1Facility: Date of Exam:

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 and Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18	3	3	6
	2	1	1	2				2	2			1	9	2	2	4
	Tier Totals	4	4	5				5	5			4	27	5	5	10
2. Plant Systems	1	2	3	3	3	2	2	3	3	3	2	2	28	3	2	5
	2	1	1	1	1	1	1	1	1	0	1	1	10	0 2	1	3
	Tier Totals	3	4	4	4	3	3	4	4	3	3	3	38	5	3	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2	3 4	7
					2		3		2		3			2 2	2 1X	

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it 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 outline. Systems or evolutions that do not apply at the facility should be deleted with justification. Operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible. Sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7. The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' IRs for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. If fuel-handling equipment is sampled in a category other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2. (Note 1 does not apply). Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

G* Generic K/As

* These systems/evolutions must be included as part of the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan. They are not required to be included when using earlier revisions of the K/A catalog.

** These systems/evolutions may be eliminated from the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan.

ES-401 PWR Examination Outline Form ES-401-2 Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/SRO)									
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1	X						007EK1.02: Shutdown margin	3.4	1
000008 (APE 8) Pressurizer Vapor Space Accident / 3		X					008AK2.02: Sensors and detectors	2.7	2
000009 (EPE 9) Small Break LOCA / 3	X						009EK1 .02: Use of steam tables	3.5	3
000011 (EPE 11) Large Break LOCA / 3						X	011EG2.4.21: Knowledge of the parameters and logic used to assess the status of safety functions	4.0	4
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4						X	015AG2.2.42 : Ability to recognize system parameters that are entry level conditions for Technical Specifications	3.9	5
000022 (APE 22) Loss of Reactor Coolant Makeup / 2			X				022AK3.06: RCP thermal barrier cooling	3.2	6
000025 (APE 25) Loss of Residual Heat Removal System / 4		X					025AK2.01: RHR heat exchangers	2.9	7
000026 (APE 26) Loss of Component Cooling Water / 8				X			026AA1.04: CRDM high-temperature alarm system	2.7	8
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3						X	027AG2.2.4: (multi-unit) Ability to explain the variations in control board layouts, systems, instrumentation and procedural actions between units at a facility.	3.6	76
000029 (EPE 29) Anticipated Transient Without Scram / 1				X			029EA1.11: Manual opening of the CRDS breakers	3.9	9
000038 (EPE 38) Steam Generator Tube Rupture / 3					X		038EA2.17: RCP restart criteria	4.4	77
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4					X		040AA2.02: Conditions requiring a reactor trip	4.6	10
000054 (APE 54; CE E06) Loss of Main Feedwater /4	X						054AK1.01: MFW line break depressurizes the SIG (similar to a steam line break)	4.1	11
000055 (EPE 55) Station Blackout / 6					X		055EA2.03: Actions necessary to restore power	4.7	78
000056 (APE 56) Loss of Offsite Power / 6						X	056AG2.1.30	4.4	12
000057 (APE 57) Loss of Vital AC Instrument Bus / 6						X	057AG2.2.44: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions	4.2	79
000058 (APE 58) Loss of DC Power / 6						X	058AG2.4.41: Knowledge of the emergency action level thresholds and classifications	2.9	80
000062 (APE 62) Loss of Nuclear Service Water / 4					X		062AA2.01: Location of a leak in the SWS	2.9	13
000065 (APE 65) Loss of Instrument Air / 8			X				065AK3.04: Cross-over to backup air supplies	3.0	14
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6				X			077 AA1.03: Voltage regulator controls	3.8	15
(W E04) LOCA Outside Containment / 3					X		WE04EA2.2: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.6	16

(W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4		X					WE05EK2.2: 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	3.9	17
(W E11) Loss of Emergency Coolant Recirculation / 4			X		X		WE11EK3.4 : Loss of Emergency Coolant Recrc./4 WET1EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	3.6 4.2	18 81
K/A Category Totals:	3	3	3	3	3/3	3/3	Group Point Total:		18/6

ES-401 PWR Examination Outline Form ES-401-2 Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/SRO)									
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1							N/A		
000003 (APE 3) Dropped Control Rod / 1							N/A		
000005 (APE 5) Inoperable/Stuck Control Rod / 1						X	005AG2.2.12:	4.1	82
000024 (APE 24) Emergency Boration / 1							N/A		
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2					X		028AA2.11: Leak in PZR	3.6	83
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7							N/A		
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7						X	033AG2.4.46: Ability to verify that the alarms are consistent with the plant conditions.	4.2	84
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8	X						036AK1.03: Indications of approaching criticality	4.0	19
000037 (APE 37) Steam Generator Tube Leak / 3				X			037AA1.07: CVCS letdown flow indicator	3.1	20
000051 (APE 51) Loss of Condenser Vacuum / 4			X				051AK3.01: Loss of steam dump capability upon loss of condenser vacuum	2.8	21
000059 (APE 59) Accidental Liquid Radwaste Release / 9					X		059AA2.03: Failure modes, their symptoms and the causes of misleading indications on a radioactive-liquid monitor	3.1	22
000060 (APE 60) Accidental Gaseous Radwaste Release / 9							N/A		
000061 (APE 61) Area Radiation Monitoring System Alarms / 7							N/A		
000067 (APE 67) Plant Fire On Site / 8					X		067AA2.07: Whether malfunction is due to common-mode electrical failures	3.1	85
000068 (APE 68; BW A06) Control Room Evacuation / 8							N/A		
000069 (APE 69; W E14) Loss of Containment Integrity / 5			X				WE14EK3.2: Normal, abnormal and emergency operating procedures associated with (High Containment Pressure).	3.1	23
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4		X					074EK2.09: Controllers and positioners	2.6	24
000076 (APE 76) High Reactor Coolant Activity / 9							N/A		
000078 (APE 78*) RCS Leak / 3							N/A		
(W E02) SI Termination / 3				X			WEO2EA1.3: Desired operating results during abnormal and emergency situations	3.8	25
(W E13) Steam Generator Overpressure / 4							N/A		
(W E15) Containment Flooding / 5					X		WE15EA2.2: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	2.9	26
(W E16) High Containment Radiation /9							N/A		
(W E03) LOCA Cooldown—Depressurization / 4						X	WE03EG2.1.20: Ability to execute procedure steps	4.6	27
K/A Category Point Totals:	1	1	2	2	2/2	1/2	Group Point Total:		9/4

ES-401 PWR Examination Outline Form ES-401-2
Plant Systems—Tier 2/Group 1 (RO/SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump									X			003A3.01: Seal injection flow	3.3	28
004 (SF1; SF2 CVCS) Chemical and Volume Control		X										004K2.06: Control instrumentation	2.6	29
											X	004G2.2.22: Knowledge of limiting conditions for operations and safety limits.	4.7	86
005 (SF4P RHR) Residual Heat Removal						X						005K6.03: RHR heat exchanger	2.5	30
006 (SF2; SF3 ECCS) Emergency Core Cooling							X					006A1 .15: RWST Level and temperature	3.3	31
007 (SF5 PRTS) Pressurizer Relief/Quench Tank			X									007K3.01: Containment	3.3	32
				X								007K4.01: Quench tank cooling	2.6	33
008 (SF8 CCW) Component Cooling Water				X								008K4.02: Operation of the surge tank, including the associated valves and controls	2.9	34
010 (SF3 PZR PCS) Pressurizer Pressure Control		X										010K2.01: PZR heaters	3.0	35
						X						010K6.01: Pressure detection systems	2.7	36
012 (SF7 RPS) Reactor Protection										X		012A4.04: Bistable, trips, reset and test switches	3.3	37
013 (SF2 ESFAS) Engineered Safety Features Actuation	X											013K1.13: HVAC	2.8	38
								X				013A2.04: Loss of instrument bus	4.2	87
022 (SF5 CCS) Containment Cooling										X		022A4.04: Valves in the CCS	3.1	39
025 (SF5 ICE) Ice Condenser							X					025A1.03: Glycol flow to ice condenser air handling units	2.5	40
					X							025K5 01: Relationships between pressure and temperature	3.0	41
026 (SF5 CSS) Containment Spray								X				026A2.07: Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding) or sump level below cutoff (interlock) limit	3.6	42
			X									026K3.02: Recirculation spray system	4.2	43
039 (SF4S MSS) Main and Reheat Steam							X					039A1 .06: Main steam pressure	3.0	44
								X				039A2.03: Indications and alarms for main steam and area radiation monitors (during SGTR)	3.4	45

059 (SF4S MFW) Main Feedwater				X								059K4.08: Feedwater regulatory valve operation (on basis of steam flow, feed flow mismatch)	2.5	46
								X				059A2.11: Failure of feedwater control system	3.3	88
061 (SF4S AFW) Auxiliary/Emergency Feedwater					X							061K5.02: Decay heat sources and magnitude	3.2	47
											X	061G2.4.30: Knowledge of events related to system operations/status that must be reported to internal organizations or outside agencies.	4.1	89
062 (SF6 ED AC) AC Electrical Distribution		X										062K2.01: Major system loads	3.3	48
063 (SF6 ED DC) DC Electrical Distribution									X			063A3.01: Meters, annunciators, dials, recorders and indicating lights	2.7	49
064 (SF6 EDG) Emergency Diesel Generator								X				064A2.15: Water buildup in cylinders	2.6	50
								X				064A2.06: Operating unloaded, lightly loaded and highly loaded time limit	3.3	90
073 (SF7 PRM) Process Radiation Monitoring											X	073G2.1.25: Ability to interpret reference materials such as graphs, monographs and tables which contain performance data	3.9	51
076 (SF4S SW) Service Water	X											076K1.19: SWS emergency heat loads	3.6	52
078 (SF8 IAS) Instrument Air			X									078K3.03: Cross-tied units	3.0	53
103 (SF5 CNT) Containment									X			103A3.01: Containment isolation	3.9	54
											X	103G2.4.50: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.2	55
053 (SF1; SF4P ICS*) Integrated Control														
K/A Category Point Totals:	2	3	3	3	2	2	3	3/3	3	2	2/2	Group Point Total:		28/5

ES-401 PWR Examination Outline Form ES-401-2

Plant Systems—Tier 2/Group 2 (RO/SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive		X										001K2.02: One-line diagram of power supply to trip breakers	3.6	56
002 (SF2; SF4P RCS) Reactor Coolant								X				002A2.04: Loss of heat sinks	4.6	91
011 (SF2 PZR LCS) Pressurizer Level Control				X								011K4.07: Cold-calibrated channel	2.9	57
014 (SF1 RPI) Rod Position Indication								X				014A2.03: Dropped rod	4.1	92
015 (SF7 NI) Nuclear Instrumentation	X											015K1 .02: Vital ac systems	3.4	58

016 (SF7 NNI) Nonnuclear Instrumentation											X	016G2.2.39: Knowledge of less than one hour technical specification action statements for systems	3.9	59
017 (SF7 ITM) In-Core Temperature Monitor											X	017G2.2.3: (multi-unit license) Knowledge of the design, procedural and operational differences between units.		93
027 (SF5 CIRS) Containment Iodine Removal												N/A		
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control					X							028K5.03: Sources of hydrogen within containment	2.9	60
029 (SF8 CPS) Containment Purge												N/A		
033 (SF8 SFPCS) Spent Fuel Pool Cooling												N/A		
034 (SF8 FHS) Fuel-Handling Equipment												N/A		
035 (SF 4P SG) Steam Generator												N/A		
041 (SF4S SDS) Steam Dump/Turbine Bypass Control												N/A		
045 (SF 4S MTG) Main Turbine Generator								X				045A2.12: Control rod insertion limits exceeded (stabilize secondary	2.5	61
055 (SF4S CARS) Condenser Air Removal			X									055K3.01: Main condenser	2.5	62
056 (SF4S CDS) Condensate												N/A		
068 (SF9 LRS) Liquid Radwaste												N/A		
071 (SF9 WGS) Waste Gas Disposal												N/A		
072 (SF7 ARM) Area Radiation Monitoring							X					072A1.01: Radiation levels	3.4	63
075 (SF8 CW) Circulating Water										X		075A4.01: Emergency/essential SWS pumps	3.2	64
079 (SF8 SAS**) Station Air												N/A		
086 Fire Protection						X						086K6.04: Fire, smoke and heat detectors	2.6	65
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:	1	1	1	1	1	1	1	1/2	0	1	1/1	Group Point Total:		10/3

Facility:		Date of Exam:				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.15	Knowledge of administrative requirements for temporary management directives such as standing orders, night orders, Operations memos, etc.	2.7	66		
	2.1.37	Knowledge of procedures, guidelines or limitations associated with reactivity management	4.3	67		
	2.1.1	Knowledge of conduct of operations requirements			4.2	94
	2.1.25	Ability to interpret reference materials such as graphs, curves, tables, etc			4.2	95
	Subtotal					
2. Equipment Control	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	4.5	68		
	2.2.22	Knowledge of limiting conditions for operations and safety limits.	4.0	69		
	2.2.43	Knowledge of the process used to track inoperable alarms	3.0	70		
	2.2.13	Knowledge of tagging and clearance procedures			4.3	96
	2.2.18	Knowledge of the process for managing maintenance activities during shutdown operations.			3.8	97
	Subtotal					
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal and emergency conditions	3.2	71		
	2.3.5	Ability to use radiation monitoring systems	2.9	72		
	2.3.11	Ability to control radiation releases			4.3	98
	2.3.12	Knowledge'of radiological safety principles pertaining to licensed operator duties			3.7	99
	2.3.					
	Subtotal					
4. Emergency Procedures/Plan	2.4.29	Knowledge of the emergency plan.	3.1	73		
	2.4.3	Ability to identify post-accident instrumentation	3.7	74		
	2.4.9	Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies	3.8	75		
	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal and emergency evolutions.			4.3	100
	Subtotal					
Tier 3 Point Total				10		7

Draft Operating Test Review

Facility: Catawba Nuclear Station   Date of Examination: Sep 2021

Examination Level: RO ☒ SRO ☐   Operating Test Number: 2021301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, N	**Calculate Reactor Vessel Head Venting Time** G 2.1.23 Ability to perform specific system and integrated plant procedures during al modes of plant operation.
Conduct of Operations	R, N	**Determine Rod Insertion Limit Boration** G 2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.
Equipment Control	R, D	**Determine NC Subcooling on a Loss of OAC** G 2.2.12 Knowledge of surveillance procedures.
Radiation Control	R, D, P	**Determine RP Requirements** G 2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
Emergency Plan		

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

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

Admin JPMs

JPM A.1-1R Calculate Reactor Vessel Head Vent Time – New JPM.

