves to restore at least <u>one</u> AFW Train.

ATTACHMENT 2
<u>Faulted S/G Isolation</u>
(Page 2 of 2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**<u>IF</u> 3C S/G is faulted, <u>THEN</u> perform the following:**	
	a. Locally place Breaker 3D01-27, 3C Steam Supply To Aux Feedwater Pumps MOV-3-1405, in OFF	
	b. Locally close MOV-3-1405	
	c. Check MOV-3-1404 – ALIGNED TO TRAIN 1	**c.** <u>IF</u> MOV-3-1403 open, <u>THEN</u> perform the following: **1)** Locally unlock and open AFSS-3-007. **2)** Locally unlock and close AFSS-3-006.
	d. Check MOV-3-1404 – OPEN	**d.** Manually or locally open MOV-3-1404.
	e. Check AFW Trains – AT LEAST <u>ONE</u> AVAILABLE	**e.** Manually or locally align valves to restore at least <u>one</u> AFW Train.

End of Attachment 2

ATTACHMENT 3
Continuous Action Summary
(Page 1 of 1)

Step 7, Check If Reactor Is Subcritical:

WHEN both of the following conditions are met:

- Power Range Channels – LESS THAN 5%
- Intermediate Range Channels – NEGATIVE STARTUP RATE

THEN observe CAUTION prior to Step 16 and go to Step 16.

Step 8, Check S/G Level:

IF Narrow Range Level in at least one S/G **NOT** greater than 7%[27%],
THEN establish total feedwater flow greater than 800 gpm.

IF feed flow **NOT** greater than 800 gpm, THEN manually start pumps and align valves to establish greater than 800 gpm.

Maintain total feedwater flow greater than 800 gpm until Narrow Range Level greater than 7%[27%] in at least one S/G.

Control feed flow to maintain Narrow Range Level between 21%[27%] and 50%.

End of Attachment 3

FOLDOUT PAGE
For Procedure 3-EOP-FR-S.1

1. ADVERSE CONTAINMENT CONDITIONS

a. IF either condition listed below occurs, THEN use [Adverse Containment Setpoints]:

* Containment atmosphere temperature ≥ 180°F

  OR

* Containment radiation levels ≥ 1.3×10^5 R/hr

b. WHEN Containment atmosphere temperature returns to less than 180°F, THEN Normal Setpoints can again be used.

c. WHEN Containment radiation levels return to less than 1.3×10^5 R/hr, THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.

FPL	TURKEY POINT UNIT 4 EMERGENCY OPERATING PROCEDURE SAFETY RELATED **CONTINUOUS USE**	Procedure No. **4-EOP-FR-S.1** Revision No. **3**

Title:

RESPONSE TO NUCLEAR POWER GENERATION/ATWS

Responsible Department: OPERATIONS

Special Considerations:

Last page of this procedure contains fold out page

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED__today___ INITIAL________

Revision	Approved By	Approval Date
3	Tom Wall	08/06/14

UNIT #	UNIT 4
DATE	
DOCT	PROCEDURE
DOCN	4-EOP-FR-S.1
SYS	
STATUS	COMPLETED
REV	3
# OF PGS	

REVISION SUMMARY	
Rev. No.	**Description**
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1.0 PURPOSE

This procedure provides actions to add negative reactivity to a core which is observed to be critical when expected to be Shutdown.

2.0 SYMPTOMS AND ENTRY CONDITIONS

This procedure is entered from:

1) E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1, <u>when</u> Reactor Trip is **NOT** verified <u>AND</u> manual trip is **NOT** effective.

2) F-0.1, SUBCRITICALITY, CRITICAL SAFETY FUNCTION STATUS TREE on <u>either</u> a RED <u>or</u> an ORANGE condition.

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.0 OPERATOR ACTIONS

CAUTION

RCPs should **NOT** be tripped with reactor power greater than 5%.

NOTE

Step 1 and Step 2 are IMMEDIATE ACTION steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Verify Reactor Trip:** • Rod Bottom Lights – ON • Reactor Trip and Bypass Breakers – OPEN • Rod Position Indicators – AT ZERO • Neutron flux – DECREASING	Perform the following: **a.** Manually trip reactor. **b.** IF reactor will **NOT** trip, THEN ensure Control Rod Insertion in Auto or Manual.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**Verify Turbine Trip:**	
	a. All Turbine Stop OR associated Control Valves – CLOSED	**a.** Manually trip turbine. IF unable to verify Turbine Trip, THEN close Main Steamline Isolation and Bypass Valves. Observe NOTE prior to Step 3 and go to Step 3.
	b. Moisture Separator Reheater Steam Valves – CLOSED:	
	1) MSR Main Steam Supply Stop MOVs	**1)** Manually close valves. IF any valve can **NOT** be closed, THEN close Main Steamline Isolation and Bypass valves. Observe NOTE prior to Step 3 and go to Step 3.
	2) Reheater Timing Valves	**2)** Close Main Steamline Isolation and Bypass valves. Observe NOTE prior to Step 3 and go to Step 3.
	3) MSR Purge Steam Valves	**3)** Manually close valves. IF any valve can **NOT** be closed, THEN close Main Steamline Isolation and Bypass Valves.

NOTE

FOLDOUT Page shall be monitored for the remainder of this procedure.

3.	**Check AFW Pumps – ALL RUNNING**	Manually open steam supply valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Initiate Emergency Boration Of RCS:**	
	a. Verify SI – RESET	
	b. Verify Charging Pumps – AT LEAST ONE RUNNING IN MANUAL	
	c. Stop Makeup System	
	d. Manually start Boric Acid Pump 4A or 4B	**d.** Align Charging Pump suction to the RWST as follows: **1)** Hold closed LCV-4-115C Control switch. **2)** Direct an operator to open Breaker 40669 for LCV-4-115C. **3)** WHEN 40669 is open, THEN release LCV-4-115C Control switch. **4)** Go to Step 4.f.
	e. Open MOV-4-350, Emergency Boration Valve	**e.** Perform the following: **1)** Open FCV-4-113A, Boric Acid To Blender. **2)** Open FCV-4-113B, Blender Flow To Charging Pump. **3)** Locally open 4-356, Manual Emergency Boration Valve. **4)** WHEN 4-356, Manual Emergency Boration Valve is open, THEN close FCV-4-113B, Blender To Charging Pump. **5)** Continue with Step 4.f.
	f. Open HCV-4-121, Charging Flow To Regen Heat Exchanger	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**(continued)**	
	g. Verify CV-4-310A, Loop A Charging Isolation – OPEN	**g.** Open CV-4-310B, Loop C Charging Isolation.
	h. Establish Emergency Boration flow: • FI-4-110 – GREATER THAN 60 GPM • FI-4-122A – GREATER THAN 45 GPM	**h.** Perform one or more of the following as necessary to establish Emergency Boration flow: * Adjust operating Charging Pump(s) speed controller(s). * Start additional Charging Pumps. * Manually align valves.
5.	**Verify Containment Ventilation Isolation:**	
	a. Verify Unit 4 Containment Purge Exhaust AND Supply Fans – OFF	
	b. Verify Containment Purge Supply AND Exhaust Isolation Valves – CLOSED: • POV-4-2600 • POV-4-2601 • POV-4-2602 • POV-4-2603	**b.** IF any Purge Valve can **NOT** be closed, THEN pull fuses for any open Purge Valves from behind VPB: • XEP for POV-4-2600 • XLAG for POV-4-2601 • XEQ for POV-4-2602 • XLAH for POV-4-2603
	c. Verify Containment Instrument Air Bleed Isolation Valves – CLOSED • CV-4-2819 • CV-4-2826	**c.** IF **neither** valve can be closed, THEN locally close MPAS-4-005, Containment Air Bleed to Purge Air Return Line Isolation.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If an SI signal exists or occurs AND the reactor is subcritical, proper safeguards equipment alignment is required to be verified using Attachment 3 of 4-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, while continuing with this procedure.

6. Check If The Following Trips Have Occurred:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Reactor Trip	**a.** In 4B MCC Room, locally trip reactor as follows: • Open 4A and 4B Reactor Trip Breakers. • Open 4A and 4B Reactor Trip Bypass Breakers. • Open A/B MG Set Generator Output Breakers. • Open A/B MG Set Motor Input Breakers
b. Turbine Trip	**b.** Locally trip turbine at Turbine Front Standard.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**(continued)**	
	c. Mid and East GCBs – OPEN (Normally 30 seconds delay)	**c.** Perform the following: **1)** Manually open breakers. **2)** <u>IF</u> breakers do **NOT** open, <u>THEN</u> actuate Emergency Gen Bkr Trip Switch for the affected breaker(s). **3)** <u>IF</u> breaker position indication **NOT** available <u>AND</u> turbine speed is **NOT** decreasing, <u>THEN</u> direct field operator to perform the following: **a)** Obtain key 17 from the Shift Manager key locker. **b)** Locally trip Mid and East GCBs from the switchyard: • 8W88 • 8W65

<u>NOTE</u>

When Adverse Containment conditions exist, Gamma-Metrics indication needs to be used.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
➔ **7.**	**Check If Reactor Is Subcritical:**	
	a. Power Range Channels – LESS THAN 5%	**a.** Observe CAUTION prior to Step 8 and go to Step 8.
	b. Intermediate Range Channels – NEGATIVE STARTUP RATE	**b.** Observe CAUTION prior to Step 8 and go to Step 8.
	c. Observe CAUTION prior to Step 16 and go to Step 16	

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

CAUTION

If CST level decreases to less than 12%, makeup water sources for CST will be necessary to maintain secondary heat sink.

➔ **8. Check S/G Level:**

a. Narrow Range Level in at least one S/G – GREATER THAN 7%[27%]

RNO **a.** Perform the following:

1) Establish total feedwater flow greater than 800 gpm.

2) IF feed flow **NOT** greater than 800 gpm, THEN manually start pumps and align valves to establish greater than 800 gpm.

3) Maintain total feedwater flow greater than 800 gpm until Narrow Range Level greater than 7%[27%] in at least one S/G.

b. Control feed flow to maintain Narrow Range Level between 21%[27%] and 50%

9. Verify All Dilution Paths – ISOLATED:

a. Check FR-4-113 – **NO** PRIMARY WATER FLOW

RNO **a.** Perform the following:

1) Close FCV-4-114A, Demin Water To Blender.

2) Locally close the following valves:

- 4-359A, Primary Water To Chemical Addition Tank
- 4-272, Primary Water From Chemical Addition Tank
- 4-353A, Manual Dilution Valve

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	**Check For Reactivity Insertion From Uncontrolled RCS Cool Down:** * RCS temperatures – DECREASING IN AN UNCONTROLLED MANNER OR * Any S/G pressure – DECREASING IN AN UNCONTROLLED MANNER	Perform the following: **a.** Stop any controlled cool down. **b.** Go to Step 14.
11.	**Check Main Steamline Isolation AND Bypass Valves – CLOSED**	Manually close valves.
12.	**Identify Faulted S/G(s):**	
	a. Check pressures in all S/Gs: * ANY S/G PRESSURE DECREASING IN AN UNCONTROLLED MANNER OR * ANY S/G COMPLETELY DEPRESSURIZED	**a.** Go to Step 14.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

- At least one S/G must be maintained available for RCS cool down.
- If all S/Gs are faulted, at least 50 gpm feed flow should be maintained to each S/G.
- If the AFW Pumps are the only available source of feed flow, the steam supply to the AFW Pumps must be maintained from at least one S/G.

13. Isolate Faulted S/G(s):

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Isolate Main Feed Line: • Close Feedwater Isolation valve(s) * MOV-4-1407 OR FCV-4-478 for S/G A * MOV-4-1408 OR FCV-4-488 for S/G B * MOV-4-1409 OR FCV-4-498 for S/G C • Close Feedwater Bypass valve(s) * POV-4-477 OR FCV-4-479 for S/G A * POV-4-487 OR FCV-4-489 for S/G B * POV-4-497 OR FCV-4-499 for S/G C	**a.** Locally isolate Main Feed Line.
b. Isolate AFW flow	**b.** Locally isolate using Attachment 1.
c. Dispatch operator to isolate AFW steam supply from faulted S/G(s) using Attachment 2	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**(continued)**	
	d. Verify S/G Dump To Atmosphere valve(s) – CLOSED	**d.** Perform the following: **1)** Place Steam Dump To Atmosphere controller in MANUAL and close the Steam Dump To Atmosphere valve. **2)** IF Steam Dump To Atmosphere can **NOT** be closed, THEN locally isolate Steam Dump To Atmosphere valve. * 4-10-001 for S/G A * 4-10-002 for S/G B * 4-10-003 for S/G C
	e. Verify S/G Blowdown Isolation valve(s) – CLOSED	
	f. Verify S/G Sample Line(s) – ISOLATED	
14.	**Check Core Exit TCs – LESS THAN 1200°F**	IF Core Exit temperatures greater than 1200°F and increasing, THEN go to SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE, Step 1.

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

CAUTION

- These CAUTIONS apply to AFW Pump operation throughout all the EOPs.
- If two AFW Pumps are operating on a single train, one of the pumps SHALL be shut down within one hour of the initial start signal using 4-NOP-075, AUXILIARY FEEDWATER SYSTEM, Section for Shutdown of AFW Pump(s) from Emergency Plant Operation.
- If two AFW Trains are operating AND one of the AFW Pumps has been operating at a low flow of less than 80 gpm, the pump SHALL be shut down within one hour of operating at less than 80 gpm using 4-NOP-075, AUXILIARY FEEDWATER SYSTEM.

15. Verify Reactor Subcritical:

a. Power Range Channels – LESS THAN 5%

b. Intermediate Range Channels – NEGATIVE STARTUP RATE

Perform the following:

1. Continue to borate.

2. IF boration **NOT** available, THEN allow RCS to heat up.

3. Perform actions of other Function Restoration Procedures in effect which do **NOT** cool down or otherwise add positive reactivity to the core.

4. Return to Step 4.

CAUTION

Boration should continue during subsequent actions until adequate Shutdown Margin is obtained.

16. Return To Procedure And Step In Effect

End of Section 3.0

4.0 REFERENCES AND COMMITMENTS

4.1 References

4.1.1 Implementing References

1. SACRG-1, Severe Accident Control Room Guideline Initial Response

2. 4-NOP-075, Auxiliary Feedwater System

4.1.2 Developmental References

1. Technical Specifications for Turkey Point Unit 3 and Unit 4

2. Turkey Point Unit 3 and Unit 4 Final Safety Analysis Report

3. As-Built Plant Drawings

4. BD-EOP-FR-S.1, Response to Nuclear Power Generations/ATWS

5. Plant Change/Modifications/Engineering Changes:

- **a.** PC/M 92-073, Addition of Reverse Power Relays and Main Generator Protection Modifications
- **b.** PC/M 09-140, EPU EC 247009, LAR Umbrella Mod
- **c.** EC 277336, Instrument Air Bleed Line Modification
- **d.** EC 242095, PC/M 05029, Justification of Increased AFW Pump Performance

6. Miscellaneous Documents:

- **a.** Generic Technical Guidelines developed by the Westinghouse Owners Group (WOG), Revision 3. This consists of the following documents:
  - **1)** Low pressure version of the WOG Optimal Recovery Guidelines, Status Trees, and Functional Restoration Guidelines
  - **2)** Background documents for each low pressure version Optimal Recovery Guidelines, Status Trees, and Functional Restoration Guidelines
  - **3)** WOG Emergency Response Guidelines Executive Volume
  - **4)** WOG Emergency Response Guidelines Maintenance Program Summary
- **b.** Calculation 514.2, Turkey Point EOP Setpoints - Miscellaneous (J, K, L, P, Q, X, Y Series)

4.1.2 Developmental References (continued)

6. (continued)

c. Calculation 511.2, Turkey Point EOP Setpoints - Tank Levels (U Series)

d. Calculation 505.3, Turkey Point EOP Setpoints - Steam Generator Level (M, N Series, X.1, X.2)

e. Calculation 509.2, Turkey Point EOP Setpoints - Flows (S Series)

f. Calculation 502.2, Turkey Point EOP Setpoints - RCS Temperature (E Series, F Series, G Series, H Series, I Series)

g. Calculation 510.2, Turkey Point EOP Setpoints-Containment Parameters (T Series)

h. PTN-ENG-SENS-98-047, AFW Pump Low-Flow Operating Restrictions

4.1.3 Management Directives

None

4.2 <u>Commitments</u>

None

End of Section 4.0

ATTACHMENT 1
<u>Local AFW Isolation of Faulted S/Gs</u>
(Page 1 of 1)

<u>IF</u> this valve will NOT close:	**<u>THEN</u> unlock and close:**
CV-4-2816	4-20-141, AFW To S/G 4A Train 1
CV-4-2831	AFPD-4-008, AFW To S/G 4A Train 2 Isol.
CV-4-2817	4-20-241, AFW To S/G 4B Train 1 Isol.
CV-4-2832	AFPD-4-007, AFW To S/G 4B Train 2 Isol.
CV-4-2818	4-20-341, AFW To S/G 4C Train 1 Isol.
CV-4-2833	AFPD-4-006, AFW To S/G 4C Train 2 Isol.

End of Attachment 1

ATTACHMENT 2
<u>Faulted S/G Isolation</u>
(Page 1 of 2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**<u>IF</u> 4A S/G Is Faulted, <u>THEN</u> Perform The Following:**	
	a. Locally place Breaker 4D01-26, 4A Steam Supply To Aux Feedwater Pumps MOV-4-1403, in OFF	
	b. Locally close MOV-4-1403	
	c. Check MOV-4-1404 – ALIGNED TO TRAIN 2	**c.** <u>IF</u> MOV-4-1405 open, <u>THEN</u> perform the following: **1)** Locally unlock and open AFSS-4-006. **2)** Locally unlock and close AFSS-4-007.
	d. Check MOV-4-1404 – OPEN	**d.** Manually or locally open MOV-4-1404.
	e. Check AFW Trains – AT LEAST <u>ONE</u> AVAILABLE	**e.** Manually or locally align valves to restore at least <u>one</u> AFW Train.
2.	**<u>IF</u> 4B S/G Is Faulted, <u>THEN</u> Perform The Following:**	
	a. Locally place Breaker 40806, 4B Steam Supply To Aux Feedwater Pumps MOV-4-1404, in OFF	
	b. Locally close MOV-4-1404	
	c. Check AFW Trains – AT LEAST <u>ONE</u> AVAILABLE	**c.** Manually or locally align valves to restore at least <u>one</u> AFW Train.

ATTACHMENT 2
Faulted S/G Isolation
(Page 2 of 2)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.	**IF 4C S/G is faulted, THEN perform the following:**	
	a. Locally place Breaker 3D01-28, 4C Steam Supply To Aux Feedwater Pumps MOV-4-1405, in OFF	
	b. Locally close MOV-4-1405	
	c. Check MOV-4-1404 – ALIGNED TO TRAIN 1	**c.** IF MOV-4-1403 open, THEN perform the following: **1)** Locally unlock and open AFSS-4-007. **2)** Locally unlock and close AFSS-4-006.
	d. Check MOV-4-1404 – OPEN	**d.** Manually or locally open MOV-4-1404.
	e. Check AFW Trains – AT LEAST ONE AVAILABLE	**e.** Manually or locally align valves to restore at least one AFW Train.

End of Attachment 2

ATTACHMENT 3
Continuous Action Summary
(Page 1 of 1)

Step 7, Check If Reactor Is Subcritical:

WHEN both of the following conditions are met:

- Power Range Channels – LESS THAN 5%
- Intermediate Range Channels – NEGATIVE STARTUP RATE

THEN observe CAUTION prior to Step 16 and go to Step 16.

Step 8, Check S/G Level:

IF Narrow Range Level in at least one S/G **NOT** greater than 7%[27%],
THEN establish total feedwater flow greater than 800 gpm.

IF feed flow **NOT** greater than 800 gpm, THEN manually start pumps and align valves to establish greater than 800 gpm.

Maintain total feedwater flow greater than 800 gpm until Narrow Range Level greater than 7%[27%] in at least one S/G.

Control feed flow to maintain Narrow Range Level between 21%[27%] and 50%.

End of Attachment 3

FOLDOUT PAGE
For Procedure 4-EOP-FR-S.1

1. ADVERSE CONTAINMENT CONDITIONS

a. IF either condition listed below occurs, THEN use [Adverse Containment Setpoints]:

* Containment atmosphere temperature ≥ 180°F

  OR

* Containment radiation levels ≥ 1.3×10^5 R/hr

b. WHEN Containment atmosphere temperature returns to less than 180°F, THEN Normal Setpoints can again be used.

c. WHEN Containment radiation levels return to less than 1.3×10^5 R/hr, THEN Normal Setpoints can again be used if the TSC determines that Containment Integrated Dose has **NOT** exceeded 10^5 Rads.

L-16-1 NRC Exam

In-Plant - JPM J

JPM TITLE: Control S/G Level Locally with AFW Control Valve

JPM NUMBER: 04075002300 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 04075002300 / Control Steam Generator Level Locally with Auxiliary Feedwater Control Valve

K/A NUMBERS: APE 054 AA1.01 **K/A VALUE:** RO 4.5 / SRO 4.4

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: X Perform: ☐

EVALUATION LOCATION:

In-Plant:	X	Control Room:	
Simulator:		Other:	
Lab:			

Time for Completion: 20 Minutes Time Critical: No

Alternate Path [NRC]: Yes

Alternate Path [INPO]: Yes

Developed by:	**Brian Clark** Instructor/Developer	**6/20/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
1-0	Updated to fleet template; text/grammar changes	Updated for 2014 LOCT Annual Exam	1933041-8	N/A / N/A	N/A / N/A
2-0	Formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A / N/A	N/A / N/A

SIMULATOR SET-UP:

- N/A

Required Materials:	• Handout 3-ONOP-075
General References:	• 3-ONOP-075, Auxiliary Feedwater System Malfunction
Task Standards:	• Following the loss of main feedwater, operate and monitor Auxiliary Feedwater System controls • Given a failure in the Auxiliary Feedwater System, obtain an alternate source of feedwater by manually opening CV-3-2818

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 3 is in Mode 3, following a reactor trip.
- All systems responded as designed.
- The plant is in a normal electrical alignment.
- The AFW System is intact.
- CV-3-2833, Train 2 AFW Flow Control Valve to 3C S/G, is OOS for maintenance.
- An AFW auto-start occurred with all components operating as designed except Control Room operators are unable to establish feedwater flow to the 3C S/G.

INITIATING CUE:

- The Field Supervisor provides the required key and directs you to investigate and attempt to restore feedwater flow to the 3C S/G, using 3-ONOP-075.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: **When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).**

NOTE: **Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.**

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 3-ONOP-075, Auxiliary Feedwater System Malfunction.
Evaluator Note:	If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue.
Evaluator Cue:	Provide examinee with a copy of handout 3-ONOP-075.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 2 **Critical: No**	**3-ONOP-075, Step 3.2.1:** **CHECK** at least <u>one</u> of the following is energized by Offsite Power: • 3A 4KV Bus • 3B 4KV Bus
Standard:	Recognize that Unit 3 is being supplied from offsite power.
Evaluator Note:	Initial conditions stated that the plant is in a normal electrical lineup (following a reactor/turbine trip).
Evaluator Cue:	If examinee attempts to call the Control Room to verify status, state that the 3A and 3B 4kV buses are being supplied from the Startup Transformer.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 3 **Critical: No**	**3-ONOP-075, Step 3.2.2:** **ENSURE** AFW System Intact A. **CHECK** AFW steam supply line intact B. **CHECK** AFW feed line intact
Standard:	Recognize that the AFW System is intact.
Evaluator Note:	Initial conditions stated that the AFW System is intact.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 4 **Critical: No**	**3-ONOP-075, Step 3.2.3:** **ENSURE** AFW system operation A. **CHECK** AFW steam supply MOVs open: • MOV-3-1403, 3A Steam Supply to AFW Pumps • MOV-3-1404, 3B Steam Supply to AFW Pumps • MOV-3-1405, 3C Steam Supply to AFW Pumps
Standard:	Recognize that all steam supply MOVs are open.
Evaluator Note:	• Initial conditions stated that all components are operating as designed except Control Room operators are unable to establish feedwater flow to the 3C S/G. • Examinee may verify locally or call Control Room to verify positions
Evaluator Cue:	• If examinee properly identifies/checks position of steam supply MOVs, state that the valve indicators are full up and no threads are showing • If examinee attempts to call the Control Room to verify status, state that all Unit 3 AFW steam supply MOVs indicate open
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: No**	**3-ONOP-075, Step 3.2.3:** **ENSURE** AFW system operation B. **CHECK** AFW regulating valves OPEN
Standard:	Recognize that CV-3-2818, Train 1 AFW Flow Control Valve to 3C S/G, is closed.
Evaluator Note:	• Initiating cue directs use of the ONOP to restore Train 1 flow to the 3C S/G only; hence, the examinee may only check CV-3-2818 • Examinee may not perform this step until after transitioning to Attachment 3 (refer to Performance Step 7 below)
Evaluator Cue:	• When examinee properly identifies/checks the position of CV-3-2818, state that valve position indicator indicates closed • If examinee checks the position of other regulating valves, state that they are throttled open • If examinee checks local flow indicators, state that flow to the 3A and 3B S/Gs is 135 gpm in each train and flow to the 3C S/G is 0 gpm • If examinee attempts to call the Control Room for AFW flow status, state that flow to the 3A and 3B S/Gs is 135 gpm in each train, but flow to the 3C S/G is 0 gpm
Performance:	**SATISFACTORY** ______ **UNSATISFACTORY** ______
Comments:	

Performance Step: 6 **Critical: No**	**3-ONOP-075, Step 3.2.3.B (RNO):** **PERFORM** the following: 1. Manually **OPEN** the valves
Standard:	Recognize that the attempted manual opening of CV-3-2818 (i.e., from the Control Room) was NOT successful.
Evaluator Note:	Examinee may contact the Control Room to manually open the valve or may recognize from the initial conditions that prior attempts to open the valve were unsuccessful.
Evaluator Cue:	• If examinee attempts to call the Control Room to open the valve, state that the valve is demanded open but is NOT responding • Any attempt to manually open valve should be indicated as unsuccessful
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

THIS BEGINS THE ALTERNATE-PATH PORTION OF THE JPM

Performance Step: 7 **Critical: No**	**3-ONOP-075, Step 3.2.3.B (RNO):** **PERFORM** the following: 2. If valves will **NOT** OPEN, THEN **DIRECT** personnel to perform Attachment 3, Manual Control of AFW Regulating Valves, while continuing with this procedure
Standard:	Transition to Attachment 3.
Evaluator Note:	The intent is for the examinee to recognize the need for, and then perform, Attachment 3.
Evaluator Cue:	If examinee attempts to call the Control Room for direction or continue with Section 3.2, indicate that the Control Room calls/states, “Perform Attachment 3 of 3-ONOP-075, Auxiliary Feedwater System Malfunction.”
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical: Yes**	**3-ONOP-075, Attachment 3, Step 1:** **PERFORM** the following for the applicable feed flow control valve: • **TO MANUALLY OPERATE** CV-3-2818 (Train 1 AFW Flow Control Valve to 3C S/G), **CLOSE VALVES** 3-40-268 (Air/N_2 Station #1 Train 1 to AFW CV-3-2818 Root) and 3-40-269 (Air/N_2 Station #1 Train 2 to AFW CV-3-2818 Root)
Standard:	• Cut seal wire and close valve 3-40-268 • Verify that 3-40-269 is closed
Evaluator Note:	• 3-40-268 is normally sealed open and 3-40-269 is normally sealed closed • Closing 3-40-268 is the only critical activity in this step
Evaluator Cue:	• When examinee properly identifies 3-40-268 and simulates cutting the seal wire and closing the valve, tell examinee that the valve handle turned clockwise ¼ turn and is perpendicular to the piping • When examinee properly identifies 3-40-269 and simulates checking the valve closed, tell examinee that the valve handle is perpendicular to the piping
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical: Yes**	**3-ONOP-075, Attachment 3, Step 2:** **OPEN** drain valves below the two pressure regulators associated with each control valve AND **BLEED** off the air/nitrogen pressure
Standard:	Open both regulator drain valves on CV-3-2818.
Evaluator Note:	The Evaluator Cue may be given for <u>each</u> regulator drain valve or after examinee simulates opening <u>both</u> drain valves.
Evaluator Cue:	When examinee properly identifies/simulates opening the regulator drain valves, tell examinee that flow noise is initially heard from the drain valves and flow eventually stops.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 10 **Critical: Yes**	**3-ONOP-075, Attachment 3, Step 3:** **UNLOCK** control valve handwheel AND **OPEN** the valve until flow is detected on either of the following: • Upper platform: FI-3-1458A1
Standard:	Unlock the handwheel for CV-3-2818, open the valve, and verify flow to the 3C S/G.
Evaluator Cue:	• When examinee properly identifies/simulates unlocking and opening CV-3-2818, indicate that the position indicator is in the throttled position and flow noise is heard • When examinee properly identifies/simulates checking FI-3-1458A1, indicate that flow is approximately 150 gpm (or whatever flow rate examinee targeted) • If examinee identifies FI-3-1458A1 and simulates adjusting CV-3-2818 while watching flow indicator, indicate that flow is approximately 150 gpm (or whatever flow rate examinee targeted)
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 11 **Critical: No**	**3-ONOP-075, Attachment 3, Step 4:** **ADJUST** AFW flow to maintain S/G levels at approximately 60% wide range, as indicated on <u>any</u> of the following: • LI-3-497B, S/G C Wide Range Level Indicator
Standard:	Observe the 3C S/G level locally on LI-3-497B.
Evaluator Cue:	When examinee properly identifies LI-3-497B, indicate that the 3C S/G's wide range level is 60%.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When examinee attains 60% wide range level in the 3C S/G, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ____________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 3 is in Mode 3, following a reactor trip.
- All systems responded as designed.
- The plant is in a normal electrical alignment.
- The AFW System is intact.
- CV-3-2833, Train 2 AFW Flow Control Valve to 3C S/G, is OOS for maintenance.
- An AFW auto-start occurred with all components operating as designed.
- An AFW auto-start occurred with all components operating as designed except Control Room operators are unable to establish feedwater flow to the 3C S/G.

INITIATING CUE:

- The Field Supervisor provides the required key and directs you to investigate and attempt to restore feedwater flow to the 3C S/G, using 3-ONOP-075.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

FPL

TURKEY POINT UNIT 3

OFF NORMAL OPERATING PROCEDURE

SAFETY RELATED

CONTINUOUS USE

Procedure No.

3-ONOP-075

Revision No.

6

Title:

AUXILIARY FEEDWATER SYSTEM MALFUNCTION

Responsible Department: OPERATIONS

Special Considerations:

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED today INITIAL

Revision	Approved By	Approval Date
2	Rich Tucker	08/03/12
6	Grant Melin	05/03/16

UNIT #	UNIT 3
DATE	
DOCT	PROCEDURE
DOCN	3-ONOP-075
SYS	
STATUS	COMPLETED
REV	6
# OF PGS	

REVISION SUMMARY

Rev. No.	Description
6	**PCR 2113197, 05/03/16, Luis Jimenez** Updated zone lists where fire may require the entry into this procedure per **EC 282069.** Updated name of referenced procedure 0-ONOP-016.10.
5	**PCR 2095548, 12/06/15, Brian Fitzgerald** Revise Step 4 of Attachment 3 of Procedure 3-ONOP-075 in accordance with the comp action of AR 2094143.
4	**PCR 2003181, 02/02/15, Jonathan Lubert** Enhancement to provide guidance to check the DC White Light ON for AFW regulating valves. **PCR 1807331** - Referenced 3-NOP-073, Condensate System, for placing the Condensate System in service in Steps 4.D.1 RNO and 7.A.1 RNO.
3	**AR 1807143, 11/07/12, Joe Madison** Revise RNO in Section 3.2, Steps 4B, 5B, 6C, and 7C(5) to correct the reference to the BFIV control switch Bypass position. The Bypass switch position was removed in a revision to **EC 242442**, replaced with a redundant Auto position.
2A	**AR 1809001, 10/04/12, Bruce Fulbright** Editorial correction to Entry Condition, Section 2.0, Step 3, should read “Any of the following valves have failed to open.”

TABLE OF CONTENTS

1.0 PURPOSE

This procedure provides methods for obtaining alternate sources of feedwater in the unlikely event of a complete failure of the Auxiliary Feedwater System and is used in all MODES of operation resulting from a Loss of AFW.

2.0 ENTRY CONDITIONS

1. Fire in any of the following areas:

- Zone 31, Unit 4 Containment Spray Pump Room
- Zone 32, Unit 4 Sample Room
- Zone 33, Units 3 and 4 Post Accident Sampling System Room
- Zone 34, Unit 4 Boric Acid Evaporator Package Room
- Zone 35, Units 3 and 4 Valve Room
- Zone 36, Unit 3 Boric Acid Evaporator Package Room
- Zone 37, Unit 3 Sample Room and Gas Stripper Room
- Zone 38, Unit 3 Containment Spray Pump Room
- Zone 39, Unit 3 Concentrate Holding Tank Room
- Zone 48, Units 3 and 4 Deborating Demineralizer Tank Room
- Zone 49, Units 3 and 4 Base and Cation Radwaste Demineralizers
- Zone 50, Units 3 and 4 Purification Demineralizers Room
- Zone 51, Units 3 and 4 Condensate Pump and Monitor Tank Room
- Zone 58, Units 3 and 4 Auxiliary Building Hallway
- Zone 70, 4160 V Switchgear 3B Room
- Zone 79A, Units 3 and 4 Auxiliary Building North-South Breezeway
- Zone 84, Units 3 and 4 Auxiliary Feedwater Pump Area
- Zone 96, Unit 3 480V Load Centers C and D Room
- Zone 97, Units 3 and 4 Mechanical Equipment Room

2.0 ENTRY CONDITIONS (continued)

1. (continued)

- Zone 98, Units 3 and 4 Cable Spreading Room
- Zone 106, Units 3 and 4 Control Room
- Zone 106R, Units 3 and 4 Control Room Roof
- Zone 108A, Units 3 and 4 A DC Equipment Room
- Zone 108B, Units 3 and 4 B DC Equipment Room
- Zone 132, Units 3 and 4 Control Room Electrical Cable Chase

2. When flow is required, Auxiliary Feedwater flow indicators show a loss of flow or **NO** flow condition.

3. Any of the following valves have failed to OPEN:

- MOV-3-1403, 3A STM SUPPLY TO AUX FEEDWATER PUMPS
- MOV-3-1404, 3B STM SUPPLY TO AUX FEEDWATER PUMPS
- MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS

4. Personnel report a steam line or feed line break.

5. Changes in PZR level or pressure and RCS temperature indicate a degenerated heat sink.

6. Loss of offsite power.

7. Loss of all AC power.

8. Failure of the pumps, valves, and/or associated piping system.

9. 3-ECA-0.0, Loss of All AC Power.

10. 0-ONOP-016.10, Safe Shutdown Manual Actions, with a confirmed fire in Zone 31 through 39, 48 through 51, 58,70, 79A, 84, 96, 97, 98, 106, 106R, 108A, 108B, or 132 AND a loss of AFW, if required.

11. 3-ONOP-004, Loss of Offsite Power, when AFW is required.

2.0 ENTRY CONDITIONS (continued)

12. MODES 1 through 3 when an AFW System break develops, **NOT** resulting in a Reactor Trip or Safety Injection.

13. MODES 4 through 6 when an AFW break develops.

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

None

3.2 Subsequent Operator Actions

NOTE

- With Emergency Operating Procedures in effect, except when directed by 3-EOP-ECA-0.0, Loss of All AC, this procedure is Reference Use only.
- Performance of local actions may require portable emergency lighting.
- DWST minimum volume of 145,000 gallons (9' 2") is sufficient to provide 77,000 gallons to remove decay heat for 6 hours for a single unit or 2 hours for two units.

CAUTION

- A Standby S/G Feed Pump is required to be operating and providing feedwater to the steam generators within 20 minutes.
- Prior to starting the second Standby S/G Feed Pump, a minimum indicated DWST level of 8 feet is required to prevent cavitation.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. CHECK at least one of the following is energized by Offsite Power: • 3A 4KV Bus • 3B 4KV Bus	**PERFORM** the following: **A.** IF any of the following conditions apply: • AFW System is capable of maintaining a secondary heat sink • AFW is **NOT** required to maintain a secondary heat sink THEN **GO TO** Section 3.2, Step 2. **B. OBSERVE** NOTE prior to Section 3.2, Step 5 AND **GO TO** Section 3.2, Step 5.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.2	**Subsequent Operator Actions (continued)**	
2.	**ENSURE** AFW System Intact.	
A.	**CHECK** AFW steam supply line intact.	**DIRECT** personnel to perform Attachment 1, AFW Steam Supply Line Fault Isolation.
B.	**CHECK** AFW feed line intact.	**DIRECT** personnel to perform Attachment 2, AFW Line Fault Isolation.
3.	**ENSURE** AFW System operation.	
A.	**CHECK** AFW steam supply MOVs OPEN: • MOV-3-1403, 3A STM SUPPLY TO AUX FEEDWATER PUMPS • MOV-3-1404, 3B STM SUPPLY TO AUX FEEDWATER PUMPS • MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS	IF <u>any</u> of the following valves are **NOT** OPEN, THEN manually **OPEN**: • MOV-3-1403, 3A STM SUPPLY TO AUX FEEDWATER PUMPS • MOV-3-1404, 3B STM SUPPLY TO AUX FEEDWATER PUMPS • MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.2 Subsequent Operator Actions (continued)

3. (continued)

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B.	**CHECK** AFW regulating valves OPEN.	**PERFORM** the following: **1.** Manually **OPEN** the valves. **2.** IF valves will **NOT** OPEN, THEN **DIRECT** personnel to perform Attachment 3, Manual Control of AFW Regulating Valves, while continuing with this procedure. **3.** **CHECK** DC White Light ON for AFW regulating valves. **a.** IF DC White Light is **NOT** ON, THEN locally **RESET** AND **CLOSE** applicable breaker(s). • 3D01-14 for Train 1 • 3D23-6 for Train 2 • 3D23-20 for Train 2
C.	**CHECK** <u>all</u> AFW Pumps operating at approximately 6,000 rpm.	**PERFORM** the following: **1.** IF AFW Pump tripped on mechanical overspeed, THEN **DIRECT** personnel to perform Attachment 4, Resetting Mechanical Overspeed Trip Following an Auto Start Signal, while continuing with this procedure. **2.** IF malfunction of the AFW Pump Turbine Throttle and Trip Valve is suspected, THEN **DIRECT** personnel to perform Attachment 5, Manual Control of AFW Pump Turbine Throttle and Trip (T&T) Valve, while continuing with this procedure.

STEP | **ACTION/EXPECTED RESPONSE** | **RESPONSE NOT OBTAINED**

3.2 Subsequent Operator Actions (continued)

3. (continued)

D. CHECK AFW flow indicators delivering the desired flow.

PERFORM the following:

1. **CHECK** DC White Light ON for AFW regulating valves.

   a. IF DC White Light is **NOT** ON, THEN locally **RESET** AND **CLOSE** applicable breaker(s).

   - 3D01-14 for Train 1
   - 3D23-6 for Train 2
   - 3D23-20 for Train 2

2. **DISPATCH** personnel to confirm <u>all</u> valves are properly positioned to provide a flow path from the CST to the steam generators.

3. **CONTINUE** with Section 3.2, Step 4.

E. GO TO appropriate plant procedure as determined by the Shift Manager.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.2 Subsequent Operator Actions (continued)

NOTE

When Instrument Air is **NOT** available OR power to the setpoint station is lost, Feedwater Bypass Flow Control Valves can be operated manually using handwheels.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**ESTABLISH** Main Feedwater flow to at least <u>one</u> S/G.	
A.	**ENSURE** Feedwater Bypass Isolation, RESET.	
B.	**ENSURE** Feedwater Bypass Isolation valves are OPEN. • POV-3-477, 3A F/W BYPASS ISOLATION • POV-3-487, 3B F/W BYPASS ISOLATION • POV-3-497, 3C F/W BYPASS ISOLATION	**PERFORM** the following: 1. **ENSURE** Feedwater Bypass Isolation valve control switch is in either AUTO position. 2. IF applicable Feedwater Bypass Isolation valve will **NOT** OPEN in AUTO, THEN locally **OPEN** valve.
C.	**OPEN** Feedwater Bypass valves between 5% and 10%. • FCV-3-479, 3A F/W BYPASS • FCV-3-489, 3B F/W BYPASS • FCV-3-499, 3C F/W BYPASS	Manually **OPEN** applicable valve.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.2	**Subsequent Operator Actions (continued)**	
4.	**(continued)**	
D.	**CHECK** Condensate System in service.	**PERFORM** the following: **1. PLACE** Condensate System in service per 3-NOP-073, Condensate System. **2.** IF Condensate System can **NOT** be placed in service, THEN **OBSERVE** NOTE prior to Section 3.2, Step 5 and **GO TO** Section 3.2, Step 5.
E.	**ESTABLISH** Main Feedwater flow. **(1) START** a S/G Feedwater Pump.	IF Main Feedwater flow can **NOT** be established, THEN **OBSERVE** NOTE prior to Section 3.2, Step 5 and **GO TO** Section 3.2, Step 5.
	(2) ADJUST Feedwater Bypass Valves to restore S/G level to greater than 7%. • FCV-3-479, 3A F/W BYPASS • FCV-3-489, 3B F/W BYPASS • FCV-3-499, 3C F/W BYPASS	Manually **OPEN** applicable valve.
F.	**MAINTAIN** S/G levels. **(1)** Narrow range level in at least <u>one</u> S/G greater than 7%. **(2)** Control feed flow to maintain levels between 21% and 50%.	
G.	**RETURN TO** Section 3.2, Step 2.	

STEP **ACTION/EXPECTED RESPONSE** **RESPONSE NOT OBTAINED**

3.2 Subsequent Operator Actions (continued)

NOTE

Lost of AFW due to a fire in any the following Zones:

- Zone 31, Unit 4 Containment Spray Pump Room
- Zone 32, Unit 4 Sample Room
- Zone 33, Units 3 and 4 Post Accident Sampling System Room
- Zone 34, Unit 4 Boric Acid Evaporator Package Room
- Zone 35, Units 3 and 4 Valve Room
- Zone 36, Unit 3 Boric Acid Evaporator Package Room
- Zone 37, Unit 3 Sample Room and Gas Stripper Room
- Zone 38, Unit 3 Containment Spray Pump Room
- Zone 39, Unit 3 Concentrate Holding Tank Room
- Zone 48, Units 3 and 4 Deborating Demineralizer Tank Room
- Zone 49, Units 3 and 4 Base and Cation Radwaste Demineralizers
- Zone 50, Units 3 and 4 Purification Demineralizers Room
- Zone 51, Units 3 and 4 Condensate Pump and Monitor Tank Room
- Zone 58, Units 3 and 4 Auxiliary Building Hallway
- Zone 70, 4160 V Switchgear 3B Room
- Zone 84, Units 3 and 4 Auxiliary Feedwater Pump Area
- Zone 97, Units 3 and 4 Mechanical Equipment Room
- Zone 98, Units 3 and 4 Cable Spreading Room
- Zone 106, Units 3 and 4 Control Room
- Zone 106R, Units 3 and 4 Control Room Roof
- Zone 132, Units 3 and 4 Control Room Electrical Cable Chase

Will require B Standby S/G FW Pump to be started locally by placing the Master Control switch to RUN or MANUAL. Master Control switch is located on the Pump Control Panel.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.2	**Subsequent Operator Actions (continued)**	
5.	**ESTABLISH** Standby Feedwater flow to at least <u>one</u> S/G from B SSGFP.	
	A. ENSURE Feedwater Bypass Isolation, RESET.	
	B. ENSURE Feedwater Bypass Isolation valves are OPEN. • POV-3-477, 3A F/W BYPASS ISOLATION • POV-3-487, 3B F/W BYPASS ISOLATION • POV-3-497, 3C F/W BYPASS ISOLATION	**PERFORM** the following: **1. ENSURE** Feedwater Bypass Isolation valve control switch is in either AUTO position. **2.** IF applicable Feedwater Bypass Isolation valve will **NOT** OPEN in AUTO, THEN locally **OPEN** valve.
	C. CHECK PI-3-1616, FEEDWATER HEADER REMOTE PRESSURE INDICATOR, greater than 500 psig.	Locally **THROTTLE** OPEN, three turns DWDS-3-012, ISOLATION VALVE STBY SG FEED PUMPS DISCH TO UNIT 3 MAIN FW HEADER.
	D. START B Standby S/G FW Pump.	IF B Standby S/G Feedwater Pump can **NOT** be started, THEN **GO TO** Section 3.2, Step 6.
	E. Locally **OPEN** DWDS-3-012, ISOLATION VALVE STBY SG FEED PUMPS DISCH TO UNIT 3 MAIN FW HEADER.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.2 Subsequent Operator Actions (continued)

5. (continued)

F. **ADJUST** Feedwater Bypass Valves to restore S/G level to greater than 7%.

- FCV-3-479, 3A F/W BYPASS
- FCV-3-489, 3B F/W BYPASS
- FCV-3-499, 3C F/W BYPASS

RESPONSE NOT OBTAINED:

PERFORM the following:

1. Manually **OPEN** applicable valve.
2. IF standby feedwater flow can **NOT** be established, THEN **GO TO** Section 3.2, Step 6.

G. **MAINTAIN** S/G levels.

(1) Narrow range level in at least one S/G greater than 7%.

(2) Control feed flow to maintain levels between 21% and 50%.

H. **RETURN TO** Section 3.2, Step 2.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.2	**Subsequent Operator Actions (continued)**	
6.	**ESTABLISH** Standby Feedwater flow to at least <u>one</u> S/G from A SSGFP.	
A.	**CHECK** 3C 4KV Bus energized.	**GO TO** Section 3.2, Step 7.
B.	**ENSURE** Feedwater Bypass Isolation, RESET.	
C.	**ENSURE** Feedwater Bypass Isolation valves are OPEN. • POV-3-477, 3A F/W BYPASS ISOLATION • POV-3-487, 3B F/W BYPASS ISOLATION • POV-3-497, 3C F/W BYPASS ISOLATION	**PERFORM** the following: **1.** **ENSURE** Feedwater Bypass Isolation valve control switch is in either AUTO position. **2.** IF applicable Feedwater Bypass Isolation valve will **NOT** OPEN in AUTO, THEN locally **OPEN** valve.
D.	**CHECK** PI-3-1616, FEEDWATER HEADER REMOTE PRESSURE INDICATOR, greater than 500 psig.	Locally **THROTTLE** OPEN, three turns DWDS-3-012, ISOLATION VALVE STBY SG FEED PUMPS DISCH TO UNIT 3 MAIN FW HEADER.
E.	**START** the A SSGFP.	IF A SSGFP can **NOT** be started, THEN **GO TO** Section 3.2, Step 7.
F.	Locally **OPEN** DWDS-3-012, ISOLATION VALVE STBY SG FEED PUMPS DISCH TO UNIT 3 MAIN FW HEADER.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.2 Subsequent Operator Actions (continued)

6. (continued)

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
G.	**ADJUST** Feedwater Bypass Valves to restore S/G level to greater than 7%. • FCV-3-479, 3A F/W BYPASS • FCV-3-489, 3B F/W BYPASS • FCV-3-499, 3C F/W BYPASS	**PERFORM** the following: **1.** Manually **OPEN** applicable valve. **2.** IF standby feedwater flow can **NOT** be established, THEN **GO TO** Section 3.2, Step 7.
H.	**MAINTAIN** S/G levels. **(1)** Narrow range level in at least <u>one</u> S/G greater than 7%. **(2)** Control feed flow to maintain levels between 21% and 50%.	
I.	**RETURN TO** Section 3.2, Step 2.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3.2 Subsequent Operator Actions (continued)

NOTE

When S/G depressurization is required, safety injection may actuate if **NOT** previously blocked by plant procedure.

7. **ESTABLISH** feed flow from Condensate System.

A. CHECK Condensate System in service.	**PERFORM** the following: **1. PLACE** Condensate System in service per 3-NOP-073, Condensate System. **2.** IF Condensate System can **NOT** be placed in service, THEN **RETURN TO** Section 3.2, Step 2.
B. ENSURE at least <u>one</u> S/G is depressurized to less than 430 psig.	**PERFORM** the following: **1.** IF Condenser is available, THEN manually **DUMP** steam to condenser. **2.** IF Condenser **NOT** available, THEN manually **DUMP** steam to atmosphere.
C. ESTABLISH Condensate flow.	IF Condensate flow can **NOT** be established, THEN **RETURN TO** Section 3.2, Step 2.

(1) Locally **OPEN** 3-FWDR-001, FEEDWATER PUMP BYPASS VALVE.

(2) **START** a condensate pump.

(3) **ENSURE** SI RESET.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3.2	**Subsequent Operator Actions (continued)**	
7. C.	**(continued)**	
	(4) **ENSURE** Feedwater Bypass Isolation, RESET.	
	(5) **ENSURE** Feedwater Bypass Isolation valves are OPEN. • POV-3-477, 3A F/W BYPASS ISOLATION • POV-3-487, 3B F/W BYPASS ISOLATION • POV-3-497, 3C F/W BYPASS ISOLATION	**PERFORM** the following: **1.** **ENSURE** Feedwater Bypass Isolation valve control switch is in either AUTO position. **2.** IF applicable Feedwater Bypass Isolation valve will **NOT** OPEN in AUTO, THEN locally **OPEN** valve.
	(6) **ADJUST** Feedwater Bypass Valves to restore S/G level to greater than 7%. • FCV-3-479, 3A F/W BYPASS • FCV-3-489, 3B F/W BYPASS • FCV-3-499, 3C F/W BYPASS	Manually **OPEN** applicable valve.
D.	**MAINTAIN** S/G levels. **(1)** Narrow range level in at least <u>one</u> S/G greater than 7%. **(2)** Control feed flow to maintain levels between 21% and 50%.	
E.	**RETURN TO** Section 3.2, Step 2.	

4.0 RECORDS

None

5.0 REFERENCES AND COMMITMENTS

5.1 References

5.1.1 Implementing

None

5.1.2 Developmental

1. Technical Specifications
   - **A.** Section 3/4.7.1.2, Auxiliary Feedwater System, Modes 1, 2, and 3
   - **B.** Section 3/4.7.1.3, Condensate Storage Tank
2. FSAR
   - **A.** Section 9.11.1, Auxiliary Feedwater System Design Basis
   - **B.** Section 9.11.2, Auxiliary Feedwater Pumps
   - **C.** Section 9.11.3, Condensate Storage Tanks
   - **D.** Section 14.1.11, Loss of Normal Feedwater
3. Plant Procedures
   - **A.** 0-ADM-212, In-Plant Equipment Clearance Orders
   - **B.** 3-EOP-ECA-0.0, Loss of All AC Power
   - **C.** 3-NOP-075, Auxiliary Feedwater System
   - **D.** 3-ONOP-004, Loss of Offsite Power
   - **E.** 0-ONOP-016.10, Safe Shutdown Manual Actions

5.1.2 Developmental (continued)

4. Miscellaneous Documents

   A. PC/M 90-238, C Bus Switchgear Control and Protection Power Isolation for Appendix R

   B. PC/M 94-059, Diesel Engine Driver for Standby Steam Gen Feedwater Pump P82B

   C. PC/M 96-022, Thermal Power Uprate Implementation

   D. PC/M 97-033, Elimination of Electrical Trip to Auxiliary Feedwater Turbines

   E. PC/M 99-059, Outdoor Electrical Raceway Fire-Proofing Requirements

   F. PC/M 04-0145, Feedwater Warmup Line Abandonment

   G. PC/M 04-112, Emergency Response Data Acquisition and Display (ERDADS) Replacement

   H. EC 242442, PCM-08106 Feedwater Isolation Valve Upgrade

   I. EC 247008, PCM-09139 EPU LAR Umbrella Doc Only

   J. EC 247006, UNIT 3 EPU Instrument Setpoint/Indication Changes

   K. EC 282069, Transition of PTN Fire Protection Licensing Basis from 10CFR50 APPENDIX R to NFPA 805

5.1.3 Management Directives

None

5.2 <u>Commitments</u>

NIR 87-07, S. D. Stadler (Feb 2 through 6, 1987), dated 4/28/87 (87-1019-34)

ATTACHMENT 1
AFW Steam Supply Line Fault Isolation
(Page 1 of 2)

> **NOTE**
>
> It may **NOT** be possible to initially determine the location and nature of the break. Therefore, all AFW steam may be isolated until the steam break location is determined and isolated using 0-ADM-212, In-Plant Equipment Clearance Orders.

1. **CLOSE** any of the following to isolate the steam leak:

- 3-10-119, S/G A AFW PUMP STM SUPPLY UPSTREAM ISOL
- 3-10-219, S/G B AFW PUMP STM SUPPLY UPSTREAM ISOL
- 3-10-319, S/G C AFW PUMP STM SUPPLY UPSTREAM ISOL

> **NOTE**
>
> With an AFW auto start signal present, steam supply MOVs-1403, 1404, and 1405, 3A/3B/3C STM SUPPLY TO AUX FEEDWATER PUMPS, will RE-OPEN if CLOSED. Coordination between Control Room and personnel at the breakers is needed to ensure that the appropriate breaker is OPENED as soon as the valve indicates CLOSED from the Control Room.

2. IF the following valves are inaccessible OR can **NOT** be CLOSED:

- 3-10-119, S/G A AFW PUMP STM SUPPLY UPSTREAM ISOL
- 3-10-219, S/G B AFW PUMP STM SUPPLY UPSTREAM ISOL
- 3-10-319, S/G C AFW PUMP STM SUPPLY UPSTREAM ISOL

THEN:

A. **CLOSE** any of the following MOVs:

- MOV-3-1403, 3A STM SUPPLY TO AUX FEEDWATER PUMPS
- MOV-3-1404, 3B STM SUPPLY TO AUX FEEDWATER PUMPS
- MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS

ATTACHMENT 1
<u>AFW Steam Supply Line Fault Isolation</u>
(Page 2 of 2)

2. (continued)

B. DIRECT personnel to OPEN the applicable breakers:

- 4D01-28, MOV-3-1403, 3A STM SUPPLY TO AUX FEEDWATER PUMPS
- 30833, MOV-3-1404, 3B STM SUPPLY TO AUX FEEDWATER PUMPS
- 3D01-27, MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS

C. WHEN conditions permit, THEN locally **ENSURE** AFW Steam Supply MOVs are CLOSED.

3. WHEN conditions permit, THEN:

A. ISOLATE steam break per 0-ADM-212, In-Plant Equipment Clearance Orders.

B. ESTABLISH two trains of AFW per 3-NOP-075, Auxiliary Feedwater System.

4. IF only <u>one</u> train of AFW is available, THEN **ENSURE** at least <u>one</u> AFW Pump is supplying water to at least <u>two</u> S/Gs.

5. IF AFW is required AND can **NOT** be established, THEN **GO TO** Section 3.2, Step 4.

6. GO TO Section 3.2, Step 2.B.

ATTACHMENT 2
AFW Line Fault Isolation
(Page 1 of 2)

NOTE

- It may **NOT** be possible to initially determine the location and nature of the break. Therefore, all AFW steam may be isolated until the feedline break location is determined and isolated using 0-ADM-212, In-Plant Equipment Clearance Orders.
- With an AFW auto start signal present, Steam Supply MOVs-1403, 1404, and 1405, 3A/3B/3C STM SUPPLY TO AUX FEEDWATER PUMPS, will RE-OPEN if CLOSED. Coordination between Control Room and personnel at the breakers is needed to ensure that the appropriate breaker is OPENED as soon as the valve indicates CLOSED from the Control Room.

1. **STOP** the affected AFW Pumps by performing the following:

   A. **CLOSE** any of the following MOVs:

   - MOV-3-1403, 3A STM SUPPLY TO AUX FEEDWATER PUMPS
   - MOV-3-1404, 3B STM SUPPLY TO AUX FEEDWATER PUMPS
   - MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS

   B. **DIRECT** personnel to OPEN the applicable breakers:

   - 4D01-28, MOV-3-1403, 3A STM SUPPLY TO AUX FEEDWATER PUMPS
   - 30833, MOV-3-1404, 3B STM SUPPLY TO AUX FEEDWATER PUMPS
   - 3D01-27, MOV-3-1405, 3C STM SUPPLY TO AUX FEEDWATER PUMPS

   C. WHEN conditions permit, THEN locally **ENSURE** AFW Steam Supply MOVs are CLOSED.

2. WHEN conditions permit, THEN:

   A. **ISOLATE** feedline break per 0-ADM-212, In-Plant Equipment Clearance Orders.

   B. **ESTABLISH** two trains of AFW per 3-NOP-075, Auxiliary Feedwater System.

ATTACHMENT 2
AFW Line Fault Isolation
(Page 2 of 2)

3. IF only one train of AFW is available, THEN **ENSURE** at least one AFW Pump is supplying water to at least two S/Gs.

4. IF AFW is required AND can **NOT** be established, THEN **GO TO** Section 3.2, Step 4.

5. **GO TO** Section 3.2, Step 3.E.

ATTACHMENT 3
Manual Control of AFW Regulating Valves
(Page 1 of 3)

1. **PERFORM** the following for the applicable feed flow control valve:

TO MANUALLY OPERATE	CLOSE VALVES
CV-3-2816, TRAIN 1 S/G A FEED FLOW CONTROL VALVE	3-40-260, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2816 ROOT
	3-40-261, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2816 ROOT
CV-3-2817, TRAIN 1 S/G B FEED FLOW CONTROL VALVE	3-40-264, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2817 ROOT
	3-40-265, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2817 ROOT
CV-3-2818, TRAIN 1 S/G C FEED FLOW CONTROL VALVE	3-40-268, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2818 ROOT
	3-40-269, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2818 ROOT
CV-3-2831, TRAIN 2 S/G A FEED FLOW CONTROL VALVE	3-40-262, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2831 ROOT
	3-40-263, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2831 ROOT
CV-3-2832, TRAIN 2 S/G B FEED FLOW CONTROL VALVE	3-40-266, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2832 ROOT
	3-40-267, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2832 ROOT
CV-3-2833, TRAIN 2 S/G C FEED FLOW CONTROL VALVE	3-40-270, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2833 ROOT
	3-40-271, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2833 ROOT

2. **OPEN** drain valves below the two pressure regulators associated with each control valve AND **BLEED** off the air/nitrogen pressure.

ATTACHMENT 3
Manual Control of AFW Regulating Valves
(Page 2 of 3)

3. **UNLOCK** control valve handwheel AND **OPEN** the valve until flow is detected on either of the following:

   - Upper platform:

     * FI-3-1401A1, FLOW IND FOR AUX FEEDWATER TO STEAM GEN A

     * FI-3-1457A1, FLOW IND FOR AUX FEEDWATER TO STEAM GEN B

     * FI-3-1458A1, FLOW IND FOR AUX FEEDWATER TO STEAM GEN C

   - Lower Platform:

     * FI-3-1401B4, FLOW INDICATOR FOR AUXILIARY FEEDWATER TO STEAM GENERATOR A

     * FI-3-1457B4, FLOW INDICATOR FOR AUXILIARY FEEDWATER TO STEAM GENERATOR B

     * FI-3-1458B4, FLOW INDICATOR FOR AUXILIARY FEEDWATER TO STEAM GENERATOR C

4. **ADJUST** AFW flow to maintain S/G levels at approximately 60% wide range, as indicated on any of the following:

   - LI-3-477A, S/G A WIDE RANGE LVL IND

   - LI-3-487B, S/G B WIDE RANGE LVL IND

   - LI-3-497B, S/G C WIDE RANGE LVL IND

5. WHEN manual control of the AFW regulating valves is **NO** longer required, THEN **RETURN TO** auto control by performing the following:

   A. **PLACE** feed flow control valves to neutral alignment by performing 3-NOP-075, Auxiliary Feedwater System, using the section for feed flow control valve alignment.

   B. **CLOSE** pressure regulator drain valves that were opened in Attachment 3, Step 2.

ATTACHMENT 3
Manual Control of AFW Regulating Valves
(Page 3 of 3)

5. (continued)

C. PERFORM the following for the valve manually operated:

MANUALLY OPERATED	OPEN AND SEAL VALVE
CV-3-2816, TRAIN 1 S/G A FEED FLOW CONTROL VALVE	3-40-260, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2816 ROOT
CV-3-2817, TRAIN 1 S/G B FEED FLOW CONTROL VALVE	3-40-264, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2817 ROOT
CV-3-2818, TRAIN 1 S/G C FEED FLOW CONTROL VALVE	3-40-268, AIR/N2 STA NO 1 TRN 1 TO AFW CV-3-2818 ROOT
CV-3-2831, TRAIN 2 S/G A FEED FLOW CONTROL VALVE	3-40-263, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2831 ROOT
CV-3-2832, TRAIN 2 S/G B FEED FLOW CONTROL VALVE	3-40-267, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2832 ROOT
CV-3-2833, TRAIN 2 S/G C FEED FLOW CONTROL VALVE	3-40-270, AIR/N2 STA NO 1 TRN 2 TO AFW CV-3-2833 ROOT

ATTACHMENT 4
Resetting Mechanical Overspeed Trip Following an Auto Start Signal
(Page 1 of 1)

1. **ENSURE** adequate water supply greater than 10% in the CST(s) aligned to the AFW System.

2. **ENSURE** flow path from CST to pump suction by performing the following:

   A. **ENSURE** 3-20-400, AFW SUPPLY FROM CST, OPEN.

   B. **ENSURE** OPEN, the following for the affected pump:

      * A Pump: 3-20-144, AFW PUMP A SUCTION ISOL
      * B Pump: 3-20-244, AFW PUMP B SUCTION ISOL
      * C Pump: 3-20-344, AFW PUMP C SUCTION ISOL

3. **CHECK** by visual inspection the governor valve linkage mechanism is intact.

> **NOTE**
>
> - Level should be at or above the mid-level mark of the sightglass when turbine is shut down.
> - Level should be visible in sightglass with turbine in operation.

4. **CHECK** correct oil level in the governor housing.

> **CAUTION**
>
> The AFW pump could start after the mechanical overspeed trip is RESET.

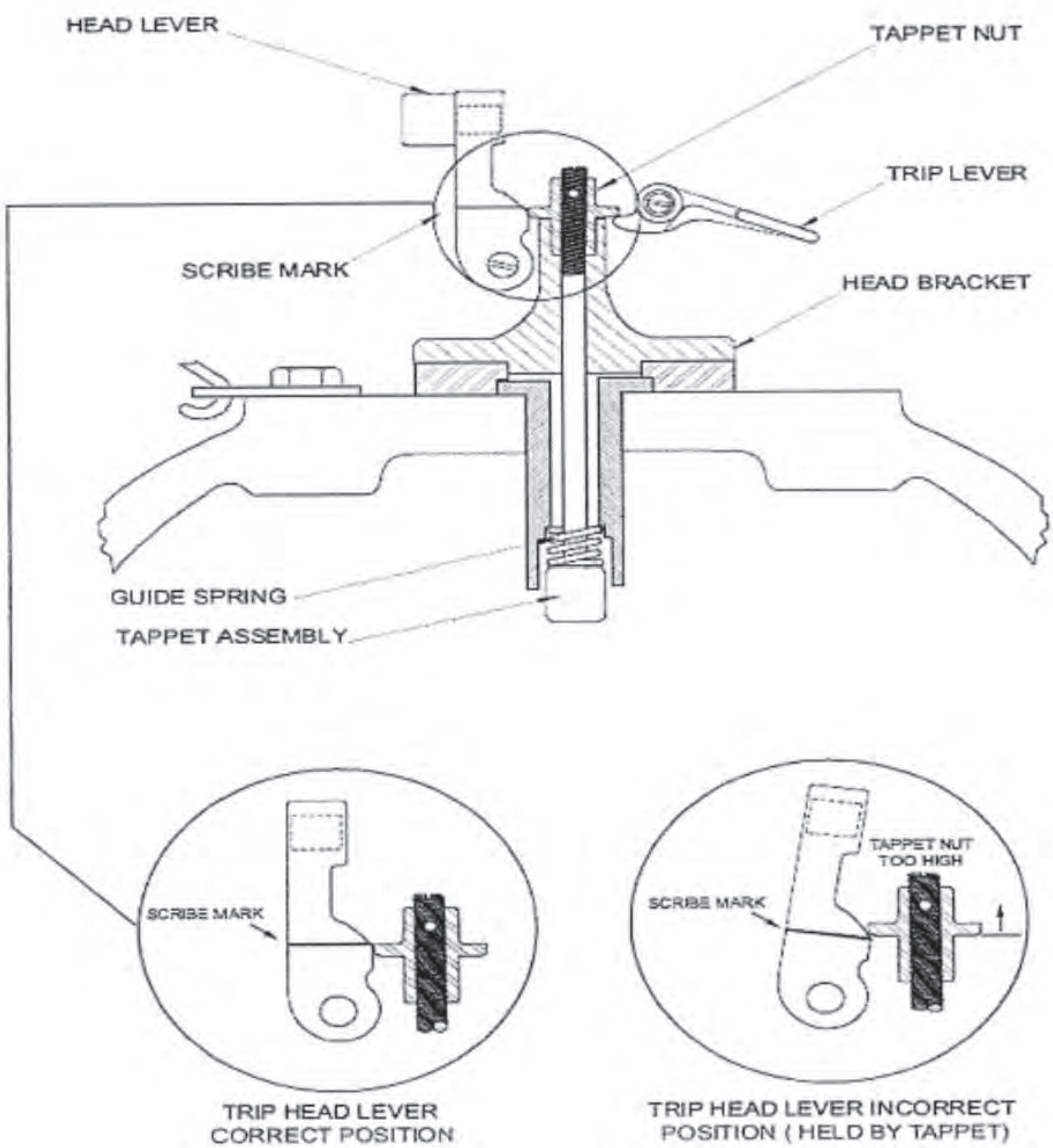
5. **RESET** mechanical overspeed trip (See Attachment 6, Mechanical Overspeed Latching Scribe Mark Location).

6. IF the AFW pump trips again on mechanical overspeed, THEN **NOTIFY** Mechanical Maintenance to determine operability of the governor.

ATTACHMENT 5
Manual Control of AFW Pump Turbine Throttle and Trip (T&T) Valve
(Page 1 of 1)

1. **CHECK** pump steam pressure available by any of the following:
   - PI-1416, STEAM SUPPLY TO AUX FW TURBINE A
   - PI-1417, STEAM SUPPLY TO AUX FW TURBINE B
   - PI-1418, STEAM SUPPLY TO AUX FW TURBINE C
2. **ENSURE** T&T valve latched.
3. **OPEN** T&T valve by engaging the handwheel AND turning the handwheel counterclockwise.
4. **ADJUST** T&T valve to obtain a pump discharge pressure approximately 150 psig greater than steam supply pressure, as indicated on any of the following:
   - PI-1429, AUX FEEDWATER PUMP A DISCH
   - PI-1430, AUX FEEDWATER PUMP A DISCH
   - PI-1431, AUX FEEDWATER PUMP C DISCH
5. **INSPECT** pump for proper operation.

ATTACHMENT 6
<u>Mechanical Overspeed Latching Scribe Mark Location</u>
(Page 1 of 1)



L-16-1 NRC Exam

In-Plant - JPM K

JPM TITLE: Align Emergency Service Water to the Charging Pumps

JPM NUMBER: 24030009300 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 24030009300 / Align Emergency Service Water to the Charging Pumps

K/A NUMBERS: APE 026 AA1.03 **K/A VALUE:** RO 3.6 / SRO 3.6

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: X Perform:

EVALUATION LOCATION: In-Plant: X Control Room:

Simulator: Other:

Lab:

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Brian Clark** Instructor/Developer	**6/20/16** Date
Reviewed by:	**Tim Hodge** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
1-0	Updated to fleet template; text/grammar changes	2015 LOCT Annual Exam	N/A	N/A N/A	N/A N/A
2-0	Formatting; text/grammar changes	L-16-1 NRC Exam	N/A	N/A N/A	N/A N/A

SIMULATOR SET-UP:

- N/A

Required Materials:	• Handout 3(4)-ONOP-030 Attachment 1
General References:	• 3(4)-ONOP-030, Component Cooling Water Malfunction
Task Standards:	• Establish emergency cooling water to the Unit 3(4) charging pumps

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Level in the CCW Surge Tank can NOT be maintained.
- The crew has entered 3(4)-ONOP-030, Component Cooling Water Malfunction.

INITIATING CUE:

- You have been directed to perform Attachment 1, Control of Emergency Cooling Water to Charging Pumps, of 3(4)-ONOP-030 for the A/B/C Charging Pump (select one and identify on Performance Step 3 and the examinee's Turnover Sheet).