K/A Generic 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR 41.10 / 43.5 / 45.2 / 45.6) RO 4.3 SRO 4.4

Initial conditions are that a LOCA has occurred on Unit 1. The applicants are directed to calculate and record the maximum reactor vessel head venting time per EP/1/A/5000/FR-I.3 (Response to Voids in Reactor Vessel) Enclosure 2 (Allowable Hydrogen Venting Time). The applicants are given Enclosure 2 of FR-I.3 and pictures of control board meters for Containment Pressure, Hydrogen Concentration, Lower Containment Air Temperature, and Wide Range NC System Pressure which will be used in the calculation. Applicants will calculate the allowable head venting time of 2.1 – 4.4 minutes based on bounding values determined.

JPM A.1-2R Determine Rod Insertion Limit Boration – New JPM.

K/A Generic 2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. (CFR 41.10 / 43.6 / 45.6) RO 4.1 SRO 4.3

Initial conditions are that Unit 1 has experienced a runback from 85% RTP following a Zone A Lockout. Applicants are given current power level, core life, current Control Bank D rod position, and current NC System boron concentration. The applicants are directed to determine the amount of boric acid required to restore control rods to 10 steps above the Rod Insertion Limits per AP/1/A/5500/003 (Load Rejection) Enclosure 3 (Rod Insertion Limit Boration). The applicants will use the Enclosure as well as the Unit 1 ROD Manual to determine that in order to restore control rods to 10 steps above the Insertion Limits that 308 – 411 gallons of boric acid must be added to the NC System.

JPM A.2R Determine NC Subcooling on a Loss of the Operator Aid Computer – Bank JPM CCM-003.

K/A Generic 2.2.12 Knowledge of surveillance procedures. (CFR 41.10 / 45.13) RO 3.7 SRO 4.1

Initial conditions are that Unit 1 is in Mode 3 and has experienced a loss of the Operator Aid Computer. PT/1/A/4600/009 (Loss of Operator Aid Computer) is in progress with both trains of the Plasma display monitors inoperable. The applicants are given a table of values for different NC system temperatures and pressures and are directed to complete Enclosure 13.8 (Subcooling Data Sheet) to determine the °F that the NC system is subcooled, and determine if the amount of subcooling margin meets the acceptance criteria of PT/1/A/4600/009. Applicants will determine the subcooling margin is 13°F – 21°F and that this does not meet the acceptance criteria of 30°F for a Mode 3 condition.

JPM A.3R Determine Radiation Protection Requirements for an activity – Bank JPM Previously used on 2019 NRC exam (JPM A.3R).

K/A Generic 2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR 41.12 / 43.4 / 45.10) RO 3.4 SRO 3.8

Initial conditions are that Unit 1 has entered AP/1/A/5500/019 (Loss of Residual Heat Removal). The CRS has sent an AO to the 1A ND pump room to stand by in a low exposure waiting area and await word to vent 1A ND pump. The applicant is given a copy of RWP # 5021 (ECCS venting) and a copy of a plan view for 1A and 2A ND pump rooms and a timeline for the evolution. The applicant will calculate total dose received during the waiting period and pump vent to be 13 mR and then calculate allowable time at LEWA before exceeding 80% of the dose specified in the RWP to be 42 minutes.

Draft Operating Test Review

Facility: ______	Date of Examination: Sep 2021
Examination Level: RO ☐ SRO ☒	Operating Test Number: 2021301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, D, P	**Calculate Boric Acid and Water Addition to FWST and determine Tech Spec actions** G 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.
Conduct of Operations	R, D	**Determine License Status** G 2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical records, "no-solo" operation, maintenance of active license status.
Equipment Control	R, N	**Determine Isolation Boundary** G 2.2.41 Ability to obtain and interpret station electrical and mechanical drawings.
Radiation Control	R, D	**Review Liquid Waste Release** G 2.3.6 Ability to approve release permits.
Emergency Plan	R, N	**Classify an Event and Fill Out the Emergency Notification Form** G 2.4.40 Knowledge of SRO responsibilities in emergency plan implementation.

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

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

Admin JPMs

JPM A.1-1S Calculate Boric Acid and Water Addition to FWST and determine Tech Spec Actions – Bank JPM previously used on 2017 NRC Exam (JPM A.1-2S).

K/A Generic 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR 41.10 / 43.5 / 45.2 / 45.6) RO 4.3 SRO 4.4

Initial conditions are that an improper valve lineup has decreased Unit 1 FWST level. The valve lineup issue has been corrected to stop the level decrease. The applicant is directed to calculate a makeup to the FWST to restore level to a value above the Tech Spec minimum. Following makeup calculation, the applicant is required to address Tech Specs action at the time of discovery and one hour later. The applicant will conclude an action statement existed at time of discovery due to level below minimum required. One hour later, with a given makeup flowrate, the applicant will determine that level remains below minimum. This will require entry into another action due to inability to restore operability within one hour, as well as remaining in original action statement.

JPM A.1-2S Determine License Status – Bank JPM (NS07-001)

K/A Generic 2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical records, "no solo" operation, maintenance of active license status. (CFR 41.10 / 43.2) RO 3.3 SRO 3.8

Initial conditions have the applicants evaluating the work histories of 3 different licensed operators with Unit 1 in Mode 1 the entire time and Unit 2 in a refueling outage. Using AD-OP-ALL-0107 (Maintenance of RO and SRO Licenses), the applicants will determine that one of the reviewed individuals will have an active license and the two others will not on July 1.

JPM A.2S Use Flow Diagrams and Electrical Prints to Determine Work Isolation Boundary – New JPM.

K/A Generic 2.2.41 Ability to obtain and interpret station electrical and mechanical drawings. (CFR 41.10 / 45.12 / 45.13) RO 3.5 SRO 3.9

Initial conditions are that 2A motor driven Auxiliary Feedwater Pump has been shutdown in accordance with OP/2/A/6250/002 (Auxiliary Feedwater System) and is to be tagged out for pump casing disassembly and impeller replacement. The applicants are given mechanical flow diagrams for the Auxiliary Feedwater System as well as electrical one-line diagrams for 2ETA and 2ETB essential busses. Applicants are directed to determine the required boundary isolation for the required work. They are instructed to use the valves closest to the work being performed to minimize drain and fill time. Applicants need to determine the mechanical/electrical isolations as well as an applicable vent and drain path.

JPM A.3S Review Liquid Waste Release – Bank JPM (WL-002)

K/A Generic 2.3.6 Ability to approve release permits. (CFR 41.10 / 45.12 / 45.13) RO 2.0 SRO 3.8

Initial conditions are given to the applicants including Unit status, RC pumps in service, RL discharge flow, RN pump status, and that the LWR integrator is operable. An LWR has been delivered to the control room and approved by the previous shifts CRS. The BOP has notified the CRS that the LWR is ready to be released per OP/0/B/6500/013 (Operations Liquid Waste Release) Enclosure 4.1 (Liquid Waste Release from a Monitor Tank). Applicants are to review the LWR and determine if the release should be generated and, if not, list any issues that would prevent the release initiation. Applicants will determine that the LWR should not be initiated due to one or all of the following: incorrect EMF-49 Trip 2 setpoint, incorrect RL flow interlock setpoint, and incorrect 1WL-124 flowrate setpoint.

Draft Operating Test Review

JPM A.4S Classify an Event and Fill Out the Emergency Notification Form – New JPM

K/A 2.4.40 Knowledge of SRO responsibilities in emergency plan implementation. (CFR 41.10 / 43.5 / 45.11) RO 2.7 SRO 4.5

Initial conditions are both Units are at 100% RTP when a seismic event is felt within the protected area. Annunciator 1AD-4, B/8 “OBE EXCEEDED” is received as well as a Loss of Offsite Power on Unit 2 with failure of 2B D/G to start. An auxiliary operator reports that the 1A NI (Safety Injection) pump discharge piping is cracked with water leaking out at 125 drops per minute. Applicants are to use AD-EP-ALL-0101 (Emergency Classification) and CSD-EP-CNS-0101-02 (EAL Wallcharts) to classify the event and fill out the Emergency Notification Form per AD-EP-ALL-0304 (State and County Notification). This JPM is time critical for both the classification ($\leq$ 15 minutes) and filling out the ENF form ($\leq$ 15 minutes).

Draft Operating Test Review

Facility:	Catawba Nuclear Station	Date of Examination:	Sep 2021
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐		Operating Test Number:	2021301

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

System/JPM Title	Type Code*	Safety Function
a. Emergency Borate the Reactor Coolant System	A, L, P, D, S	1
b. Isolate Cold Leg Accumulators During Shutdown LOCA	A, L, M, S	3
c. Restore CA flow following Feed & Bleed (w/ 1CA-4 closed)	A, EN, L, N, S	4S
d. Perform E-0 Actions to Ensure Complete Containment Isolation	A, M, S	2
e. Align the NS System for Cold Leg Recirculation	A, D, L, S	5
f. Manual Alignment of 1FTB	D, S	6
g. Reset Radiation Monitor Trip Setpoints	D, P, S	7
h. Place KC in Parallel Operation	D, S	8

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

System/JPM Title	Type Code*	Safety Function
i. Local ESPS alignment to 2ETB (2ATD) – AP-07 Encl. 52	E, L, N	6
j. Place Hydrogen Recombiner in Service	E, L, D, R	5
k. Break Main Condenser Vacuum Locally – Unit 1	E, D	4S

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

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

System JPMs

JPM A Emergency Borate the Reactor Coolant System – Bank JPM (Alternate Path) Previously used on 2017 NRC exam (JPM A)

K/A System 004 A2.14 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct control, or mitigate the consequences of those malfunctions or operations: Emergency Boration. (CFR 41.5 / 43.5 / 45.3 / 45.5) RO 3.8 SRO 3.9

Initial conditions are that an ATWS is in progress following a valid reactor trip signal and failure of the reactor to trip automatically or manually. The applicants are directed to initiate emergency boration per EP/1/A/5000/FR-S.1 (Nuclear Power Generation/ATWS) step 4. The applicants will begin by attempting to open 1NV-236B (Boric Acid to NV Pump Suction). This valve will not open and begins the alternate path for this JPM. The applicants will continue with the procedure and start both boric acid transfer pumps. With emergency boration flow not showing greater than 30 GPM, the applicants will align the suction of the NV pumps to the FWST and isolate the NV pump suction from the VCT.

JPM B Isolate Cold Leg Accumulators Following a Shutdown LOCA – Modified Bank JPM (Alternate Path). JPM was modified to change the Cold Leg Accumulators that could not be isolated.

K/A System 006 A1.13 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration). (CFR 41.5 / 45.5) RO 3.5 SRO 3.7

Initial conditions have the Unit in Mode 4 with a shutdown in progress for a refueling outage, when Pressurizer pressure and level begin to unexpectedly lower. The crew has entered AP/1/A/5500/027 (Shutdown LOCA) to address the Reactor Coolant System leak. The applicants are directed to isolate the Unit 1 Cold Leg Accumulators per AP/27 Enclosure 17 (Isolating Cold Leg Accumulators). The applicants will attempt to close all 4 Cold Leg Accumulator isolation valves, but the isolations for 1A and 1D Cold Leg Accumulators will not close. This begins the alternate path for this JPM. The applicants will go on to perform the steps to vent the 1A and 1D Cold Leg Accumulators to containment.

JPM C Restore CA flow following NC System Feed and Bleed – NEW JPM (Alternate Path)

K/A EPE05 EA1.1 Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR 41.7 / 45.5 / 45.6) RO 4.1 SRO 4.0

Initial conditions have Unit 1 in Mode 3 following a loss of all feedwater. NC system bleed and feed have been initiated per EP/1/A/5000/FR-H.1 (Loss of Secondary Heat Sink). A report from maintenance is received, informing the crew that 1A CA pump is ready to be started. The applicants are directed to perform Step 7 to establish CA flow from 1A CA pump. During the initial stages of restoring flow from 1A CA pump, it is discovered that 1CA-4 (CA Pmps Suct From UST) is closed and cannot be opened. This begins the alternate path for this JPM. The applicant will align suction from the RN system, and start the 1A CA pump. Since the CA flow control valves were previously closed when aligning for feed and bleed, the applicants will be sent to Enclosure 7 (S/G CA Flow Restoration) to initiate CA flow to 1A and/or 1B S/Gs.

JPM D Perform E-0 Actions to Ensure a Complete Containment Isolation – Modified Bank JPM (Alternate Path). JPM was modified by changing the containment penetration that the applicants will need to manually isolate.