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Evaluator Note:	**Prior to administering, determine which Unit AND pump the JPM will be performed on and provide the appropriate procedure and initiating cue.**

Performance Step: 1 **Critical: No**	**Obtain required reference materials, proceed to Unit 3(4) Charging Pump Room, and retrieve emergency cooling hoses from the storage box.**
Standard:	• Obtain Attachment 1, Control of Emergency Cooling Water to Charging Pumps, of 3(4)-ONOP-030, Component Cooling Water Malfunction. • Simulate removing the emergency cooling hoses from the storage box.
Evaluator Note:	If a peer check is requested for any of the following steps, then acknowledge the request and allow the operator to continue.
Evaluator Cue:	Provide examinee with a copy of handout 3(4)-ONOP-030 Attachment 1.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 1:** **CONNECT** cam lock fitting end of emergency cooling water supply hose to 3-70-179A (4-70-118B), Service Water Connection Inside (Outside) Unit 3(4) Charging Pump Room.
Standard:	Identify the correct valve and connect the supply hose to the cam lock fitting.
Evaluator Note:	• 3-70-179A is located inside of the Unit 3 Charging Pump Room; 4-70-118B is located outside of the Unit 4 Charging Pump Room • The emergency cooling water supply hose has a quick disconnect fitting on one end and a cam lock fitting on the other end
Evaluator Cue:	When the examinee properly simulates installing the supply hose on the valve, inform the examinee that the hose is connected.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: No**	**3(4)-ONOP-030, Attachment 1, Step 2:** **CONSULT** with Unit 3(4) Reactor Operator to determine desired charging pump.
Standard:	Recognize from the Initial Conditions that the Unit 3(4) A/B/C Charging Pump is to be cooled.
Evaluator Note:	If asked, refer the examinee to the Initiating Cue.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: No**	**3(4)-ONOP-030, Attachment 1, Step 3:** **ENSURE** desired Charging Pump is STOPPED OR running at maximum speed.
Standard:	Check the appropriate pump's status.
Evaluator Cue:	When the correct pump is identified/checked, inform the examinee that the shaft is NOT rotating and no pump or motor noise is heard.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 5 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 4:** **CONNECT** quick disconnect fitting end of emergency cooling water supply hose to emergency hose connection on desired Charging Pump: • 3(4)-10-291, Emergency Hose Connection to Charging Pump A Oil Cooler Supply • 3(4)-10-289, Emergency Hose Connection to Charging Pump B Oil Cooler Supply • 3(4)-10-299, Emergency Hose Connection to Charging Pump C Oil Cooler Supply
Standard:	Identify the correct pump and connect the quick disconnect fitting on the supply hose to the pump's quick disconnect supply fitting.
Evaluator Note:	The emergency cooling water supply hose has a quick disconnect fitting on one end and a cam lock fitting on the other end.
Evaluator Cue:	When the examinee identifies the correct fitting and properly simulates installing the supply hose on the pump, inform the examinee that the hose is connected.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 5:** **CONNECT** quick disconnect fitting end of emergency cooling water outlet hose to emergency hose connection on desired Charging Pump: • 3(4)-10-290, Emergency Hose Connection to Charging Pump A Oil Cooler Return • 3(4)-10-288, Emergency Hose Connection to Charging Pump B Oil Cooler Return • 3(4)-10-298, Emergency Hose Connection to Charging Pump C Oil Cooler Return
Standard:	Identify the correct pump and connect the quick disconnect fitting on the outlet hose to the pump's quick disconnect return fitting.
Evaluator Note:	The emergency cooling water outlet hose has a quick disconnect fitting on one end and no fitting on the other end.
Evaluator Cue:	When the examinee identifies the correct fitting, simulates removing the pipe plug, and properly simulates installing the outlet hose on the pump, inform the examinee that the hose is connected.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 7 **Critical: No**	**3(4)-ONOP-030, Attachment 1, Step 6:** **REMOVE** cover from floor drain to be used in Charging Pump Room.
Standard:	Identify the appropriate floor drain and remove its cover.
Evaluator Cue:	When the examinee identifies the appropriate floor drain and simulates removing its cover, inform the examinee that the cover is removed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical: No**	**3(4)-ONOP-030, Attachment 1, Step 7:** **ROUTE** open end of emergency cooling water outlet hose to floor drain being used in Charging Pump Room.
Standard:	Place the open end of the outlet hose near the appropriate floor drain.
Evaluator Cue:	When the examinee places the open end of the outlet hose near the appropriate floor drain, inform the examinee that the outlet hose is properly routed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 8:** **ISOLATE** CCW to hydraulic oil cooler on desired Charging Pump: • **CLOSE** 3(4)-825A, CCW to A Charging Pump Oil Cooler Inlet • **CLOSE** 3(4)-825C, CCW to B Charging Pump Oil Cooler Inlet • **CLOSE** 3(4)-825E, CCW to C Charging Pump Oil Cooler Inlet
Standard:	Identify/close the CCW valve to the appropriate charging pump.
Evaluator Cue:	When the examinee identifies the correct valve and properly simulates closing it, inform the examinee that the valve is closed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 10 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 9:** **ISOLATE** CCW from hydraulic oil cooler on desired Charging Pump: • **CLOSE** 3(4)-825B, CCW from A Charging Pump Oil Cooler Inlet • **CLOSE** 3(4)-825D, CCW from B Charging Pump Oil Cooler Inlet • **CLOSE** 3(4)-825F, CCW from C Charging Pump Oil Cooler Inlet
Standard:	Identify/close the CCW valve from the appropriate charging pump.
Evaluator Cue:	When the examinee identifies the correct valve and properly simulates closing it, inform the examinee that the valve is closed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 11 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 10 (10.a):** **OPEN** 3-70-179, Service Water Root Valve Outside Unit 3 Charging Pump Room (4-70-118B, Service Water Connection Outside Unit 4 Charging Pump Room).
Standard:	Identify/open the appropriate service water valve.
Evaluator Cue:	When the examinee identifies the correct valve and properly simulates opening it, inform the examinee that the valve is open.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 12 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 11 (10.b):** **OPEN** 3-70-179A, Service Water Connection Inside Unit 3 Charging Pump Room (4-70-118, Service Water Root Valve Outside Unit 4 Charging Pump Room).
Standard:	Identify/open the appropriate service water valve.
Evaluator Cue:	When the examinee identifies the correct valve and properly simulates opening it, inform the examinee that the valve is open.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 13 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 12 (11):** **ESTABLISH** Service Water to desired Charging Pump: • **OPEN** 3(4)-10-291, Emergency Hose Connection to Charging Pump A Oil Cooler Supply • **OPEN** 3(4)-10-289, Emergency Hose Connection to Charging Pump B Oil Cooler Supply • **OPEN** 3(4)-10-299, Emergency Hose Connection to Charging Pump C Oil Cooler Supply
Standard:	Identify/open the appropriate service water valve.
Evaluator Cue:	When the examinee identifies the correct valve and properly simulates opening it, inform the examinee that the valve is open.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 14 **Critical: Yes**	**3(4)-ONOP-030, Attachment 1, Step 13 (12):** **ADJUST** Service Water flow from desired charging pump to provide maximum flow: • **OPEN** 3(4)-10-290, Emergency Hose Connection to Charging Pump A Oil Cooler Return • **OPEN** 3(4)-10-288, Emergency Hose Connection to Charging Pump B Oil Cooler Return • **OPEN** 3(4)-10-298, Emergency Hose Connection to Charging Pump C Oil Cooler Return
Standard:	Identify/open the appropriate service water valve.
Evaluator Cue:	When the examinee identifies the correct valve and properly simulates opening it, inform the examinee that the valve is open.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 15 **Critical: No**	**3(4)-ONOP-030, Attachment 1, Step 14 (13):** IF Service Water flow is **NOT** obtained, THEN **PLACE** Service Water System in service per 0-NOP-012, Service Water System, using any available pump including the diesel driven SWP D.
Standard:	Verify that service water flow is obtained.
Evaluator Cue:	When the examinee checks the open end of the discharge hose at the floor drain, inform the examinee that service water flow is observed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When service water flow has been verified, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ RO ☐ SRO ☐ STA ☐ Non-Lic ☐ SRO CERT **Date:** ______________________

☐ LOIT RO ☐ LOIT SRO

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ____

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Level in the CCW Surge Tank can NOT be maintained.
- The crew has entered 3(4)-ONOP-030, Component Cooling Water Malfunction.

INITIATING CUE:

- You have been directed by the Unit 3(4) RCO to perform Attachment 1, Control of Emergency Cooling Water to Charging Pumps, of 3(4)-ONOP-030 for the A/B/C Charging Pump.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

ATTACHMENT 1
Control of Emergency Cooling Water to Charging Pumps
(Page 1 of 5)

NOTE

- Emergency cooling water supply hose has a quick disconnect fitting on one end and a cam lock fitting on the other end.
- Loss of off-site power in coincidence with a loss of CCW will require the diesel driven Service Water Pump to be in service in order to provide emergency cooling water to the Charging Pumps.

1. **CONNECT** cam lock fitting end of emergency cooling water supply hose to 3-70-179A, SERVICE WATER CONNECTION INSIDE UNIT 3 CHARGING PUMP ROOM.

2. **CONSULT** with Unit 3 Reactor Operator to determine desired charging pump.

3. **ENSURE** desired Charging Pump is STOPPED OR running at maximum speed.

4. **CONNECT** quick disconnect fitting end of emergency cooling water supply hose to emergency hose connection on desired Charging Pump:

   * 3-10-291, EMERGENCY HOSE CONNECTION TO CHARGING PUMP A OIL COOLER SUPPLY

   * 3-10-289, EMERGENCY HOSE CONNECTION TO CHARGING PUMP B OIL COOLER SUPPLY

   * 3-10-299, EMERGENCY HOSE CONNECTION TO CHARGING PUMP C OIL COOLER SUPPLY

ATTACHMENT 1
Control of Emergency Cooling Water to Charging Pumps
(Page 2 of 5)

NOTE

Emergency cooling water outlet hose has a quick disconnect fitting on one end and **NO** fitting on the other end.

5. **CONNECT** quick disconnect fitting end of emergency cooling water outlet hose to emergency hose connection on desired Charging Pump.

   * 3-10-290, EMERGENCY HOSE CONNECTION TO CHARGING PUMP A OIL COOLER RETURN

   * 3-10-288, EMERGENCY HOSE CONNECTION TO CHARGING PUMP B OIL COOLER RETURN

   * 3-10-298, EMERGENCY HOSE CONNECTION TO CHARGING PUMP C OIL COOLER RETURN

6. **REMOVE** cover from floor drain to be used in Charging Pump Room.

7. **ROUTE** open end of emergency cooling water outlet hose to floor drain being used in Charging Pump Room.

8. **ISOLATE** CCW to hydraulic oil cooler on desired Charging Pump:

   * **CLOSE** 3-825A, CCW TO A CHARGING PUMP OIL COOLER INLET

   * **CLOSE** 3-825C, CCW TO B CHARGING PUMP OIL COOLER INLET

   * **CLOSE** 3-825E, CCW TO C CHARGING PUMP OIL COOLER INLET

9. **ISOLATE** CCW from hydraulic oil cooler on desired Charging Pump:

   * **CLOSE** 3-825B, CCW FROM A CHARGING PUMP OIL COOLER INLET

   * **CLOSE** 3-825D, CCW FROM B CHARGING PUMP OIL COOLER INLET

   * **CLOSE** 3-825F, CCW FROM C CHARGING PUMP OIL COOLER INLET

10. **OPEN** 3-70-179, SERVICE WATER CONNECTION INSIDE UNIT 3 CHARGING PUMP ROOM ROOT VALVE.

11. **OPEN** 3-70-179A, SERVICE WATER CONNECTION INSIDE UNIT 3 CHARGING PUMP ROOM.

ATTACHMENT 1
Control of Emergency Cooling Water to Charging Pumps
(Page 3 of 5)

12. ESTABLISH Service Water to desired Charging Pump:

* **OPEN** 3-10-291, EMERGENCY HOSE CONNECTION TO CHARGING PUMP A OIL COOLER SUPPLY
* **OPEN** 3-10-289, EMERGENCY HOSE CONNECTION TO CHARGING PUMP B OIL COOLER SUPPLY
* **OPEN** 3-10-299, EMERGENCY HOSE CONNECTION TO CHARGING PUMP C OIL COOLER SUPPLY

13. ADJUST Service Water flow from desired charging pump to provide maximum flow.

* **OPEN** 3-10-290, EMERGENCY HOSE CONNECTION TO CHARGING PUMP A OIL COOLER RETURN
* **OPEN** 3-10-288, EMERGENCY HOSE CONNECTION TO CHARGING PUMP B OIL COOLER RETURN
* **OPEN** 3-10-298, EMERGENCY HOSE CONNECTION TO CHARGING PUMP C OIL COOLER RETURN

14. IF Service Water flow is **NOT** obtained, THEN **PLACE** Service Water System in service per 0-NOP-012, Service Water System, using any available pump including the diesel driven SWP D.

15. NOTIFY Unit 3 Reactor Operator that emergency cooling water has been established to the desired Charging Pump.

NOTE

Maximum Charging Pump oil temperature is 220°F to prevent oil break down. The installed temperature indicators only indicate up to 200°F. Some indicators are located on the cooler inlet and others on the cooler outlet. Maximum expected ΔT across the cooler is 20°F. At 195°F on the cooler outlet (oil to the hydraulic coupling), this would equate to 215°F on the cooler inlet (oil from the hydraulic coupling).

16. MONITOR oil temperatures on operating Charging Pump.

ATTACHMENT 1
Control of Emergency Cooling Water to Charging Pumps
(Page 4 of 5)

17. IF hydraulic coupling oil temperature on running Charging Pump exceeds 195°F, THEN:

    A. **NOTIFY** Unit 3 Reactor Operator that operating Charging Pump should be STOPPED.

    B. **CONSULT** with Unit 3 Reactor Operator to determine if emergency cooling water should be realigned to a different Charging Pump.

    C. IF Unit 3 Reactor Operator determines that emergency cooling water must be realigned to a different Charging Pump, THEN **GO TO** Attachment 1, Step 20.

18. IF Unit 3 Reactor Operator determines that emergency cooling water to Charging Pumps is **NO** longer required, THEN **GO TO** Attachment 1, Step 20.

19. **RETURN TO** Attachment 1, Step 16.

20. **ENSURE** Charging Pump being supplied with emergency cooling water is STOPPED.

21. **ISOLATE** emergency cooling water flow from previously running Charging Pump:

    * **CLOSE** 3-10-290, EMERGENCY HOSE CONNECTION TO CHARGING PUMP A OIL COOLER RETURN

    * **CLOSE** 3-10-288, EMERGENCY HOSE CONNECTION TO CHARGING PUMP B OIL COOLER RETURN

    * **CLOSE** 3-10-298, EMERGENCY HOSE CONNECTION TO CHARGING PUMP C OIL COOLER RETURN

22. **ISOLATE** emergency cooling water flow to previously running Charging Pump:

    * **CLOSE** 3-10-291, EMERGENCY HOSE CONNECTION TO CHARGING PUMP A OIL COOLER SUPPLY

    * **CLOSE** 3-10-289, EMERGENCY HOSE CONNECTION TO CHARGING PUMP B OIL COOLER SUPPLY

    * **CLOSE** 3-10-299, EMERGENCY HOSE CONNECTION TO CHARGING PUMP C OIL COOLER SUPPLY

ATTACHMENT 1
Control of Emergency Cooling Water to Charging Pumps
(Page 5 of 5)

23. **ESTABLISH** CCW to hydraulic oil cooler on previously running Charging Pump:

    * **OPEN** 3-825A, CCW TO A CHARGING PUMP OIL COOLER INLET
    * **OPEN** 3-825C, CCW TO B CHARGING PUMP OIL COOLER INLET
    * **OPEN** 3-825E, CCW TO C CHARGING PUMP OIL COOLER INLET

24. **ESTABLISH** CCW from hydraulic oil cooler on previously running Charging Pump.

    * **OPEN** 3-825B, CCW FROM A CHARGING PUMP OIL COOLER INLET
    * **OPEN** 3-825D, CCW FROM B CHARGING PUMP OIL COOLER INLET
    * **OPEN** 3-825F, CCW FROM C CHARGING PUMP OIL COOLER INLET

25. **DISCONNECT** emergency cooling water outlet hose from previously running Charging Pump.

26. **CLOSE** 3-70-179, SERVICE WATER CONNECTION INSIDE UNIT 3 CHARGING PUMP ROOM ROOT VALVE.

27. **CLOSE** 3-70-179A, SERVICE WATER CONNECTION INSIDE UNIT 3 CHARGING PUMP ROOM.

28. **DISCONNECT** emergency cooling water supply hose from previously running Charging Pump.

29. IF emergency cooling water must be aligned to a different Charging Pump, THEN **RETURN TO** Attachment 1, Step 2.

30. **DISCONNECT** emergency cooling water supply hose from 3-70-179A, SERVICE WATER CONNECTION INSIDE UNIT 3 CHARGING PUMP ROOM.

31. **RETURN** emergency cooling water supply and outlet hoses to their designated storage locations.

32. **REPLACE** cover on floor drain used for emergency cooling water.

33. **NOTIFY** Unit 3 Reactor Operator that emergency cooling water alignment has been terminated.

ATTACHMENT 1
(Page 1 of 5)

CONTROL OF EMERGENCY COOLING WATER TO CHARGING PUMP

NOTES

- *Emergency cooling water SUPPLY hose has a quick disconnect fitting on one end and a cam lock fitting on the other end.*
- *Loss of offsite power in coincidence with a loss of CCW will require the diesel driven service wtr pump to be in service to provide emergency cooling water to the charging pumps.*

1. Connect cam lock fitting end of emergency cooling water supply hose to Service Water Connection Outside Unit 4 charging Pump Room, 4-70-118B.
2. Consult with Unit 4 Reactor Operator to determine desired charging pump.
3. Verify desired charging pump is stopped, OR running at maximum speed.
4. Connect quick disconnect fitting end of emergency cooling water supply hose to emergency hose connection on desired charging pump:

   a. Emergency Hose Connection To Charging Pump A Oil Cooler, 4-10-291.

   OR

   b. Emergency Hose Connection To Charging Pump B Oil Cooler, 4-10-289.

   OR

   c. Emergency Hose Connection To Charging Pump C Oil Cooler, 4-10-299.

NOTE

Emergency cooling water OUTLET hose has a quick disconnect fitting on one end and no fitting on the other end.

5. Connect quick disconnect fitting end of emergency cooling water outlet hose to emergency hose connection on desired charging pump:

   a. Emergency Hose Connection To Charging Pump A Oil Cooler, 4-10-290.

   OR

   b. Emergency Hose Connection To Charging Pump B Oil Cooler, 4-10-288.

   OR

   c. Emergency Hose Connection To Charging Pump C Oil Cooler, 4-10-298.

ATTACHMENT 1
(Page 2 of 5)

CONTROL OF EMERGENCY COOLING WATER TO CHARGING PUMP

6. Remove cover from floor drain to be used in Charging Pump Room.

7. Route open end of emergency cooling water outlet hose to floor drain being used in Charging Pump Room.

8. Isolate CCW to hydraulic oil cooler on desired charging pump:

   a. Close CCW To A Charging Pump Oil Cooler Inlet, 4-825A.

   OR

   b. Close CCW To B Charging Pump Oil Cooler Inlet, 4-825C.

   OR

   c. Close CCW To C Charging Pump Oil Cooler Inlet, 4-825E.

9. Isolate CCW from hydraulic oil cooler on desired charging pump:

   a. Close CCW From A Charging Pump Oil Cooler Inlet, 4-825B.

   OR

   b. Close CCW From B Charging Pump Oil Cooler Inlet, 4-825D.

   OR

   c. Close CCW From C Charging Pump Oil Cooler Inlet, 4-825F.

10. a. Open Service Water Connection Outside Unit 4 Charging Pump Room, 4-70-118B.

    b. Open isol valve service water header outside Charging Pump Room 4-70-118.

11. Establish service water to desired charging pump:

    a. Open Emergency Hose Connection To Charging Pump A Oil Cooler, 4-10-291.

    OR

    b. Open Emergency Hose Connection To Charging Pump B Oil Cooler, 4-10-289.

    OR

    c. Open Emergency Hose Connection To Charging Pump C Oil Cooler, 4-10-299.

ATTACHMENT 1
(Page 3 of 5)

CONTROL OF EMERGENCY COOLING WATER TO CHARGING PUMP

12. Adjust service water flow from desired charging pump to provide maximum flow.

    a. Open Emergency Hose Connection To Charging Pump A Oil Cooler, 4-10-290.

    OR

    b. Open Emergency Hose Connection To Charging Pump B Oil Cooler, 4-10-288.

    OR

    c. Open Emergency Hose Connection To Charging Pump C Oil Cooler, 4-10-298.

13. **IF** Service Water flow is not obtained, **THEN** have the Service Water System placed in service using 0-NOP-012, Service Water System, using any available pump including the diesel driven SWP D.

14. Notify Unit 4 Reactor Operator that emergency cooling water has been established to desired charging pump.

NOTE

Maximum charging pump oil temperature is 220°F to prevent oil break down. The installed temperature indicators only indicate up to 200°F. Some indicators are located on the cooler inlet and others on the cooler outlet. Maximum expected ΔT across the cooler is 20°F. At 195°F on the cooler outlet (oil to the hydraulic coupling), this would equate to 215°F on the cooler inlet (oil from the hydraulic coupling).

15. Monitor oil temperatures on running charging pump.

16. **IF** hydraulic coupling oil outlet temperature on running charging pump exceeds 195°F, **THEN** perform the following:

    a. Notify Unit 4 Reactor Operator that operating charging pump should be stopped.

    b. Consult with Unit 4 Reactor Operator to determine if emergency cooling water should be realigned to a different charging pump.

    c. **IF** Unit 4 Reactor Operator determines that emergency cooling water must be realigned to a different charging pump, **THEN** go to Step 19 of this attachment.

17. **IF** Unit 4 Reactor Operator determines that emergency cooling water to charging pumps is no longer required, **THEN** go to Step 19 of this attachment.

18. Return to Step 15 of this attachment.

19. Verify charging pump being supplied with emergency cooling water is stopped.

ATTACHMENT 1
(Page 4 of 5)

CONTROL OF EMERGENCY COOLING WATER TO CHARGING PUMP

20. Isolate emergency cooling water flow from previously running charging pump:

    a. Close Emergency Hose Connection To Charging Pump A Oil Cooler, 4-10-290.

    OR

    b. Close Emergency Hose Connection To Charging Pump B Oil Cooler, 4-10-288.

    OR

    c. Close Emergency Hose Connection To Charging Pump C Oil Cooler, 4-10-298.

21. Isolate emergency cooling water flow to previously running charging pump:

    a. Close Emergency Hose Connection To Charging Pump A Oil Cooler, 4-10-291.

    OR

    b. Close Emergency Hose Connection To Charging Pump B Oil Cooler, 4-10-289.

    OR

    c. Close Emergency Hose Connection To Charging Pump C Oil Cooler, 4-10-299.

22. Reestablish CCW to hydraulic oil cooler on previously running charging pump:

    a. Open CCW To A Charging Pump Oil Cooler Inlet, 4-825A.

    OR

    b. Open CCW To B Charging Pump Oil Cooler Inlet, 4-825C.

    OR

    c. Open CCW To C Charging Pump Oil Cooler Inlet, 4-825E.

ATTACHMENT 1
(Page 5 of 5)

CONTROL OF EMERGENCY COOLING WATER TO CHARGING PUMP

23. Reestablish CCW from hydraulic oil cooler on desired charging pump:

    a. Open CCW From A Charging Pump Oil Cooler Inlet, 4-825B.

    OR

    b. Open CCW From B Charging Pump Oil Cooler Inlet, 4-825D.

    OR

    c. Open CCW From C Charging Pump Oil Cooler Inlet, 4-825F.

24. Disconnect emergency cooling water outlet hose from previously running charging pump

25. Close Service Water Connection Outside Unit 4 charging Pump Room, 4-70-118B.

26. Close Isol valve service water header outside Charging Pump Room, 4-70-118

27. Disconnect emergency cooling water supply hose from previously running charging pump.

28. **IF** emergency cooling water must be realigned to a different charging pump, **THEN** return to Step 2.

29. Disconnect emergency cooling water supply hose from Service Water Connection Outside Unit 4 Charging Pump Room, 4-10-118B.

30. Return emergency cooling water supply and outlet hoses to their designated storage locations.

31. Replace cover on floor drain used for emergency cooling water.

32. Notify Unit 4 Reactor Operator that emergency cooling water alignment has been terminated.

FINAL PAGE

JPM TITLE: Respond to Control Bank D Demanded Past 230 Steps

JPM NUMBER: 01028916302 **REV.** 1-0

TASK NUMBER(S) / TASK TITLE(S): 01028916300 / Respond to Control Bank D Demanded Past 230 Steps

K/A NUMBERS: 001 A4.14 **K/A VALUE:** RO 3.0 / SRO 3.4

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [X] Other: [ ]
Lab: [ ]

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: Yes
Alternate Path [INPO]: Yes

Developed by:	Brian Clark Instructor/Developer	6/21/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/21/16 Date
Validated by:	R Schoenlieb SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R Schoenlieb Training Program Owner	06/22/16 Date

JPM TITLE: Place Excess Letdown in Service

JPM NUMBER: 01047016102 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 01047016100 / Initiate Excess Letdown

K/A NUMBERS: 004 A4.06 **K/A VALUE:** RO 3.6 / SRO 3.1

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ RO ☒ SRO ☐ STA ☐ Non-Lic ☐ SRO CERT ☐ OTHER: ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [X] Other: [ ]
Lab: [ ]

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: Yes

Alternate Path [INPO]: Yes

Developed by:	Brian Clark Instructor/Developer	6/22/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/22/16 Date
Validated by:	R Dehoarle SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R Dehoarle Training Program Owner	06/22/16 Date

JPM TITLE: Establish Auxiliary Pressurizer Spray

JPM NUMBER: 01041052100 **REV.** 0-0

TASK NUMBER(S) / TASK TITLE(S): 01041052100 / Initiate Pressurizer Auxiliary Spray

K/A NUMBERS: EPE 038 EA1.04 **K/A VALUE:** RO 4.3 / SRO 4.1

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: X

EVALUATION LOCATION:

In-Plant:		Control Room:	
Simulator:	X	Other:	
Lab:			

Time for Completion: 20 Minutes Time Critical: No

Alternate Path [NRC]: Yes

Alternate Path [INPO]: Yes

Developed by:	Brian Clark Instructor/Developer	6/20/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/21/16 Date
Validated by:	R Dehoeuk SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R Dehoeuk Training Program Owner	06/22/16 Date

JPM TITLE: Respond to Loss of RHR

JPM NUMBER: 01050004301 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 01050004300 / Respond to Loss of RHR

K/A NUMBERS: APE 025 AA1.03 **K/A VALUE:** RO 3.4 / SRO 3.3

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]

Simulator: [X] Other: [ ]

Lab: [ ]

Time for Completion: 20 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Brian Clark Instructor/Developer	6/20/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/21/16 Date
Validated by:	R Schaeubh SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R Schaeubh Training Program Owner	06/22/16 Date

JPM TITLE: Manually Initiate Containment Spray

JPM NUMBER: 01068007502 **REV.** 1-0

TASK NUMBER(S) / TASK TITLE(S): 01068007500 / Manually Initiate Containment Spray

K/A NUMBERS: 026 A3.01 **K/A VALUE:** RO 4.3 / SRO 4.5

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ RO ☒ SRO ☐ STA ☐ Non-Lic ☐ SRO CERT ☐ OTHER: ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]

Simulator: [X] Other: [ ]

Lab: [ ]

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Brian Clark Instructor/Developer	6/22/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/22/16 Date
Validated by:	R Schoenl SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R Schoenl Training Program Owner	06/22/16 Date

JPM TITLE:	Restore Power to the 3A 4KV Bus
JPM NUMBER:	03005032300 **REV.** 0-0
TASK NUMBER(S) / TASK TITLE(S):	03005032300 / Cross-Tie 3D and 4D 4KV Buses
K/A NUMBERS:	EPE 055 EA1.07 **K/A VALUE:** RO 4.3 / SRO 4.5

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: ☒

EVALUATION LOCATION:

In-Plant:	☐	Control Room:	☐
Simulator:	X	Other:	☐
Lab:	☐		

Time for Completion: 15 Minutes Time Critical: Yes

Alternate Path [NRC]: Yes

Alternate Path [INPO]: Yes

Developed by:	Brian Clark Instructor/Developer	6/20/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/21/16 Date
Validated by:	R. Vehoonlink SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R. Vehoonlink Training Program Owner	06/22/16 Date

JPM TITLE: Place N-3-42 Power Range Drawer in Service

JPM NUMBER: 01059016200 **REV.** 1-0

TASK NUMBER(S) / TASK TITLE(S): 01059016200 / Place N-42 Power Range Drawer in Service

K/A NUMBERS: 015 A4.02 **K/A VALUE:** RO 3.9 / SRO 3.9

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: X

EVALUATION LOCATION: In-Plant: ☐ Control Room: ☐

Simulator: X Other: ☐

Lab: ☐

Time for Completion: 10 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Brian Clark Instructor/Developer	6/20/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/21/16 Date
Validated by:	R. Schoenbeck SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R. Schoenbeck Training Program Owner	06/22/16 Date

JPM TITLE:	Respond To Control Room Evacuation Condition – Unit 3 RO		
JPM NUMBER:	01200011301	**REV.**	2-0
TASK NUMBER(S) / TASK TITLE(S):	01200011300 / Respond To Control Room Evacuation Condition – Unit 3 RO		
K/A NUMBERS:	APE 068 AA1.23	**K/A VALUE:**	RO 4.3 / SRO 4.4

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION:

In-Plant:	[ ]	Control Room:	[ ]
Simulator:	[X]	Other:	[ ]
Lab:	[ ]		

Time for Completion: 10 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Brian Clunn (signature) Instructor/Developer	6/20/16 Date
Reviewed by:	(signature) Instructor (Instructional Review)	6/21/16 Date
Validated by:	R Schoenlink (signature) SME (Technical Review)	06/22/16 Date
Approved by:	(signature) Training Supervision	6/22/16 Date
Approved by:	R Schoenlink (signature) Training Program Owner	06/22/16 Date

JPM TITLE: Locally Trip the Reactor and Turbine

JPM NUMBER: 14028009501 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 14028009500 / Respond to an ATWS

K/A NUMBERS: EPE 029 EA1.12 **K/A VALUE:** RO 4.1 / SRO 4.0

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: X Perform: ☐

EVALUATION LOCATION: In-Plant: X Control Room: ☐

Simulator: ☐ Other: ☐

Lab: ☐

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Brian Clark Instructor/Developer	6/20/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/22/16 Date
Validated by:	R. Schoenbeck SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R. Schoenbeck Training Program Owner	06/22/16 Date

JPM TITLE: Control S/G Level Locally with AFW Control Valve

JPM NUMBER: 04075002300 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 04075002300 / Control Steam Generator Level Locally with Auxiliary Feedwater Control Valve

K/A NUMBERS: APE 054 AA1.01 **K/A VALUE:** RO 4.5 / SRO 4.4

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ RO ☒ SRO ☐ STA ☐ Non-Lic ☐ SRO CERT ☐ OTHER: ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [X] Perform: [ ]

EVALUATION LOCATION: In-Plant: [X] Control Room: [ ]
Simulator: [ ] Other: [ ]
Lab: [ ]

Time for Completion: 20 Minutes Time Critical: No

Alternate Path [NRC]: Yes
Alternate Path [INPO]: Yes

Developed by:	Brian Clark Instructor/Developer	6/20/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/22/16 Date
Validated by:	R Schoenherr SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R Schoenherr Training Program Owner	06/22/16 Date

JPM TITLE: Align Emergency Service Water to the Charging Pumps

JPM NUMBER: 24030009300 **REV.** 2-0

TASK NUMBER(S) / TASK TITLE(S): 24030009300 / Align Emergency Service Water to the Charging Pumps

K/A NUMBERS: APE 026 AA1.03 **K/A VALUE:** RO 3.6 / SRO 3.6

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: X Perform: ☐

EVALUATION LOCATION: In-Plant: X Control Room: ☐

Simulator: ☐ Other: ☐

Lab: ☐

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Brian Clark Instructor/Developer	6/20/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/22/16 Date
Validated by:	R Schaalb SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R Schaalb Training Program Owner	06/22/16 Date

L-16-1 NRC Exam

Admin - JPM RO A1a

JPM TITLE: Calculate a Manual Makeup to the VCT

JPM NUMBER: 01046046101 **REV.** 1-1

TASK NUMBER(S) / TASK TITLE(S): 01046046100/ Calculate a Manual Makeup to the VCT

K/A NUMBERS: 2.1.25 **K/A VALUE:** RO 3.9 / SRO 4.2

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION:

In-Plant:	[ ]	Control Room:	[ ]
Simulator:	[ ]	Other:	[X]
Lab:	[ ]		

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

<u>Protected Content:</u> (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
0	New JPM		N/A	N/A	N/A
				N/A	N/A
1-0	Added task for determining controller settings; formatting; text/grammar changes		N/A	Schilk	
				Wilson	

SIMULATOR SET-UP:

- N/A

Required Materials:

- 4-OP-046, CVCS – Boron Concentration Control
- Plant Curve Book, Section III
- Calculator

General References:

- 4-OP-046, CVCS – Boron Concentration Control
- Plant Curve Book, Section III

Task Standards:

- Calculate the boric acid and primary water flow rates, volumes, and controller settings as required to makeup to the VCT, using Method 2 of the Plant Curve Book (Section III)

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 4 is at 100% steady-state power.
- VCT level is 20%.
- Boron concentrations in the RCS and BAST are 874 ppm and 5687 ppm, respectively.
- A manual makeup to the VCT is to be performed, with a desired boric acid flow rate of 11.0 gpm.
- All relevant prerequisites, precautions/limitations, and associated attachments in 0-OP-046, CVCS – Boron Concentration Control, have been addressed.

INITIATING CUES:

- VCT level is to be raised to 37%, while maintaining a constant VCT/RCS boron concentration.
- You are directed to perform Section 5.4 (Manual Makeup) of 0-OP-046, using Method 2 (Calculation) from Section III of the Plant Curve Book, to calculate the following parameters:
  - Primary water flow rate: ___________________ (to the nearest tenth of a gpm)
  - Primary water volume: ___________________ (to the nearest gallon)
  - Boric acid volume: ___________________ (to the nearest gallon)
- Based on the available information, determine the potentiometer settings for the following controllers:
  - Boric Acid Flow Controller (FC-4-113A): ___________________
  - Primary Water Flow Controller (FC-4-114A): ___________________

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 0-OP-046, CVCS – Boron Concentration Control.
Evaluator Cue:	Provide examinee with a copy of 0-OP-046, CVCS – Boron Concentration Control.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: No**	**0-OP-046, Step 5.4.1.1:** Applicable Prerequisites in Section 3.0 are satisfied.
Standard:	Recognize, from the Initial Conditions, that all relevant prerequisites have been addressed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 3 **Critical: No**	**0-OP-046, prior to Step 5.4.2.1:** **CAUTION** ***Instrument uncertainties for the Boric Acid and Primary Water flow transmitters can result in the actual amount of Boric Acid or Primary Water added to be either more or less than the amount calculated. Thus, care is needed to ensure that excessive reduction in RCS boron concentration does NOT occur due to the uncertainties.*** **NOTE** ***VCT level is 14.15 gallons per percent level indication.***
Standard:	Read CAUTION/NOTE and recognize that it is safe to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: No**	**0-OP-046, Step 5.4.2.1:** Determine the approximate boric acid and primary water flows and volumes needed to obtain the desired blend concentration from the boron change tables in Section III of the Plant Curve Book. The primary water flow rate should be determined in order to ensure all primary water is injected prior to completion of the manual make-up.
Standard:	Obtain Section III of the Plant Curve Book and locate Figure 4 (Blended Flow), Method 2 (Calculation).
Evaluator Cue:	Provide examinee with a copy of Section III of the Plant Curve Book.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: Yes**	**Determine the appropriate primary water flow needed to obtain the desired blend, using the blended flow calculation in Section III of the Plant Curve Book.**
Standard:	Determine the primary water flow rate and record the value on the Turnover Sheet. • Primary water flow rate: 60.6 gpm (60.5 to 60.7 gpm)
Evaluator Note:	• From Section III of Plant Curve Book: o $Boron_{ppm} = (Acid_{gpm})(BAST_{ppm})/(Acid_{gpm} + Water_{gpm})$, where $Boron_{ppm}$ is the desired blended boron concentration • Therefore, $Water_{gpm} = [(Acid_{gpm})(BAST_{ppm})/(Boron_{ppm})] - (Acid_{gpm})$: o $Water_{gpm} = [(11.0)(5687)/(874)] - (11.0) =$ 60.6 gpm
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: Yes**	**Determine the appropriate boric acid and primary water volumes and to raise VCT level from 20% to 37%.**
Standard:	Determine the required boric acid and primary water volumes and record the values on Turnover Sheet. • Primary water volume: 204 gallons (±6%; 192 to 216 gallons) • Boric acid volume: 37 gallons (±6%; 35 to 39 gallons)
Evaluator Note:	• From NOTE prior to Step 5.4.2.1 of 0-OP-046 (i.e., 14.15 gallons/%): o (37% - 20%)(14.15 gallons/%) = 240.55 gallons • Therefore, with 11.0 gpm of boric acid and 60.6 gpm of primary water: o (240.55 gallons)[(11.0)/(11.0 + 60.6)] = 37.0 gallons of boric acid o (240.55 gallons)[(60.6)/(11.0 + 60.6)] = 203.6 gallons of primary water • Various methods may be used to determine the fluid volumes • Answer bands are based on potential rounding error (e.g., 14.15 gallons/% rounded up to 15 gallons/% would yield 216 gallons of primary water)
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: Yes**	**Based on the available information, determine the potentiometer settings for the following controllers:** • **Boric Acid Flow Controller (FC-3-113A)** • **Primary Water Flow Controller (FC-3-114A)**
Standard:	Determine the associated potentiometer settings and record the values on Turnover Sheet. • Boric Acid Flow Controller (FC-4-113A): 2.2 (2.1 to 2.3) • Primary Water Flow Controller (FC-4-114A): 4.0 (3.9 to 4.1)
Evaluator Note:	• From Step 4.23 of 0-OP-046 (ratio of 5 gpm to 1; i.e., 50 gpm maximum), a boric acid flow rate of 11.0 gpm is equivalent to a controller setting of 2.2 on the ten-turn potentiometer. • From Step 4.24 of 0-OP-046 (ratio of 15 gpm to 1; i.e., 150 gpm maximum), a primary water flow rate of 60.6 gpm is equivalent to a controller setting of 4.0 on the ten-turn potentiometer.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When the examinee completes Step 7, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ____________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 4 is at 100% steady-state power.
- VCT level is 20%.
- Boron concentrations in the RCS and BAST are 874 ppm and 5687 ppm, respectively.
- A manual makeup to the VCT is to be performed, with a desired boric acid flow rate of 11.0 gpm.
- All relevant prerequisites, precautions/limitations, and associated attachments in 0-OP-046, CVCS – Boron Concentration Control, have been addressed.

INITIATING CUES:

- VCT level is to be raised to 37%, while maintaining a constant VCT/RCS boron concentration.
- You are directed to perform Section 5.4 (Manual Makeup) of 0-OP-046, using Method 2 (Calculation) from Section III of the Plant Curve Book, to calculate the following parameters:
  - Primary water flow rate: ___________________ (to the nearest tenth of a gpm)
  - Primary water volume: ___________________ (to the nearest gallon)
  - Boric acid volume: ___________________ (to the nearest gallon)
- Based on the available information, determine the potentiometer settings for the following controllers:
  - Boric Acid Flow Controller (FC-4-113A): ___________________ turns
  - Primary Water Flow Controller (FC-4-114A): ___________________ turns

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

DRAFT L-16-1 NRC EXAM SECURE INFORMATION

L-16-1 NRC Exam

Admin - JPM RO A1b

JPM TITLE: Determine Heatup of the RCS

JPM NUMBER: 01041046101 **REV.** 0-1

TASK NUMBER(S) / TASK TITLE(S): 01041046100/ Determine Heatup of the RCS

K/A NUMBERS: 2.1.20 **K/A VALUE:** RO 4.6 / SRO 4.6

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [ ] Other: [X]
Lab: [ ]

Time for Completion: 30 Minutes Time Critical: No

Alternate Path [NRC]: No
Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
0	New JPM		N/A	N/A	N/A
				N/A	N/A
0-1	Formatting; text/grammar changes		N/A	Schilk	
				Wilson	

SIMULATOR SET-UP:

- N/A

Required Materials:

- Handout 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification
- Technical Specifications
- Calculator

General References:

- 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification
- Technical Specifications
- Plant Curve Book, Section V, Figure 3D

Task Standards:

- Identify discrepancy in heatup determination and list any subsequent procedural actions and/or Technical Specification actions that apply

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Operators at Unit 3 have just completed an RCS heatup.
- The RCS is stable at 380°F and 499 psig.
- 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification, is complete through step 4.2.14.
- All relevant data was recorded on Attachment 2, Heatup Data Sheet.

INITIATING CUES:

- You are directed to review the heatup data, complete the remaining procedural steps, and record any discrepancies and required subsequent actions in Section 5.2.
- [SRO only] Record any relevant Technical Specification actions in Section 5.2.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification.
Evaluator Cue:	Provide examinee with a copy of handout 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 2 **Critical: No**	**Review heatup data (Attachment 2) and identify discrepancy.**
Standard:	Recognize that the ΔT value for RCS T_{hot} was miscalculated at 1030 (i.e., the actual value is 101°F, rather than 74°F).
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: Yes**	**Complete remaining procedural steps and record any discrepancies and required subsequent actions in Section 5.2.**
Standard:	Per Step 4.1.1, immediately notify the Unit Supervisor and document the condition in Section 5.2: • Mark "Acceptance Criteria of Section 6.1" as UNSAT. • Mark "Functional Criteria of Section 6.2" as UNSAT. • In the "Remarks" section, indicate that the heatup rate exceeded the Administrative (<90°F/hour) and Technical Specification (<100°F/hour) limits at 1030. [SRO only] Identify Technical Specification 3.4.9.1, Action a, with the following requirements: • Restore the temperature and/or pressure to within the limit within 30 minutes (effectively accomplished). • Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the RCS. • Determine that the RCS remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.
Evaluator Cue:	When requested, provide examinee with a copy of the Technical Specifications.
Evaluator Note:	Examinee may base required subsequent actions on Attachment 1: • When the Administrative limit is exceeded, immediately reduce the heatup rate (irrelevant at this time) and notify the Shift Manager or Unit Supervisor – ACTION 1. • When the Technical Specification limit is exceeded, immediately reduce the heatup rate (irrelevant at this time), notify the Shift Manager or Unit Supervisor, and take actions required by the Technical Specifications – ACTION 2.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When the examinee completes Step 3, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Operators at Unit 3 have just completed an RCS heatup.
- The RCS is stable at 380°F and 499 psig.
- 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification, is complete through step 4.2.14.
- All relevant data was recorded on Attachment 2, Heatup Data Sheet.

INITIATING CUES:

- You are directed to <u>review</u> the heatup data, <u>complete</u> the remaining procedural steps, and <u>record</u> any discrepancies and required subsequent actions in Section 5.2.
- [SRO only] <u>Record</u> any relevant Technical Specification actions in Section 5.2.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

DRAFT L-16-1 NRC EXAM SECURE INFORMATION

FPL	**TURKEY POINT UNIT 3** **OPERATIONS SURVEILLANCE PROCEDURE** SAFETY RELATED **CONTINUOUS USE**	Procedure No. **3-OSP-041.7** Revision No. **5**

Title:

REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN TEMPERATURE VERIFICATION

Responsible Department: OPERATIONS

Special Considerations:

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED today INITIAL S

Revision	Approved By	Approval Date
0	Michael Murphy	05/26/10
5	Mike Murphy	07/14/15

UNIT #	UNIT 3
DATE	
DOCT	PROCEDURE
DOCN	3-OSP-041.7
SYS	
STATUS	COMPLETED
REV	5
# OF PGS	

REVISION SUMMARY	
Rev. No.	**Description**
5	**PCR 1997526, 07/14/15, Brian Fitzgerald** Removed functional criteria for delta T reading between PZR and hottest RCS hot leg reading for RCS heatup in accordance with AR 1960410.
4	**PCR 2017385, 04/30/15, Michael Hargis** Revised RCS Pressurization Rate as recommended by AR 1709722.
3	**AR 1710614, 07/31/12, Joseph Madison** Revised procedure to provide administrative limits on RCS pressurization rate during heatup in accordance with EC 247008 and AR 1644725.
2	**AR 1652046, 06/08/11, Brian Fitzgerald** Revised Functional Criteria for subcooling in accordance with **AR 1627155**.
1	**AR 590917, 1/27/11, Brian Fitzgerald** Revised Acceptance Criteria to prompt user to consider using pressurizer vapor temp when calculating the difference in PZR and spray water temperatures. Addressed/corrected inconsistencies for heatup and cooldown monitoring.
0	**PCR 09-3331, 05/26/10, Dennis Bonsall** Upgraded procedure format to AD-AA-100-1003, FPL Fleet Procedure Writer's Guide standards. Added new section for Scope to cover frequency of performance, applicability, and mode restrictions. Revised step wording to apply human factors in accordance with Writers Guide. Split Heatup and Cooldown guidance into separate sections and made Heatup and Cooldown data sheets separate Attachments to improve Human Performance to reduce potential for error. This procedure supersedes 3-OSP-041.7, approval date 3/25/08

TABLE OF CONTENTS

1.0 PURPOSE AND SCOPE

1.1 Purpose

1. This procedure provides guidance to satisfy the requirements of Technical Specifications:
   - 4.4.9.1.1 - RCS Pressure/Temperature Limits
   - 4.4.9.2 - Pressurizer Temperature Limits
2. This procedure provides guidance on RCS Pressurization rates during RCS heatup to reduce the probability of Pressurizer Safety Valve leakage.

1.2 Scope

1.2.1 Frequency

At least once every 30 minutes during:

- RCS heatup
- RCS cooldown
- In-service leak and hydrostatic testing operations

1.2.2 Applicability

At all times

1.2.3 MODE Restrictions

None

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

1. This surveillance may be performed at the discretion of the Shift Manager/Unit Supervisor when RCS temperature is rising or lowering.

2. During collapsing of the Pressurizer bubble or any RCS heatup or cooldown operation, both RCS and Pressurizer cooldown and heatup limits shall be observed due to possible temperature stratifications, insurges, or outsurges of water in the Pressurizer.

3. If DCS points are used, periodic validation against other equivalent indications is required.

2.2 Limitations

1. The Reactor Coolant System (RCS), excluding the Pressurizer (PRZ), temperature and pressure shall be limited per the heatup and cooldown curves in the Plant Curve Book.

2. To reduce the probability of Pressurizer Safety Valve leakage, RCS pressurization rate during RCS heatup should be limited to 50 psi / hr for RCS pressure between 2000 psi and NOP.

3. The RCS pressurization rate during RCS heatup shall be limited to 400 psi/hr for RCS pressure between 1500 psig and 2235 psig.

3.0 PREREQUISITES

1. **ENSURE** Shift Manager or designee permission is obtained for data collection. S

End of Section 3.0

4.0 INSTRUCTIONS

4.1 General Requirements

1. IF during performance of this procedure any of the following occur:

   - Acceptance/Functional Criteria is UNSAT
   - A malfunction occurs
   - An abnormal condition is found

   THEN:

   - Immediately **NOTIFY** the Unit Supervisor. ______
   - **DOCUMENT** condition in Section 5.2. ______

2. WHEN during RCS heatup or cooldown, three consecutive readings within two degrees are obtained on each recorded RCS and Pressurizer temperature, THEN **DISCONTINUE** this surveillance. S

3. WHEN during in service hydrostatic and leak testing operations, RCS temperature and pressure are below and to the right of the heatup and cooldown limit curves, THEN **DISCONTINUE** this surveillance. N/A

4. **PERFORM** the following in Attachment 1, Reason for Performance of Data Sheet:

   A. **CHECK** the appropriate block to indicate which requirement(s) is (are) being met by completion. S

   B. **RECORD** start date and time. today 0900 S

5. **GO TO** the appropriate Section:

   - Section 4.2, Heatup S
   - Section 4.3, Cooldown N/A

End of Section 4.1

4.2 Heatup

NOTE

- DCS readings for the listed instruments should be used, if available, for recording temperatures and pressures. If one or more of these instruments are **NOT** available, an alternate instrument or its associated DCS point may be used. Proper notation should be made under Remarks if alternate instruments are used.
- If DCS points are used, periodic validation against other equivalent indications is required.

1. Using DCS or at VPA, **RECORD** the following indications every 15 minutes on Attachment 2, Heatup Data Sheet: S
   - TR-3-413 Pen 1 Loop A, if RCP A is in operation (DCS T413A_A)
   - TR-3-413 Pen 2 Loop B, if RCP B is in operation (DCS T423A_A)
   - TR-3-413 Pen 3 Loop C (DCS T433A_A)
   - TI-3-453 PRZ Liquid Temp (DCS T453_A)
   - TI-3-454 PRZ Vapor Temp (DCS T454_A)
   - PI-3-403 RCS Pressure (DCS P403_A)
   - PI-3-405 RCS Pressure (DCS P405_A)
   - TI-3-123 REGEN Hx Outlet Temp (DCS T123_A)
2. **RECORD** RCS temperature change every 15 minutes on Attachment 2, Heatup Data Sheet. S
3. For times of less than 1 hour, **RECORD** maximum RCS temperature change every 15, 30, and 45 minutes, while continuing to look back one hour. S
4. **DETERMINE** maximum RCS temperature change from the last 60 minutes, every 15 minutes. S

4.2 Heatup (continued)

5. **RECORD** PRZ liquid temperature change every 15 minutes on Attachment 2, Heatup Data Sheet. S

6. For times of less than 1 hour, **RECORD** PRZ liquid temperature change every 15, 30, and 45 minutes, while continuing to look back 1 hour. S

7. **DETERMINE** PRZ liquid temperature change from the last 60 minutes, every 15 minutes. S

8. **RECORD** PRZ vapor temperature change every 15 minutes on Attachment 2, Heatup Data Sheet. S

9. For times of less than 1 hour, **RECORD** PRZ vapor temperature change every 15, 30, and 45 minutes, while continuing to look back 1 hour. S

10. **DETERMINE** PRZ vapor temperature change from the last 60 minutes, every 15 minutes. S

11. IF using Loop-B, Loop-C, or Auxiliary sprays during RCS heatup or cooldown, THEN **RECORD** the ΔT between the lowest indicating spray water source in service and the highest indicating Pressurizer temperature on Attachment 2, Heatup Data Sheet. S

12. Record RCS Pressure every 15 minutes on Attachment 2, Heatup Data Sheet. S

13. IF RCS Pressure is greater than 1500 psig, **DETERMINE** RCS Pressure change from the last 60 minutes, every 15 minutes. N/A

14. WHEN data recording is **NO** longer required, THEN **RECORD** completion date and time in Attachment 1, Reason for Performance of Data Sheet. S

End of Section 4.2

4.3 Cooldown

NOTE

- DCS readings for the listed instruments should be used, if available, for recording temperatures and pressures. If one or more of these instruments are **NOT** available, an alternate instrument or its associated DCS point may be used. Proper notation should be made under Remarks if alternate instruments are used.
- If DCS points are used, periodic validation against other equivalent indications is required.

1. Using DCS or at VPA, **RECORD** the following indications every 15 minutes on Attachment 3, Cooldown Data Sheet: ______
   - TR-3-410 Pen 1 Loop A, if RCP A is in operation (DCS TE410A_A)
   - TR-3-410 Pen 2 Loop B, if RCP B is in operation (DCS TE420A_A)
   - TR-3-410 Pen 3 Loop C (DCS TE430A_A)
   - TI-3-453 PRZ Liquid Temp (DCS T453_A)
   - TI-3-454 PRZ Vapor Temp (DCS T454_A)
   - PI-3-403 RCS Pressure (DCS P403_A)
   - PI-3-405 RCS Pressure (DCS P405_A)
   - TI-3-123 REGEN Hx Outlet Temp (DCS T123_A)
2. **RECORD** RCS temperature change every 15 minutes on Attachment 3, Cooldown Data Sheet. ______
3. For times of less than 1 hour, **RECORD** the maximum RCS temperature change every 15, 30 and 45 minutes, while continuing to look back one hour. ______
4. **DETERMINE** maximum RCS temperature change from the last 60 minutes, every 15 minutes. ______

INITIAL

4.3 Cooldown (continued)

5. **RECORD** PRZ liquid temperature change every 15 minutes on Attachment 3, Cooldown Data Sheet. _______

6. For times of less than one hour, **RECORD** PRZ liquid temperature change every 15, 30, and 45 minutes, while continuing to look back 1 hour. _______

7. **DETERMINE** PRZ liquid temperature change from the last 60 minutes, every 15 minutes. _______

8. **RECORD** PRZ vapor temperature change every 15 minutes on Attachment 3, Cooldown Data Sheet. _______

9. For times of less than one hour, **RECORD** PRZ vapor temperature change every 15, 30, and 45 minutes, while continuing to look back one hour. _______

10. **DETERMINE** PRZ vapor temperature change from the last 60 minutes, every 15 minutes. _______

11. IF the PRZ is **NOT** solid during RCS cooldown, THEN **RECORD** the current ΔT between the highest reading hot leg temperature and Pressurizer liquid temperature, every 15 minutes on Attachment 3, Cooldown Data Sheet. _______

12. IF using Loop-B, Loop-C, or Auxiliary sprays during RCS heatup or cooldown, THEN **RECORD** the ΔT between the lowest indicating spray water source in service and the highest indicating Pressurizer temperature on Attachment 3, Cooldown Data Sheet. _______

13. WHEN data recording is **NO** longer required, THEN **RECORD** completion date and time in Attachment 1, Reason for Performance of Data Sheet. _______

End of Section 4.3

INITIAL

5.0 RESTORATION AND DOCUMENTATION

5.1 Restoration

None

5.2 Documentation

1. Acceptance Criteria of Section 6.1: ______

   ☐ SAT ☐ UNSAT

2. Functional Criteria of Section 6.2: ______

   ☐ SAT ☐ UNSAT

Remarks: ______________________________

Performed By:

(Signature)	(Print)	(Init)	(Date)
x x x	operator	S	today

Reviewed By: ______ (Shift Manager or SRO Designee) ______ (Print) ______ (Date)

Approved By: ______ (Shift Manager or SRO Designee) ______ (Print) ______ (Date)

Reviewed By: ______ (Reactor Engineering Supervisor/Designee) ______ (Print) ______ (Date)

6.0 ACCEPTANCE AND FUNCTIONAL CRITERIA

6.1 Acceptance Criteria

1. RCS heatup does **NOT** exceed 100°F in any one hour.
2. Pressurizer heatup does **NOT** exceed 100°F in any one hour.
3. RCS cooldown does **NOT** exceed 100°F in any one hour.
4. Pressurizer cooldown does **NOT** exceed 200°F in any one hour.
5. During in service hydrostatic and leak testing operations above the heatup and cooldown limit curves, RCS temperature change does **NOT** exceed 5°F in any one hour.
6. ΔT between Pressurizer (liquid and vapor) and Pressurizer spray water shall **NOT** exceed 320°F.

6.2 Functional Criteria

1. RCS heatup does **NOT** exceed 90°F in any one hour.
   [Section 8.1.2, Developmental 4.B]
2. Pressurizer heatup does **NOT** exceed 90°F in any one hour.
3. RCS cooldown does **NOT** exceed 90°F in any one hour.
   [Section 8.1.2, Developmental 4.B]
4. Pressurizer cooldown does **NOT** exceed 190°F in any one hour.
5. PRZ liquid temperature is maintained at least 100°F greater than the highest reading hot leg temperature during RCS cooldown. The minimum 100°F Δ T limit between RCS and pressurizer is to ensure a safe subcooling margin such that any steam formation will occur in the pressurizer.
   - IF PRZ is solid, THEN this criteria is **NOT** applicable.
   - IF RCS pressure indication is greater than or equal to 2235 psig, THEN this criteria is **NOT** applicable.
6. RCS pressurization rate during heatup does **NOT** exceed 400 psi in any one hour for RCS Pressure from 1500 psig to NOP.

7.0 RECORDS

1. Date, time, and section(s) completed shall be entered in the Unit Narrative Log.

2. Problems encountered while performing the procedure shall be entered in the Unit Narrative Log; i.e., malfunctioning equipment, delays due to change in plant conditions, etc.

3. Completed copies of the below listed items document compliance with Technical Specification surveillance requirements and shall be transmitted to QA Records for retention per QA Records Program:

    - Section 3.0
    - Section 5.2
    - Attachment 1, Reason for Performance of Data Sheet
    - Attachment 2, Heatup Data Sheet
    - Attachment 3, Cooldown Data Sheet

8.0 REFERENCES AND COMMITMENTS

8.1 References

8.1.1 Implementing

None

8.1.2 Developmental

1. Technical Specifications
   - A. 4.4.9.1.1 RCS Pressure/Temperature Limits
   - B. 4.4.9.2 Pressurizer Temperature Limits
2. FSAR
   - A. Chapter 4.2.6
3. Plant Procedures
   - A. 3-GOP-305, Hot Standby to Cold Shutdown
   - B. 3-GOP-503, Cold Shutdown to Hot Standby
4. Miscellaneous Documents
   - A. PC/M 04-112, Emergency Response Data Acquisition and Display System (ERDADS) Replacement
   - B. JPN-PTN-SEMJ-89-067, Change to Administrative Temperature Limits on RCS Heatup and Cooldown Rates
   - C. EC 247008, PCM-09139 EPU Umbrella Doc Only PC/M
   - D. AR 1644725-05, Administrative limit for heat up rate during plant startup.
   - E. AR 1709722, RV-4-551A Safety valve leakage.

8.1.3 Management Directives

None

8.2 Commitments

None

ATTACHMENT 1
Reason for Performance of Data Sheet
(Page 1 of 2)

Acceptance Criteria (Section 6.0)

Section 4.1, Step 4

☒ Heatup	RCS -	(1)	Administrative - Less than 90°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 100°F in any 1 hour (**ACTION 2**)
		(3)	Administrative - Less than 400 psi in any 1 hour for RCS pressure from 1500 psig to 2235 psig
		(4)	Administrative Less than 50 psi in any 1 hour for RCS Pressure from 2000 psig to NOP
	PRZ -	(1)	Administrative - Less than 90°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 100°F in any 1 hour (**ACTION 2**)
		(3)	Tech. Spec. - Less than 320°F ΔT between PRZ (liquid and vapor) and PRZ spray water
☐ Cooldown	RCS -	(1)	Administrative - Less than 90°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 100°F in any 1 hour (**ACTION 2**)
	PRZ -	(1)	Administrative - Less than 190°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 200°F in any 1 hour (**ACTION 2**)
		(3)	Administrative - If PRZ is **NOT** solid, PRZ liquid temp at least 100°F greater than the highest reading hot leg temperature (see Section 6.2, Step 5)
		(4)	Tech. Spec. - Less than 320°F ΔT between PRZ (liquid and vapor) and PRZ spray water
☐ Inservice Hydrostatic and Leak Test	RCS -	(1)	Tech. Spec. - Less than or equal to 5°F in any 1 hour (**ACTION 2**)
	PRZ -	(1)	Administrative - Less than 190°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 200°F in any 1 hour (**ACTION 2**)

ATTACHMENT 1
Reason for Performance of Data Sheet
(Page 2 of 2)

ACTION 1 - IF the Administrative heatup or cooldown rate is exceeded, THEN:

- Immediately **REDUCE** the rate to less than the allowable rate.
- **NOTIFY** the Shift Manager or Unit Supervisor.

ACTION 2 - If the Technical Specification heatup or cooldown rate is exceeded, THEN:

- Immediately **REDUCE** the rate to less than the allowable rate.
- **NOTIFY** the Shift Manager or Unit Supervisor.
- **TAKE** actions required by Technical Specifications.

today / 1200 / S

Date/Time Complete / Initials

ATTACHMENT 2
Heatup Data Sheet
(Page 1 of 1)

NOTE 1: Pressurizer spray may be from B-loop, C-loop, or Regen Hx Outlet, using TE452 (T452_A), TE451 (T451_A) or TI-3-123 (T123_A) respectively, whichever spray source in service indicates lowest temperature and instrument used should be indicated.
NOTE 2: NA for RCS pressure < 1500 psig.

Time	RCS T-Hot				Pressurizer						RCS Pressure		
	Loop A	Loop B	Loop C	ΔT	Liquid		Vapor		Spray				
	TR-3-413 RED (T413A_A)	TR-3-413 BLUE (T423A_A)	TR-3-413 GREEN (T433A_A)	Highest in 60 Min	Temp TI-3-453 (T453_A)	ΔT in 60 Min	Temp TI-3-454 (T454_A)	ΔT in 60 Min	Water Temp (NOTE 1) T 451 A	ΔT Water Temp to LIQ or VAP	PI-3-403 (P403_A)	PI-3-405 (P405_A)	ΔP in 60 Min (NOTE 2)
0900	202	203	202	—	333	—	338	—	200	138	338	334	N/A
0915	223	224	223	21	346	13	351	13	220	131	353	348	
0930	234	235	234	32	355	22	360	22	231	129	372	372	
0945	251	252	251	49	367	34	371	33	249	122	377	373	
1000	261	262	260	59	372	39	376	38	258	118	382	378	
1015	299	300	298	76	407	61	412	61	296	116	393	389	
1030	335	336	334	74	441	86	447	87	331	116	431	428	
1045	340	341	340	89	446	79	450	79	338	112	461	458	
1100	349	350	348	88	452	80	456	80	347	109	470	467	
1115	373	375	374	76	478	71	484	72	370	114	493	490	
1130	380	381	380	46	479	38	485	38	378	107	501	498	
1145	379	380	379	39	481	35	486	36	378	108	500	497	
1200	379	381	380	32	481	29	485	29	378	107	500	497	

ATTACHMENT 3
Cooldown Data Sheet
(Page 1 of 1)

NOTE 1: Pressurizer spray may be from B-loop, C-loop, or Regen Hx Outlet, using TE452 (T452_A), TE451 (T451_A) or TI-3-123 (T123_A) respectively, whichever spray source in service indicates lowest temperature and instrument used should be indicated.

Time	RCS T-Cold				Pressurizer						RCS Pressure		PRZ to RCS ΔT	
	Loop A	Loop B	Loop C	ΔT	Liquid		Vapor		Spray					
	TR-3-410 RED (T410A_A)	TR-3-410 BLUE (T420A_A)	TR-3-410 GREEN (T430A_A)	Highest in 60 Min	Temp TI-3-453 (T453_A)	ΔT in 60 Min	Temp TI-3-454 (T454_A)	ΔT in 60 Min	Water Temp (**NOTE 1**) T______	ΔT Water Temp to LIQ or VAP	PI-3-403 (P403_A)	PI-3-405 (P405_A)	Highest Hot Leg T______	ΔT PRZ LIQ to Highest Hot Leg

L-16-1 NRC Exam

Admin - JPM RO A2

JPM TITLE: Review an ECO for the B AFW Pump

JPM NUMBER: 01201013103 **REV.** 1-1

TASK NUMBER(S) / TASK TITLE(S): 01201013100/ Write Equipment Clearance Orders

K/A NUMBERS: 2.2.13 **K/A VALUE:** RO 4.1 / SRO 4.3

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: X

EVALUATION LOCATION:

In-Plant: ☐ Control Room: ☐

Simulator: ☐ Other: X

Lab: ☐

Time for Completion: 35 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☐	☐	☒
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER SUPERVISOR	DATE DATE
1-0	Updated to fleet template; text/grammar changes	2015 LOCT Annual Exam		N/A N/A	N/A N/A
1-1	Formatting; text/grammar changes		N/A	Schilk Wilson	

SIMULATOR SET-UP:

- N/A

Required Materials:

- Handout ECO package
- OP-AA 101-1000, Clearance and Tagging
- 3-NOP-075, Auxiliary Feedwater System
- 4-NOP-075, Auxiliary Feedwater System

General References:

- OP-AA 101-1000, Clearance and Tagging
- 3-NOP-075, Auxiliary Feedwater System
- 4-NOP-075, Auxiliary Feedwater System
- 5610-E-855, Breaker List
- 5614-E-321, Vital DC Bus 4D01 and 4D01A Load List

Task Standards:

- Given a prepared ECO, identify any existing errors

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- MOV-6459B, B AFW Pump T&T Valve, has a damaged linkage that requires replacement.
- Mechanical Maintenance has requested that the B AFW Pump's turbine be disabled from starting.
- Maintenance activities will NOT breach the piping to the turbine.
- The Admin RCO prepared the attached ECO package (without marked-up drawings) and submitted it for review.
- The eSOMS database is NOT available for clearance research and preparation.
- AFW is in its normal alignment.

INITIATING CUES:

- The Shift Manager directs you to review the ECO package for MOV-6459B.
- If the ECO is correct and complete, <u>sign</u> as the reviewer; if the ECO is NOT correct or complete, <u>revise</u> it as appropriate. (Note: Walkdown, hanging, and restoration steps are NOT required.)

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain the completed ECO package.
Evaluator Cue:	Provide examinee with the following: • Handout OP-AA-101-1000-F01, Clearance Development and Implementation • Handout OP-AA-101-1000-F02, Paper Based Tagging Instructions • 5610-M-3075 (Sheets 1 - 2), Auxiliary Feedwater System • 5613-M-3075 (Sheets 1 - 3), Auxiliary Feedwater System • 5614-M-3075 (Sheets 1 - 3), Auxiliary Feedwater System • 3-NOP-075, Auxiliary Feedwater System • 4-NOP-075, Auxiliary Feedwater System
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: Yes**	**Review ECO package for completeness/correctness and revise as appropriate.**
Standard:	Determine the following: • MOV-6459B's T&T valve must be in the <u>tripped</u>, rather than the reset, position. • AFSS-002B must be in a <u>lock-closed-plus</u>, rather than just a lock-closed, condition. • <u>AFSS-001B</u> must be closed/tagged, but was omitted from the ECO.
Evaluator Note:	Refer to the key; <u>all</u> of the above elements are critical.
Evaluator Cue:	If requested, provide examinee with a copy of 5610-E-855, Breaker List.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Terminating Cue: **When the examinee completes the ECO revision, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- MOV-6459B, B AFW Pump T&T Valve, has a damaged linkage that requires replacement.
- Mechanical Maintenance has requested that the B AFW Pump's turbine be disabled from starting.
- Maintenance activities will NOT breach the piping to the turbine.
- The Admin RCO prepared the attached ECO package (without marked-up drawings) and submitted it for review.
- The eSOMS database is NOT available for clearance research and preparation.
- AFW is in its normal alignment.

INITIATING CUES:

- The Shift Manager directs you to review the ECO package for MOV-6459B.
- If the ECO is correct and complete, <u>sign</u> as the reviewer; if the ECO is NOT correct or complete, <u>revise</u> it as appropriate. (Note: Walkdown, hanging, and restoration steps are NOT required.)

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

CLEARANCE DEVELOPMENT AND IMPLEMENTATION
Steps within a section may be performed in any logical order

> **NOTE**
> A facsimile of the following forms may be used to improve usability provided the content of the form is maintained

Clearance ID XXXXXXX **Unit** 3 and 4

SECTION 1	Development Actions	Preparer	Reviewer	N/A
1	**Determine** the work scope by reviewing Clearance request.	☒	☐	☐
2	**Review** work order steps for applicability.	☒	☐	☐
3	**Identify** potential stored energy sources (steam, air, electricity, etc).	☒	☐	☐
4	**Have** all power supplies been considered such as motive force, control power, interlocks, computer points, annunciators, back feed sources, etc.?	☒	☐	☐
5	**Reference** a master or copy a previous Clearance and adjust as required. Identify the Master or copied Clearance___________________	☒	☐	☐
6	**Ensure** that the proper work order task(s) have been included on the Clearance.	☒	☐	☐
7	**Review** Operations Instructions for the affected system(s). Do not “cherry pick” specific steps of a procedure. Direct the action to be accomplished as referenced in a procedure.	☒	☐	☐
8	**Ensure** all procedures that place a component in a configuration required for isolation or restoration are referenced along with the revision number.	☐	☐	☒
9	**Determine** proper tags to be used. (Caution, Danger, Operating Permit, Information, etc.). If specified, do they match the clearance request?	☒	☐	☐
10	**Verify** adequacy of the isolation points when using sliding links or lifted leads for electrical isolation, using connection or wiring diagrams, when available.	☐	☐	☒
11	**Verify** any potential electrical back-feed sources to a bus are isolated, keeping in mind it is not just the physical electrical bus but also the switchgear cubicle(s) that are part of any inspection.	☐	☐	☒
12	**Ensure** the proposed boundary components are not being worked on or have a known deficiency which affects the ability to control the energy.	☒	☐	☐
13	**Ensure** pump discharge is closed before closing the pump suction (if applicable).	☐	☐	☒
14	**Ensure** a vent path is established.	☐	☐	☒
15	**Ensure** a drain path is established.	☐	☐	☒
16	**Identify** approximate liquid volume to drain.	☐	☐	☒
17	**Identify** seals and locks to be removed/ installed.	☒	☐	☐
18	**Identify** breaker configuration (ganged or single).	☒	☐	☐
19	**Identify** additional requirements if purging gaseous volume.	☐	☐	☒
20	**Ensure** positions of the throttle valves are directed to be recorded.	☐	☐	☒

SECTION 1	**Development Actions**	**Preparer**	**Reviewer**	**N/A**
21	**Ensure** adequate detail to maintain plant configuration control.	☒	☐	☐
22	**Identify** proper sequence for hanging tags including ALARA considerations	☒	☐	☐
23	**Identify** Maintenance in Progress (MIP) tags that need to be hung.	☐	☐	☒
24	**Perform** conflict checks If a conflict is identified or special hanging sequence is required, coordinate with the Work Week Manager/Coordinator to incorporate concerns into the schedule.	☐	☐	☒
25	**Identify** proper component position for tag removal.	☒	☐	☐
26	**Ensure** the Restoration Instructions contain adequate detail when returning equipment to service. Identify recommended restoration positions based upon expected plant conditions.	☒	☐	☐
27	**Identify** in the Attributes if any components being tagged affect containment integrity. Determine if a separate Configuration Control clearance is necessary to comply with TS Action Statements.	☐	☐	☒
28	**Are** all LCO's, etc. listed for Equipment Tech Spec, TRM, Fire Plan, ODAM	☒	☐	☐
29	**Identify** alarms or annunciators that are affected.	☐	☐	☒
30	**Identify SERT** criteria.	☒	☐	☐
31	**Does** Clearance affect more than one system or train? If so ensure all affected systems are listed.	☒	☐	☐
32	**Ensure** the "Clearance Hazards" clearly identifies in CAPITAL LETTERING potential safety hazards such as remaining fluids in piping systems, hot/energized work and hazards in the job location, etc. Special equipment needed to accomplish the Clearance may be listed here or as a Comment Step.	☒	☐	☐
33	**Will** the support of another work group be required (i.e., Security, Maintenance, HP)? If YES, consider having them scheduled to provide the required support.	☐	☐	☒
34	**Consider** including any marked up drawings depicting the isolation boundaries.	☒	☐	☐
35	**Can** isolation be accomplished without interfering with protected system equipment? If not, then coordinate with Work Week Manager/Coordinator to comply with WM-AA-1000 for risk management.	☒	☐	☐

***Prepared by*: XXX / today** Initial/ Date	***Reviewed by*:_________________** Initial/ Date

CLEARANCE DEVELOPMENT AND IMPLEMENTATION
Steps within a section may be performed in any logical order

Clearance ID ____________________ **Unit** __________

SECTION 2	**Clearance Walkdown Checklist**	**YES**	**NO**	**N/A**
1	Are locks or seals properly identified? If NO*, list changes needed or what is required	☐	☐	☐
2	Are there any adjacent hazards that would affect hanging the Clearance? If YES*, list hazards or what is needed to safely hang the Clearance.	☐	☐	☐
3	Is the Energy Release Criteria identified, appropriate, and capable of being performed? If NO*, list recommendations and return for re-write.	☐	☐	☐
4	Are hose /fittings needed for draining/ venting? If YES*, list materials needed.	☐	☐	☐
5	Are locations correct? Are they specific enough? If NO*, initiate an ECR (Engineering Change Request) or Condition Report as needed	☐	☐	☐
6	Are there any elevated Clearance points that require ladders or scaffolding that is not or will not be available? If YES*, list what is required.	☐	☐	☐
7	Is an entry into High-Rad or Contamination Area required? If YES*, determine RWP required and/or HP support required.	☐	☐	☐
8	Is the tag sequence proper? If NO*, list concerns and return for re-write.	☐	☐	☐
9	Are procedures listed that are needed during Clearance? If No*, update the Clearance to reflect any required procedure.	☐	☐	☐
10	Will any off normal "Tags Plus" equipment be necessary to prevent inadvertent operations of danger tagged components (Handwheel clam shells, etc)? If YES*, list what would be necessary.	☐	☐	☐
11	Are label requests required? If YES*, initiate label request and indicate labels requested.	☐	☐	☐
12	Consider including photographs to assist the implementation of the Clearance	☐	☐	☐
13	Do sliding links or lifted lead points have adequate thread or clearance?	☐	☐	☐
14	Identify other tools/equipment necessary to perform Clearance.	☐	☐	☐

COMMENTS (mandatory for any "*" response) WALK DOWN WAIVED ☐

***Completed by*:** ____________________ Initial/ Date	***Reviewed by*:** __________________ Initial/ Date

Clearance ID _____________________ **Unit** __________

SECTION 3	Clearance Hanging Authorization	Shift Supervision
1	**Single** valve isolations require OSM authorization.	☐
2	**Ensure** that equipment condition supports clearance to be hung.	☐
3	**Ensure** that current plant conditions support clearance to be hung.	☐
4	**Validate** LCO Action Statements, Fire Impairments, Plant Risk (if applicable).	☐
5	**Ensure** that the Control Room is aware of clearance to be hung	☐
6	**Ensure** that the proper log entries are made when required.	☐
7	**SRO Sign** onto Configuration Control clearance	☐
8	**Determine** if Maintenance support is required.	☐
9	**Ensure** clearance briefing performed.	☐

SECTION 4	Restoration Actions	Preparer	Approver	N/A
1	**Review** work order for coordination or PMT requirements.	☐	☐	☐
2	**Review** procedural lineup positions.	☐	☐	☐
3	**Review** spring return switch positions.	☐	☐	☐
4	**Verify** flowpaths will be properly restored.	☐	☐	☐
5	**Identify** correct component positions, including recorded throttle valve positions.	☐	☐	☐
6	**To prevent potential water hammer, provide direction for restoring system to operation, if unable to ensure system is filled and vented,**	☐	☐	☐
7	**Identify** any filling and venting requirements including I&C support for instrumentation.	☐	☐	☐
8	**Grounds** are removed first in relationship to other electrical isolation components.	☐	☐	☐
9	**Identify** correct restoration sequence including ALARA considerations.	☐	☐	☐
10	**Identify** special verification requirements.	☐	☐	☐
11	**Identify** special hazards / instructions when restoring.	☐	☐	☐

Clearance ID ______________________ **Unit** ____________

SECTION 5	Clearance Removal Authorization	Shift Supervision
1	**Verify** that all work assigned to the Clearance is complete or no longer requires tags and that equipment condition supports restoration. (ready to receive energy).	☐
2	**Ensure** that current plant conditions support restoration of equipment.	☐
3	**Consider** ALARA when reviewing sequence of steps.	☐
4	**Ensure** that the Control Room is aware of equipment to be restored.	☐
5	**Determine** if Maintenance support is required.	☐
6	**Ensure** restoration briefing performed.	☐
7	**Verify** all tags are authorized to be removed, otherwise **verify** the Clearance instructions for the remaining tags are updated to reflect any instructions unable to be performed.	☐

SECTION 6	Clearance Closure	Shift Supervision
1	**Ensure** eSOMS is updated to reflect tags are removed and verified.	☐
2	**Status** the Clearance as COMPLETED and ARCHIVE (if desired)	☐
3	**Update** the Control Room	☐

PAPER BASED TAGGING INSTRUCTIONS

> **NOTE**
> A facsimile of the following forms may be used to improve usability provided the content of the form is maintained

1. Work may continue on jobs where the Clearance Holders had been signed on to a Clearance prior to the eSOMS software failure.