K/A System 013 A4.01 Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate. (CFR 41.7 / 45.5 to 45.8) RO 4.5 SRO 4.8

Initial conditions have the Unit in Mode 1, 100% RTP. The applicants are informed that they are the OATC, that the BOP has stepped out of the control room, and that the CRS is performing an IPTE brief on Unit 2, and to monitor their control boards. Once the applicants are ready, a Large Break LOCA is inserted. The applicants will verify the immediate actions of EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). The applicants are then directed to continue the actions of E-0. When checking for proper Phase A containment isolation, the applicants will discover failure of automatic actuation and will manually initiate Phase A. This begins the alternate path for this JPM. When the applicants check for proper Monitor Light Panel alignment, they will discover that NV letdown isolation valves 1NV-10A and 1NV-15B and liquid waste penetration isolation valves 1WL-805A and 1WL-807B did not close on the Phase A initiation signal. The applicants will manually close these valves to complete the JPM.

JPM E Align the NS System for Cold Leg Recirculation – Bank JPM (Alternate Path)

K/A System 026 A4.01 Ability to manually operate and/or monitor in the control room: CSS controls (CFR 41.7 / 45.5 to 45.8) RO 4.5 SRO 4.3

Initial conditions are that a LOCA has occurred on Unit 1 and EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation) is in progress. The applicants are directed to align NS for Cold Leg Recirculation per ES-1.3 Enclosure 2 (Aligning NS for Recirculation). The applicants will close the NS pump suctions from the FWST and open the NS Spray header 1A containment isolation valves. The applicants will attempt to open the 1A NS pump suction from the containment sump, but the valve will not open. This begins the alternate path for this JPM. The applicants will then perform the steps to align and start the 1B NS pump and align cooling water to the 1B NS Heat Exchanger.

JPM F Manually Align Essential power to 1FTB from 1ETB per AP/1/A/5500/007 Case 1 Step 10 – Bank JPM

K/A System 062 A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard). (CFR 41.7 / 45.5 to 45.8) RO 3.3 SRO 3.1

Initial conditions are that Unit 1 is operating at 100% RTP when a blackout occurs on 1ETB due to a failure of 1ATD. The 1B D/G is supplying 1ETB. The crew has implemented AP/1/A/5500/007 Case 1 (Loss of Normal Power to an Essential Train). The applicants are directed to perform Step 10. Applicants will determine that 1FTB is de-energized and will use the procedure to reset the 1B D/G load sequencer and close breakers to energize 1FTB from 1ETB.

JPM G Reset Radiation Monitor Trip Setpoints – Bank JPM previously used on 2017 NRC Exam (JPM F)

K/A System 073 A4.02 Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel. (CFR 41.7 / 45.5 to 45.8) RO 3.7 SRO 3.7

Initial conditions are that following a discussion with RP on a premature gaseous waste release termination, that EMF-50L trip setpoints need to be changed. The applicants are directed to change the trip 1 setpoint to 6300 cpm and the trip 2 setpoint to 9000 cpm using OP/0/A/6500/080 (EMF RP86A Output Modules) Enclosure 4.2 (EMF RP86A and RM1000 Trip Setpoint Adjustment).

JPM H Place KC in Parallel Operation per OP/1/A/6400/005 – Bank JPM

K/A System 008 A4.01 Ability to manually operate and/or monitor in the control room: CCW indications and controls. (CFR 41.7 / 45.5) RO 3.3 SRO 3.1

Initial conditions are that Unit 1 is at 100% RTP and that a worklist item has been generated to place Unit 1 KC system in parallel operation in preparation for Aux Safeguards Testing early next shift. The applicants are directed to place KC in parallel operation per OP/1/A/6400/005 (Component Cooling System) Enclosure 4.4 (Operation of Additional KC Pumps/Parallel Operation). The applicants will ensure RN system miniflow from Unit 2, ensure proper flowpath on Unit 1, place the 1A KC outlet mode switch in the "KC TEMP" position, bypass the letdown mixed bed demineralizers, and start an 'A' train KC pump.

JPM I Local ESPS Alignment to 2ETB through 2ATD – NEW JPM

K/A Generic Emergency 055 EA2.03 Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power. (CFR 43.5 / 45.13) RO 3.9 SRO 4.7

Initial conditions are that Unit 2 is in Mode 3 following a Loss of All AC Power. EP/2/A/5000/ECA-0.0 (Loss of All AC Power) has been entered. Neither of the Emergency D/Gs could be started. Station management has determined that power will be restored to 2ETB from ESPS (Emergency Supplemental Power Supply) through 2ATD. The applicants are directed to perform ECA-0.0 Enclosure 52 (Local ESPS Alignment to 2ETB through 2ATD) beginning at Step 3. Applicants will ensure required breakers are open, start ESPS D/Gs 1 and 2, and close breakers required to align power from the ESPS D/Gs through 2ATD to 2ETB.

JPM J Place 1B Hydrogen Recombiner in Service – Bank JPM

K/A System 028 A2.01 Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Hydrogen Recombiner power setting, determined by using plant data book. (CFR 41.5 / 43.5 / 45.3 / 45.13) RO 3.4 SRO 3.6

Initial conditions are that a LOCA has occurred on Unit 1. The applicants are directed to place Hydrogen Recombiner 1B in service at the required power level per OP/1/A/6450/010 (Containment Hydrogen Control System) Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA) steps 3.1 through 3.3.14. Applicants are given current containment pressure and containment hydrogen concentration. When performing the steps to place 1B Hydrogen Recombiner in service, the applicants will be required to determine the required power level using Unit 1 Revised Databook Figure 10. With hydrogen concentration exceeding 3.5%, the applicants will be required to add 4 KW to the value determined from Figure 10 to determine the final power level.

JPM K Break Main Condenser Vacuum Locally – Bank JPM

K/A System 045 A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of secondary plant parameters following a T/G trip.

Initial conditions are that Unit 1 is in Mode 3 following a reactor trip. The applicants are directed to perform AP/1/A/5500/006 (Loss of S/G Feedwater) Enclosure 3 (Local Actions to Break Condenser Vacuum). This JPM is time critical and must be performed in < 10 minutes. The time critical time will end when the first condenser vacuum breaker valve is opened. Once all 3 vacuum breaker valves are opened, the applicant will close Main and Aux Steam to the Air Ejectors.

Draft Operating Test Review

Facility: Catawba Nuclear Station Date of Examination: Sep 2021

Exam Level: RO ☐ SRO-I ☐ SRO-U ☒ Operating Test Number: 2021301

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

System/JPM Title	Type Code*	Safety Function
a. Emergency Borate the Reactor Coolant System	A, L, P, D, S	1
b. Isolate Cold Leg Accumulators During Shutdown LOCA	A, L, M, S	3
c. Restore CA flow following Feed & Bleed (w/ 1CA-4 closed)	A, EN, L, N, S	4S

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

System/JPM Title	Type Code*	Safety Function
i. Local ESPS alignment to 2ETB (2ATD) – AP-07 Encl. 52	E, L, N	6
j. Place Hydrogen Recombiner in Service	E, L, D, R	5

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

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

System JPMs

JPM A Emergency Borate the Reactor Coolant System – Bank JPM (Alternate Path) Previously used on 2017 NRC exam (JPM A)

K/A System 004 A2.14 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct control, or mitigate the consequences of those malfunctions or operations: Emergency Boration. (CFR 41.5 / 43.5 / 45.3 / 45.5) RO 3.8 SRO 3.9

Initial conditions are that an ATWS is in progress following a valid reactor trip signal and failure of the reactor to trip automatically or manually. The applicants are directed to initiate emergency boration per EP/1/A/5000/FR-S.1 (Nuclear Power Generation/ATWS) step 4. The applicants will begin by attempting to open 1NV-236B (Boric Acid to NV Pump Suction). This valve will not open and begins the alternate path for this JPM. The applicants will continue with the procedure and start both boric acid transfer pumps. With emergency boration flow not showing greater than 30 GPM, the applicants will align the suction of the NV pumps to the FWST and isolate the NV pump suction from the VCT.

JPM B Isolate Cold Leg Accumulators Following a Shutdown LOCA – Modified Bank JPM (Alternate Path). JPM was modified to change the Cold Leg Accumulators that could not be isolated.

K/A System 006 A1.13 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration). (CFR 41.5 / 45.5) RO 3.5 SRO 3.7

Initial conditions have the Unit in Mode 4 with a shutdown in progress for a refueling outage, when Pressurizer pressure and level begin to unexpectedly lower. The crew has entered AP/1/A/5500/027 (Shutdown LOCA) to address the Reactor Coolant System leak. The applicants are directed to isolate the Unit 1 Cold Leg Accumulators per AP/27 Enclosure 17 (Isolating Cold Leg Accumulators). The applicants will attempt to close all 4 Cold Leg Accumulator isolation valves, but the isolations for 1A and 1D Cold Leg Accumulators will not close. This begins the alternate path for this JPM. The applicants will go on to perform the steps to vent the 1A and 1D Cold Leg Accumulators to containment.

JPM C Restore CA flow following NC System Feed and Bleed – NEW JPM (Alternate Path)

K/A EPE05 EA1.1 Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR 41.7 / 45.5 / 45.6) RO 4.1 SRO 4.0

Initial conditions have Unit 1 in Mode 3 following a loss of all feedwater. NC system bleed and feed have been initiated per EP/1/A/5000/FR-H.1 (Loss of Secondary Heat Sink). A report from maintenance is received, informing the crew that 1A CA pump is ready to be started. The applicants are directed to perform Step 7 to establish CA flow from 1A CA pump. During the initial stages of restoring flow from 1A CA pump, it is discovered that 1CA-4 (CA Pmps Suct From UST) is closed and cannot be opened. This begins the alternate path for this JPM. The applicant will align suction from the RN system, and start the 1A CA pump. Since the CA flow control valves were previously closed when aligning for feed and bleed, the applicants will be sent to Enclosure 7 (S/G CA Flow Restoration) to initiate CA flow to 1A and/or 1B S/Gs.

JPM I Local ESPS Alignment to 2ETB through 2ATD – NEW JPM

K/A Generic Emergency 055 EA2.03 Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power. (CFR 43.5 / 45.13) RO 3.9 SRO 4.7

Initial conditions are that Unit 2 is in Mode 3 following a Loss of All AC Power. EP/2/A/5000/ECA-0.0 (Loss of All AC Power) has been entered. Neither of the Emergency D/Gs could be started. Management has determined that power will be restored to 2ETB from ESPS (Emergency Supplemental Power Supply) through 2ATD. The applicants are directed to perform ECA-0.0 Enclosure 52 (Local ESPS Alignment to 2ETB through 2ATD) beginning at Step 3. Applicants will ensure required breakers are open, start ESPS D/Gs 1 and 2, and close breakers required to align power from the ESPS D/Gs through 2ATD to 2ETB.

JPM J Place 1B Hydrogen Recombiner in Service – Bank JPM

K/A System 028 A2.01 Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Hydrogen Recombiner power setting, determined by using plant data book. (CFR 41.5 / 43.5 / 45.3 / 45.13) RO 3.4 SRO 3.6

Initial conditions are that a LOCA has occurred on Unit 1. The applicants are directed to place Hydrogen Recombiner 1B in service at the required power level per OP/1/A/6450/010 (Containment Hydrogen Control System) Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA) steps 3.1 through 3.3.14. Applicants are given current containment pressure and containment hydrogen concentration. When performing the steps to place 1B Hydrogen Recombiner in service, the applicants will be required to determine the required power level using Unit 1 Revised Databook Figure 10. With hydrogen concentration exceeding 3.5%, the applicants will be required to add 4 KW to the value determined from Figure 10 to determine the final power level.

Draft Operating Test Review

Facility: Catawba Nuclear Station	Date of Examination: Sep 2021
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐	Operating Test Number: 2021301

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

System/JPM Title	Type Code*	Safety Function
a. Emergency Borate the Reactor Coolant System	A, L, P, D, S	1
b. Isolate Cold Leg Accumulators During Shutdown LOCA	A, L, M, S	3
c. Restore CA flow following Feed & Bleed (w/ 1CA-4 closed)	A, EN, L, N, S	4S
d. Perform E-0 Actions to Ensure Complete Containment Isolation	A, M, S	2
e. Align the NS System for Cold Leg Recirculation	A, D, L, S	5
f. Manual Alignment of 1FTB	D, S	6
g. Reset Radiation Monitor Trip Setpoints	D, P, S	7

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

System/JPM Title	Type Code*	Safety Function
i. Local ESPS alignment to 2ETB (2ATD) – AP-07 Encl. 52	E, L, N	6
j. Place Hydrogen Recombiner in Service	E, L, D, R	5
k. Break Main Condenser Vacuum Locally – Unit 1	E, D	4S

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

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

System JPMs

JPM A Emergency Borate the Reactor Coolant System – Bank JPM (Alternate Path) Previously used on 2017 NRC exam (JPM A)

K/A System 004 A2.14 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct control, or mitigate the consequences of those malfunctions or operations: Emergency Boration. (CFR 41.5 / 43.5 / 45.3 / 45.5) RO 3.8 SRO 3.9

Initial conditions are that an ATWS is in progress following a valid reactor trip signal and failure of the reactor to trip automatically or manually. The applicants are directed to initiate emergency boration per EP/1/A/5000/FR-S.1 (Nuclear Power Generation/ATWS) step 4. The applicants will begin by attempting to open 1NV-236B (Boric Acid to NV Pump Suction). This valve will not open and begins the alternate path for this JPM. The applicants will continue with the procedure and start both boric acid transfer pumps. With emergency boration flow not showing greater than 30 GPM, the applicants will align the suction of the NV pumps to the FWST and isolate the NV pump suction from the VCT.