2. No new Clearance Holders may sign on to the eSOMS Clearance until the software is available.

3. At least one qualified Clearance Owner shall accept the paper based clearance as a Clearance Holder before remaining Clearance Holders sign on.

4. The qualified Clearance Owner shall be the last Clearance Holder to release the paper based Clearance.

5. If the Paper-Based Tagging Form is used to remove tags previously hung using the tagging software, then update the tagging software as soon as it is restored.

6. If the Paper-Based Tagging process is used to hang tags, then the Paper-Based system SHALL be used to remove the same tags.

7. Tags that are issued by the Paper-Based Tagging Form should be replaced with the tagging software tags at the earliest opportunity.

8. Tags hung using the Paper-Based Tagging Form SHALL not share tags.

9. Tags hung using the Paper-Based Tagging Form SHALL be handwritten using a permanent marker and SHALL contain the following minimum information:
7.1 Component ID number
7.2 Component description
7.3 Component position
7.4 Clearance Number

10. Use Paper-Based Tagging Form to perform the following actions:
8.1 Prepare tagging
8.2 Review tagging
8.3 Approve tagging
8.4 Authorize Clearance
8.5 Hang / Remove tags
8.6 Complete the entire Clearance
8.7 Sign on/off as a Holder

WORK AGAINST/PURPOSE OF TAGGING: Replace T&T Valve linkage on B AFW Pump

CLEARANCE #: (on file)

CLEARANCE TYPE: WO (Work Order) ☒ CC (Configuration Control) ☐

WORK ORDER NUMBER (S): (on file)

PREPARED BY: (on file)

NOTES and SERT REQUIREMENTS: None

HAZARDS: None

STORED ENERGY RELEASED VERIFIED BY:

____

PAPER BASED TAGGING INSTRUCTIONS
(Page 3 of 5)

WORK AGAINST/PURPOSE OF TAGGING: Replace T&T Valve linkage on B AFW Pump

CLEARANCE #: (on file)

WORK ORDER NUMBER (S): (on file)

HOLDER SIGN ON/OFF (use additional sheets as needed)

Page ___ of ___

Holder Name	Sign On	Date/Time	Sign Off	Date/Time

PAPER BASED TAGGING INSTRUCTIONS

WORK AGAINST / PURPOSE OF TAGGING: Replace T&T Valve linkage on B AFW Pump

CLEARANCE #: (on file)

CLEARANCE TYPE: (HANG) ☒ (CLEAR) ☐

PREPARED BY: (on file)

REVIEWED BY:________________________

APPROVED BY: ________________________ AUTHORIZED BY:_______________

STEP NO.	TAG NO.	TAG TYPE	ACTION	COMPONENT ID	COMPONENT DESCRIPTION	REQUIRED POSITION	STEP COMPLETED BY	STEP VERIFIED BY
1	--	Info	Hang Info Tag	MOV-6459B Unit 3 control switch	AFWP B T&T control switch (Unit 3 console)	N/A		
2	--	Info	Hang Info Tag	MOV-6459B Unit 4 control switch	AFWP B T&T control switch (Unit 4 console)	N/A		
3	1	Danger	Hang Danger Tag	4D01-4	Control power B AFW Pump	Off plus		
4	--	No tag	--	MOV-6459B mechanical trip lever	AFWP B T&T valve	Reset		
5	2	Danger	Hang Danger Tag	3-10-084A	AFWP B train-1 upstream steam isolation valve (Unit 3)	Lock closed plus		

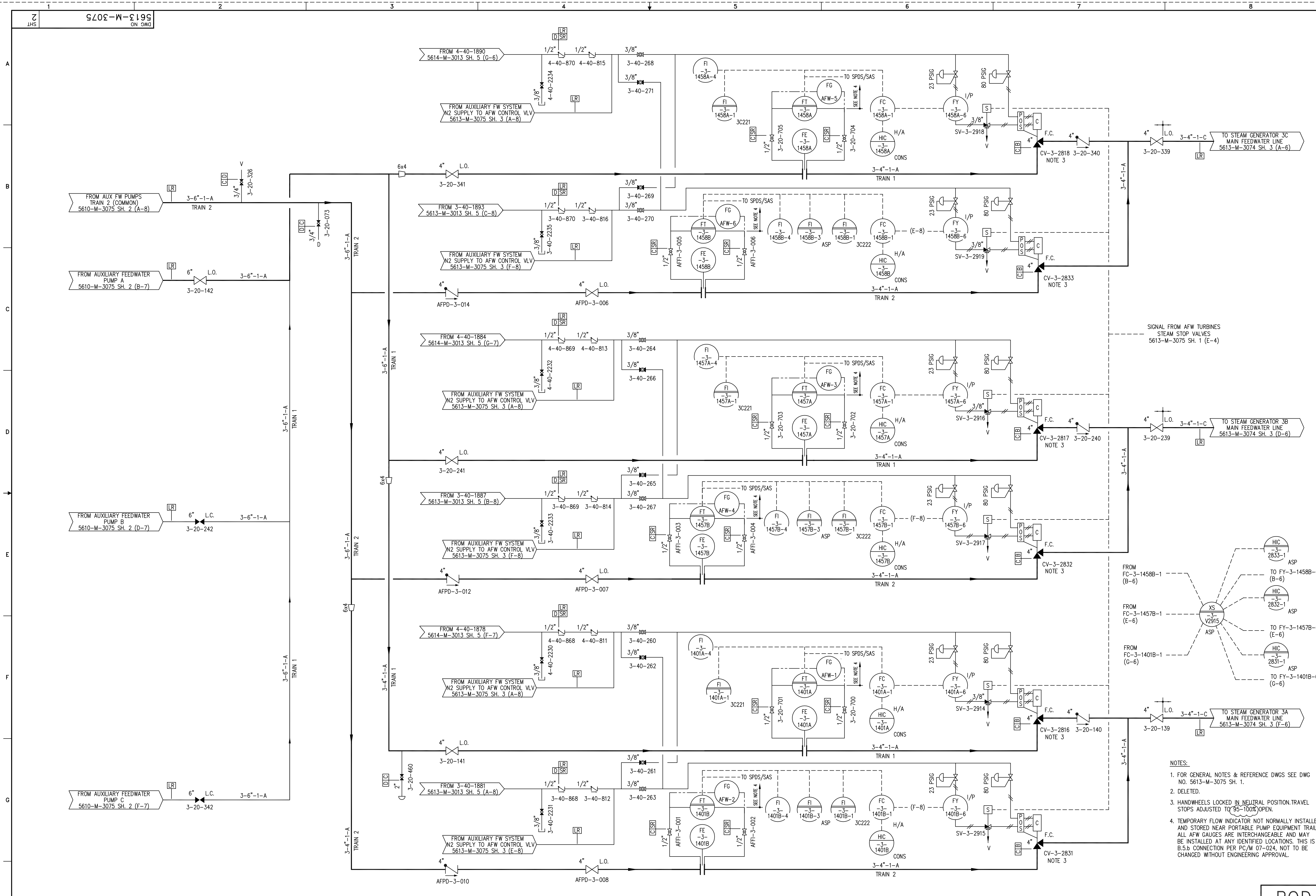
6	3	Danger	Hang Danger Tag	3-10-084B	AFWP B train-1 downstream steam isolation valve (Unit 3)	Lock closed plus		
7	4	Danger	Hang Danger Tag	4-10-084A	AFWP B train-1 upstream steam isolation valve (Unit 4)	Lock closed plus		
8	5	Danger	Hang Danger Tag	4-10-084B	AFWP B train-1 downstream steam isolation valve (Unit 4)	Lock closed plus		
9	6	Danger	Hang Danger Tag	AFSS-002B	AFWP B train-2 downstream steam isolation valve (Unit 3,4)	Lock closed		
10	7	Danger	Hang Danger Tag	AFSS-013	AFWP B train-2 steam trap (ST-47) isolation valve	Closed plus		

KEY

STEP NO.	TAG NO.	TAG TYPE	ACTION	COMPONENT ID	COMPONENT DESCRIPTION	REQUIRED POSITION	NOTES
1	--	Info	Hang Info Tag	MOV-6459B Unit 3 control switch	AFWP B T&T control switch (Unit 3 console)	N/A	None
2	--	Info	Hang Info Tag	MOV-6459B Unit 4 control switch	AFWP B T&T control switch (Unit 4 console)	N/A	None
3	1	Danger	Hang Danger Tag	4D01-4	Control power B AFW Pump	Off plus	None
4	--	No tag	--	MOV-6459B mechanical trip lever	AFWP B T&T valve	~~Reset~~ Tripped	T&T valve must be in tripped condition (**critical**)
5	2	Danger	Hang Danger Tag	3-10-084A	AFWP B train-1 upstream steam isolation valve (Unit 3)	Lock closed plus	None
6	3	Danger	Hang Danger Tag	3-10-084B	AFWP B train-1 downstream steam isolation valve (Unit 3)	Lock closed plus	None
7	4	Danger	Hang Danger Tag	4-10-084A	AFWP B train-1 upstream steam isolation valve (Unit 4)	Lock closed plus	None
8	5	Danger	Hang Danger Tag	4-10-084B	AFWP B train-1 downstream steam isolation valve (Unit 4)	Lock closed plus	None

KEY

9	6	Danger	Hang Danger Tag	AFSS-002B	AFWP B train-2 downstream steam isolation valve (Unit 3,4)	Lock closed plus	None
10	7	Danger	Hang Danger Tag	3-20-244	AFWP B suction isolation valve (Unit 3)	Lock closed plus	None
11	8	Danger	Hang Danger Tag	4-20-244	AFWP B suction isolation valve (Unit 4)	Lock closed plus	"Lock closed" is NOT sufficient (**critical**)
12	9	Danger	Hang Danger Tag	AFSS-013	AFWP B train-2 steam trap (ST-47) isolation valve	Closed plus	None
13	10	Danger	Hang Danger Tag	AFSS-001B	AFWP B train-2 upstream steam isolation valve (Unit 3,4)	Lock closed plus	Valve omitted from ECO (**critical**)



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5613-M-3075 SH. 1.
2. DELETED.
3. HANDWHEELS LOCKED IN NEUTRAL POSITION.TRAVEL STOPS ADJUSTED TO 95-100% OPEN.
4. TEMPORARY FLOW INDICATOR NOT NORMALLY INSTALLED AND STORED NEAR PORTABLE PUMP EQUIPMENT TRAILER. ALL AFW GAUGES ARE INTERCHANGEABLE AND MAY BE INSTALLED AT ANY IDENTIFIED LOCATIONS. THIS IS A B.5.b CONNECTION PER PC/M 07-024, NOT TO BE CHANGED WITHOUT ENGINEERING APPROVAL.

THIS DRAWING SUPERSEDES DRAWINGS:	
5610-M-361	REV. 3
5610-M-1301	REV. 7
5610-M-1303	REV. 5

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4062 SH 3 REV. 64

REV	DATE	REVISION	BY	CH	APP	APP
15	8-8-12	ISSUED AS-BUILT PER EC 275011.	RV	AFG	JL	JD
14	7-28-12	ISSUED AS-BUILT PER EC 247006 (PC/M 09-137).	RV	AFG	–	PSB
13	6-30-09	ISSUED AS-BUILT PER PC/M 07-024 AND INCORP. CRN-M-12376.	RV	BB	JKP	PRB
12	3-6-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061) AND CRN-M-9997	JPG	RH	BSG	JMM
—	—	(PC/M 99-061).	—	—	—	—
11	10-14-98	ISSUED AS-BUILT PER PC/M 98-005.	RH	RV	CMZ	JJC
10	5-26-98	ISSUED AS-BUILT PER CRN-I-3712(PC/M 98-013).	RV	RH	BP	TS
9	5-19-98	ISSUED AS-BUILT PER PC/M 98-005.	RH	RV	JZ	JJC
8	04-12-95	INCORPORATED CRN-I-3159 FOR PC/M 95-031.	RV	RH	DSR	JCW
0	5-14-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	RNB	MD	AM/RS	LRB
—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-137.	—	—	—	—

FPL

TURKEY POINT NUCLEAR UNIT 3

P & ID

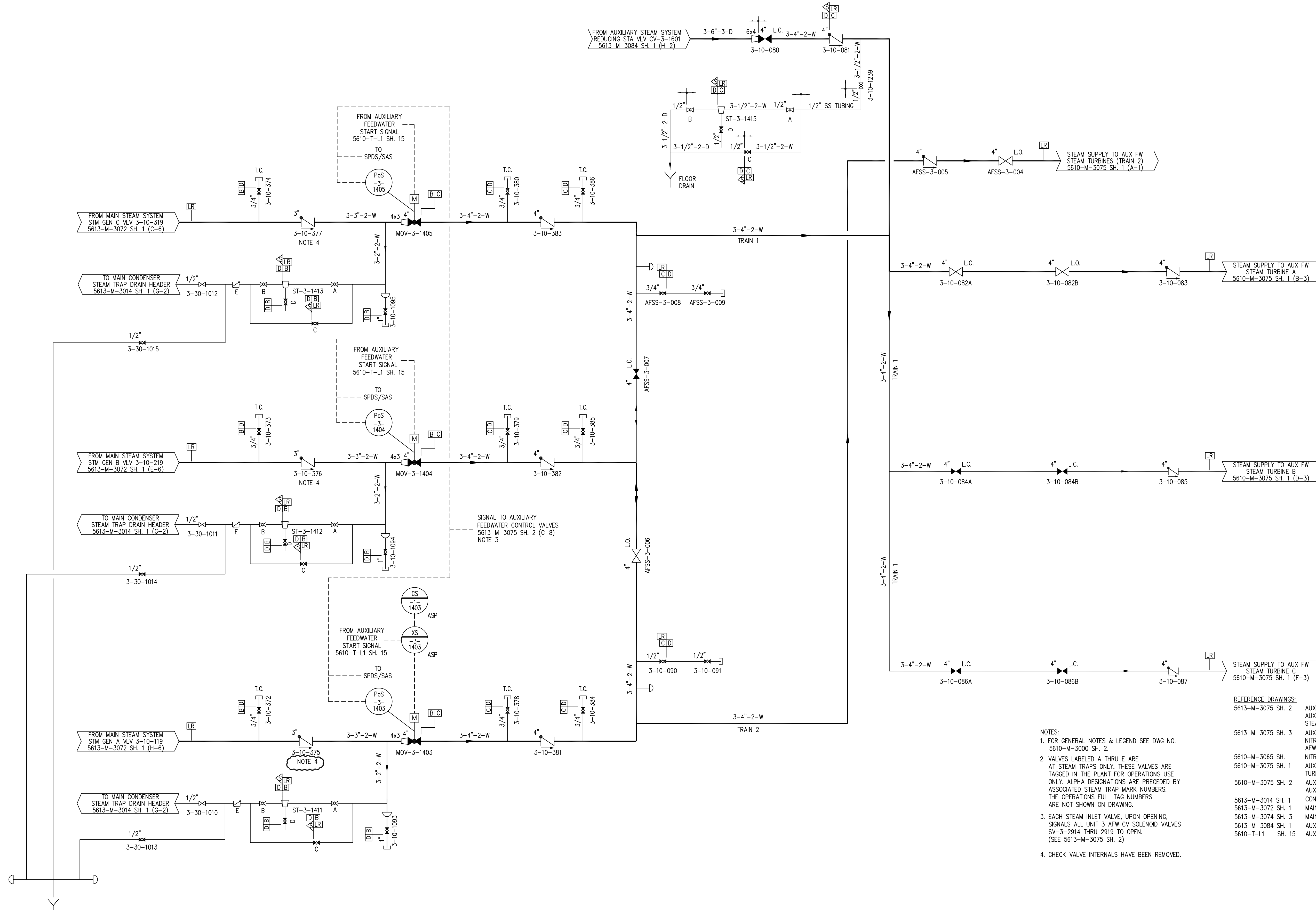
AUXILIARY FEEDWATER SYSTEM
AUXILIARY FEEDWATER TO
STEAM GENERATORS

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3075
SYS 075
SHEET 2
REV 15

5613-M-3075_SH 2.DWG

DWG NO 5613-M-3075 SHT 1



REFERENCE DRAWINGS:

5613-M-3075 SH. 2	AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER TO STEAM GENERATORS
5613-M-3075 SH. 3	AUXILIARY FEEDWATER SYSTEM NITROGEN SUPPLY TO AFW CONTROL VALVES
5610-M-3065 SH.	NITROGEN & HYDROGEN SYSTEMS
5610-M-3075 SH. 1	AUXILIARY FEEDWATER SYSTEM TURBINE DRIVE FOR AFW PUMPS
5610-M-3075 SH. 2	AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER PUMPS
5613-M-3014 SH. 1	CONDENSER SYSTEM
5613-M-3072 SH. 1	MAIN STEAM SYSTEM
5613-M-3074 SH. 3	MAIN FEEDWATER SYSTEM
5613-M-3084 SH. 1	AUXILIARY STEAM SYSTEM
5610-T-L1 SH. 15	AUXILIARY FEEDWATER PUMP START

NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2.
2. VALVES LABELED A THRU E ARE AT STEAM TRAPS ONLY. THESE VALVES ARE TAGGED IN THE PLANT FOR OPERATIONS USE ONLY. ALPHA DESIGNATIONS ARE PRECEDED BY ASSOCIATED STEAM TRAP MARK NUMBERS. THE OPERATIONS FULL TAG NUMBERS ARE NOT SHOWN ON DRAWING.
3. EACH STEAM INLET VALVE, UPON OPENING, SIGNALS ALL UNIT 3 AFW CV SOLENOID VALVES SV-3-2914 THRU 2919 TO OPEN. (SEE 5613-M-3075 SH. 2)
4. CHECK VALVE INTERNALS HAVE BEEN REMOVED.

POD

THIS DRAWING SUPERSEDES DRAWINGS:
5610-M-1 SH. 2 REV. 5
5610-M-1302 REV. 5

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4061 SH. 4 REV. 30

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
8	9-5-95	INCORPORATED CRN-M-8346 FOR PC/M 95-099.	RH	RV	DJB	BD	12	03-29-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061).	JPG	RGR	——	JRH
17	5-13-10	ISSUED AS-BUILT PER PC/M 05-036.	RH	RV	BSG	ASD	11	11-24-98	ISSUED AS-BUILT PER CRN-M-9439 (PC/M 98-036).	RH	RV	JMM	JDL
16	8-30-05	ISSUED AS-BUILT PER PC/M 05-036. (PARTIAL)	RH	RV	AAP	RSV	10	6-3-97	ISSUED AS-BUILT PER PC/M 96-029 AND INCORPORATED CRN-M-8922.	RH	JK	JZ	JJC
15	8-15-05	ISSUED AS-BUILT PER PC/M 05-036. (PARTIAL)	RH	RRW	AAP	JTL	9	6-18-96	INCORPORATED CRN-M-8684 FOR PC/M 96-023.	RH	RV	PDK	BD
14	02-07-02	ISSUED AS-BUILT PER CRN-C-10939 (PC/M 00-016).	RH	RV	CMZ	JTL	0	5-14-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	GFH	MD	AM/RAS	LRB
13	05-09-01	ISSUED AS-BUILT PER CRN-M-10292 (PC/M 00-016).	RH	RV	AAP	JM	——	——	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-137.	——	——	——	——

FPL

TURKEY POINT NUCLEAR UNIT 3

P & I D

AUXILIARY FEEDWATER SYSTEM
STEAM TO AUXILIARY FEEDWATER
PUMP TURBINES

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

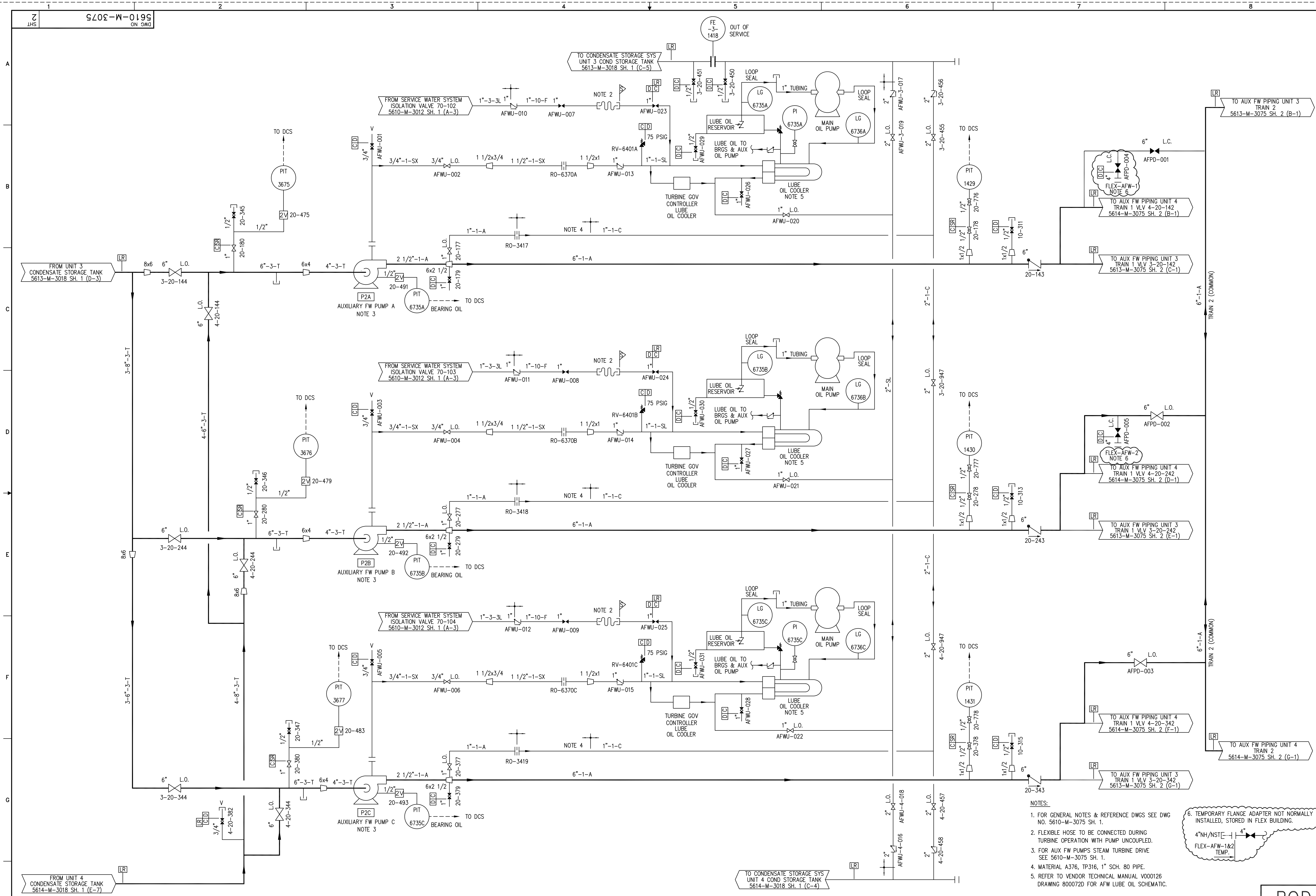
DRAWING NUMBER 5613-M-3075

SHEET 1

SYS 075

REV 17

5613-M-3075_SH 1.DWG



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5610-M-3075 SH. 1.
2. FLEXIBLE HOSE TO BE CONNECTED DURING TURBINE OPERATION WITH PUMP UNCOUPLED.
3. FOR AUX FW PUMPS STEAM TURBINE DRIVE SEE 5610-M-3075 SH. 1.
4. MATERIAL A376, TP316, 1" SCH. 80 PIPE.
5. REFER TO VENDOR TECHNICAL MANUAL V000126 DRAWING 800072D FOR AFW LUBE OIL SCHEMATIC.
6. TEMPORARY FLANGE ADAPTER NOT NORMALLY INSTALLED, STORED IN FLEX BUILDING.

4"NH/NST 4" FLEX-AFW-1&2 TEMP.

POD

THIS DRAWING SUPERSEDES DRAWINGS:
5610-M-1303 REV. 5

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4062 SH. 3 REV. 64

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
17	7-10-08	ISSUED AS-BUILT PER PC/M 04-112 (Partial); INCORP. CRN M-12369 (Partial)	RH	RV	–	PMB	21	4-8-15	ISSUED AS-BUILT PER EC 280631. (PARTIAL)	RH	BB	–	SB
--	---	AND CRN M-12189 (Partial).	--	--	--	--	20	6-6-12	ISSUED AS-BUILT PER EC 276270. (PARTIAL)	RV	AFG	–	PRB
16	6-11-08	ISSUED AS-BUILT PER PC/M 04-112 (Partial) & INCORP. CRN M-12369 (Partial)	RH	RV	–	PMB	19	1-19-11	ISSUED AS-BUILT PER EC-DCR 249651.	RH	RV	BSG	PRB
15	12-19-03	ISSUED AS-BUILT PER PC/M 03-102.	RV	JK	TEJ	JTL	18	7-22-08	ISSUED AS-BUILT PER PC/M 04-112; INCORP. CRN'S M-12189 & M-12369.	RH	RV	–	PMB
14	10-17-03	ISSUED AS-BUILT PER PC/M 03-055.	RH	JK	JH	JTL	0	5-14-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	GFH	MD	AM/RS	LRB
22	4-22-15	ISSUED AS-BUILT PER EC 280631.	RH	BB	–	LOG	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-137.	—	—	—	—

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

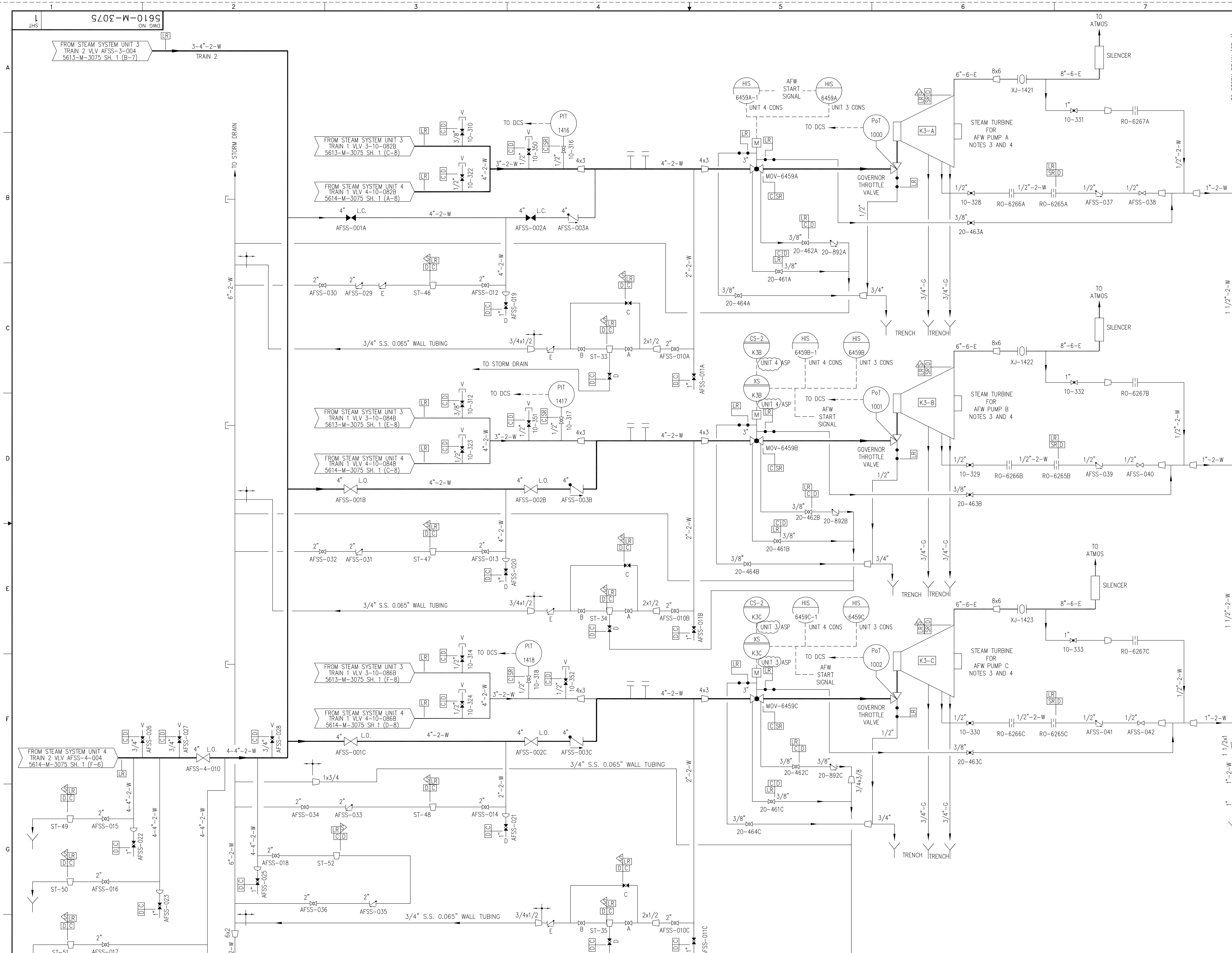
P & ID

AUXILIARY FEEDWATER SYSTEM

AUXILIARY FEEDWATER PUMPS

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER	SYS
5610-M-3075	075
SHEET 2	REV 22



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2.
2. VALVES LABELED A THRU J (EXCLUDING I) ARE AT STEAM TRAPS ONLY. THESE VALVES ARE TAGGED IN THE PLANT FOR OPERATIONS USE ONLY. ALPHA DESIGNATIONS ARE PRECEDED BY ASSOCIATED STEAM TRAP MARK NUMBERS AND ARE NOT IN TEDB. THE OPERATIONS FULL TAG NUMBERS ARE NOT SHOWN ON DRAWING.
3. FOR AUXILIARY FEEDWATER PUMPS SEE DWG NO. 5610-M-3075 SH. 2.
4. TURBINE OVERSPEED TRIP SETPOINTS: MECHANICAL: 6490 ± 59 RPM
5. ALL SAFETY-RELATED PIPING AND ASSOCIATED COMPONENTS DEPICTED ON THIS DRAWING ARE WITHIN THE SCOPE OF LICENSE RENEWAL UNLESS NOTED OTHERWISE.

REFERENCE DRAWINGS:

5613-M-3014 SH. 1	CONDENSER SYSTEM
5613-M-3018 SH. 1	CONDENSATE STORAGE SYSTEM
5613-M-3075 SERIES	AUXILIARY FEEDWATER SYSTEM
5614-M-3018 SH. 1	CONDENSATE STORAGE SYSTEM
5614-M-3075 SERIES	AUXILIARY FEEDWATER SYSTEM
5610-M-3012 SH. 1	SERVICE WATER SYSTEM

POD

THIS DRAWING SUPERSEDES DRAWINGS:
5610-M-1 SH. 2 REV. 5
5610-M-1302 REV. 5

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4061 SH. 4 REV. 30

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
25	6-11-08	ISSUED AS-BUILT PER PC/M 04-112. (PARTIAL)	RH	RV	–	PMB	29	3-31-10	ISSUED AS-BUILT PER CRN E-18036 (PC/M 09-106).	RH	RV	–	LG
24	2-23-07	ISSUED AS-BUILT PER CRN M-11948 (PC/M 06-108)	RH	RV	EM	PRB	28	9-16-08	ISSUED AS-BUILT PER CRN M-12440 (PC/M 08-038).	RH	BB	PJV	SB
23	6-8-04	ISSUED AS-BUILT PER CRN-M-10757 (PC/M 02-065).	RH	RV	CMZ	JTL	27	7-22-08	ISSUED AS-BUILT PER PC/M 04-112.	RH	RV	–	PMB
22	11-01-03	ISSUED AS-BUILT PER CRN-M-10854 (PC/M 03-050).	RH	JK	–	JTL	26	7-9-08	ISSUED AS-BUILT PER PC/M 04-112 (PARTIAL).	RH	RV	–	PMB
21	1-18-02	ISSUED AS-BUILT PER CRN-M-10470 (PC/M 01-038).	RH	RV	CMZ	JTL	0	5-14-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	TAM	MD	AM/RS	LRB
20	03-29-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061).	JPG	RGR	——	JRH	——	——	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-137.	——	——	——	——

FPL

TURKEY POINT NUCLEAR UNITS 3 & 4

P & I D

AUXILIARY FEEDWATER SYSTEM

TURBINE DRIVE FOR AFW PUMPS

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5610-M-3075

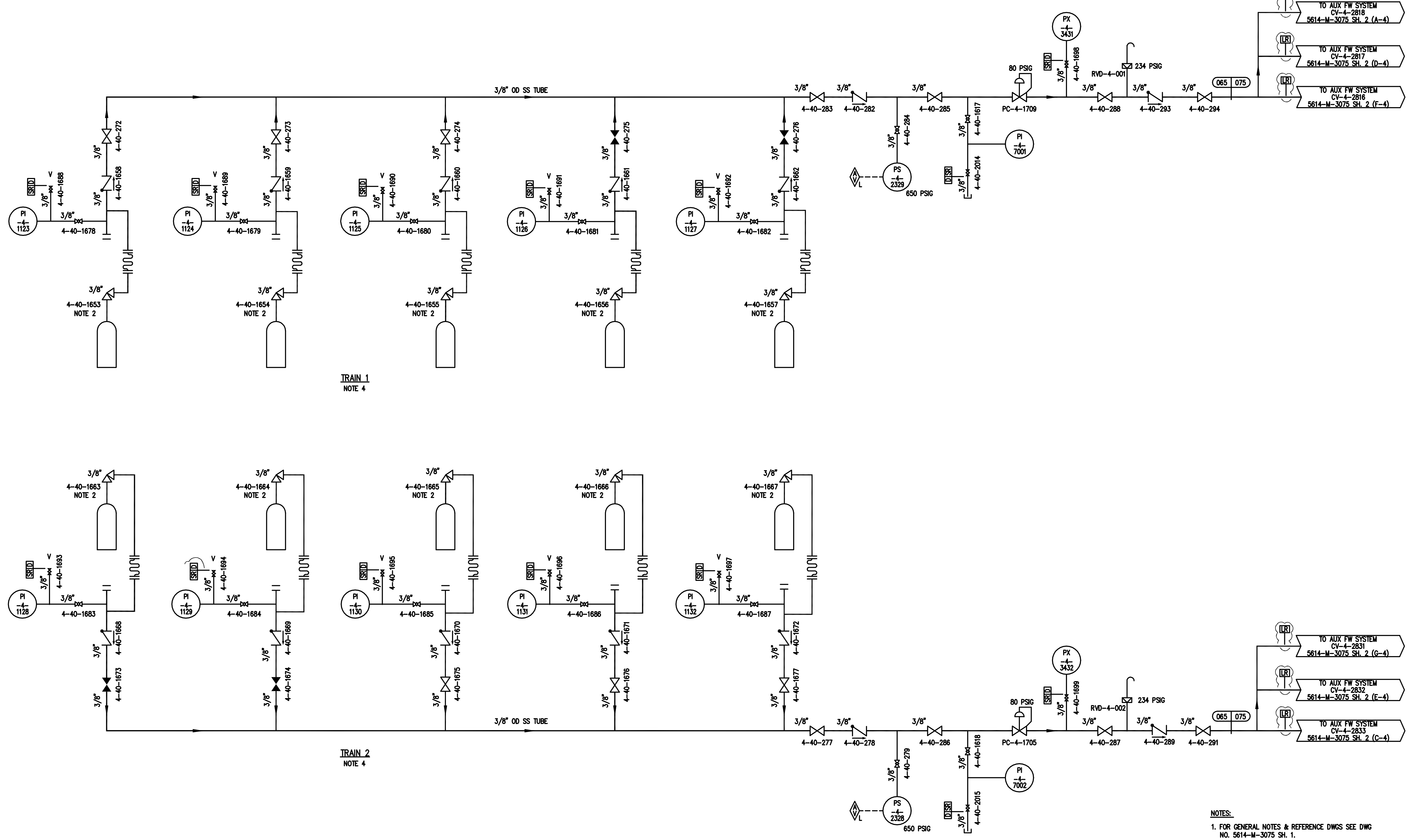
SHEET 1

SYS 075

REV 29

30x42 E SIZE 05/17/93

5610-M-3075_SH 1.DWG



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5614-M-3075 SH. 1.
2. FURNISHED WITH ASSOCIATED EQUIPMENT.
3. ALL PORTIONS OF THIS SYSTEM WITHIN THE SEISMIC FLAG ARE NUCLEAR SAFETY RELATED.
4. N2 HEADER ISOLATION VALVES (272,273,274,275 & 276 FOR TRAIN 1 AND 1673,1674,1675,1676 & 1677 FOR TRAIN 2) ARE SHOWN IN A TYPICAL VALVE LINEUP. ANY THREE OF THE FIVE VALVES PER TRAIN SHALL BE OPEN WITH THE REMAINING TWO VALVES CLOSED.
5. ALL SAFETY-RELATED PIPING AND ASSOCIATED COMPONENTS DEPICTED ON THIS DRAWING ARE WITHIN THE SCOPE OF LICENSE RENEWAL UNLESS NOTED OTHERWISE.

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-M-339 SH. 3 REV. 4

POD

REV	DATE	REVISION	BY	CH	APP	APP
2	03-29-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061).	JPG	RGR	——	JRH
1	12-14-92	ISSUED AS-BUILT FOR PC/M 89-542.	TAM	MD	LRB	HAR
0	03-20-92	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-92-030.	SM	MD	KW AM	LRB

FPL

TURKEY POINT NUCLEAR UNIT 4

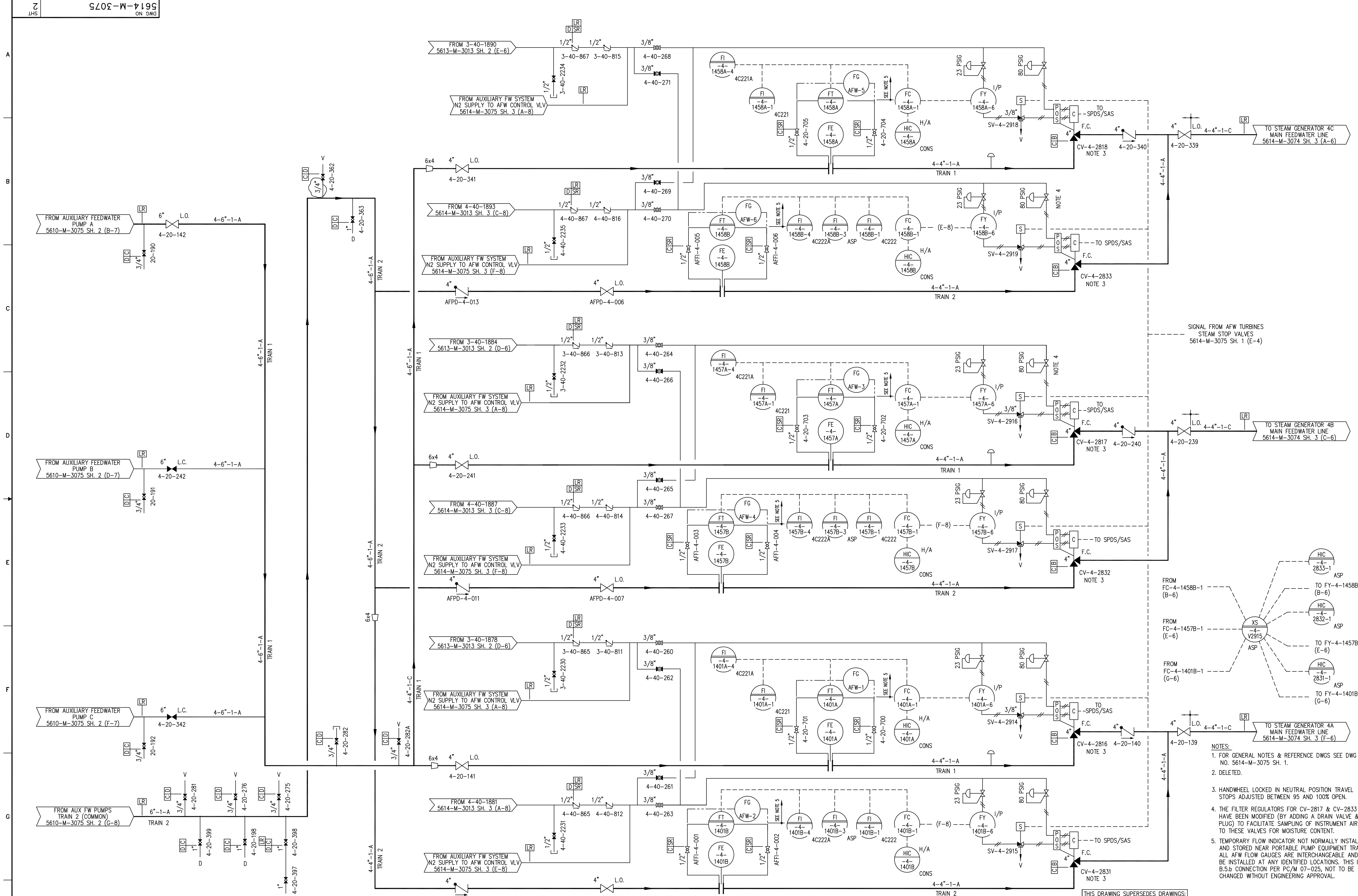
P & I D

AUXILIARY FEEDWATER SYSTEM
NITROGEN SUPPLY TO
AFW CONTROL VALVES

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5614-M-3075
SYS 075
SHEET 3
REV 2

UTM00211.DWG



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5614-M-3075 SH. 1.
2. DELETED.
3. HANDWHEEL LOCKED IN NEUTRAL POSITION TRAVEL STOPS ADJUSTED BETWEEN 95 AND 100% OPEN.
4. THE FILTER REGULATORS FOR CV-2817 & CV-2833 HAVE BEEN MODIFIED (BY ADDING A DRAIN VALVE & PLUG) TO FACILITATE SAMPLING OF INSTRUMENT AIR TO THESE VALVES FOR MOISTURE CONTENT.
5. TEMPORARY FLOW INDICATOR NOT NORMALLY INSTALLED AND STORED NEAR PORTABLE PUMP EQUIPMENT TRAILER. ALL AFW FLOW GAUGES ARE INTERCHANGEABLE AND MAY BE INSTALLED AT ANY IDENTIFIED LOCATIONS. THIS IS A B.5.b CONNECTION PER PC/M 07-025, NOT TO BE CHANGED WITHOUT ENGINEERING APPROVAL.

THIS DWG SUPERSEDES DRAWINGS:	
5610-M-1301	REV. 7
5610-M-1303	REV. 5
5610-M-341	REV. 11
5610-M-361	REV. 3

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-T-E-4062 SH 3 REV. 64

POD

REV	DATE	REVISION	BY	CH	APP	APP
17	3-29-13	ISSUED AS-BUILT PER EC 275011 (ITOP 13-03-304)	RV	RH	-	RDP
16	7-1-12	ISSUED AS-BUILT PER EC 275011. (PARTIAL)	RH	BB	-	JG
15	3-24-12	ISSUED AS-BUILT PER EC 275011 & INCORP. CRN-002 (PARTIAL).	RV	SB	-	PSB
14	3-23-12	ISSUED AS-BUILT PER EC 275011 & INCORP. CRN-002 (PARTIAL).	RV	RH	JL	JD
13	6-30-09	ISSUED AS-BUILT PER PC/M 07-025 AND INCORP. CRN-M-12361.	RV	BB	JKP	PRB
12	05-09-01	ISSUED AS-BUILT PER CRN-M-10117 (PC/M 00-016).	RH	RV	AAP	JM
11	3-6-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061) AND CRN-M-9997	JPG	RH	BSG	BSG
		(PC/M 99-061).				
10	3-27-99	ISSUED AS-BUILT PER PC/M 98-005.	RH	JK	RSV	JLD
18	10-28-13	ISSUED AS-BUILT PER EC-DCR 278758.	RH	RV	-	ELB
0	5-14-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	RNB	MD	AM/RS	LRB
		AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-137.				

FPL

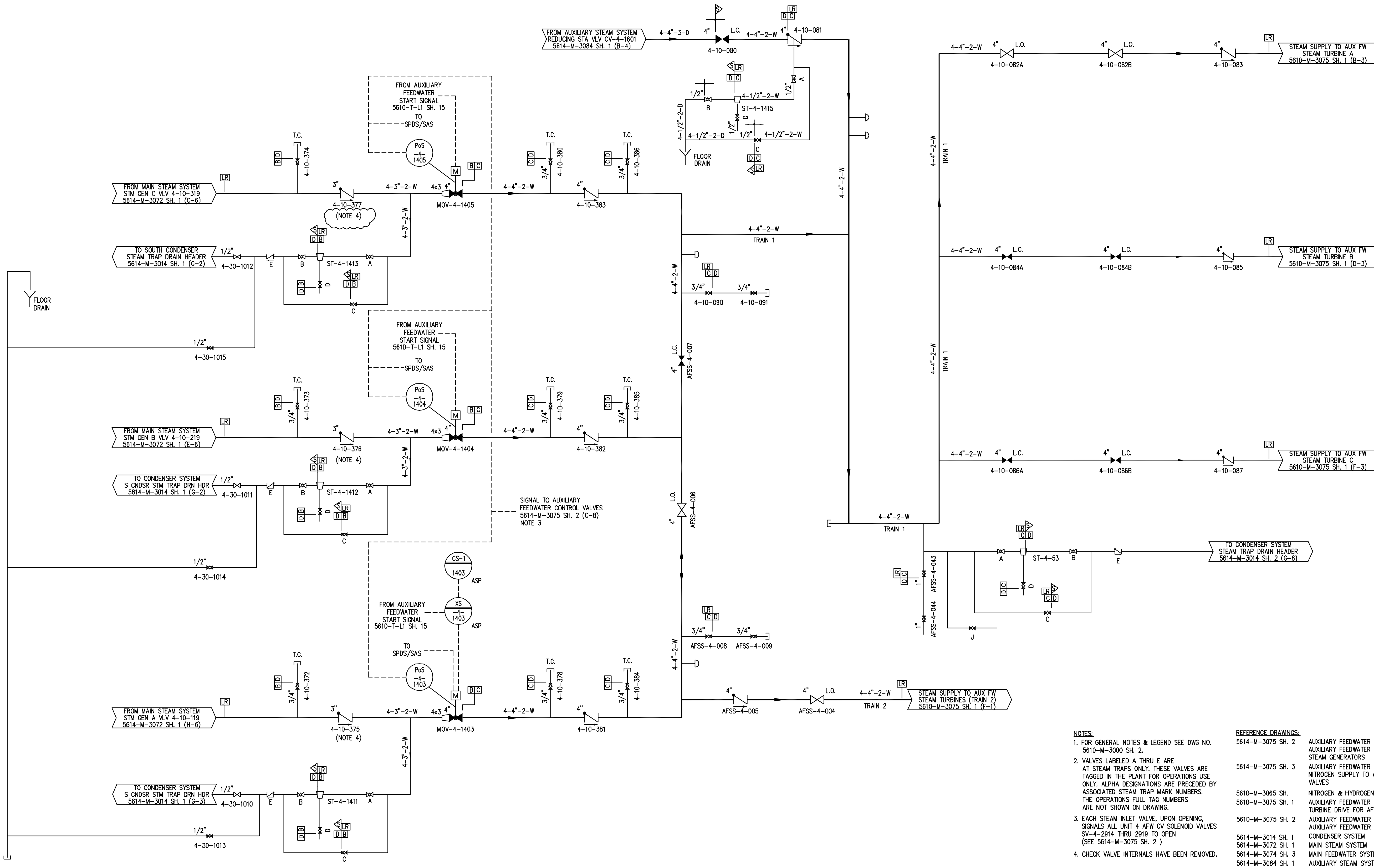
TURKEY POINT NUCLEAR UNIT 4

P & ID

AUXILIARY FEEDWATER SYSTEM
AUXILIARY FEEDWATER TO
STEAM GENERATORS

STONE & WEBSTER ENGINEERING CORP.
FT. LAUDERDALE, FLORIDA

DRAWING NUMBER	SYS
5614-M-3075	075
SHEET 2	REV 18



NOTES:

1. FOR GENERAL NOTES & LEGEND SEE DWG NO. 5610-M-3000 SH. 2.
2. VALVES LABELED A THRU E ARE AT STEAM TRAPS ONLY. THESE VALVES ARE TAGGED IN THE PLANT FOR OPERATIONS USE ONLY. ALPHA DESIGNATIONS ARE PRECEDED BY ASSOCIATED STEAM TRAP MARK NUMBERS. THE OPERATIONS FULL TAG NUMBERS ARE NOT SHOWN ON DRAWING.
3. EACH STEAM INLET VALVE, UPON OPENING, SIGNALS ALL UNIT 4 AFW CV SOLENOID VALVES SV-4-2914 THRU 2919 TO OPEN (SEE 5614-M-3075 SH. 2)
4. CHECK VALVE INTERNALS HAVE BEEN REMOVED.

REFERENCE DRAWINGS:

5614-M-3075 SH. 2	AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER TO STEAM GENERATORS
5614-M-3075 SH. 3	AUXILIARY FEEDWATER SYSTEM NITROGEN SUPPLY TO AFW CONTROL VALVES
5610-M-3065 SH.	NITROGEN & HYDROGEN SYSTEMS
5610-M-3075 SH. 1	AUXILIARY FEEDWATER SYSTEM TURBINE DRIVE FOR AFW PUMPS
5610-M-3075 SH. 2	AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER PUMPS
5614-M-3014 SH. 1	CONDENSER SYSTEM
5614-M-3072 SH. 1	MAIN STEAM SYSTEM
5614-M-3074 SH. 3	MAIN FEEDWATER SYSTEM
5614-M-3084 SH. 1	AUXILIARY STEAM SYSTEM
5610-T-L1 SH. 15	LOGIC DIAGRAM, AUXILIARY FEEDWATER PUMP START

POD

THIS DRAWING SUPERSEDES DRAWINGS:
5610-M-1 SH. 2 REV. 5
5610-M-1302 REV. 5

NOTE: THIS DWG IS MADE FROM: FPL POD 5610-T-E-4061 SH. 4 REV. 30

REV	DATE	REVISION	BY	CH	APP	APP	REV	DATE	REVISION	BY	CH	APP	APP
9	11-24-98	ISSUED AS-BUILT PER CRN-M-9439 (PC/M 98-036).	RH	RV	JMM	JDL	13	6-21-05	ISSUED AS-BUILT PER PC/M 05-036 (PARTIAL).	RH	RV	JK	ASD
8	7-8-97	ISSUED AS-BUILT PER PC/M 96-029 AND INCORPORATED CRN-M-8922.	RH	BSG	JZ	JMM	12	02-06-03	ISSUED AS-BUILT PER CRN-M-10634 (PC/M 02-065).	RH	RV	CMZ	JTL
7	2-25-94	ISSUED AS-BUILT FOR DCR-TPM-93-548.		INITIALS	ON FILE		11	05-09-01	ISSUED AS-BUILT PER CRN-M-10292 (PC/M 00-016).	RH	RV	AAP	JM
16	3-2-10	ISSUED AS-BUILT PER PC/M 05-036.	RH	RV	BSG	PRB	10	03-29-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061).	JPG	RGR	—	JRH
15	8-17-05	ISSUED AS-BUILT PER CRN-M-11477 (PC/M 03-106).	RH	RV	AAP	JTL	0	5-14-91	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE	MAB	MD	AM/RS	LRB
14	7-21-05	ISSUED AS-BUILT PER PC/M 05-036 (PARTIAL).	RV	JK	AAP	RSV	—	—	AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-91-137.	—	—	—	—

FPL

TURKEY POINT NUCLEAR UNIT 4

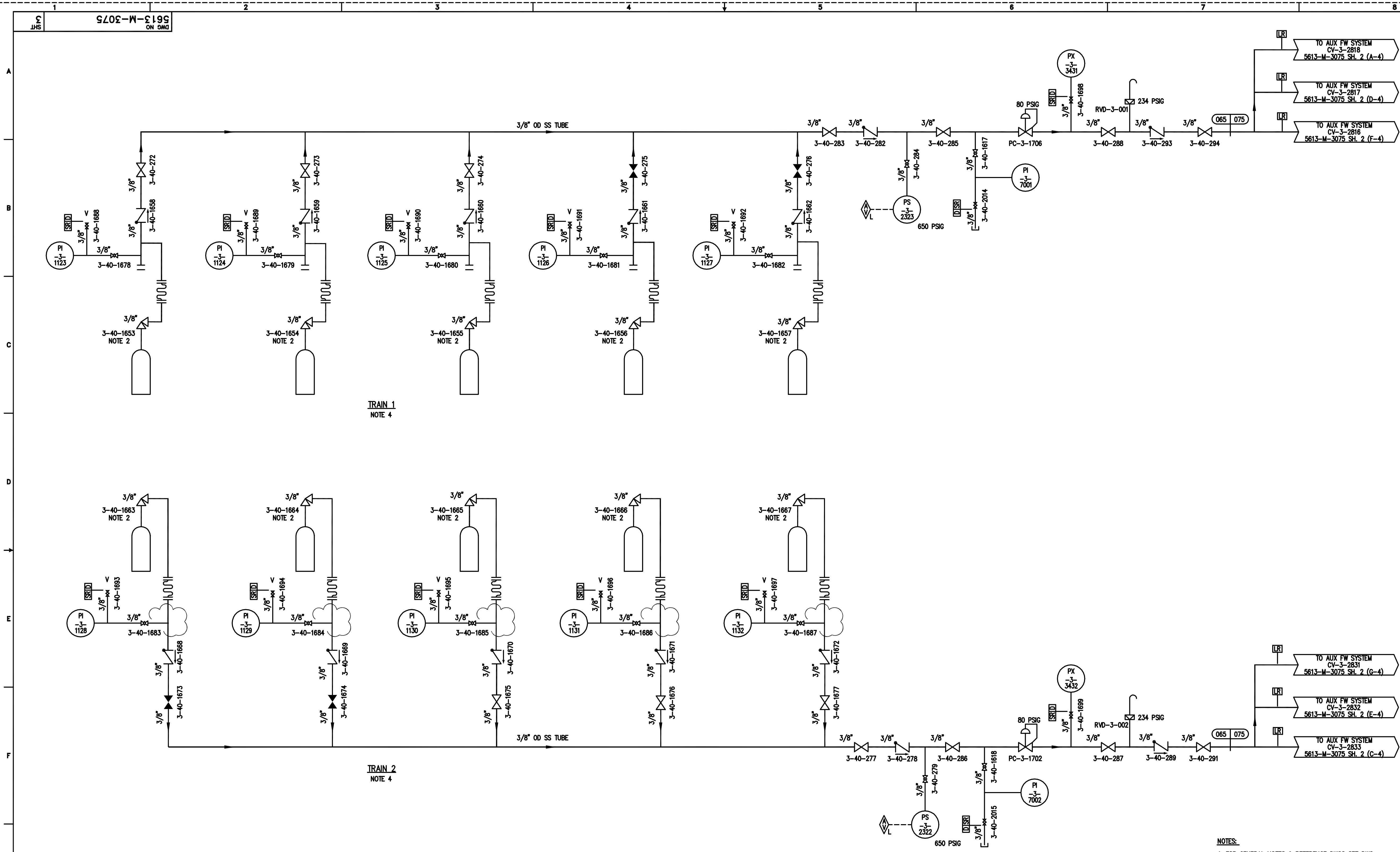
P & ID

AUXILIARY FEEDWATER SYSTEM STEAM TO AUXILIARY FEEDWATER PUMP TURBINES

STONE & WEBSTER ENGINEERING CORP. FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5614-M-3075 | SYS 075 | SHEET 1 | REV 16

5614-M-3075_SH 1



NOTES:

1. FOR GENERAL NOTES & REFERENCE DWGS SEE DWG NO. 5613-M-3075 SH. 1.
2. FURNISHED WITH ASSOCIATED EQUIPMENT.
3. ALL PORTIONS OF THIS SYSTEM WITHIN THE SEISMIC FLAG ARE NUCLEAR SAFETY RELATED.
4. N2 HEADER ISOLATION VALVES (272,273,274,275 & 276 FOR TRAIN 1 AND 1673,1674,1675,1676 & 1677 FOR TRAIN 2) ARE SHOWN IN A TYPICAL VALVE LINEUP. ANY THREE OF THE FIVE VALVES PER TRAIN SHALL BE OPEN WITH THE REMAINING TWO VALVES CLOSED.
5. ALL SAFETY-RELATED PIPING AND ASSOCIATED COMPONENTS DEPICTED ON THIS DRAWING ARE WITHIN THE SCOPE OF LICENSE RENEWAL UNLESS NOTED OTHERWISE.

NOTE: THIS DWG IS MADE FROM:
FPL POD 5610-M-339 SH. 1 REV. 46

POD

REV	DATE	REVISION	BY	CH	APP	APP
4	8-2-00	ISSUED AS-BUILT PER PC/M 99-024.	RH	JK	IJR	RR

REV	DATE	REVISION	BY	CH	APP	APP
3	03-29-00	ISSUED AS-BUILT PER CRN-M-9993 (PC/M 99-061).	JPG	RGR	—	JRH
2	3-6-93	ISSUED AS-BUILT FOR DCR-TPM-93-075.	KT	EMV	LIK	MFM
1	12-14-92	ISSUED AS-BUILT FOR PC/M 89-542.	TAM	INITIALS ON FILE		
0	3-20-92	THIS DWG CREATED PER THE P & ID RECONSTITUTION PROJECT SCOPE AND ISSUED INTO THE FPL DWG SYSTEM PER DCR-TPM-92-030.	SM	MD	KW AM	LRB

FPL

TURKEY POINT NUCLEAR UNIT 3

P & ID

AUXILIARY FEEDWATER SYSTEM NITROGEN SUPPLY TO AFW CONTROL VALVES

STONE & WEBSTER ENGINEERING CORP. FT. LAUDERDALE, FLORIDA

DRAWING NUMBER 5613-M-3075

SYS 075

SHEET 3

REV 4

UTM00208.DWG

L-16-1 NRC Exam

Admin - JPM RO A3

JPM TITLE: Evaluate Conditions for Restart of Refueling Pre-shuffle in the Spent Fuel Pit

JPM NUMBER: 01038034100 **REV.** 0-1

TASK NUMBER(S) / TASK TITLE(S): 01038034100/ Evaluate Conditions for Restart of Refueling Pre-shuffle in the Spent Fuel Pit

K/A NUMBERS: 2.3.12 **K/A VALUE:** RO 3.2 / SRO 3.7

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]

Simulator: [ ] Other: [X]

Lab: [ ]

Time for Completion: 20 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☐	☐	☒
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER SUPERVISOR	DATE DATE
0	New JPM			N/A N/A	N/A N/A
0-1	Formatting; text/grammar changes		N/A	Schilk Wilson	

DRAFT L-16-1 NRC EXAM SECURE INFORMATION

SIMULATOR SET-UP:

- N/A

Required Materials:

- 3-NOP-040.03, Fuel Handling and Insert Shuffle in the Spent Fuel Pit
- Technical Specifications

General References:

- 3-NOP-040.03, Fuel Handling and Insert Shuffle in the Spent Fuel Pit
- Technical Specifications

Task Standards:

- Given a set of plant conditions, identify those that will prevent the recommencement of a refueling pre-shuffle

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Refueling pre-shuffle activities in the Unit 3 Spent Fuel Pit were interrupted and a recommencement is desired.
- The following conditions exist in the Spent Fuel Pit:
  - The associated demineralizer is aligned and water clarity is good.
  - Pool level is 56' 8".
  - 3-12-031 (Unit 3 Fuel Transfer Tube Gate Valve) is closed.
  - Boron concentration is 2330 ppm, per the last sample (taken 36 hours ago).
  - The Spent Fuel Pit’s ventilation system is fully operational.
  - RAD-3-6418 [Unit 3 SFP Vent Stack Radiation Monitor (SPING)], RI-3-1407B (Unit 3 SFP Transfer Canal Radiation Indicator), and RI-3-1421B (Unit 3 SFP North Area Radiation Indicator) are in operation.
  - RI-3-1419B (Unit 3 SFP Exhaust Duct Radiation Indicator) is OOS; a non-alarming portable monitor has been substituted in its place.
  - A Radiation Protection technician is present.
  - An FME Monitor has been contacted and is in route to the Spent Fuel Pit.
  - All three communications systems are functional.
- E-16C (Control Room Ventilation Air Handling Unit C) is OOS.

INITIATING CUE:

- The Shift Manager directs you to perform Step 4.2.1.20 of 3-NOP-040.03 (Fuel Handling and Insert Shuffle in the Spent Fuel Pit), determine whether refueling pre-shuffle activities may recommence, and <u>document</u> any discrepancies.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain a copy of 3-NOP-040.03, Fuel Handling and Insert Shuffle in the Spent Fuel Pit.
Evaluator Cue:	Provide examinee with a copy of 3-NOP-040.03, Fuel Handling and Insert Shuffle in the Spent Fuel Pit.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: No**	**3-NOP-040.03, Step 4.2.1.20:** IF fuel movement is stopped, THEN **PERFORM** Attachment 2, Restart Minimum Equipment Checklist, prior to restart.
Standard:	Transition to Attachment 2.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: No**	**3-NOP-040.03, Attachment 2:** Spent Fuel Pit Ventilation System – OPERABLE
Standard:	Recognize that the ventilation system is fully operational and initial the table.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: Yes**	**3-NOP-040.03, Attachment 2:** Spent Fuel Pit Level – 57' 0"
Standard:	Recognize that the pool level does NOT meet the minimum level requirement and document this as a reason for NOT restarting the fuel shuffle.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: No**	**3-NOP-040.03, Attachment 2:** Spent Fuel Pit Water surface and subsurface is clear enough to allow good visibility during refueling operations – CLEAR
Standard:	Recognize that the pool's water clarity is satisfactory and initial the table.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: No**	**3-NOP-040.03, Attachment 2:** Spent Fuel Pool Demin aligned to SFP – In service
Standard:	Recognize that the demineralization system is fully operational and initial the table.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: Yes**	**3-NOP-040.03, Attachment 2:** RI-3-1407B, RI-3-1419B, and RI-3-1421 Remote/Local Indications and Alarms – OPERABLE (Remarks: IF area monitor is **NOT** operable, THEN **INSTALL** a portable monitor with an alarm.)
Standard:	Recognize that RI-3-1419B is OOS and a <u>non-alarming</u> portable monitor has been substituted, which does NOT meet the minimum requirement; document this as a reason for NOT restarting the fuel shuffle.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 8 **Critical: No**	**3-NOP-040.03, Attachment 2:** Spent Fuel Pit Boron Concentration – Greater than or equal to 2300 ppm
Standard:	Recognize that the most recent boron concentration is satisfactory and initial the table.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 9 **Critical: No**	**3-NOP-040.03, Attachment 2:** Spent Fuel Storage Pool Area High Gaseous Radioactivity Monitor – OPERABLE
Standard:	Recognize that RAD-3-6418 (SPING) is in operation and initial the table.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 10 **Critical: No**	**3-NOP-040.03, Attachment 2:** Communication Headsets/other reliable communication system - Control Room to SFP – CONTINUOUS
Standard:	Recognize that communications are sufficient and initial the table.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 11 **Critical: No**	**3-NOP-040.03, Attachment 2:** RP Coverage – CONTINUOUS
Standard:	Recognize that a Radiation Protection technician is present and initial the table.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 13 **Critical: Yes**	**3-NOP-040.03, Attachment 2:** FME Monitor – CONTINUOUS
Standard:	Recognize that an FME Monitor is NOT present and document this as a reason for NOT restarting the fuel shuffle.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 14 **Critical: Yes**	**3-NOP-040.03, Attachment 2:** Control Room Emergency Ventilation System – OPERABLE
Standard:	Recognize that E-16C is OOS, which does NOT meet the minimum Technical Specification requirement of three Control Room air-handling units; document this as a reason for NOT restarting the fuel shuffle.
Evaluator Cue:	Provide examinee with a copy of the Technical Specifications, if requested.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Terminating Cue: **When the examinee completes the attachment, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ____

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Refueling pre-shuffle activities in the Unit 3 Spent Fuel Pit were interrupted and a recommencement is desired.
- The following conditions exist in the Spent Fuel Pit:
    - The associated demineralizer is aligned and water clarity is good.
    - Pool level is 56' 8".
    - 3-12-031 (Unit 3 Fuel Transfer Tube Gate Valve) is closed.
    - Boron concentration is 2330 ppm, per the last sample (taken 36 hours ago).
    - The Spent Fuel Pit's ventilation system is fully operational.
    - RAD-3-6418 [Unit 3 SFP Vent Stack Radiation Monitor (SPING)], RI-3-1407B (Unit 3 SFP Transfer Canal Radiation Indicator), and RI-3-1421B (Unit 3 SFP North Area Radiation Indicator) are in operation.
    - RI-3-1419B (Unit 3 SFP Exhaust Duct Radiation Indicator) is OOS; a non-alarming portable monitor has been substituted in its place.
    - A Radiation Protection technician is present.
    - An FME Monitor has been contacted and is in route to the Spent Fuel Pit.
    - All three communications systems are functional.
- E-16C (Control Room Ventilation Air Handling Unit C) is OOS.

INITIATING CUE:

- The Shift Manager directs you to perform Step 4.2.1.20 of 3-NOP-040.03 (Fuel Handling and Insert Shuffle in the Spent Fuel Pit), determine whether refueling pre-shuffle activities may recommence, and <u>document</u> any discrepancies.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

DRAFT L-16-1 NRC EXAM SECURE INFORMATION

L-16-1 NRC Exam

Admin - JPM SRO A1a

JPM TITLE: Calculate a Manual Makeup to the VCT

JPM NUMBER: 01046046101 **REV.** 1-1

TASK NUMBER(S) / TASK TITLE(S): 01046046100/ Calculate a Manual Makeup to the VCT

K/A NUMBERS: 2.1.25 **K/A VALUE:** RO 3.9 / SRO 4.2

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION:

In-Plant:	[ ]	Control Room:	[ ]
Simulator:	[ ]	Other:	[X]
Lab:	[ ]		

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

<u>**Protected Content:**</u> (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER SUPERVISOR	DATE DATE
0	New JPM		N/A	N/A N/A	N/A N/A
1-0	Added task for determining controller settings; formatting; text/grammar changes		N/A	Schilk Wilson	

SIMULATOR SET-UP:

- N/A

Required Materials:

- 4-OP-046, CVCS – Boron Concentration Control
- Plant Curve Book, Section III
- Calculator

General References:

- 4-OP-046, CVCS – Boron Concentration Control
- Plant Curve Book, Section III

Task Standards:

- Calculate the boric acid and primary water flow rates, volumes, and controller settings as required to makeup to the VCT, using Method 2 of the Plant Curve Book (Section III)

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 4 is at 100% steady-state power.
- VCT level is 20%.
- Boron concentrations in the RCS and BAST are 874 ppm and 5687 ppm, respectively.
- A manual makeup to the VCT is to be performed, with a desired boric acid flow rate of 11.0 gpm.
- All relevant prerequisites, precautions/limitations, and associated attachments in 0-OP-046, CVCS – Boron Concentration Control, have been addressed.