JPM B Isolate Cold Leg Accumulators Following a Shutdown LOCA – Modified Bank JPM (Alternate Path). JPM was modified to change the Cold Leg Accumulators that could not be isolated.

K/A System 006 A1.13 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration). (CFR 41.5 / 45.5) RO 3.5 SRO 3.7

Initial conditions have the Unit in Mode 4 with a shutdown in progress for a refueling outage, when Pressurizer pressure and level begin to unexpectedly lower. The crew has entered AP/1/A/5500/027 (Shutdown LOCA) to address the Reactor Coolant System leak. The applicants are directed to isolate the Unit 1 Cold Leg Accumulators per AP/27 Enclosure 17 (Isolating Cold Leg Accumulators). The applicants will attempt to close all 4 Cold Leg Accumulator isolation valves, but the isolations for 1A and 1D Cold Leg Accumulators will not close. This begins the alternate path for this JPM. The applicants will go on to perform the steps to vent the 1A and 1D Cold Leg Accumulators to containment.

JPM C Restore CA flow following NC System Feed and Bleed – NEW JPM (Alternate Path)

K/A EPE05 EA1.1 Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR 41.7 / 45.5 / 45.6) RO 4.1 SRO 4.0

Initial conditions have Unit 1 in Mode 3 following a loss of all feedwater. NC system bleed and feed have been initiated per EP/1/A/5000/FR-H.1 (Loss of Secondary Heat Sink). A report from maintenance is received, informing the crew that 1A CA pump is ready to be started. The applicants are directed to perform Step 7 to establish CA flow from 1A CA pump. During the initial stages of restoring flow from 1A CA pump, it is discovered that 1CA-4 (CA Pmps Suct From UST) is closed and cannot be opened. This begins the alternate path for this JPM. The applicant will align suction from the RN system, and start the 1A CA pump. Since the CA flow control valves were previously closed when aligning for feed and bleed, the applicants will be sent to Enclosure 7 (S/G CA Flow Restoration) to initiate CA flow to 1A and/or 1B S/Gs.

JPM D Perform E-0 Actions to Ensure a Complete Containment Isolation – Modified Bank JPM (Alternate Path). JPM was modified by changing the containment penetration that the applicants will need to manually isolate.

K/A System 013 A4.01 Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate. (CFR 41.7 / 45.5 to 45.8) RO 4.5 SRO 4.8

Initial conditions have the Unit in Mode 1, 100% RTP. The applicants are informed that they are the OATC, that the BOP has stepped out of the control room, and that the CRS is performing an IPTE brief on Unit 2, and to monitor their control boards. Once the applicants are ready, a Large Break LOCA is inserted. The applicants will verify the immediate actions of EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). The applicants are then directed to continue the actions of E-0. When checking for proper Phase A containment isolation, the applicants will discover failure of automatic actuation and will manually initiate Phase A. This begins the alternate path for this JPM. When the applicants check for proper Monitor Light Panel alignment, they will discover that NV letdown isolation valves 1NV-10A and 1NV-15B and liquid waste penetration isolation valves 1WL-805A and 1WL-807B did not close on the Phase A initiation signal. The applicants will manually close these valves to complete the JPM.

JPM E Align the NS System for Cold Leg Recirculation – Bank JPM (Alternate Path)

K/A System 026 A4.01 Ability to manually operate and/or monitor in the control room: CSS controls (CFR 41.7 / 45.5 to 45.8) RO 4.5 SRO 4.3

Initial conditions are that a LOCA has occurred on Unit 1 and EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation) is in progress. The applicants are directed to align NS for Cold Leg Recirculation per ES-1.3 Enclosure 2 (Aligning NS for Recirculation). The applicants will close the NS pump suctions from the FWST and open the NS Spray header 1A containment isolation valves. The applicants will attempt to open the 1A NS pump suction from the containment sump, but the valve will not open. This begins the alternate path for this JPM. The applicants will then perform the steps to align and start the 1B NS pump and align cooling water to the 1B NS Heat Exchanger.

JPM F Manually Align Essential power to 1FTB from 1ETB per AP/1/A/5500/007 Case 1 Step 10 – Bank JPM

K/A System 062 A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard). (CFR 41.7 / 45.5 to 45.8) RO 3.3 SRO 3.1

Initial conditions are that Unit 1 is operating at 100% RTP when a blackout occurs on 1ETB due to a failure of 1ATD. The 1B D/G is supplying 1ETB. The crew has implemented AP/1/A/5500/007 Case 1 (Loss of Normal Power to an Essential Train). The applicants are directed to perform Step 10. Applicants will determine that 1FTB is de-energized and will use the procedure to reset the 1B D/G load sequencer and close breakers to energize 1FTB from 1ETB.

JPM G Reset Radiation Monitor Trip Setpoints – Bank JPM previously used on 2017 NRC Exam (JPM F)

K/A System 073 A4.02 Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel. (CFR 41.7 / 45.5 to 45.8) RO 3.7 SRO 3.7

Initial conditions are that following a discussion with RP on a premature gaseous waste release termination, that EMF-50L trip setpoints need to be changed. The applicants are directed to change the trip 1 setpoint to 6300 cpm and the trip 2 setpoint to 9000 cpm using OP/0/A/6500/080 (EMF RP86A Output Modules) Enclosure 4.2 (EMF RP86A and RM1000 Trip Setpoint Adjustment).

JPM I Local ESPS Alignment to 2ETB through 2ATD – NEW JPM

K/A Generic Emergency 055 EA2.03 Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power. (CFR 43.5 / 45.13) RO 3.9 SRO 4.7

Initial conditions are that Unit 2 is in Mode 3 following a Loss of All AC Power. EP/2/A/5000/ECA-0.0 (Loss of All AC Power) has been entered. Neither of the Emergency D/Gs could be started. Management has determined that power will be restored to 2ETB from ESPS (Emergency Supplemental Power Supply) through 2ATD. The applicants are directed to perform ECA-0.0 Enclosure 52 (Local ESPS Alignment to 2ETB through 2ATD) beginning at Step 3. Applicants will ensure required breakers are open, start ESPS D/Gs 1 and 2, and close breakers required to align power from the ESPS D/Gs through 2ATD to 2ETB.

JPM J Place 1B Hydrogen Recombiner in Service – Bank JPM

K/A System 028 A2.01 Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Hydrogen Recombiner power setting, determined by using plant data book. (CFR 41.5 / 43.5 / 45.3 / 45.13) RO 3.4 SRO 3.6

Initial conditions are that a LOCA has occurred on Unit 1. The applicants are directed to place Hydrogen Recombiner 1B in service at the required power level per OP/1/A/6450/010 (Containment Hydrogen Control System) Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA) steps 3.1 through 3.3.14. Applicants are given current containment pressure and containment hydrogen concentration. When performing the steps to place 1B Hydrogen Recombiner in service, the applicants will be required to determine the required power level using Unit 1 Revised Databook Figure 10. With hydrogen concentration exceeding 3.5%, the applicants will be required to add 4 KW to the value determined from Figure 10 to determine the final power level.

JPM K Break Main Condenser Vacuum Locally – Bank JPM

K/A System 045 A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of secondary plant parameters following a T/G trip.

Initial conditions are that Unit 1 is in Mode 3 following a reactor trip. The applicants are directed to perform AP/1/A/5500/006 (Loss of S/G Feedwater) Enclosure 3 (Local Actions to Break Condenser Vacuum). This JPM is time critical and must be performed in < 10 minutes. The time critical time will end when the first condenser vacuum breaker valve is opened. Once all 3 vacuum breaker valves are opened, the applicant will close Main and Aux Steam to the Air Ejectors.

Facility: Catawba Nuclear Station	Date of Examination: Sep 2021
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐	Operating Test Number: 2021301

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

System/JPM Title	Type Code*	Safety Function
a. Emergency Borate the Reactor Coolant System	A, L, P, D, S	1
b. Isolate Cold Leg Accumulators During Shutdown LOCA	A, L, M, S	3
c. Restore CA flow following Feed & Bleed (w/ 1CA-4 closed)	A, EN, L, N, S	4S
d. Perform E-0 Actions to Ensure Complete Containment Isolation	A, M, S	2
e. Align the NS System for Cold Leg Recirculation	A, D, L, S	5
f. Manual Alignment of 1FTB	D, S	6
g. Reset Radiation Monitor Trip Setpoints	D, P, S	7
h. Place KC in Parallel Operation	D, S	8

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

System/JPM Title	Type Code*	Safety Function
i. Local ESPS alignment to 2ETB (2ATD) – AP-07 Encl. 52	E, L, N	6
j. Place Hydrogen Recombiner in Service	E, L, D, R	5
k. Break Main Condenser Vacuum Locally – Unit 1	E, D	4S

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

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

System JPMs

JPM A Emergency Borate the Reactor Coolant System – Bank JPM (Alternate Path) Previously used on 2017 NRC exam (JPM A)

K/A System 004 A2.14 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct control, or mitigate the consequences of those malfunctions or operations: Emergency Boration. (CFR 41.5 / 43.5 / 45.3 / 45.5) RO 3.8 SRO 3.9

Initial conditions are that an ATWS is in progress following a valid reactor trip signal and failure of the reactor to trip automatically or manually. The applicants are directed to initiate emergency boration per EP/1/A/5000/FR-S.1 (Nuclear Power Generation/ATWS) step 4. The applicants will begin by attempting to open 1NV-236B (Boric Acid to NV Pump Suction). This valve will not open and begins the alternate path for this JPM. The applicants will continue with the procedure and start both boric acid transfer pumps. With emergency boration flow not showing greater than 30 GPM, the applicants will align the suction of the NV pumps to the FWST and isolate the NV pump suction from the VCT.

JPM B Isolate Cold Leg Accumulators Following a Shutdown LOCA – Modified Bank JPM (Alternate Path). JPM was modified to change the Cold Leg Accumulators that could not be isolated.

K/A System 006 A1.13 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration). (CFR 41.5 / 45.5) RO 3.5 SRO 3.7

Initial conditions have the Unit in Mode 4 with a shutdown in progress for a refueling outage, when Pressurizer pressure and level begin to unexpectedly lower. The crew has entered AP/1/A/5500/027 (Shutdown LOCA) to address the Reactor Coolant System leak. The applicants are directed to isolate the Unit 1 Cold Leg Accumulators per AP/27 Enclosure 17 (Isolating Cold Leg Accumulators). The applicants will attempt to close all 4 Cold Leg Accumulator isolation valves, but the isolations for 1A and 1D Cold Leg Accumulators will not close. This begins the alternate path for this JPM. The applicants will go on to perform the steps to vent the 1A and 1D Cold Leg Accumulators to containment.

JPM C Restore CA flow following NC System Feed and Bleed – NEW JPM (Alternate Path)

K/A EPE05 EA1.1 Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR 41.7 / 45.5 / 45.6) RO 4.1 SRO 4.0

Initial conditions have Unit 1 in Mode 3 following a loss of all feedwater. NC system bleed and feed have been initiated per EP/1/A/5000/FR-H.1 (Loss of Secondary Heat Sink). A report from maintenance is received, informing the crew that 1A CA pump is ready to be started. The applicants are directed to perform Step 7 to establish CA flow from 1A CA pump. During the initial stages of restoring flow from 1A CA pump, it is discovered that 1CA-4 (CA Pmps Suct From UST) is closed and cannot be opened. This begins the alternate path for this JPM. The applicant will align suction from the RN system, and start the 1A CA pump. Since the CA flow control valves were previously closed when aligning for feed and bleed, the applicants will be sent to Enclosure 7 (S/G CA Flow Restoration) to initiate CA flow to 1A and/or 1B S/Gs.

JPM D Perform E-0 Actions to Ensure a Complete Containment Isolation – Modified Bank JPM (Alternate Path). JPM was modified by changing the containment penetration that the applicants will need to manually isolate.

K/A System 013 A4.01 Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate. (CFR 41.7 / 45.5 to 45.8) RO 4.5 SRO 4.8

Initial conditions have the Unit in Mode 1, 100% RTP. The applicants are informed that they are the OATC, that the BOP has stepped out of the control room, and that the CRS is performing an IPTE brief on Unit 2, and to monitor their control boards. Once the applicants are ready, a Large Break LOCA is inserted. The applicants will verify the immediate actions of EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). The applicants are then directed to continue the actions of E-0. When checking for proper Phase A containment isolation, the applicants will discover failure of automatic actuation and will manually initiate Phase A. This begins the alternate path for this JPM. When the applicants check for proper Monitor Light Panel alignment, they will discover that NV letdown isolation valves 1NV-10A and 1NV-15B and liquid waste penetration isolation valves 1WL-805A and 1WL-807B did not close on the Phase A initiation signal. The applicants will manually close these valves to complete the JPM.

JPM E Align the NS System for Cold Leg Recirculation – Bank JPM (Alternate Path)

K/A System 026 A4.01 Ability to manually operate and/or monitor in the control room: CSS controls (CFR 41.7 / 45.5 to 45.8) RO 4.5 SRO 4.3

Initial conditions are that a LOCA has occurred on Unit 1 and EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation) is in progress. The applicants are directed to align NS for Cold Leg Recirculation per ES-1.3 Enclosure 2 (Aligning NS for Recirculation). The applicants will close the NS pump suctions from the FWST and open the NS Spray header 1A containment isolation valves. The applicants will attempt to open the 1A NS pump suction from the containment sump, but the valve will not open. This begins the alternate path for this JPM. The applicants will then perform the steps to align and start the 1B NS pump and align cooling water to the 1B NS Heat Exchanger.

JPM F Manually Align Essential power to 1FTB from 1ETB per AP/1/A/5500/007 Case 1 Step 10 – Bank JPM

K/A System 062 A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard). (CFR 41.7 / 45.5 to 45.8) RO 3.3 SRO 3.1

Initial conditions are that Unit 1 is operating at 100% RTP when a blackout occurs on 1ETB due to a failure of 1ATD. The 1B D/G is supplying 1ETB. The crew has implemented AP/1/A/5500/007 Case 1 (Loss of Normal Power to an Essential Train). The applicants are directed to perform Step 10. Applicants will determine that 1FTB is de-energized and will use the procedure to reset the 1B D/G load sequencer and close breakers to energize 1FTB from 1ETB.

JPM G Reset Radiation Monitor Trip Setpoints – Bank JPM previously used on 2017 NRC Exam (JPM F)

K/A System 073 A4.02 Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel. (CFR 41.7 / 45.5 to 45.8) RO 3.7 SRO 3.7

Initial conditions are that following a discussion with RP on a premature gaseous waste release termination, that EMF-50L trip setpoints need to be changed. The applicants are directed to change the trip 1 setpoint to 6300 cpm and the trip 2 setpoint to 9000 cpm using OP/0/A/6500/080 (EMF RP86A Output Modules) Enclosure 4.2 (EMF RP86A and RM1000 Trip Setpoint Adjustment).

JPM H Place KC in Parallel Operation per OP/1/A/6400/005 – Bank JPM

K/A System 008 A4.01 Ability to manually operate and/or monitor in the control room: CCW indications and controls. (CFR 41.7 / 45.5) RO 3.3 SRO 3.1

Initial conditions are that Unit 1 is at 100% RTP and that a worklist item has been generated to place Unit 1 KC system in parallel operation in preparation for Aux Safeguards Testing early next shift. The applicants are directed to place KC in parallel operation per OP/1/A/6400/005 (Component Cooling System) Enclosure 4.4 (Operation of Additional KC Pumps/Parallel Operation). The applicants will ensure RN system miniflow from Unit 2, ensure proper flowpath on Unit 1, place the 1A KC outlet mode switch in the “KC TEMP” position, bypass the letdown mixed bed demineralizers, and start an ‘A’ train KC pump.

JPM I Local ESPS Alignment to 2ETB through 2ATD – NEW JPM

K/A Generic Emergency 055 EA2.03 Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power. (CFR 43.5 / 45.13) RO 3.9 SRO 4.7

Initial conditions are that Unit 2 is in Mode 3 following a Loss of All AC Power. EP/2/A/5000/ECA-0.0 (Loss of All AC Power) has been entered. Neither of the Emergency D/Gs could be started. Station management has determined that power will be restored to 2ETB from ESPS (Emergency Supplemental Power Supply) through 2ATD. The applicants are directed to perform ECA-0.0 Enclosure 52 (Local ESPS Alignment to 2ETB through 2ATD) beginning at Step 3. Applicants will ensure required breakers are open, start ESPS D/Gs 1 and 2, and close breakers required to align power from the ESPS D/Gs through 2ATD to 2ETB.

JPM J Place 1B Hydrogen Recombiner in Service – Bank JPM

K/A System 028 A2.01 Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Hydrogen Recombiner power setting, determined by using plant data book. (CFR 41.5 / 43.5 / 45.3 / 45.13) RO 3.4 SRO 3.6

Initial conditions are that a LOCA has occurred on Unit 1. The applicants are directed to place Hydrogen Recombiner 1B in service at the required power level per OP/1/A/6450/010 (Containment Hydrogen Control System) Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA) steps 3.1 through 3.3.14. Applicants are given current containment pressure and containment hydrogen concentration. When performing the steps to place 1B Hydrogen Recombiner in service, the applicants will be required to determine the required power level using Unit 1 Revised Databook Figure 10. With hydrogen concentration exceeding 3.5%, the applicants will be required to add 4 KW to the value determined from Figure 10 to determine the final power level.

JPM K Break Main Condenser Vacuum Locally – Bank JPM

K/A System 045 A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of secondary plant parameters following a T/G trip.

Initial conditions are that Unit 1 is in Mode 3 following a reactor trip. The applicants are directed to perform AP/1/A/5500/006 (Loss of S/G Feedwater) Enclosure 3 (Local Actions to Break Condenser Vacuum). This JPM is time critical and must be performed in < 10 minutes. The time critical time will end when the first condenser vacuum breaker valve is opened. Once all 3 vacuum breaker valves are opened, the applicant will close Main and Aux Steam to the Air Ejectors.

Facility:	Catawba Nuclear Station	Date of Examination:	Sep 2021
Exam Level: RO ☐ SRO-I ☐ SRO-U ☒		Operating Test Number:	2021301

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

System/JPM Title	Type Code*	Safety Function
a. Emergency Borate the Reactor Coolant System	A, L, P, D, S	1
b. Isolate Cold Leg Accumulators During Shutdown LOCA	A, L, M, S	3
c. Restore CA flow following Feed & Bleed (w/ 1CA-4 closed)	A, EN, L, N, S	4S

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

i. Local ESPS alignment to 2ETB (2ATD) – AP-07 Encl. 52	E, L, N	6
j. Place Hydrogen Recombiner in Service	E, L, D, R	5

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

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

System JPMs

JPM A Emergency Borate the Reactor Coolant System – Bank JPM (Alternate Path) Previously used on 2017 NRC exam (JPM A)

K/A System 004 A2.14 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct control, or mitigate the consequences of those malfunctions or operations: Emergency Boration. (CFR 41.5 / 43.5 / 45.3 / 45.5) RO 3.8 SRO 3.9

Initial conditions are that an ATWS is in progress following a valid reactor trip signal and failure of the reactor to trip automatically or manually. The applicants are directed to initiate emergency boration per EP/1/A/5000/FR-S.1 (Nuclear Power Generation/ATWS) step 4. The applicants will begin by attempting to open 1NV-236B (Boric Acid to NV Pump Suction). This valve will not open and begins the alternate path for this JPM. The applicants will continue with the procedure and start both boric acid transfer pumps. With emergency boration flow not showing greater than 30 GPM, the applicants will align the suction of the NV pumps to the FWST and isolate the NV pump suction from the VCT.

JPM B Isolate Cold Leg Accumulators Following a Shutdown LOCA – Modified Bank JPM (Alternate Path). JPM was modified to change the Cold Leg Accumulators that could not be isolated.

K/A System 006 A1.13 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration). (CFR 41.5 / 45.5) RO 3.5 SRO 3.7

Initial conditions have the Unit in Mode 4 with a shutdown in progress for a refueling outage, when Pressurizer pressure and level begin to unexpectedly lower. The crew has entered AP/1/A/5500/027 (Shutdown LOCA) to address the Reactor Coolant System leak. The applicants are directed to isolate the Unit 1 Cold Leg Accumulators per AP/27 Enclosure 17 (Isolating Cold Leg Accumulators). The applicants will attempt to close all 4 Cold Leg Accumulator isolation valves, but the isolations for 1A and 1D Cold Leg Accumulators will not close. This begins the alternate path for this JPM. The applicants will go on to perform the steps to vent the 1A and 1D Cold Leg Accumulators to containment.

JPM C Restore CA flow following NC System Feed and Bleed – NEW JPM (Alternate Path)

K/A EPE05 EA1.1 Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR 41.7 / 45.5 / 45.6) RO 4.1 SRO 4.0

Initial conditions have Unit 1 in Mode 3 following a loss of all feedwater. NC system bleed and feed have been initiated per EP/1/A/5000/FR-H.1 (Loss of Secondary Heat Sink). A report from maintenance is received, informing the crew that 1A CA pump is ready to be started. The applicants are directed to perform Step 7 to establish CA flow from 1A CA pump. During the initial stages of restoring flow from 1A CA pump, it is discovered that 1CA-4 (CA Pmps Suct From UST) is closed and cannot be opened. This begins the alternate path for this JPM. The applicant will align suction from the RN system, and start the 1A CA pump. Since the CA flow control valves were previously closed when aligning for feed and bleed, the applicants will be sent to Enclosure 7 (S/G CA Flow Restoration) to initiate CA flow to 1A and/or 1B S/Gs.

JPM I Local ESPS Alignment to 2ETB through 2ATD – NEW JPM

K/A Generic Emergency 055 EA2.03 Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power. (CFR 43.5 / 45.13) RO 3.9 SRO 4.7

Initial conditions are that Unit 2 is in Mode 3 following a Loss of All AC Power. EP/2/A/5000/ECA-0.0 (Loss of All AC Power) has been entered. Neither of the Emergency D/Gs could be started. Management has determined that power will be restored to 2ETB from ESPS (Emergency Supplemental Power Supply) through 2ATD. The applicants are directed to perform ECA-0.0 Enclosure 52 (Local ESPS Alignment to 2ETB through 2ATD) beginning at Step 3. Applicants will ensure required breakers are open, start ESPS D/Gs 1 and 2, and close breakers required to align power from the ESPS D/Gs through 2ATD to 2ETB.

JPM J Place 1B Hydrogen Recombiner in Service – Bank JPM

K/A System 028 A2.01 Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Hydrogen Recombiner power setting, determined by using plant data book. (CFR 41.5 / 43.5 / 45.3 / 45.13) RO 3.4 SRO 3.6

Initial conditions are that a LOCA has occurred on Unit 1. The applicants are directed to place Hydrogen Recombiner 1B in service at the required power level per OP/1/A/6450/010 (Containment Hydrogen Control System) Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA) steps 3.1 through 3.3.14. Applicants are given current containment pressure and containment hydrogen concentration. When performing the steps to place 1B Hydrogen Recombiner in service, the applicants will be required to determine the required power level using Unit 1 Revised Databook Figure 10. With hydrogen concentration exceeding 3.5%, the applicants will be required to add 4 KW to the value determined from Figure 10 to determine the final power level.

Facility: Catawba Nuclear Station Date of Examination: Sep 2021

Exam Level: RO ☐ SRO-I ☒ SRO-U ☐ Operating Test Number: 2021301

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

System/JPM Title	Type Code*	Safety Function
a. Emergency Borate the Reactor Coolant System	A, L, P, D, S	1
b. Isolate Cold Leg Accumulators During Shutdown LOCA	A, L, M, S	3
c. Restore CA flow following Feed & Bleed (w/ 1CA-4 closed)	A, EN, L, N, S	4S
d. Perform E-0 Actions to Ensure Complete Containment Isolation	A, M, S	2
e. Align the NS System for Cold Leg Recirculation	A, D, L, S	5
f. Manual Alignment of 1FTB	D, S	6
g. Reset Radiation Monitor Trip Setpoints	D, P, S	7

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

i. Local ESPS alignment to 2ETB (2ATD) – AP-07 Encl. 52	E, L, N	6
j. Place Hydrogen Recombiner in Service	E, L, D, R	5
k. Break Main Condenser Vacuum Locally – Unit 1	E, D	4S

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

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

System JPMs

JPM A Emergency Borate the Reactor Coolant System – Bank JPM (Alternate Path) Previously used on 2017 NRC exam (JPM A)

K/A System 004 A2.14 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct control, or mitigate the consequences of those malfunctions or operations: Emergency Boration. (CFR 41.5 / 43.5 / 45.3 / 45.5) RO 3.8 SRO 3.9

Initial conditions are that an ATWS is in progress following a valid reactor trip signal and failure of the reactor to trip automatically or manually. The applicants are directed to initiate emergency boration per EP/1/A/5000/FR-S.1 (Nuclear Power Generation/ATWS) step 4. The applicants will begin by attempting to open 1NV-236B (Boric Acid to NV Pump Suction). This valve will not open and begins the alternate path for this JPM. The applicants will continue with the procedure and start both boric acid transfer pumps. With emergency boration flow not showing greater than 30 GPM, the applicants will align the suction of the NV pumps to the FWST and isolate the NV pump suction from the VCT.

JPM B Isolate Cold Leg Accumulators Following a Shutdown LOCA – Modified Bank JPM (Alternate Path). JPM was modified to change the Cold Leg Accumulators that could not be isolated.

K/A System 006 A1.13 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration). (CFR 41.5 / 45.5) RO 3.5 SRO 3.7

Initial conditions have the Unit in Mode 4 with a shutdown in progress for a refueling outage, when Pressurizer pressure and level begin to unexpectedly lower. The crew has entered AP/1/A/5500/027 (Shutdown LOCA) to address the Reactor Coolant System leak. The applicants are directed to isolate the Unit 1 Cold Leg Accumulators per AP/27 Enclosure 17 (Isolating Cold Leg Accumulators). The applicants will attempt to close all 4 Cold Leg Accumulator isolation valves, but the isolations for 1A and 1D Cold Leg Accumulators will not close. This begins the alternate path for this JPM. The applicants will go on to perform the steps to vent the 1A and 1D Cold Leg Accumulators to containment.

JPM C Restore CA flow following NC System Feed and Bleed – NEW JPM (Alternate Path)

K/A EPE05 EA1.1 Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR 41.7 / 45.5 / 45.6) RO 4.1 SRO 4.0

Initial conditions have Unit 1 in Mode 3 following a loss of all feedwater. NC system bleed and feed have been initiated per EP/1/A/5000/FR-H.1 (Loss of Secondary Heat Sink). A report from maintenance is received, informing the crew that 1A CA pump is ready to be started. The applicants are directed to perform Step 7 to establish CA flow from 1A CA pump. During the initial stages of restoring flow from 1A CA pump, it is discovered that 1CA-4 (CA Pmps Suct From UST) is closed and cannot be opened. This begins the alternate path for this JPM. The applicant will align suction from the RN system, and start the 1A CA pump. Since the CA flow control valves were previously closed when aligning for feed and bleed, the applicants will be sent to Enclosure 7 (S/G CA Flow Restoration) to initiate CA flow to 1A and/or 1B S/Gs.