INITIATING CUES:

- VCT level is to be raised to 37%, while maintaining a constant VCT/RCS boron concentration.
- You are directed to perform Section 5.4 (Manual Makeup) of 0-OP-046, using <u>Method 2</u> (Calculation) from Section III of the Plant Curve Book, to calculate the following parameters:
  - Primary water flow rate: ___________________ (to the nearest <u>tenth of a gpm</u>)
  - Primary water volume: ___________________ (to the nearest <u>gallon</u>)
  - Boric acid volume: ___________________ (to the nearest <u>gallon</u>)
- Based on the available information, determine the potentiometer settings for the following controllers:
  - Boric Acid Flow Controller (FC-4-113A): ___________________
  - Primary Water Flow Controller (FC-4-114A): ___________________

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: __________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 0-OP-046, CVCS – Boron Concentration Control.
Evaluator Cue:	Provide examinee with a copy of 0-OP-046, CVCS – Boron Concentration Control.
Performance:	**SATISFACTORY _____ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: No**	**0-OP-046, Step 5.4.1.1:** Applicable Prerequisites in Section 3.0 are satisfied.
Standard:	Recognize, from the Initial Conditions, that all relevant prerequisites have been addressed.
Performance:	**SATISFACTORY _____ UNSATISFACTORY _____**
Comments:	

Performance Step: 3 **Critical: No**	**0-OP-046, prior to Step 5.4.2.1:** **CAUTION** ***Instrument uncertainties for the Boric Acid and Primary Water flow transmitters can result in the actual amount of Boric Acid or Primary Water added to be either more or less than the amount calculated. Thus, care is needed to ensure that excessive reduction in RCS boron concentration does NOT occur due to the uncertainties.*** **NOTE** ***VCT level is 14.15 gallons per percent level indication.***
Standard:	Read CAUTION/NOTE and recognize that it is safe to proceed.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: No**	**0-OP-046, Step 5.4.2.1:** Determine the approximate boric acid and primary water flows and volumes needed to obtain the desired blend concentration from the boron change tables in Section III of the Plant Curve Book. The primary water flow rate should be determined in order to ensure all primary water is injected prior to completion of the manual make-up.
Standard:	Obtain Section III of the Plant Curve Book and locate Figure 4 (Blended Flow), Method 2 (Calculation).
Evaluator Cue:	Provide examinee with a copy of Section III of the Plant Curve Book.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: Yes**	**Determine the appropriate primary water flow needed to obtain the desired blend, using the blended flow calculation in Section III of the Plant Curve Book.**
Standard:	Determine the primary water flow rate and record the value on the Turnover Sheet. • Primary water flow rate: 60.6 gpm (60.5 to 60.7 gpm)
Evaluator Note:	• From Section III of Plant Curve Book: o $Boron_{ppm} = (Acid_{gpm})(BAST_{ppm})/(Acid_{gpm} + Water_{gpm})$, where $Boron_{ppm}$ is the desired blended boron concentration • Therefore, $Water_{gpm} = [(Acid_{gpm})(BAST_{ppm})/(Boron_{ppm})] - (Acid_{gpm})$: o $Water_{gpm} = [(11.0)(5687)/(874)] - (11.0) =$ 60.6 gpm
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 6 **Critical: Yes**	**Determine the appropriate boric acid and primary water volumes and to raise VCT level from 20% to 37%.**
Standard:	Determine the required boric acid and primary water volumes and record the values on Turnover Sheet. • Primary water volume: <u>204 gallons</u> (±6%; 192 to 216 gallons) • Boric acid volume: <u>37 gallons</u> (±6%; 35 to 39 gallons)
Evaluator Note:	• From NOTE prior to Step 5.4.2.1 of 0-OP-046 (i.e., 14.15 gallons/%): o (37% - 20%)(14.15 gallons/%) = 240.55 gallons • Therefore, with 11.0 gpm of boric acid and 60.6 gpm of primary water: o (240.55 gallons)[(11.0)/(11.0 + 60.6)] = <u>37.0 gallons of boric acid</u> o (240.55 gallons)[(60.6)/(11.0 + 60.6)] = <u>203.6 gallons of primary water</u> • Various methods may be used to determine the fluid volumes • Answer bands are based on potential rounding error (e.g., 14.15 gallons/% rounded up to 15 gallons/% would yield 216 gallons of primary water)
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 7 **Critical: Yes**	**Based on the available information, determine the potentiometer settings for the following controllers:** • **Boric Acid Flow Controller (FC-3-113A)** • **Primary Water Flow Controller (FC-3-114A)**
Standard:	Determine the associated potentiometer settings and record the values on Turnover Sheet. • Boric Acid Flow Controller (FC-4-113A): 2.2 (2.1 to 2.3) • Primary Water Flow Controller (FC-4-114A): 4.0 (3.9 to 4.1)
Evaluator Note:	• From Step 4.23 of 0-OP-046 (ratio of 5 gpm to 1; i.e., 50 gpm maximum), a boric acid flow rate of 11.0 gpm is equivalent to a controller setting of 2.2 on the ten-turn potentiometer. • From Step 4.24 of 0-OP-046 (ratio of 15 gpm to 1; i.e., 150 gpm maximum), a primary water flow rate of 60.6 gpm is equivalent to a controller setting of 4.0 on the ten-turn potentiometer.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When the examinee completes Step 7, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ____________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ____

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

L-16-1 NRC Exam

Admin - JPM SRO A1b

JPM TITLE: Determine Heatup of the RCS

JPM NUMBER: 01041046101 **REV.** 0-1

TASK NUMBER(S) / TASK TITLE(S): 01041046100/ Determine Heatup of the RCS

K/A NUMBERS: 2.1.20 **K/A VALUE:** RO 4.6 / SRO 4.6

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION:

In-Plant:	[ ]	Control Room:	[ ]
Simulator:	[ ]	Other:	[X]
Lab:	[ ]		

Time for Completion: 30 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☒	☐	☐
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

<u>Protected Content:</u> (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
0	New JPM		N/A	N/A	N/A
				N/A	N/A
0-1	Formatting; text/grammar changes		N/A	Schilk	
				Wilson	

SIMULATOR SET-UP:

- N/A

Required Materials:

- Handout 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification
- Technical Specifications
- Calculator

General References:

- 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification
- Technical Specifications
- Plant Curve Book, Section V, Figure 3D

Task Standards:

- Identify discrepancy in heatup determination and list any subsequent procedural actions and/or Technical Specification actions that apply

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Operators at Unit 3 have just completed an RCS heatup.
- The RCS is stable at 380°F and 499 psig.
- 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification, is complete through step 4.2.14.
- All relevant data was recorded on Attachment 2, Heatup Data Sheet.

INITIATING CUES:

- You are directed to <u>review</u> the heatup data, <u>complete</u> the remaining procedural steps, and <u>record</u> any discrepancies and required subsequent actions in Section 5.2.
- [SRO only] <u>Record</u> any relevant Technical Specification actions in Section 5.2.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification.
Evaluator Cue:	Provide examinee with a copy of handout 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: No**	**Review heatup data (Attachment 2) and identify discrepancy.**
Standard:	Recognize that the ΔT value for RCS T_{hot} was miscalculated at 1030 (i.e., the actual value is 101°F, rather than 74°F).
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 3 **Critical: Yes**	**Complete remaining procedural steps and record any discrepancies and required subsequent actions in Section 5.2.**
Standard:	Per Step 4.1.1, immediately notify the Unit Supervisor and document the condition in Section 5.2: • Mark "Acceptance Criteria of Section 6.1" as UNSAT. • Mark "Functional Criteria of Section 6.2" as UNSAT. • In the "Remarks" section, indicate that the heatup rate exceeded the Administrative (<90°F/hour) and Technical Specification (<100°F/hour) limits at 1030. [SRO only] Identify Technical Specification 3.4.9.1, Action a, with the following requirements: • Restore the temperature and/or pressure to within the limit within 30 minutes (effectively accomplished). • Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the RCS. • Determine that the RCS remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.
Evaluator Cue:	When requested, provide examinee with a copy of the Technical Specifications.
Evaluator Note:	Examinee may base required subsequent actions on Attachment 1: • When the Administrative limit is exceeded, immediately reduce the heatup rate (irrelevant at this time) and notify the Shift Manager or Unit Supervisor – ACTION 1. • When the Technical Specification limit is exceeded, immediately reduce the heatup rate (irrelevant at this time), notify the Shift Manager or Unit Supervisor, and take actions required by the Technical Specifications – ACTION 2.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When the examinee completes Step 3, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Operators at Unit 3 have just completed an RCS heatup.
- The RCS is stable at 380°F and 499 psig.
- 3-OSP-041.7, Reactor Coolant System Heatup and Cooldown Temperature Verification, is complete through step 4.2.14.
- All relevant data was recorded on Attachment 2, Heatup Data Sheet.

INITIATING CUES:

- You are directed to <u>review</u> the heatup data, <u>complete</u> the remaining procedural steps, and <u>record</u> any discrepancies and required subsequent actions in Section 5.2.
- [SRO only] <u>Record</u> any relevant Technical Specification actions in Section 5.2.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

FPL	**TURKEY POINT UNIT 3** **OPERATIONS SURVEILLANCE PROCEDURE** SAFETY RELATED **CONTINUOUS USE**	Procedure No. **3-OSP-041.7** Revision No. **5**

Title:

REACTOR COOLANT SYSTEM HEATUP AND COOLDOWN TEMPERATURE VERIFICATION

Responsible Department: OPERATIONS

Special Considerations:

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED today INITIAL S

Revision	Approved By	Approval Date
0	Michael Murphy	05/26/10
5	Mike Murphy	07/14/15

UNIT #	UNIT 3
DATE	
DOCT	PROCEDURE
DOCN	3-OSP-041.7
SYS	
STATUS	COMPLETED
REV	5
# OF PGS	

REVISION SUMMARY	
Rev. No.	**Description**
5	**PCR 1997526, 07/14/15, Brian Fitzgerald** Removed functional criteria for delta T reading between PZR and hottest RCS hot leg reading for RCS heatup in accordance with AR 1960410.
4	**PCR 2017385, 04/30/15, Michael Hargis** Revised RCS Pressurization Rate as recommended by AR 1709722.
3	**AR 1710614, 07/31/12, Joseph Madison** Revised procedure to provide administrative limits on RCS pressurization rate during heatup in accordance with EC 247008 and AR 1644725.
2	**AR 1652046, 06/08/11, Brian Fitzgerald** Revised Functional Criteria for subcooling in accordance with **AR 1627155**.
1	**AR 590917, 1/27/11, Brian Fitzgerald** Revised Acceptance Criteria to prompt user to consider using pressurizer vapor temp when calculating the difference in PZR and spray water temperatures. Addressed/corrected inconsistencies for heatup and cooldown monitoring.
0	**PCR 09-3331, 05/26/10, Dennis Bonsall** Upgraded procedure format to AD-AA-100-1003, FPL Fleet Procedure Writer's Guide standards. Added new section for Scope to cover frequency of performance, applicability, and mode restrictions. Revised step wording to apply human factors in accordance with Writers Guide. Split Heatup and Cooldown guidance into separate sections and made Heatup and Cooldown data sheets separate Attachments to improve Human Performance to reduce potential for error. This procedure supersedes 3-OSP-041.7, approval date 3/25/08

TABLE OF CONTENTS

1.0 PURPOSE AND SCOPE

1.1 Purpose

1. This procedure provides guidance to satisfy the requirements of Technical Specifications:
   - 4.4.9.1.1 - RCS Pressure/Temperature Limits
   - 4.4.9.2 - Pressurizer Temperature Limits
2. This procedure provides guidance on RCS Pressurization rates during RCS heatup to reduce the probability of Pressurizer Safety Valve leakage.

1.2 Scope

1.2.1 Frequency

At least once every 30 minutes during:

- RCS heatup
- RCS cooldown
- In-service leak and hydrostatic testing operations

1.2.2 Applicability

At all times

1.2.3 MODE Restrictions

None

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

1. This surveillance may be performed at the discretion of the Shift Manager/Unit Supervisor when RCS temperature is rising or lowering.

2. During collapsing of the Pressurizer bubble or any RCS heatup or cooldown operation, both RCS and Pressurizer cooldown and heatup limits shall be observed due to possible temperature stratifications, insurges, or outsurges of water in the Pressurizer.

3. If DCS points are used, periodic validation against other equivalent indications is required.

2.2 Limitations

1. The Reactor Coolant System (RCS), excluding the Pressurizer (PRZ), temperature and pressure shall be limited per the heatup and cooldown curves in the Plant Curve Book.

2. To reduce the probability of Pressurizer Safety Valve leakage, RCS pressurization rate during RCS heatup should be limited to 50 psi / hr for RCS pressure between 2000 psi and NOP.

3. The RCS pressurization rate during RCS heatup shall be limited to 400 psi/hr for RCS pressure between 1500 psig and 2235 psig.

3.0 PREREQUISITES

1. **ENSURE** Shift Manager or designee permission is obtained for data collection. S

End of Section 3.0

4.0 INSTRUCTIONS

4.1 General Requirements

1. IF during performance of this procedure any of the following occur:
   - Acceptance/Functional Criteria is UNSAT
   - A malfunction occurs
   - An abnormal condition is found

   THEN:
   - Immediately **NOTIFY** the Unit Supervisor. ______
   - **DOCUMENT** condition in Section 5.2. ______

2. WHEN during RCS heatup or cooldown, three consecutive readings within two degrees are obtained on each recorded RCS and Pressurizer temperature, THEN **DISCONTINUE** this surveillance. S

3. WHEN during in service hydrostatic and leak testing operations, RCS temperature and pressure are below and to the right of the heatup and cooldown limit curves, THEN **DISCONTINUE** this surveillance. N/A

4. **PERFORM** the following in Attachment 1, Reason for Performance of Data Sheet:
   - A. **CHECK** the appropriate block to indicate which requirement(s) is (are) being met by completion. S
   - B. **RECORD** start date and time. today 0900 S

5. **GO TO** the appropriate Section:
   - Section 4.2, Heatup S
   - Section 4.3, Cooldown N/A

End of Section 4.1

4.2 Heatup

NOTE

- DCS readings for the listed instruments should be used, if available, for recording temperatures and pressures. If one or more of these instruments are **NOT** available, an alternate instrument or its associated DCS point may be used. Proper notation should be made under Remarks if alternate instruments are used.
- If DCS points are used, periodic validation against other equivalent indications is required.

1. Using DCS or at VPA, **RECORD** the following indications every 15 minutes on Attachment 2, Heatup Data Sheet: S
   - TR-3-413 Pen 1 Loop A, if RCP A is in operation (DCS T413A_A)
   - TR-3-413 Pen 2 Loop B, if RCP B is in operation (DCS T423A_A)
   - TR-3-413 Pen 3 Loop C (DCS T433A_A)
   - TI-3-453 PRZ Liquid Temp (DCS T453_A)
   - TI-3-454 PRZ Vapor Temp (DCS T454_A)
   - PI-3-403 RCS Pressure (DCS P403_A)
   - PI-3-405 RCS Pressure (DCS P405_A)
   - TI-3-123 REGEN Hx Outlet Temp (DCS T123_A)
2. **RECORD** RCS temperature change every 15 minutes on Attachment 2, Heatup Data Sheet. S
3. For times of less than 1 hour, **RECORD** maximum RCS temperature change every 15, 30, and 45 minutes, while continuing to look back one hour. S
4. **DETERMINE** maximum RCS temperature change from the last 60 minutes, every 15 minutes. S

4.2 Heatup (continued)

5.	**RECORD** PRZ liquid temperature change every 15 minutes on Attachment 2, Heatup Data Sheet.	S
6.	For times of less than 1 hour, **RECORD** PRZ liquid temperature change every 15, 30, and 45 minutes, while continuing to look back 1 hour.	S
7.	**DETERMINE** PRZ liquid temperature change from the last 60 minutes, every 15 minutes.	S
8.	**RECORD** PRZ vapor temperature change every 15 minutes on Attachment 2, Heatup Data Sheet.	S
9.	For times of less than 1 hour, **RECORD** PRZ vapor temperature change every 15, 30, and 45 minutes, while continuing to look back 1 hour.	S
10.	**DETERMINE** PRZ vapor temperature change from the last 60 minutes, every 15 minutes.	S
11.	IF using Loop-B, Loop-C, or Auxiliary sprays during RCS heatup or cooldown, THEN **RECORD** the ΔT between the lowest indicating spray water source in service and the highest indicating Pressurizer temperature on Attachment 2, Heatup Data Sheet.	S
12.	Record RCS Pressure every 15 minutes on Attachment 2, Heatup Data Sheet.	S
13.	IF RCS Pressure is greater than 1500 psig, **DETERMINE** RCS Pressure change from the last 60 minutes, every 15 minutes.	N/A
14.	WHEN data recording is **NO** longer required, THEN **RECORD** completion date and time in Attachment 1, Reason for Performance of Data Sheet.	S

End of Section 4.2

4.3 Cooldown

NOTE

- DCS readings for the listed instruments should be used, if available, for recording temperatures and pressures. If one or more of these instruments are **NOT** available, an alternate instrument or its associated DCS point may be used. Proper notation should be made under Remarks if alternate instruments are used.
- If DCS points are used, periodic validation against other equivalent indications is required.

1. Using DCS or at VPA, **RECORD** the following indications every 15 minutes on Attachment 3, Cooldown Data Sheet: _______

   - TR-3-410 Pen 1 Loop A, if RCP A is in operation (DCS TE410A_A)
   - TR-3-410 Pen 2 Loop B, if RCP B is in operation (DCS TE420A_A)
   - TR-3-410 Pen 3 Loop C (DCS TE430A_A)
   - TI-3-453 PRZ Liquid Temp (DCS T453_A)
   - TI-3-454 PRZ Vapor Temp (DCS T454_A)
   - PI-3-403 RCS Pressure (DCS P403_A)
   - PI-3-405 RCS Pressure (DCS P405_A)
   - TI-3-123 REGEN Hx Outlet Temp (DCS T123_A)

2. **RECORD** RCS temperature change every 15 minutes on Attachment 3, Cooldown Data Sheet. _______

3. For times of less than 1 hour, **RECORD** the maximum RCS temperature change every 15, 30 and 45 minutes, while continuing to look back one hour. _______

4. **DETERMINE** maximum RCS temperature change from the last 60 minutes, every 15 minutes. _______

INITIAL

4.3 Cooldown (continued)

5. **RECORD** PRZ liquid temperature change every 15 minutes on Attachment 3, Cooldown Data Sheet. _______

6. For times of less than one hour, **RECORD** PRZ liquid temperature change every 15, 30, and 45 minutes, while continuing to look back 1 hour. _______

7. **DETERMINE** PRZ liquid temperature change from the last 60 minutes, every 15 minutes. _______

8. **RECORD** PRZ vapor temperature change every 15 minutes on Attachment 3, Cooldown Data Sheet. _______

9. For times of less than one hour, **RECORD** PRZ vapor temperature change every 15, 30, and 45 minutes, while continuing to look back one hour. _______

10. **DETERMINE** PRZ vapor temperature change from the last 60 minutes, every 15 minutes. _______

11. IF the PRZ is **NOT** solid during RCS cooldown, THEN **RECORD** the current ΔT between the highest reading hot leg temperature and Pressurizer liquid temperature, every 15 minutes on Attachment 3, Cooldown Data Sheet. _______

12. IF using Loop-B, Loop-C, or Auxiliary sprays during RCS heatup or cooldown, THEN **RECORD** the ΔT between the lowest indicating spray water source in service and the highest indicating Pressurizer temperature on Attachment 3, Cooldown Data Sheet. _______

13. WHEN data recording is **NO** longer required, THEN **RECORD** completion date and time in Attachment 1, Reason for Performance of Data Sheet. _______

End of Section 4.3

5.0 RESTORATION AND DOCUMENTATION

5.1 Restoration

None

5.2 Documentation

1. Acceptance Criteria of Section 6.1: ______

   ☐ SAT ☐ UNSAT

2. Functional Criteria of Section 6.2: ______

   ☐ SAT ☐ UNSAT

Remarks: ______________________________

Performed By:

(Signature)	(Print)	(Init)	(Date)
x x x	operator	S	today

Reviewed By: ______ (Shift Manager or SRO Designee) ______ (Print) ______ (Date)

Approved By: ______ (Shift Manager or SRO Designee) ______ (Print) ______ (Date)

Reviewed By: ______ (Reactor Engineering Supervisor/Designee) ______ (Print) ______ (Date)

6.0 ACCEPTANCE AND FUNCTIONAL CRITERIA

6.1 Acceptance Criteria

1. RCS heatup does **NOT** exceed 100°F in any one hour.
2. Pressurizer heatup does **NOT** exceed 100°F in any one hour.
3. RCS cooldown does **NOT** exceed 100°F in any one hour.
4. Pressurizer cooldown does **NOT** exceed 200°F in any one hour.
5. During in service hydrostatic and leak testing operations above the heatup and cooldown limit curves, RCS temperature change does **NOT** exceed 5°F in any one hour.
6. ΔT between Pressurizer (liquid and vapor) and Pressurizer spray water shall **NOT** exceed 320°F.

6.2 Functional Criteria

1. RCS heatup does **NOT** exceed 90°F in any one hour.
   [Section 8.1.2, Developmental 4.B]
2. Pressurizer heatup does **NOT** exceed 90°F in any one hour.
3. RCS cooldown does **NOT** exceed 90°F in any one hour.
   [Section 8.1.2, Developmental 4.B]
4. Pressurizer cooldown does **NOT** exceed 190°F in any one hour.
5. PRZ liquid temperature is maintained at least 100°F greater than the highest reading hot leg temperature during RCS cooldown. The minimum 100°F Δ T limit between RCS and pressurizer is to ensure a safe subcooling margin such that any steam formation will occur in the pressurizer.
   - IF PRZ is solid, THEN this criteria is **NOT** applicable.
   - IF RCS pressure indication is greater than or equal to 2235 psig, THEN this criteria is **NOT** applicable.
6. RCS pressurization rate during heatup does **NOT** exceed 400 psi in any one hour for RCS Pressure from 1500 psig to NOP.

7.0 RECORDS

1. Date, time, and section(s) completed shall be entered in the Unit Narrative Log.

2. Problems encountered while performing the procedure shall be entered in the Unit Narrative Log; i.e., malfunctioning equipment, delays due to change in plant conditions, etc.

3. Completed copies of the below listed items document compliance with Technical Specification surveillance requirements and shall be transmitted to QA Records for retention per QA Records Program:

   - Section 3.0
   - Section 5.2
   - Attachment 1, Reason for Performance of Data Sheet
   - Attachment 2, Heatup Data Sheet
   - Attachment 3, Cooldown Data Sheet

8.0 REFERENCES AND COMMITMENTS

8.1 References

8.1.1 Implementing

None

8.1.2 Developmental

1. Technical Specifications
   - A. 4.4.9.1.1 RCS Pressure/Temperature Limits
   - B. 4.4.9.2 Pressurizer Temperature Limits
2. FSAR
   - A. Chapter 4.2.6
3. Plant Procedures
   - A. 3-GOP-305, Hot Standby to Cold Shutdown
   - B. 3-GOP-503, Cold Shutdown to Hot Standby
4. Miscellaneous Documents
   - A. PC/M 04-112, Emergency Response Data Acquisition and Display System (ERDADS) Replacement
   - B. JPN-PTN-SEMJ-89-067, Change to Administrative Temperature Limits on RCS Heatup and Cooldown Rates
   - C. EC 247008, PCM-09139 EPU Umbrella Doc Only PC/M
   - D. AR 1644725-05, Administrative limit for heat up rate during plant startup.
   - E. AR 1709722, RV-4-551A Safety valve leakage.

8.1.3 Management Directives

None

8.2 Commitments

None

ATTACHMENT 1
Reason for Performance of Data Sheet
(Page 1 of 2)

Acceptance Criteria (Section 6.0)

Section 4.1, Step 4

☒ Heatup	RCS -	(1)	Administrative - Less than 90°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 100°F in any 1 hour (**ACTION 2**)
		(3)	Administrative - Less than 400 psi in any 1 hour for RCS pressure from 1500 psig to 2235 psig
		(4)	Administrative Less than 50 psi in any 1 hour for RCS Pressure from 2000 psig to NOP
	PRZ -	(1)	Administrative - Less than 90°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 100°F in any 1 hour (**ACTION 2**)
		(3)	Tech. Spec. - Less than 320°F ΔT between PRZ (liquid and vapor) and PRZ spray water
☐ Cooldown	RCS -	(1)	Administrative - Less than 90°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 100°F in any 1 hour (**ACTION 2**)
	PRZ -	(1)	Administrative - Less than 190°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 200°F in any 1 hour (**ACTION 2**)
		(3)	Administrative - If PRZ is **NOT** solid, PRZ liquid temp at least 100°F greater than the highest reading hot leg temperature (see Section 6.2, Step 5)
		(4)	Tech. Spec. - Less than 320°F ΔT between PRZ (liquid and vapor) and PRZ spray water
☐ Inservice Hydrostatic and Leak Test	RCS -	(1)	Tech. Spec. - Less than or equal to 5°F in any 1 hour (**ACTION 2**)
	PRZ -	(1)	Administrative - Less than 190°F in any 1 hour (**ACTION 1**)
		(2)	Tech. Spec. - Less than 200°F in any 1 hour (**ACTION 2**)

ATTACHMENT 1
Reason for Performance of Data Sheet
(Page 2 of 2)

ACTION 1 - IF the Administrative heatup or cooldown rate is exceeded, THEN:

- Immediately **REDUCE** the rate to less than the allowable rate.
- **NOTIFY** the Shift Manager or Unit Supervisor.

ACTION 2 - If the Technical Specification heatup or cooldown rate is exceeded, THEN:

- Immediately **REDUCE** the rate to less than the allowable rate.
- **NOTIFY** the Shift Manager or Unit Supervisor.
- **TAKE** actions required by Technical Specifications.

today / 1200 / S

Date/Time Complete / Initials

ATTACHMENT 2
Heatup Data Sheet
(Page 1 of 1)

NOTE 1: Pressurizer spray may be from B-loop, C-loop, or Regen Hx Outlet, using TE452 (T452_A), TE451 (T451_A) or TI-3-123 (T123_A) respectively, whichever spray source in service indicates lowest temperature and instrument used should be indicated.
NOTE 2: NA for RCS pressure < 1500 psig.

Time	RCS T-Hot				Pressurizer						RCS Pressure		
	Loop A	Loop B	Loop C	ΔT	Liquid		Vapor		Spray				
	TR-3-413 RED (T413A_A)	TR-3-413 BLUE (T423A_A)	TR-3-413 GREEN (T433A_A)	Highest in 60 Min	Temp TI-3-453 (T453_A)	ΔT in 60 Min	Temp TI-3-454 (T454_A)	ΔT in 60 Min	Water Temp (**NOTE 1**) T 451 A	ΔT Water Temp to LIQ or VAP	PI-3-403 (P403_A)	PI-3-405 (P405_A)	ΔP in 60 Min (NOTE 2)
0900	202	203	202	—	333	—	338	—	200	138	338	334	N/A
0915	223	224	223	21	346	13	351	13	220	131	353	348	
0930	234	235	234	32	355	22	360	22	231	129	372	372	
0945	251	252	251	49	367	34	371	33	249	122	377	373	
1000	261	262	260	59	372	39	376	38	258	118	382	378	
1015	299	300	298	76	407	61	412	61	296	116	393	389	
1030	335	336	334	74	441	86	447	87	331	116	431	428	
1045	340	341	340	89	446	79	450	79	338	112	461	458	
1100	349	350	348	88	452	80	456	80	347	109	470	467	
1115	373	375	374	76	478	71	484	72	370	114	493	490	
1130	380	381	380	46	479	38	485	38	378	107	501	498	
1145	379	380	379	39	481	35	486	36	378	108	500	497	
1200	379	381	380	32	481	29	485	29	378	107	500	497	

ATTACHMENT 3
Cooldown Data Sheet
(Page 1 of 1)

NOTE 1: Pressurizer spray may be from B-loop, C-loop, or Regen Hx Outlet, using TE452 (T452_A), TE451 (T451_A) or TI-3-123 (T123_A) respectively, whichever spray source in service indicates lowest temperature and instrument used should be indicated.

Time	RCS T-Cold				Pressurizer						RCS Pressure		PRZ to RCS ΔT	
	Loop A	Loop B	Loop C	ΔT	Liquid		Vapor		Spray					
	TR-3-410 RED (T410A_A)	TR-3-410 BLUE (T420A_A)	TR-3-410 GREEN (T430A_A)	Highest in 60 Min	Temp TI-3-453 (T453_A)	ΔT in 60 Min	Temp TI-3-454 (T454_A)	ΔT in 60 Min	Water Temp (**NOTE 1**) T______	ΔT Water Temp to LIQ or VAP	PI-3-403 (P403_A)	PI-3-405 (P405_A)	Highest Hot Leg T______	ΔT PRZ LIQ to Highest Hot Leg

L-16-1 NRC Exam

Admin - JPM SRO A2

JPM TITLE: Evaluate Technical Specification Conditions While Performing a Valve Operability Test

JPM NUMBER: 02051013101 **REV.** 1-2

TASK NUMBER(S) / TASK TITLE(S): 02051013100 / Verify/Set Containment Integrity

K/A NUMBERS: 2.2.40 **K/A VALUE:** RO 3.4 / SRO 4.7

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☐ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION:

In-Plant:		Control Room:	
Simulator:		Other:	X
Lab:			

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☐	☐	☒
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER SUPERVISOR	DATE DATE
1-0	Updated to fleet template; text/grammar changes	2015 LOCT Annual Exam	N/A	N/A N/A	N/A N/A
1-1	Typos on cover page	Editorial	N/A	Hodge Wilson	8/27/15 8/27/15
1-2	Formatting; text/grammar changes		N/A	Schilk Wilson	

SIMULATOR SET-UP:

- N/A

Required Materials:

- 4-OSP-047.1E, Letdown Line Isolation Valve Test
- Technical Specifications

General References:

- 4-OSP-047.1E, Letdown Line Isolation Valve Test
- Technical Specifications
- 5614-M-3047, Sheets 1 and 2

Task Standards:

- Determine required Technical Specification actions for accident monitoring instrumentation

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 4 is in Mode 3.
- 4-OSP-047.1E, Letdown Line Isolation Valve Test, is in progress at Step 4.3.2.11.
- When the handswitch for CV-4-200A, 45 GPM L/D Isolation Valve, was taken to CLOSE, the following indications were observed:
  - Red indicating light at CV-4-200A's handswitch – OFF
  - Green indicating light at CV-4-200A's handswitch – OFF
  - "Phase A Isolation" status light for CV-4-200A (VPB) – NOT LIT
- With the exception of notifying the Unit Supervisor and Shift Manager, no other actions have been taken.

INITIATING CUE:

- The Shift Manager directs you to review the following Technical Specifications and identify the required actions, if any (list the appropriate required actions or indicate that none are required):
  - 3.3.2, Engineered Safety Features Actuation System Instrumentation

    ____
    ____
    ____
    ____

  - 3.3.3.3, Accident Monitoring Instrumentation

    ____
    ____
    ____
    ____

  - 3.6.4, Containment Isolation Valves

    ____
    ____
    ____
    ____

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain 4-OSP-047.1E, Letdown Line Isolation Valve Test.
Evaluator Cue:	Provide examinee with a copy of handout 4-OSP-047.1E, Letdown Line Isolation Valve Test.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: No**	**Review Technical Specification 3.3.2, Engineered Safety Features Actuation System Instrumentation, and identify any required actions.**
Standard:	• Review Technical Specification 3.3.2 and Table 3.3-2 (Functional Unit 3a, Phase A Isolation). • Recognize that the manual and automatic actuation capabilities are NOT impacted and <u>no actions are required</u>.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: Yes**	**Review Technical Specification 3.3.3.3, Accident Monitoring Instrumentation, and identify any required actions.**
Standard:	• Review Technical Specification 3.3.3.3 and Table 3.3-5 (Instrument 22). • Recognize that the valve-position-indication requirement for CV-4-200A is NOT met and <u>action 39 applies</u>. o With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, verify position by an alternate means (e.g. administrative controls, ERDADS, alternate position indication, or visual observation) within 2 hours, and restore the inoperable channel(s) within 7 days, or comply with the provisions of Specification 3.6.4 for an inoperable containment isolation valve.
Evaluator Note:	Examinee may summarize the above requirement on the Turnover Sheet or simply identify the appropriate action by number.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: Yes**	**Review Technical Specification 3.6.4, Containment Isolation Valves, and identify any required actions.**
Standard:	• Review Technical Specification 3.6.4 and recognize that, if the valve-position indications for CV-4-200A are NOT restored within 7 days, the <u>valve must be declared inoperable</u> and <u>actions a through d apply</u>. o With one or more isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either: a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic containment isolation valve secured in the isolation position, or c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
Evaluator Cue:	If examinee indicates that Technical Specification 3.6.4 does NOT apply <u>at this time</u> (i.e., within 7 days), inform examinee that repairs are expected to take more than one week.
Evaluator Note:	Examinee may summarize the above requirement on the Turnover Sheet or simply identify the appropriate actions by letter.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Terminating Cue: **When the examinee completes Step 4, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 4 is in Mode 3.
- 4-OSP-047.1E, Letdown Line Isolation Valve Test, is in progress at Step 4.3.2.11.
- When the handswitch for CV-4-200A, 45 GPM L/D Isolation Valve, was taken to CLOSE, the following indications were observed:
  - Red indicating light at CV-4-200A's handswitch – OFF
  - Green indicating light at CV-4-200A's handswitch – OFF
  - "Phase A Isolation" status light for CV-4-200A (VPB) – NOT LIT
- With the exception of notifying the Unit Supervisor and Shift Manager, no other actions have been taken.

INITIATING CUE:

- The Shift Manager directs you to review the following Technical Specifications and identify the required actions, if any (list the appropriate required actions or indicate that none are required):
  - 3.3.2, Engineered Safety Features Actuation System Instrumentation

    ____
    ____
    ____
    ____

  - 3.3.3.3, Accident Monitoring Instrumentation

    ____
    ____
    ____
    ____

  - 3.6.4, Containment Isolation Valves

    ____
    ____
    ____
    ____

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

FPL	**TURKEY POINT UNIT 4** **OPERATIONS SURVEILLANCE PROCEDURE** SAFETY RELATED **CONTINUOUS USE**	Procedure No. **4-OSP-047.1E** Revision No. **2**

Title:

LETDOWN LINE ISOLATION VALVE TEST

Responsible Department: ENGINEERING

Special Considerations:

Performance of this procedure may affect core reactivity.

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED today INITIAL S

Revision	Approved By	Approval Date
0	Steve Murano	02/22/11
2	Mike Mowbray	04/03/16

UNIT #	UNIT 4
DATE	
DOCT	PROCEDURE
DOCN	4-OSP-047.1E
SYS	
STATUS	COMPLETED
REV	2
# OF PGS	

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

1. If test results indicate a component is inoperable, component shall be removed from service and **NOT** returned to service until the condition is corrected.

2. Valves with remote position indication are required to be observed at least once every 2 years to verify that valve operation is accurately indicated. As the valve is operated, actual valve position will be compared to that indicated at the remote location. If a valve fails the Remote Position Indication Verification Test, the valve shall immediately be declared inoperable and appropriate corrective action initiated.