JPM D Perform E-0 Actions to Ensure a Complete Containment Isolation – Modified Bank JPM (Alternate Path). JPM was modified by changing the containment penetration that the applicants will need to manually isolate.

K/A System 013 A4.01 Ability to manually operate and/or monitor in the control room: ESFAS-initiated equipment which fails to actuate. (CFR 41.7 / 45.5 to 45.8) RO 4.5 SRO 4.8

Initial conditions have the Unit in Mode 1, 100% RTP. The applicants are informed that they are the OATC, that the BOP has stepped out of the control room, and that the CRS is performing an IPTE brief on Unit 2, and to monitor their control boards. Once the applicants are ready, a Large Break LOCA is inserted. The applicants will verify the immediate actions of EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). The applicants are then directed to continue the actions of E-0. When checking for proper Phase A containment isolation, the applicants will discover failure of automatic actuation and will manually initiate Phase A. This begins the alternate path for this JPM. When the applicants check for proper Monitor Light Panel alignment, they will discover that NV letdown isolation valves 1NV-10A and 1NV-15B and liquid waste penetration isolation valves 1WL-805A and 1WL-807B did not close on the Phase A initiation signal. The applicants will manually close these valves to complete the JPM.

JPM E Align the NS System for Cold Leg Recirculation – Bank JPM (Alternate Path)

K/A System 026 A4.01 Ability to manually operate and/or monitor in the control room: CSS controls (CFR 41.7 / 45.5 to 45.8) RO 4.5 SRO 4.3

Initial conditions are that a LOCA has occurred on Unit 1 and EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation) is in progress. The applicants are directed to align NS for Cold Leg Recirculation per ES-1.3 Enclosure 2 (Aligning NS for Recirculation). The applicants will close the NS pump suctions from the FWST and open the NS Spray header 1A containment isolation valves. The applicants will attempt to open the 1A NS pump suction from the containment sump, but the valve will not open. This begins the alternate path for this JPM. The applicants will then perform the steps to align and start the 1B NS pump and align cooling water to the 1B NS Heat Exchanger.

JPM F Manually Align Essential power to 1FTB from 1ETB per AP/1/A/5500/007 Case 1 Step 10 – Bank JPM

K/A System 062 A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard). (CFR 41.7 / 45.5 to 45.8) RO 3.3 SRO 3.1

Initial conditions are that Unit 1 is operating at 100% RTP when a blackout occurs on 1ETB due to a failure of 1ATD. The 1B D/G is supplying 1ETB. The crew has implemented AP/1/A/5500/007 Case 1 (Loss of Normal Power to an Essential Train). The applicants are directed to perform Step 10. Applicants will determine that 1FTB is de-energized and will use the procedure to reset the 1B D/G load sequencer and close breakers to energize 1FTB from 1ETB.

JPM G Reset Radiation Monitor Trip Setpoints – Bank JPM previously used on 2017 NRC Exam (JPM F)

K/A System 073 A4.02 Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel. (CFR 41.7 / 45.5 to 45.8) RO 3.7 SRO 3.7

Initial conditions are that following a discussion with RP on a premature gaseous waste release termination, that EMF-50L trip setpoints need to be changed. The applicants are directed to change the trip 1 setpoint to 6300 cpm and the trip 2 setpoint to 9000 cpm using OP/0/A/6500/080 (EMF RP86A Output Modules) Enclosure 4.2 (EMF RP86A and RM1000 Trip Setpoint Adjustment).

JPM I Local ESPS Alignment to 2ETB through 2ATD – NEW JPM

K/A Generic Emergency 055 EA2.03 Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power. (CFR 43.5 / 45.13) RO 3.9 SRO 4.7

Initial conditions are that Unit 2 is in Mode 3 following a Loss of All AC Power. EP/2/A/5000/ECA-0.0 (Loss of All AC Power) has been entered. Neither of the Emergency D/Gs could be started. Management has determined that power will be restored to 2ETB from ESPS (Emergency Supplemental Power Supply) through 2ATD. The applicants are directed to perform ECA-0.0 Enclosure 52 (Local ESPS Alignment to 2ETB through 2ATD) beginning at Step 3. Applicants will ensure required breakers are open, start ESPS D/Gs 1 and 2, and close breakers required to align power from the ESPS D/Gs through 2ATD to 2ETB.

JPM J Place 1B Hydrogen Recombiner in Service – Bank JPM

K/A System 028 A2.01 Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Hydrogen Recombiner power setting, determined by using plant data book. (CFR 41.5 / 43.5 / 45.3 / 45.13) RO 3.4 SRO 3.6

Initial conditions are that a LOCA has occurred on Unit 1. The applicants are directed to place Hydrogen Recombiner 1B in service at the required power level per OP/1/A/6450/010 (Containment Hydrogen Control System) Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA) steps 3.1 through 3.3.14. Applicants are given current containment pressure and containment hydrogen concentration. When performing the steps to place 1B Hydrogen Recombiner in service, the applicants will be required to determine the required power level using Unit 1 Revised Databook Figure 10. With hydrogen concentration exceeding 3.5%, the applicants will be required to add 4 KW to the value determined from Figure 10 to determine the final power level.

JPM K Break Main Condenser Vacuum Locally – Bank JPM

K/A System 045 A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of secondary plant parameters following a T/G trip.

Initial conditions are that Unit 1 is in Mode 3 following a reactor trip. The applicants are directed to perform AP/1/A/5500/006 (Loss of S/G Feedwater) Enclosure 3 (Local Actions to Break Condenser Vacuum). This JPM is time critical and must be performed in < 10 minutes. The time critical time will end when the first condenser vacuum breaker valve is opened. Once all 3 vacuum breaker valves are opened, the applicant will close Main and Aux Steam to the Air Ejectors.

C

EPE007 EK1.02 - Reactor Trip
Knowledge of the operational implications of the following concepts as they apply to the reactor trip: (CFR 41.8 / 41.10 / 45.3)
Shutdown margin ....

Given the following Unit 1 initial conditions:

- Unit 1 was at 100% RTP
- DRPI Data B power supply failure is being investigated by IAE
- A Turbine runback has occurred due to a trip of 1A CFPT on lowering vacuum

Subsequently:

- Unit 1 Reactor is manually tripped following the loss of 1B CFPT
- NC Temperature is 548 degrees
- The CRS has transitioned to EP/1/A/5000/ES-0.1 (Reactor Trip Response) and reached step 8:

8. **Verify adequate shutdown margin as follows:**

In accordance with ES-0.1:

DRPI indication _____(1)_____ require emergency boration.

NC Temperature _____(2)_____ require emergency boration.

Which ONE of the following correctly completes the statement above?

A. 1. will
2. will

B. 1. will
2. will NOT

C. 1. will NOT
2. will

D. 1. will NOT
2. will NOT

General Discussion

Following a loss of feedpump turbine runback, Control Rod insertion limits will be violated requiring initiation of boration within 1 hour in order to restore shutdown margin.

Following a reactor trip, ES-0.1 (step 8) will verify adequate shutdown margin by directing a boration if (1) DRPI indication is unavailable, (2) two or more control rods are not fully inserted, or (3) RCS temperature is less than 545 degrees and less than allowable limit.

With the given conditions, DRPI data B power supply failure does not constitute a loss of DRPI and, therefore, does not require emergency boration. RCS temperature is above required value and does not require emergency boration.

Answer A Discussion

Part 1 is plausible if the applicant believes one power supply renders DRPI unavailable.

Part 2 is plausible if the aplicant confuses the temperature requirement of ES-0.1.

Answer B Discussion

Part 1 is plausible if the applicant believes one power supply renders DRPI unavailable.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible if the aplicant confuses the temperature requirement of ES-0.1.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

This K/A is matched because applicants are required to demonstrate knowledge of the step which verifies adequate shutdown margin following a reactor trip (operational implications).

Basis for Hi Cog

The applicant must analyze given conditions as compared to actions which are recalled from memory in order to determine the correct answer.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
ES-0.1, step 8 AP/03, step 12 and Encl. 3

Student References Provided

EPE007 EK1.02 - Reactor Trip

Knowledge of the operational implications of the following concepts as they apply to the reactor trip: (CFR 41.8 / 41.10 / 45.3)

Shutdown margin ....

Remarks/Status

APE008 AK2.02 - Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: (CFR 41.7 / 45.7)
Sensors and detectors ....

Given the following Unit 2 conditions:

- The unit is operating at 50% RTP
- Pzr pressure is 2235 PSIG
- Pzr Relief Tank (PRT) pressure is 20 PSIG
- PRT temperature is 125°F
- A Pzr code safety valve is leaking by its seat

Which ONE of the following correctly identifies the approximate temperature that is indicated on the leaking safety valve discharge RTD?

REFERENCE PROVIDED

A. 123 - 127°F

B. 161 - 165°F

C. 227 - 231°F

D. 258 - 262°F

ILT 21 CNS RO NRC Examination **QUESTION 2** **D**

General Discussion

Throttling is a constant enthalpy process. For this condition, go to the 2250 psia point on the saturation line on the Mollier diagram. Cross the constant enthalpy line to the 35 psia line (20 psig + 15 psi atmos = 35 psia). Follow that line up to the saturation curve. The constant temperature line that ends at that point on the curve establishes the temperature of the fluid.

Answer A Discussion

Temp is too low - the correct temp is 260 degrees F. Plausible: If the applicant thinks that the discharge temperature will be at the same temperature as the PRT fluid.

Answer B Discussion

Temp is too low - the correct temp is 260 degrees F. Plausible: If the applicant reverses the correction for atmospheric pressure by subtracting 14.6 psi from PRT pressure of 20 psig to get 5 psia.

Answer C Discussion

Temp is too low - the correct temp is 260 degrees F. Plausible: If the applicant makes the mistake of not correcting for atmospheric pressure by failing to adding 14.6 psi to the PRT pressure and uses 20 psia.

Answer D Discussion

CORRECT - See discussion above.

Basis for meeting the KA

The KA is matched because when given a PZR vapor space accident the applicant is required to demonstrate knowledge of the detector interrelation by determining its indication.

Basis for Hi Cog

Applicant must use the numerous parameters given, and determine by use of the Mollier Diagram, what the leaking safety valve discharge temperature will be.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6305

Development References
OP-BNT-TH04 pg 35-37

Student References Provided
Steam Tables with Mollier Diagram

APE008 AK2.02 - Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: (CFR 41.7 / 45.7)

Sensors and detectors ....

Remarks/Status

EPE009 EK1.02 - Small Break LOCA
Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: (CFR 41.8 / 41.10 / 45.3)
Use of steam tables ....

Given the following Unit 1 initial conditions:

- A small break LOCA has occurred
- EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant) has been entered
- Neither train of ICCM is available

Subsequently:

- The crew has transitioned to EP/1/A/5000/ES-1.2 (Post LOCA Cooldown and Depressurization)
- S/G PORVs are being used for cooldown
- Current NC pressure is 665 psig
- Core exit thermocouple temperatures are 490°F
- T-Colds are 487.7°F

In accordance with E-1, the value of subcooling is _____(1)_____ .

Based on current conditions, steam header pressure is _____(2)_____ .

Which ONE of the following correctly completes the statements above?

REFERENCE PROVIDED

A. 1. - 10° F
   2. 608 psig

B. 1. - 10° F
   2. 593 psig

C. 1. + 10° F
   2. 608 psig

D. 1. + 10° F
   2. 593 psig

General Discussion

Using Databook Figure 57 the applicant should determine that saturation temperature for 665 psig is ~ 480°F. Therefore, -10°F subcooling is correct. The second part is determined by use of steam tables and obtaining the saturation pressure associated with Tcold and then converting psia to psig.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because the saturation pressure for 487.8°F is ~610 psia. The applicant must convert psia to psig.

Answer B Discussion

CORRECT: See discussion above.

Answer C Discussion

Part 1 is plausible because it is the correct value of saturation. E-1 requires subcooling to be determined based on ICCM (per core exit thermocouples). ICCM includes a 20°F conservatism and requires use of Databook Figure 57 if ICCM is not available.

Part 2 is plausible because the saturation pressure for 487.8°F is ~610 psia. The applicant must convert psia to psig.

Answer D Discussion

Part 1 is plausible because it is the correct value of saturation. E-1 requires subcooling based on ICCM (per core exit thermocouples). ICCM includes a 20°F conservatism and requires use of Databook Figure 57 if not available.

Part 2 is correct.

Basis for meeting the KA

The applicant is required to demonstrate proper use of steam tables, and appropriate conversions, and also demonstrate knowledge of when steam tables may not used per the procedure requirements of a small break LOCA, for determining the correct value of subcooling (operational implication).

Basis for Hi Cog

The applicant is required to analyze supplied information, determine the correct application to use (i.e. graph vs. steam tables), and then chart data separately to arrive at the correct answer.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6899

Development References

Revised Data Book Figure 57 (Reactor Coolant Saturation Cure, Narrow Range),
Steam Tables
OP-CN-PS-CCM, Lesson Plan for Inadequate Core Cooling Monitor, Section 2.10
EP/1/A/5000/E-1, Loss of Reactor or Secondary Coolant, step 2
EP/1/A/5000/ES-1.2 (Post LOCA Cooldown and Depressurization), Step 11

Student References Provided

Databook Figure 57
Steam Tables

EPE009 EK1.02 - Small Break LOCA

Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: (CFR 41.8 / 41.10 / 45.3)

Use of steam tables ....

Remarks/Status

EPE011 2.4.21 - Large Break LOCA
EPE011 GENERIC
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12)

Given the following Unit 1 conditions:

- The crew has entered EP/1/A/5000/FR-P.1 (Response to Imminent Pressurized Thermal Shock) due to a RED path condition on the Reactor Coolant Integrity CSF Status Tree.

FR-P.1 utilizes the parameters of NC pressure and __________ to determine if a large break LOCA has occurred.

Which ONE of the following correctly completes the statement above?

A. RVLIS level

B. S/G pressure

C. NC T-cold temperatures

D. ND flow rate to cold legs

General Discussion

The first step in FR-P.1 determines whether a large break LOCA exists by verifying NC pressure is > 285 psig. If not, and ND flow to C-legs is > 675 gpm, then the procedure guidance is to return to procedure and step in effect due to the accident being a large break LOCA and doesn't present a PTS concern.

Answer A Discussion

Plausible because RVLIS level would be an indicator of a LBLOCA, but could also be abnormal for a SBLOCA, which is a significant PTS concern.

Answer B Discussion

Plausible because low S/G pressure would be an indicator of a LBLOCA, but S/G pressure may be going down for either event, whether a secondary break or primary break.

Answer C Discussion

Plausible because low NC temperature would be an indicator of a LBLOCA, but NC temperature will be reduced as well for a PTS event.

Answer D Discussion

CORRECT. See discussion above.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of parameters used to assess safety function (i.e. PTS).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6307

Development References
EP/1/A/5000/FR-P.1 and bases

Student References Provided

EPE011 2.4.21 - Large Break LOCA

EPE011 GENERIC

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12)

Remarks/Status

APE015/017 2.2.42 - Reactor Coolant Pump (RCP) Malfunctions
APE015/017 GENERIC
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Given the following Unit 1 conditions:

- Unit is in Mode 3
- Rod control is capable of rod withdrawal
- NC loops 1A, 1B, and 1D are in operation
- The crew has entered AP/1/A/5500/008 (Malfunction of Reactor Coolant Pump)
- 1A NC Pump Lower Bearing temperature is currently 190ºF and rising 5ºF per minute

1A NC Pump Lower Bearing temperature will reach trip setpoint in _____(1)_____ .

Following the trip of 1A NCP, entry into the action statement of TS 3.4.5 (RCS Loops – MODE 3) _____(2)_____ required.

Which ONE of the following correctly completes the statements above?

A. 1. 7 minutes
   2. is

B. 1. 7 minutes
   2. is NOT

C. 1. 1 minute
   2. is

D. 1. 1 minute
   2. is NOT

General Discussion

Since the Unit is in Mode 3, AND rod control is capable of rod withdrawal, TS 3.4.5 requires that at least 3 loops be operable (and in operation). The first part of the question is knowledge of RCP trip criteria (lower bearing temp of 225 degrees). The TS required action is per Condition C "One or two required loops not in operation and rod control capable of withdrawal" which has a completion time of 1 hour.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because this would be the correct answer if control rods were not capable of withdrawal.

Answer C Discussion

Part 1 is plausible because this would be the correct answer if given the same conditions for the motor bearing (195 degrees).

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because this would be the correct answer if given the same conditions for the motor bearing (195 degrees).

Part 2 is plausible because this would be the correct answer if control rods were not capable of withdrawal.

Basis for meeting the KA

The applicant is required to recognize TS entry condition upon loss (malfunction) of a Reactor Coolant Pump.

Basis for Hi Cog

This question requires more than one mental step. The applicant must analyze given conditions and then perform a calculation in order to determine the correct answer.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	Bank Question 6606

Development References

AP/1/A/5500/008 (Malfunction of Reactor Coolant Pump), Case I, Step 2
TS 3.4.5 (RCS Loops - Mode 3), Condition C
OP-CN-PS-NCP (Reactor Coolant Pumps Lesson Plan), Section 6.1

Student References Provided

APE015/017 2.2.42 - Reactor Coolant Pump (RCP) Malfunctions
APE015/017 GENERIC
Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

Remarks/Status

APE022 AK3.06 - Loss of Reactor Coolant Makeup

Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: (CFR 41.5, 41.10 / 45.6 / 45.13)

RCP thermal barrier cooling . . . .

Concerning the NCP Thermal Barrier Heat Exchanger:

The Thermal Barrier Heat Exchanger is the _____(1)_____ cooling source for NC Pump seals.

Thermal Barrier Heat Exchanger cooling _____(2)_____ be available following a Hi-Hi Containment pressure signal.

Which ONE of the following correctly completes the statements above?

A. 1. primary
   2. will NOT

B. 1. primary
   2. will

C. 1. backup
   2. will NOT

D. 1. backup
   2. will

General Discussion

Per the NCP lesson plan "The primary design purpose of the thermal barrier heat exchanger is to provide backup cooling in the event injection flow is lost." The component cooling water reactor building isolations close on Hi-Hi Containment pressure, isolating cooling water flow to the thermal barrier heat exchanger.

Answer A Discussion

Part 1 is plausible because this is an aligned source of cooling during normal operation, but seal injection flow is the primary means of pump seal cooling.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because this is an aligned source of cooling during normal operation, but seal injection flow is the primary means of pump seal cooling.

Part 2 is plausible because flow will continue following a Hi Containment pressure signal (not Hi Hi).

Answer C Discussion

CORRECT - See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because flow will continue following a Hi Containment pressure signal (not Hi Hi).

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the use of thermal barrier cooling upon a loss of seal injection and also recall a condition which isolates this cooling medium. The "reasons" aspect is met as follows: the question tests knowledge of what is the function (or reason for) a thermal barrier cooling component - the heat exchanger - it is the backup, not the primary, source of cooling for the RCP seals. Another way that the "reason" aspect is tested is as follows: given the loss of reactor coolant makeup (which is evaluated in both parts of the question, by testing knowledge of the backup source upon a loss of the normal source; i.e., reactor coolant makeup - charging), what is the function of the thermal barrier heat exchanger.

Basis for Hi Cog

Applicant must apply knowledge of system design (thermal barrier cooling) and RCP seals to arrive at the correct answer, including assessment of containment conditions and its effect on thermal barrier cooling.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6901

Development References

OP-CN-PS-NCP (Reactor Coolant Pump Lesson Plan), Section 2.3.1
OP-CN-PSS-KC (Component Cooling Water System Lesson Plan)

Student References Provided

APE022 AK3.06 - Loss of Reactor Coolant Makeup

Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: (CFR 41.5, 41.10 / 45.6 / 45.13)

RCP thermal barrier cooling . . . .

Remarks/Status

APE025 AK2.01 - Loss of Residual Heat Removal System (RHRS)
Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: (CFR 41.7 / 45.7)
RHR heat exchangers ....

Given the following Unit 1 initial conditions:

- Refueling was in progress when a loss of ND occurred
- CRS has implemented AP/1/A/5500/019 (Loss of Residual Heat Removal System) Case III (Loss of ND With Large Vent Path Established)

Subsequently:

- The reason for the loss of ND has been corrected
- Crew is performing Enclosure 8 (Restoring an ND Train To Operation) to place 1A ND train in service

In accordance with Enclosure 8:

The MINIMUM KC flow established to the ND heat exchanger is _____(1)_____ GPM.

Prior to starting the 1A ND pump, 1ND-27 (ND Hx 1A Bypass Ctrl) is placed in the _____(2)_____ position.

Which ONE of the following correctly completes the statements above?

A. 1. 5000
2. CLOSED

B. 1. 5000
2. OPEN

C. 1. 3000
2. CLOSED

D. 1. 3000
2. OPEN

General Discussion

AP/19 Enclosure 8 is used when restoring an ND train to service following a loss of ND. Enclosure 8 step 10 states "Verify KC flow to desired ND heat exchanger - GREATER THAN 5000 GPM."
Enclosure 8 step 17 & 33 state to close the associated ND heat exchanger outlet control and bypass control valves prior to starting the affected ND pump.

Answer A Discussion

CORRECT - See discussion above.

Answer B Discussion

Part 1 is CORRECT.

Part 2 is plausible because this is normally part of the flowpath for ND system flow. However, when starting an ND pump, this valve is closed and the recirc valve is used as the flowpath for the ND pump to ensure that runout conditions do not occur on pump start.

Answer C Discussion

Part 1 is plausible because this is a flow requirement in AP/19. It is the normal minimum ND flow set on the affected heat exchanger bypass control valve. See Enclosure 8 step 24.c.

Part 2 is CORRECT.

Answer D Discussion

Part 1 is plausible because this is a flow requirement in AP/19. It is the normal minimum ND flow set on the affected heat exchanger bypass control valve. See Enclosure 8 step 24.c.

Part 2 is plausible because this is normally part of the flowpath for ND system flow. However, when starting an ND pump, this valve is closed and the recirc valve is used as the flowpath for the ND pump to ensure that runout conditions do not occur on pump start.

Basis for meeting the KA

The KA is matched because applicants are required to demonstrate knowledge of RHR Heat Exchanger operation (interrelation) upon restoration from a loss of RHR.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6309

Development References
AP/1/A/5500/019 Encl. 8

Student References Provided

APE025 AK2.01 - Loss of Residual Heat Removal System (RHRS)

Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: (CFR 41.7 / 45.7)

RHR heat exchangers ....

Remarks/Status

APE026 AA1.06 - Loss of Component Cooling Water (CCW)
Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: (CFR 41.7 / 45.5 / 45.6)
Control of flow rates to components cooled by the CCWS ....

Given the following Unit 1 timeline:

1200
- The Unit is at 100% RTP
- 1B2 KC Pump is in service
- 1B KC flow is 5000 gpm and stable
- 1B KF is in service

1203
- 1KC-15 (1B2 KC Pump Disch) is inadvertently closed

1205
- 1KC-15 is reopened
- No additional KC pumps have been started

Following closure of 1KC-15, 1KC-156 (KF HX 1B Cool Wtr Otlt) _____(1)_____ automatically change position.

Assuming no operator action, once 1KC-15 is reopened 1AD-9 F/6 “KC Train B Single Pump Runout” _____(2)_____ alarm.

Which ONE of the following correctly completes the statements above?

A. 1. will
   2. will

B. 1. will NOT
   2. will

C. 1. will
   2. will NOT

D. 1. will NOT
   2. will NOT

General Discussion

Component Cooling flow to the Spent Fuel Cooling Heat Exchanger is controlled via a manual loader.

Minimum flow valve opens at 3150 gpm decreasing and closes at 5800 gpm rising. Annunciator alarms at 5700 gpm. The minimum flow valve will automatically open following valve closure as flow lowers below 3150 gpm. Once valve is re-opened flow will rise through 5700 gpm (Annunciator) to 5800 gpm and minimum valve closure.

Answer A Discussion

Part 1 is plausible because Component Cooling water is automatically controlled to the majority of components cooled by this system.

Part 2 is correct.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because Component Cooling water is automatically controlled to the majority of components cooled by this system.

Part 2 is plausible because it is reasonable to believe that operation of the mini-flow valve would not cause a pump runout condition.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because it is reasonable to believe that operation of the mini-flow valve would not cause a pump runout condition.

Basis for meeting the KA

The applicant is required to monitor control of flowrates of the CCWS following loss/restoration of CCW.

Basis for Hi Cog

This question requires a calculation (existing flowrate plus miniflow) along with comparison of information recalled from memory (single pump runout alarm setpoint).

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	Modified 19 NRC #33 Bank Question 7105

Development References
OP/1/A/6100/010J (Annunciator Response for Panel 1AD-9), Rev. 70, F/6 OP-CN-PSS-KC (Component Cooling System LP), Rev. 102a, Pg. 20

Student References Provided

APE026 AA1.06 - Loss of Component Cooling Water (CCW)

Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: (CFR 41.7 / 45.5 / 45.6)

Control of flow rates to components cooled by the CCWS ....

Remarks/Status

EPE029 EA1.11 - Anticipated Transient Without Scram (ATWS)
Ability to operate and monitor the following as they apply to a ATWS: (CFR 41.7 / 45.5 / 45.6)
Manual opening of the CRDS breakers ....

Given the following Unit 1 initial conditions:

- Unit is at 100% RTP with surveillance testing in progress
- Reactor Trip Breaker 'A' (RTA) and Bypass Breaker 'B' (BYB) are racked-in and closed

Subsequently:

- A complete loss of feedwater occurred
- All efforts to trip the reactor from the control room were unsuccessful
- Annunciator 1AD-1 A/5 (P-14, S/I OR RX TRIP CAUSES TURBINE TRIP) is LIT
- Operators entered EP/1/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS)
- An AO was dispatched to locally trip the Reactor

Based on the given conditions, the Reactor Trip Breaker 'A' (RTA) _____(1)_____ failed to operate as designed.

If successful in opening all Reactor Trip and Bypass breakers, then per FR-S.1, the AO _____(2)_____ required to open the MG set breakers locally.

Which ONE of the following correctly completes the statements above?

A. 1. shunt trip coil ONLY
   2. is NOT

B. 1. shunt trip AND undervoltage coils
   2. is NOT

C. 1. shunt trip coil ONLY
   2. is

D. 1. shunt trip AND undervoltage coils
   2. is

General Discussion

The following occur when each reactor trip breaker is taken to "TRIP":
Associated Reactor Trip Breaker shunt trip coil is energized
Associated Reactor Trip Bypass Breaker shunt trip coil is energized
An automatic trip signal is generated by SSPS for the respective train and the following occurs:
Associated Reactor Trip Breaker undervoltage coil is de-energized
Associated Reactor Trip Breaker shunt trip coil is energized
Opposite train Reactor Trip Bypass Breaker undervoltage coil is de-energized

Since neither reactor trip breaker opened from the control room, both the shunt trip and undervoltage trip coils failed to operate as designed.