3. Technical Specification Table 4.3-4, Item 22 requires Containment Isolation Valve accident monitoring channels indication to be calibrated every 18 months. Reg Guide 1.97 requires this indication to distinguish between valve closed and **NOT** closed conditions. FSAR Table 6.6-3 lists the containment isolation valves. When performing this procedure to satisfy that Technical Specification, the status panel indication shall be compared to actual valve position and Control Room indication to verify accurate indication of Phase A status, including indication during valve travel (intermediate position), if possible. If the Phase A Containment Isolation indicator (white) light is found to be outside the Acceptance Criteria, Technical Specification Table 3.3-5, Item 22 contains appropriate actions.

4. If this surveillance is performed as a post-maintenance test, results shall be evaluated by the IST Coordinator to determine if new Reference Values are required per 0-ADM-502, In-Service Testing (IST) Program prior to returning the valve to service. [Section 8.2, Commitment 1]

2.2 Limitations

1. Due to the heat recovery function of the Regenerative Heat Exchanger, letdown temperature is dependent on Charging flowrate. When adding a Letdown Orifice, an equivalent increase in Charging flowrate must be made to maintain letdown temperature and prevent flashing.

2. If during Post Maintenance Testing, stroke timing is outside of the Acceptable Range, then test results are evaluated per 0-ADM-502, In-Service Testing (IST) Program, prior to returning the valve to service.

3. If any valve stroke exercise testing stroke timing is outside of the IST Acceptable Range AND less than the Required Action Range, it may be declared inoperable or immediately retested.

4. The following matrix may be used as an aide to evaluate test results:

RESULTS RANGE	**ACTIONS**	
	1st Stroke	2nd Stroke
In Acceptable Range	• Test Is SAT AND • 2nd Stroke **NOT** required	IF 1st stroke time deviation is **NOT** due to degraded valve, THEN **RECORD** deviation in Section 5.2, Documentation.
• Outside Acceptable Range AND • **NOT** in Required Action Range	• Immediately **PERFORM** 2nd stroke OR • **DECLARE** valve INOPERABLE	**GENERATE** 3-day OPERABILITY CR to determine if: • Valve operation is acceptable. • New reference stroke time may be established from this test.
In Required Action Range	**DECLARE** valve INOPERABLE	

3.0 PREREQUISITES

- **CHECK** Charging and Letdown System aligned per 4-OP-047, CVCS - Charging and Letdown. S

- IF Remote Position Indication Verification is required, THEN **ENSURE** an observer is available. S

- **OBTAIN** the following M&TE and **RECORD** M&TE tag number and calibration due dates: S

Instrument	Equipment No.	Cal Due Date
Stopwatch	X X	current

- **CHECK** M&TE calibration due date current. S

- **OBTAIN** permission from the Shift Manager to perform this test. S

End of Section 3.0

INITIAL

4.0 INSTRUCTIONS

4.1 General Requirements

1. IF during performance of this procedure any Acceptance Criteria is UNSAT, THEN Immediately **NOTIFY** Unit Supervisor that component is INOPERABLE. ______

2. IF during performance of this procedure any of the following occur:

   - Acceptance Criteria is UNSAT
   - A malfunction occurs
   - An abnormal condition is found

   THEN:

   - Immediately **NOTIFY** Unit Supervisor. ______
   - **DOCUMENT** condition in Section 5.1, Step 1, ENSURE Independent Verifications complete. ______ . ______
   - **NOTIFY** the IST Coordinator or designee. ______

3. IF in MODE 4, 5 OR 6, THEN **GO TO** any of the following:

N/A

   - Section 4.4, Letdown Line Isolation Valves Test (Quarterly - Mode 4, 5, and 6)
   - Section 4.5, Letdown Line Isolation Valves Exercise and Remote Position Indication 2-Year Test (Mode 4, 5, and 6)

End of Section 4.1

4.3 Letdown Line Isolation Valves Exercise and Remote Position Indication (2-Year 2-Pump 2-Orifice)

4.3.1 Test Preparations

1. **NOTIFY** RP of intended changes in letdown flow. S
2. **POSITION** TCV-4-143, L/D DEMINERALIZER DIVERT VALVE, in VCT DIVERT. S
3. **STATION** an observer in the vicinity of the valves in communications with Control Room. S
4. IF at any time Pressurizer level needs to be restored, THEN **RESTORE** level per 4-OP-047, CVCS - Charging and Letdown. S
5. IF less than two Charging Pumps are in operation, THEN **PERFORM** the following per 4-OP-047, CVCS - Charging and Letdown:
   - **START** an additional Charging Pump. S
   - **PLACE** an additional Letdown Orifice in operation. S
6. **ENSURE** CONTAINMENT ISOLATION VALVE POSITION display is selected on DCS. S
7. **PERFORM** any of the following sections first based on initial Letdown Orifice valve positions:
   - Section 4.3.2, CV-4-200A, 45 GPM L/D Isolation Valve Test (2-Year 2-Pump 2-Orifice) _____
   - Section 4.3.3, CV-4-200B, 60 GPM L/D Isolation Valve Test (2-Year 2-Pump 2-Orifice) _____
   - Section 4.3.4, CV-4-200C, 60 GPM L/D Isolation Valve Test (2-Year 2-Pump 2-Orifice) _____

End of Section 4.3.1

4.3.2 CV-4-200A, 45 GPM L/D Isolation Valve Test (2-Year 2-Pump 2-Orifice)

1. **ENSURE** CV-4-200A, 45 GPM L/D ISOLATION VALVE, OPEN. S

2. **REQUEST** observer locally check CV-4-200A, LTDN ORIFICE STOP VALVE-45 GPM, OPEN. S

3. **OBSERVE** the following for OPEN position indication of CV-4-200A, 45 GPM L/D ISOLATION VALVE: S

Location	CV-4-200A Position Indicators	Expected OPEN Indication
C03	CV-4-200A, 45 GPM L/D ISOLATION VALVE handswitch	red light ON
		green light OFF
VPB	L/D ORIFICE CV-4-200A, PHASE "A" ISOLATION status light	DIM
DCS	CV200A, on CONTAINMENT ISOLATION VALVE POSITION screen	100

4. **COMPARE** results to Acceptance Criteria and **DOCUMENT**: S

BC
IV

CV-4-200A Position Indicator Acceptance Criteria	Results
L/D ORIFICE CV-4-200A, PHASE "A" ISOLATION status light and CV-4-200A, 45 GPM L/D ISOLATION VALVE handswitch lights and local position, all indicate OPEN	☒ SAT ☐ UNSAT

5. **COMPARE** (DCS) OPEN position indication results to Functional Criteria and **DOCUMENT**: S

CV-4-200A DCS Position Indicator Functional Criteria	Results
CV200A, on DCS CONTAINMENT ISOLATION VALVE POSITION screen indicates OPEN	☒ SAT ☐ UNSAT

6. **ENSURE** PCV-4-145, LOW PRESSURE LTDN CONTROLLER, in MAN. S

7. **ADJUST** PCV-4-145, LOW PRESSURE LTDN CONTROLLER in MAN to raise Letdown pressure to 300 psig as indicated on PI-4-145, LOW PRESS LTDN PRESS. S

4.3.2 CV-4-200A, 45 GPM L/D Isolation Valve Test (2-Year 2-Pump 2-Orifice) (continued)

8. **ENSURE** CV-4-200A, 45 GPM L/D ISOLATION VALVE is OPEN for at least 3 minutes. S

9. **ENSURE** LC-4-459G, MASTER CHARGING PUMP CONTROLLER, in MAN. S

10. **REQUEST** observer monitor CV-4-200A, LTDN ORIFICE STOP VALVE-45 GPM for abnormalities in valve motion or operation, such as excessive vibration, jerky motion, binding or unusual noise. S

NOTE

Section 4.3.2, Step 11 should be performed using two operators, one to perform valve test requirements and the other to perform any PCV-4-145 adjustments.

11. Simultaneously **PERFORM** the following: ______

- **PLACE** CV-4-200A, 45 GPM L/D ISOLATION VALVE, in CLOSE.
- **START** stopwatch.
- **ADJUST** PCV-4-145, LOW PRESSURE LTDN CONTROLLER, in MAN, to maintain Letdown pressure greater than 150 psig, as indicated on PI-4-145, LOW PRESS LTDN PRESS.
- **OBSERVE** the following for INTERMEDIATE position indication of CV-4-200A, 45 GPM L/D ISOLATION VALVE:

Location	CV-4-200A Position Indicator	Expected INTERMEDIATE Indication
C03	CV-4-200A, 45 GPM L/D ISOLATION VALVE handswitch	red light ON
		green light ON
VPB	L/D ORIFICE CV-4-200A, PHASE “A” ISOLATION status light	DIM
DCS	CV200A, on CONTAINMENT ISOLATION VALVE POSITION screen	between 0 and 100

L-16-1 NRC Exam

Admin - JPM SRO A3

JPM TITLE: Authorize Emergency Exposure Limits

JPM NUMBER: 02200010501 **REV.** 0-1

TASK NUMBER(S) / TASK TITLE(S): 02200010500/ Direct Emergency Response as the Emergency Coordinator

K/A NUMBERS: 2.3.4 **K/A VALUE:** RO 3.2 / SRO 3.7

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☐ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION:

In-Plant: [ ] Control Room: [ ]

Simulator: [ ] Other: [X]

Lab: [ ]

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☐	☐	☒
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☐	☐	☒
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered "YES" or "N/A" or the JPM is not valid for use. If all questions/statements are answered "YES" or "N/A," then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

Protected Content: (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
0	New JPM			N/A	N/A
				N/A	N/A
0-1	Incorporate procedural changes; formatting; text/grammar changes		N/A	Schilk	
				Wilson	

SIMULATOR SET-UP:

- N/A

Required Materials:

- 0-EPIP-20101, Duties of the Emergency Coordinator
- 0-EPIP-20111,Re-Entry
- Technical Specifications

General References:

- 0-EPIP-20101, Duties of the Emergency Coordinator
- 0-EPIP-20111,Re-Entry
- Technical Specifications

Task Standards:

- Given a set of plant conditions and available personnel, select appropriate individuals to serve as rescue team members and elect to issue them potassium iodide

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Operators at Unit 3 were performing a plant shutdown due to elevated RCS activity (failed fuel), when a LOCA occurred outside of containment.
- A General Emergency was declared and an Owner Controlled Area evacuation was initiated in accordance with Section 5.8 of 0-EPIP-20101, Duties of the Emergency Coordinator.
- An RP technician reports that there is an unconscious operator in the Pipe and Valve Room with a life-threatening injury; the rescue will require two individuals and is expected to take 10 minutes.
- Local dose rates in the Pipe and Valve Room are >40 rem/hour.
- The TSC has NOT been activated.

INITIATING CUE:

- As the Emergency Coordinator, you are required to select two individuals from the list of available Rescue Team members and determine whether they should be issued potassium iodide.
  - Selected Rescue Team members: ____
  - Issue potassium iodide? Yes __________ No __________

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	• Obtain copies of 0-EPIP-20111, Re-Entry, and 0-EPIP-20101, Duties of the Emergency Coordinator. • Obtain the list of available Rescue Team members.
Evaluator Cue:	Provide examinee with copies of 0-EPIP-20101, Duties of the Emergency Coordinator; 0-EPIP-20111, Re-Entry; and the list of available Rescue Team members.
Performance:	**SATISFACTORY _____ UNSATISFACTORY _____**
Comments:	

Performance Step: 2 **Critical: No**	**Determine whether the rescuers could receive radiation exposures in excess of the regulatory limit (5 rem/year).**
Standard:	Recognize that the rescuers will likely receive radiation exposures in excess of the regulatory limit.
Evaluator Note:	(40 rem/hour)(10 minutes)(1 hour/60 minutes) = <u>6.7 rem (per individual)</u> <u>From 0-EPIP-20101, Step 3.1.4</u>: The Emergency Coordinator shall authorize any radiation exposures in excess of regulatory limits. This authorization should be in accordance with 0-EPIP-20111, Re-Entry.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: Yes**	**Exclude individuals as Rescue Team members, based on volunteer and declared-pregnancy status.**
Standard:	• Exclude member #1, who has chosen NOT to volunteer. • Exclude member #5, who is a declared pregnant adult.
Evaluator Note:	• **This and the following steps may be performed in any order.** • From 0-EPIP-20101, Step 3.1.5: The Emergency Coordinator shall authorize personnel exposures in excess of regulatory limits only for volunteers who are familiar with the risks involved and the tasks to be performed. Declared pregnant adults should not be used as on-site emergency workers. • From 0-EPIP-20111, Step 5.1.1.1: Re-entry personnel that have been authorized to exceed regulatory exposure limits should be volunteers, familiar with the risks involved (radiosensitivity of fetuses, effects of acute exposures, etc.), and whose normal duties have trained them for such missions. • From 0-EPIP-20111, Step 5.1.1.2: Declared pregnant adults should not be used as on-site emergency workers.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: Yes**	**Exclude individuals as Rescue Team members, based on age/youth.**
Standard:	Exclude members #3 and #7, who are NOT over 45 years old.
Evaluator Note:	From 0-EPIP-20111, Enclosure 1, Page 1: Rescue of persons from a life-threatening situation (Volunteers should be above the age of 45).
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 5 **Critical: Yes**	**Determine that the rescuers could receive thyroid exposures as high as 5 rem and exclude individuals as Rescue Team members, based on iodine sensitivity.**
Standard:	• Recognize that the rescuers could receive thyroid exposures (CDEs) of ≥5 rem, which warrants the issuance of potassium iodide. • Exclude member #8, who has an iodide/iodine sensitivity.
Evaluator Note:	• From 0-EPIP-20101, Step 3.1.6, and 0-EPIP-20111, Enclosure 1, Item 1: The Emergency Coordinator shall authorize the issuance of Potassium Iodide (KI) to emergency workers…based on a thyroid CDE of greater than or equal to 5 rem actual or estimated. • From 0-EPIP-20111, Enclosure 1, Item 9: Caution emergency response personnel of potential KI side effects if they are allergic to iodide. Emergency response personnel who know they have such allergies should be replaced in lieu of directing them to ingest KI. KI should **NOT** be given to individuals with known iodine sensitivity.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 6 **Critical: Yes**	**Exclude individuals as Rescue Team members, based on staffing requirements.**
Standard:	Exclude member #6, based on the Technical Specification requirements for minimum shift crew composition.
Evaluator Cue:	If requested, provide examinee with a copy of the Technical Specifications.
Evaluator Note:	From Technical Specifications, Table 6.2-1: An STA is required to meet the minimum shift crew composition, as long as at least one unit is in Mode 1, 2, 3, or 4.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Performance Step: 7 **Critical: Yes**	**Identify selected Rescue Team members and the need to issue potassium iodide to same.**
Standard:	• Identify team members #2 and #4 for rescue efforts. • Recognize the need to issue potassium iodide to the rescuers.
Evaluator Note:	From 0-EPIP-20101, Step 3.1.6, and 0-EPIP-20111, Enclosure 1, Item 1: The Emergency Coordinator shall authorize the issuance of Potassium Iodide (KI) to emergency workers…based on a thyroid CDE of greater than or equal to 5 rem actual or estimated.
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Terminating Cue: **When the examinee completes Step 7, state "This completes the JPM."**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: __________________________ **Evaluator:** ____________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ____________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ____________________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Operators at Unit 3 were performing a plant shutdown due to elevated RCS activity (failed fuel), when a LOCA occurred outside of containment.
- A General Emergency was declared and an Owner Controlled Area evacuation was initiated in accordance with Section 5.8 of 0-EPIP-20101, Duties of the Emergency Coordinator.
- An RP technician reports that there is an unconscious operator in the Pipe and Valve Room with a life-threatening injury; the rescue will require two individuals and is expected to take 10 minutes.
- Local dose rates in the Pipe and Valve Room are >40 rem/hour.
- The TSC has NOT been activated.

INITIATING CUE:

- As the Emergency Coordinator, you are required to select two individuals from the list of available Rescue Team members and determine whether they should be issued potassium iodide.
  - Selected Rescue Team members: ____
  - Issue potassium iodide? Yes __________ No __________

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

TURNOVER SHEET ADDENDUM

List of Available Rescue Team Members

1. Non-Licensed Operator, 49-year-old male, fair physical condition, prefers not to volunteer, lifetime exposure of 5.2 rem.
2. Reactor Engineer, 48-year-old female, good physical condition, volunteer, lifetime exposure of 400 mrem.
3. Maintenance Technician, 43-year-old male, fair physical condition, volunteer, lifetime exposure of 35.4 rem.
4. Licensed Operator (no current Control Room duties), 52-year-old male, fair physical condition, volunteer, lifetime exposure of 1.4 rem.
5. Security Supervisor, 32-year-old female, good physical condition, declared pregnant adult, volunteer, lifetime exposure of 65 mrem.
6. Shift Technical Advisor (the only qualified STA on shift), 46-year-old female, good physical condition, volunteer, lifetime exposure of 120 mrem.
7. Licensed Operator (current BOP operator), 34-year-old male, good physical condition, volunteer, lifetime exposure of 1.7 rem.
8. Security Officer, 47-year-old male, good physical condition, iodine sensitivity, volunteer, lifetime exposure of 287 mrem.

NOTE: Ensure the addendum sheet is returned to the evaluator when the JPM is complete.

DRAFT L-16-1 NRC EXAM SECURE INFORMATION

L-16-1 NRC Exam

Admin - JPM SRO A4

JPM TITLE: Classify Event and Fill Out SNF

JPM NUMBER: 02201052320 **REV.** 1-0

TASK NUMBER(S) / TASK TITLE(S): 02201052300/ Classify Significant Events

K/A NUMBERS: 2.4.41 **K/A VALUE:** RO 2.9 / SRO 4.6

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☐ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [ ] Other: [X]
Lab: [ ]

Time for Completion: 30 Minutes Time Critical: Yes

Alternate Path [NRC]: No
Alternate Path [INPO]: No

Developed by:	**Alan Schilk** Instructor/Developer	**6/22/16** Date
Reviewed by:	**Luis Sagion** Instructor (Instructional Review)	**6/22/16** Date
Validated by:	**Rocky Schoenhals** SME (Technical Review)	**6/22/16** Date
Approved by:	**Mark Wilson** Training Supervision	**6/22/16** Date
Approved by:	**Rocky Schoenhals** Training Program Owner	**6/22/16** Date

JOB PERFORMANCE MEASURE VALIDATION CHECKLIST

ALL STEPS IN THIS CHECKLIST ARE TO BE PERFORMED PRIOR TO USE.

REVIEW STATEMENTS	YES	NO	N/A
1. Are all items on the signature page filled in correctly?	☒	☐	☐
2. Has the JPM been reviewed and validated by SMEs?	☒	☐	☐
3. Can the required conditions for the JPM be appropriately established in the simulator if required?	☒	☐	☐
4. Do the performance steps accurately reflect trainee's actions in accordance with plant procedures?	☒	☐	☐
5. Is the standard for each performance item specific as to what controls, indications and ranges are required to evaluate if the trainee properly performed the step?	☒	☐	☐
6. Has the completion time been established based on validation data or incumbent experience?	☒	☐	☐
7. If the task is time critical, is the time critical portion based upon actual task performance requirements?	☒	☐	☐
8. Is the job level appropriate for the task being evaluated if required?	☒	☐	☐
9. Is the K/A appropriate to the task and to the licensee level if required?	☒	☐	☐
10. Is justification provided for tasks with K/A values less than 3.0?	☐	☐	☒
11. Have the performance steps been identified and classified (Critical / Sequence / Time Critical) appropriately?	☒	☐	☐
12. Have all special tools and equipment needed to perform the task been identified and made available to the trainee?	☐	☐	☒
13. Are all references identified, current, accurate, and available to the trainee?	☒	☐	☐
14. Have all required cues (as anticipated) been identified for the evaluator to assist task completion?	☒	☐	☐
15. Are all critical steps supported by procedural guidance? (e.g., if licensing, EP or other groups were needed to determine correct actions, then the answer should be NO.)	☒	☐	☐
16. If the JPM is to be administered to an LOIT student, has the required knowledge been taught to the individual prior to administering the JPM? TPE does not have to be completed, but the JPM evaluation may not be valid if they have not been taught the required knowledge.	☒	☐	☐

All questions/statements must be answered “YES” or “N/A” or the JPM is not valid for use. If all questions/statements are answered “YES” or “N/A,” then the JPM is considered valid and can be performed as written. The individual(s) performing the initial validation shall sign and date the cover sheet.

<u>Protected Content:</u> (CAPRs, corrective actions, licensing commitments, etc. associated with this material)

N/A

UPDATE LOG: Indicate in the following table any minor changes or major revisions (as defined in TR-AA-230-1003) made to the material after initial approval. Or use separate Update Log form TR-AA-230-1003-F16.

#	DESCRIPTION OF CHANGE	REASON FOR CHANGE	AR/TWR#	PREPARER / SUPERVISOR	DATE / DATE
0	New JPM		N/A	N/A	N/A
				N/A	N/A
1-0	Formatting; text/grammar changes		N/A	Schilk	
				Wilson	

SIMULATOR SET-UP:

- N/A

Required Materials:

- 0-EPIP-20101, Duties of Emergency Coordinator
- 0-EPIP-20134, Offsite Notifications and Protective Action Recommendations
- F439, Florida Nuclear Plant Emergency Notification Form
- F444, Guidance For Determining Protective Action Recommendations (PARS)
- F668, Turkey Point EAL Classification Tables (Hot)
- F669, Turkey Point EAL Classification Tables (Cold)

General References:

- 0-EPIP-20101, Duties of Emergency Coordinator
- 0-EPIP-20134, Offsite Notifications and Protective Action Recommendations

Task Standards:

- Within 15 minutes, declare an Alert (CA3)
- Within 15 minutes of declaration, complete a Florida Nuclear Plant Emergency Notification Form per 0-EPIP-20134 (Offsite Notifications and Protective Action Recommendations) with no errors on required items that are marked with an asterisk (with the exception of Item 2B)

HAND JPM BRIEFING SHEET TO EXAMINEE AT THIS TIME

I will explain the initial conditions, which step(s) to simulate or discuss, and provide initiating cues. When you complete the task successfully, the objective for this job performance measure will be satisfied.

DURING THE JPM, ENSURE PROPER SAFETY PRECAUTIONS, FME, AND/OR RADIOLOGICAL CONCERNS AS APPLICABLE ARE FOLLOWED.

INITIAL CONDITIONS:

- Unit 4 is at 100% power.
- Unit 3 is shut down, with an RCS temperature of 185°F.
- Wind speed is 10 mph and wind direction is from 180 degrees.

SUBSEQUENT CONDITIONS (UNIT 3):

- At 0800, Unit 3 experienced a loss of the Startup Transformer and the crew entered 3-ONOP-004, Loss of Offsite Power.
- At 0810, both the 3A and 3B EDGs tripped and repair teams began investigating.
- At 0825, no power is available from any source and RCS temperature is 195°F and slowly rising.

INITIATING CUES:

- You are the Emergency Coordinator in the Control Room. Based on the subsequent conditions, classify the event and, if necessary, determine protective action recommendations (do NOT use EC judgment).
- By raising your hand, you signify that you have completed the event declaration. At that time, inform the Examiner of the event classification and the Examiner will provide you with a Florida Nuclear Plant Emergency Notification Form, which you will complete.
- When you have completed the Florida Nuclear Plant Emergency Notification Form, raise your hand to inform the Examiner that you are done.
- There are elements of this task that are Time Critical.

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

JPM PERFORMANCE INFORMATION

Start Time: ___________

NOTE: When providing "Evaluator Cues" to the examinee, care must be exercised to avoid prompting the examinee. Typically, cues are only provided when the examinee's actions warrant receiving the information (i.e., the examinee looks or asks for the indication).

NOTE: Critical steps are marked with a "Yes" below the performance step number. Failure to meet the standard for any critical step shall result in failure of this JPM.

Performance Step: 1 **Critical: No**	**Obtain required reference materials.**
Standard:	Obtain the following references: • 0-EPIP-20101, Duties of Emergency Coordinator • 0-EPIP-20134, Offsite Notifications and Protective Action Recommendations • F668, Turkey Point EAL Classification Tables (Hot) • F669, Turkey Point EAL Classification Tables (Cold)
Evaluator Cue:	Provide examinee with the following: • 0-EPIP-20101, Duties of Emergency Coordinator • 0-EPIP-20134, Offsite Notifications and Protective Action Recommendations • F668, Turkey Point EAL Classification Tables (Hot) • F669, Turkey Point EAL Classification Tables (Cold)
Performance:	**SATISFACTORY** ______ **UNSATISFACTORY** _____
Comments:	

Performance Step: 2 **Critical: No**	**Review Turkey Point EAL Classification Tables (F668 and F669) and 0-EPIP-20101 (Duties of Emergency Coordinator), based on the given conditions.**
Standard:	Review F668, F669, and 0-EPIP-20101, based on the given conditions.
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 3 **Critical: Yes**	**Determine the highest emergency classification level and EAL number using F669 (Turkey Point EAL Classification Tables [Cold]).**
Standard:	• Determine that the highest emergency classification level is an Alert and the EAL number is CA3. • Examinee raises his/her hand, within 15 minutes, to signify completion of the event declaration.
Evaluator Note:	• Log event declaration time and classification: ____________________ • Declaration time is the start time for completion of the Florida Nuclear Plant Emergency Notification Form.
Evaluator Cue:	Upon receiving the event declaration, provide examinee with the following: • F439, Florida Nuclear Plant Emergency Notification Form • F444, Guidance For Determining Protective Action Recommendations (PARS)
Performance:	**SATISFACTORY ______ UNSATISFACTORY ______**
Comments:	

Performance Step: 4 **Critical: Yes**	**Complete a Florida Nuclear Plant Emergency Notification Form (F439), in accordance with 0-EPIP-20134 (Offsite Notifications and Protective Action Recommendations).**
Standard:	Within 15 minutes of event declaration, a Florida Nuclear Plant Emergency Notification Form (F439) is completed, in accordance 0-EPIP-20134 (Offsite Notifications and Protective Action Recommendations), with no errors on required items identified with an asterisk. • In Item 1 of F439, examinee is allowed to select either “This is A Drill” or “This is An Actual Event.” • Item 2B is N/A, until offsite agencies are contacted.
Evaluator Note:	Log form completion time: _____________
Performance:	**SATISFACTORY ______ UNSATISFACTORY _____**
Comments:	

Terminating Cue: **When the examinee submits the Florida Nuclear Plant Emergency Notification Form (F439), state “This completes the JPM.”**

NOTE: Ensure the turnover sheet that was given to the examinee is returned to the evaluator.

Stop Time: ___________

Examinee: ______________________ **Evaluator:** ______________________

☐ **RO** ☐ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** **Date:** ______________________

☐ **LOIT RO** ☐ **LOIT SRO**

PERFORMANCE RESULTS: **SAT:** ☐ **UNSAT:** ☐

Remediation required: **YES** ☐ **NO** ☐

COMMENTS/FEEDBACK: (Comments shall be made for any steps graded unsatisfactory).

EXAMINER NOTE: ENSURE ALL EXAM MATERIAL IS COLLECTED AND PROCEDURES CLEANED, AS APPROPRIATE.

EVALUATOR'S SIGNATURE: ______________________________

NOTE: Only this page needs to be retained in examinee's record if completed satisfactorily. If unsatisfactory performance is demonstrated, the entire JPM should be retained.

TURNOVER SHEET

INITIAL CONDITIONS:

- Unit 4 is at 100% power.
- Unit 3 is shut down, with an RCS temperature of 185°F.
- Wind speed is 10 mph and wind direction is from 180 degrees.

SUBSEQUENT CONDITIONS (UNIT 3):

- At 0800, Unit 3 experienced a loss of the Startup Transformer and the crew entered 3-ONOP-004, Loss of Offsite Power.
- At 0810, both the 3A and 3B EDGs tripped and repair teams began investigating.
- At 0825, no power is available from any source and RCS temperature is 195°F and slowly rising.

INITIATING CUES:

- You are the Emergency Coordinator in the Control Room. Based on the subsequent conditions, classify the event and, if necessary, determine protective action recommendations (do NOT use EC judgment).
- By raising your hand, you signify that you have completed the event declaration. At that time, inform the Examiner of the event classification and the Examiner will provide you with a Florida Nuclear Plant Emergency Notification Form, which you will complete.
- When you have completed the Florida Nuclear Plant Emergency Notification Form, raise your hand to inform the Examiner that you are done.
- There are elements of this task that are Time Critical.

NOTE: Ensure the turnover sheet is returned to the evaluator when the JPM is complete.

JPM TITLE: Calculate a Manual Makeup to the VCT

JPM NUMBER: 01046046101 **REV.** 1-1

TASK NUMBER(S) / TASK TITLE(S): 01046046100/
Calculate a Manual Makeup to the VCT

K/A NUMBERS: 2.1.25 **K/A VALUE:** RO 3.9 / SRO 4.2

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: ☒

EVALUATION LOCATION:
In-Plant: ☐ Control Room: ☐
Simulator: ☐ Other: ☒
Lab: ☐

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Instructor/Developer	6/22/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by:	R. Schoenlub SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	R. Schoenlub Training Program Owner	06/22/16 Date

JPM TITLE: Review an ECO for the B AFW Pump

JPM NUMBER: 01201013103 **REV.** 1-1

TASK NUMBER(S) / TASK TITLE(S): 01201013100/ Write Equipment Clearance Orders

K/A NUMBERS: 2.2.13 **K/A VALUE:** RO 4.1 / SRO 4.3

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: X

EVALUATION LOCATION:
In-Plant: ☐ Control Room: ☐
Simulator: ☐ Other: X
Lab: ☐

Time for Completion: 35 Minutes Time Critical: No

Alternate Path [NRC]: No
Alternate Path [INPO]: No

Developed by:	[signature] Instructor/Developer	6/22/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/22/16 Date
Validated by:	[signature] SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	[signature] Training Program Owner	06/22/16 Date

JPM TITLE:	Evaluate Conditions for Restart of Refueling Pre-shuffle in the Spent Fuel Pit
JPM NUMBER:	01038034100 **REV.** 0-1
TASK NUMBER(S) / TASK TITLE(S):	01038034100/ Evaluate Conditions for Restart of Refueling Pre-shuffle in the Spent Fuel Pit
K/A NUMBERS:	2.3.12 **K/A VALUE:** RO 3.2 / SRO 3.7

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: X

EVALUATION LOCATION:

In-Plant:		Control Room:	
Simulator:		Other:	X
Lab:			

Time for Completion: 20 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Instructor/Developer	6/22/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by:	SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	Training Program Owner	06/22/16 Date

JPM TITLE: Calculate a Manual Makeup to the VCT

JPM NUMBER: 01046046101 **REV.** 1-1

TASK NUMBER(S) / TASK TITLE(S): 01046046100/ Calculate a Manual Makeup to the VCT

K/A NUMBERS: 2.1.25 **K/A VALUE:** RO 3.9 / SRO 4.2

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ RO ☒ SRO ☐ STA ☐ Non-Lic ☐ SRO CERT ☐ OTHER: _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: ☒

EVALUATION LOCATION:

In-Plant:	☐	Control Room:	☐
Simulator:	☐	Other:	☒
Lab:	☐		

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Instructor/Developer	6/22/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by:	SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	Training Program Owner	06/22/16 Date

JPM TITLE: Determine Heatup of the RCS

JPM NUMBER: 01041046101 **REV.** 0-1

TASK NUMBER(S) / TASK TITLE(S): 01041046100/ Determine Heatup of the RCS

K/A NUMBERS: 2.1.20 **K/A VALUE:** RO 4.6 / SRO 4.6

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: X

EVALUATION LOCATION:

In-Plant:	☐	Control Room:	☐
Simulator:	☐	Other:	X
Lab:	☐		

Time for Completion: 30 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	[signature] Instructor/Developer	6/22/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/22/16 Date
Validated by:	R. Schoenlink SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R. Schoenlink Training Program Owner	06/22/16 Date

JPM TITLE: Evaluate Technical Specification Conditions While Performing a Valve Operability Test

JPM NUMBER: 02051013101 **REV.** 1-2

TASK NUMBER(S) / TASK TITLE(S): 02051013100 / Verify/Set Containment Integrity

K/A NUMBERS: 2.2.40 **K/A VALUE:** RO 3.4 / SRO 4.7

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☐ RO ☒ SRO ☐ STA ☐ Non-Lic ☐ SRO CERT ☐ OTHER: _____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION:

In-Plant:	[ ]	Control Room:	[ ]
Simulator:	[ ]	Other:	[X]
Lab:	[ ]		

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Instructor/Developer	6/22/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by:	SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	Training Program Owner	06/22/16 Date

JPM TITLE: Authorize Emergency Exposure Limits

JPM NUMBER: 02200010501 **REV.** 0-1

TASK NUMBER(S) / TASK TITLE(S): 02200010500/
Direct Emergency Response as the Emergency Coordinator

K/A NUMBERS: 2.3.4 **K/A VALUE:** RO 3.2 / SRO 3.7

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☐ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: ☐ Perform: X

EVALUATION LOCATION:

In-Plant: ☐ Control Room: ☐

Simulator: ☐ Other: X

Lab: ☐

Time for Completion: 15 Minutes Time Critical: No

Alternate Path [NRC]: No

Alternate Path [INPO]: No

Developed by:	Instructor/Developer	6/22/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by:	SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	Training Program Owner	06/22/16 Date

JPM TITLE: Classify Event and Fill Out SNF

JPM NUMBER: 02201052320 **REV.** 1-0

TASK NUMBER(S) / TASK TITLE(S): 02201052300/ Classify Significant Events

K/A NUMBERS: 2.4.41 **K/A VALUE:** RO 2.9 / SRO 4.6

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☐ RO ☒ SRO ☐ STA ☐ Non-Lic ☐ SRO CERT ☐ OTHER: ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [ ] Other: [X]
Lab: [ ]

Time for Completion: 30 Minutes Time Critical: Yes

Alternate Path [NRC]: No
Alternate Path [INPO]: No

Developed by:	[signature] Instructor/Developer	6/22/16 Date
Reviewed by:	[signature] Instructor (Instructional Review)	6/22/16 Date
Validated by:	R Schoenfeld [signature] SME (Technical Review)	06/22/16 Date
Approved by:	[signature] Training Supervision	6/22/16 Date
Approved by:	R Schoenfeld [signature] Training Program Owner	06/22/16 Date

JPM TITLE: Determine Heatup of the RCS

JPM NUMBER: 01041046101 **REV.** 0-1

TASK NUMBER(S) / TASK TITLE(S): 01041046100/ Determine Heatup of the RCS

K/A NUMBERS: 2.1.20 **K/A VALUE:** RO 4.6 / SRO 4.6

Justification (FOR K/A VALUES <3.0): N/A

TASK APPLICABILITY:
☒ **RO** ☒ **SRO** ☐ **STA** ☐ **Non-Lic** ☐ **SRO CERT** ☐ **OTHER:** ____

APPLICABLE METHOD OF TESTING: Simulate/Walkthrough: [ ] Perform: [X]

EVALUATION LOCATION: In-Plant: [ ] Control Room: [ ]
Simulator: [ ] Other: [X]
Lab: [ ]

Time for Completion: 30 Minutes Time Critical: No

Alternate Path [NRC]: No
Alternate Path [INPO]: No

Developed by:	Instructor/Developer	6/22/16 Date
Reviewed by:	Instructor (Instructional Review)	6/22/16 Date
Validated by:	R. Schoenlank SME (Technical Review)	06/22/16 Date
Approved by:	Training Supervision	6/22/16 Date
Approved by:	R. Schoenlank Training Program Owner	06/22/16 Date