Per FR-S.1, the operator is dispatched to open reactor trip breakers and the M/G set "Motor" and "Generator" breakers.

Answer A Discussion

Part 1 is plausible because the associated RTB will only get a direct signal to energize the shunt trip coil. The undervoltage coil is de-energized by the automatic reactor trip signal generated from SSPS.

Part 2 is plausible because the reactor trip breakers were the only breakers that failed to open from the control room and are normally the only breakers used to determine that the reactor is tripped.

Answer B Discussion

Part 1 is CORRECT.

Part 2 is plausible because the reactor trip breakers were the only breakers that failed to open from the control room and are normally the only breakers used to determine that the reactor is tripped.

Answer C Discussion

Part 1 is plausible because the associated RTB will only get a direct signal to energize the shunt trip coil. The undervoltage coil is de-energized by the automatic reactor trip signal generated from SSPS.

Part 2 is CORRECT.

Answer D Discussion

CORRECT. See discussion above

Basis for meeting the KA

The applicant is required to demonstrate the ability to operate CRDS breakers (MG set motor and generator breakers) as applied to an ATWS condition.

Basis for Hi Cog

Question is of a higher cognitive level due to requiring more than one mental step to determine the correct answer. Given the conditions, the applicant must determine that the action taken to manually trip the reactor from the control room generated a signal sent directly to the shunt trip coil of each breaker, but also generated an automatic reactor trip signal which causes SSPS to generate signals sent to both the shunt trip coil and the undervoltage coil of each reactor trip breaker.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6311

Development References

EP/1/A/5000/FR-S.1 page 6
OP-CN-IC-IPX pages 16 & 17

Student References Provided

EPE029 EA1.11 - Anticipated Transient Without Scram (ATWS)
Ability to operate and monitor the following as they apply to a ATWS: (CFR 41.7 / 45.5 / 45.6)
Manual opening of the CRDS breakers ....

Remarks/Status

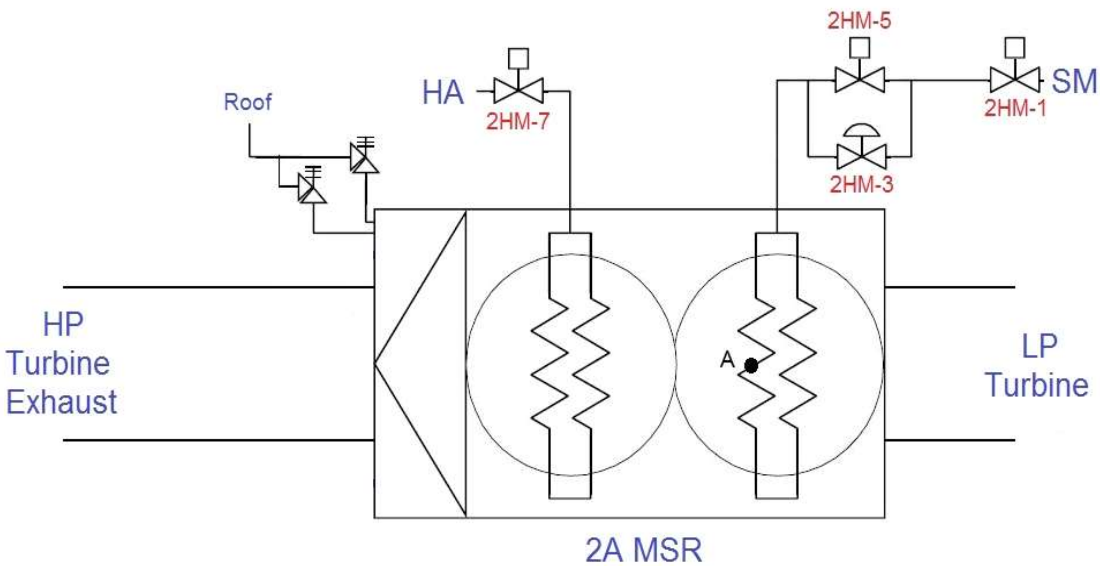
APE040 AA2.02 - Steam Line Rupture

Ability to determine and interpret the following as they apply to the Steam Line Rupture: (CFR: 43.5 / 45.13)

Conditions requiring a reactor trip ....

Given the following Unit 2 conditions:

- The Unit is at 45% RTP
- The crew has entered AP/2/A/5500/028 (Secondary Steam Leak)



A steam leak at location A, will cause Main Turbine Megawatts to _____(1)_____ .

In order to isolate the leak, AP/28 will direct the crew to _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. rise
   2. trip the Main Turbine

B. 1. rise
   2. trip the Reactor and close MSIVs

C. 1. lower
   2. trip the Main Turbine

D. 1. lower
   2. trip the Reactor and close MSIVs

General Discussion

A steam leak inside the moisture separator reheater (MSR) shell from 2nd stage reheat steam will cause a pressure and temperature increase to the low pressure (LP) turbine. This will cause turbine megawatts to increase.
AP-28 Step 15 e. attempts to isolate the steam leak based on location relative to the main turbine. If it is isolable by tripping the main turbine and power is less than 69% the crew will trip the turbine and transition to AP-2. Even though the leak is in the MSR, which is located in the turbine train between the HP and LP turbines, the source of the leak is from Main Steam and is not isolable by tripping the main turbine.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because the physical location of the leak is downstream of the main turbine stop valves, and RX power is less than 69%.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because a steam leak external to the MSR shell, or from the MSR shell itself would result in this response.

Part 2 is plausible because the physical location of the leak is downstream of the main turbine stop valves, and RX power is less than 69%.

Answer D Discussion

Part 1 is plausible because a steam leak external to the MSR shell, or from the MSR shell itself would result in this response.

Part 2 is correct.

Basis for meeting the KA

The KA is met because the applicant is required to determine conditions requiring a reactor trip when given a steam leak by interpreting that the leak is unisolable via turbine trip.

Basis for Hi Cog

The applicant must compare provided information with knowledge recalled from memory (main turbine train layout) in order to determine the location of the steam leak and its effects of turbine operation. This requires more than one mental step.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	17 NRC #10 Bank Question 7107

Development References
OP-CN-MT-MSR (Main Steam LP), Figure 10.1,10.3 AP/2/A/5500/028, Step 16 e

Student References Provided

APE040 AA2.02 - Steam Line Rupture

Ability to determine and interpret the following as they apply to the Steam Line Rupture: (CFR: 43.5 / 45.13)

Conditions requiring a reactor trip ....

Remarks/Status

APE054 AK1.01 - Loss of Main Feedwater (MFW)

Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): (CFR 41.8 / 41.10 / 45.3)

MFW line break depressurizes the S/G (similar to a steam line break) ....

Given the following Unit 2 conditions:

- A seismic event has resulted in the following:
  - 2A S/G has experienced a complete shear of the Main Steam line at the S/G outlet
  - 2D S/G has experienced a complete shear of the Main Feed line at the S/G inlet

Steam Generator _____(1)_____ will lower to 0% WR level FIRST.

Procedural guidance to isolate 2D S/G is contained in _____(2)_____.

Which ONE of the following correctly completes the statements above?

A. 1. 2A
   2. EP/2/A/5000/E-1 (Loss of Reactor or Secondary Coolant)

B. 1. 2A
   2. EP/2/A/5000/E-2 (Faulted Steam Generator Isolation)

C. 1. 2D
   2. EP/2/A/5000/E-1 (Loss of Reactor or Secondary Coolant)

D. 1. 2D
   2. EP/2/A/5000/E-2 (Faulted Steam Generator Isolation)

General Discussion

S/G level will decrease more rapidly for a Main Feed Line Break (as compared to a Main Steam Line Break). Level will decrease to 0% in approximately half of the time.

Following the event, a transition from E-0 to E-2 will be directed based on decreasing S/G pressure. E-2 contains steps to identify and isolate any faulted S/G. Therefore, 2A and 2D S/Gs will be isolated in this procedure. Following isolation, if SI termination is not met the crew will transition to E-1.

Answer A Discussion

Part 1 is plausible if the applicant reasons that the loss of high energy steam depletes S/G level faster than liquid loss.

Part 2 is plausible because E-1 addresses a loss of secondary coolant. Additionally, the crew will enter E-1 following S/G isolation if SI termination is not met.

Answer B Discussion

Part 1 is plausible if the applicant reasons that the loss of high energy steam depletes S/G level faster than liquid loss.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because E-1 addresses a loss of secondary coolant. Additionally, the crew will enter E-1 following S/G isolation if SI termination is not met.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the operational implications and similarities of a Main Feed Line Break (i.e. Loss of Main Feedwater) as compared to a Main Steam Line Break.

Basis for Hi Cog

The applicant is required to analyze given information and compare to knowledge recalled from memory.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6412

Development References

OP-CN-EP-E2 (E-2 Series Lesson Plan), Section 3.1
OP-CN-EFPAM-E2 (Faulted Steam Generator Isolation Exercise), Section D

Student References Provided

APE054 AK1.01 - Loss of Main Feedwater (MFW)

Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): (CFR 41.8 / 41.10 / 45.3)

MFW line break depressurizes the S/G (similar to a steam line break) ....

Remarks/Status

APE056 2.1.30 - Loss of Offsite Power
APE056 GENERIC
Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)

Given the following Unit 2 conditions:

- A Loss of All Offsite Power has occurred
- Both Unit 2 D/Gs started and loaded their associated bus
- While monitoring D/G operating parameters, the local operator notes that D/G 2B "VOLTS" indicates 3925 V

In order to adjust 2B D/G Voltage, local controls will be operated on the 2B Diesel _____(1)_____ Control Panel .

Following this adjustment, 2B D/G 'AMPS' will be _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. Engine
   2. lower

B. 1. Generator
   2. lower

C. 1. Engine
   2. higher

D. 1. Generator
   2. higher

General Discussion

Although most local controls associated with operating a D/G are located on the engine control panel, D/G voltage control is located on the generator control panel.
Additionally, when voltage is increased, amps will decrease.

Answer A Discussion

Part 1 is plausible because most controls and indications necessary for D/G operation are located on the engine control panel.

Part 2 is correct.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because most controls and indications necessary for D/G operation are located on the engine control panel.

Part 2 is plausible since voltage needs to be increased and applicant may conclude that amps will increase if voltage and current relationship is misunderstood.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible since voltage needs to be increased and applicant may conclude that amps will increase if voltage and current relationship is misunderstood.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the location of local controls and determine the effect of operating these controls has on other indications when faced with a loss of offsite power.

Basis for Hi Cog

The applicant must recall a setpoint from memory (normal Essential bus voltage), compare to the value provided, determine required voltage adjustment, and then determine the effects of that adjustment.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	17 NRC #12 Modified

Development References

OP-CN-DG-DG3 (D/G LP), Sect. 6.3, Figure 10

Student References Provided

APE056 2.1.30 - Loss of Offsite Power
APE056 GENERIC
Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)

Remarks/Status

APE062 AA2.01 - Loss of Nuclear Service Water
Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: (CFR: 43.5 / 45.13)
Location of a leak in the SWS ....

Given the following initial conditions:

- 1B RN Pump in service
- 1B1 KC Pump in service
- 2A1 KC Pump in service

Subsequently:

- Both units enter AP/0/A/5500/030 (Plant Flooding), Enclosure 8 (Flooding From RN) following discovery of a large RN leak on the 1A Essential Header
- Per AP/30 guidance, the crew has isolated the 1A RN Essential Header (ONLY)

Based on current conditions:

Cooling water supply _____(1)_____ available to the 1A KD Heat Exchanger.

Mini-Flow protection _____(2)_____ available for the 1B RN Pump.

Which ONE of the following correctly completes the statements above?

A. 1. is
 2. is

B. 1. is
 2. is NOT

C. 1. is NOT
 2. is

D. 1. is NOT
 2. is NOT

General Discussion

Following discovery of flooding, the crew will enter AP/30, Case II (Auxiliary/Diesel Building Flooding) which will direct performance of Enclosure 8 (RN Flooding). Step 4 of Enclosure 8 will direct the crew to perform step 13, based on leak location. This step will isolate the 1A Essential Header. Following this isolation, the 1A D/G cooling water supply will be available due to the system layout (see RN diagram).

RN Mini-flow protection is determined by operation of the CCW system (KC). KC trains in service require the associated HX to be aligned in temperature control mode. The alternate HX will be aligned for mini-flow protection. With the provided conditions, 1A and 2B KC HX are aligned for miniflow protections, since the alternate train KC pumps are in service. Following the isolation of 1A KC HX (within the essential header boundary), and with trains remaining cross connected, the 2B KC HX remains available as mini-flow protection for both units.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because the 1A KC Heat Exchanger was aligned for mini-flow protection. This flowpath is now isolated. The applicant may not realize that miniflow protection remains available through the 2B KC Heat Exchanger.

Answer C Discussion

Part 1 is plausible because the 1A D/G is an essential SWS load. However, this cooling water flow path is not part of the associated essential header.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because the 1A D/G is an essential SWS load. However, this cooling water flow path is not part of the associated essential header.

Part 2 is plausible because the 1A KC Heat Exchanger was aligned for mini-flow protection. This flow path is now isolated. The applicant may not realize that miniflow protection remains available through the 2B KC Heat Exchanger.

Basis for meeting the KA

With a given leak location, the applicant is required to interpret the significance of the leak location (interpret availability of D/G cooling) and the significance of actions taken to address the leak (interpret availability of miniflow protection following mitigating actions)

NOTE: Although the Loss of NSW AOP is not referenced in this question, the listed conditions would meet entry criteria. If entered, the Loss of NSW AOP would direct the crew to the Plant Flooding AOP for mitigation of the given event.

Basis for Hi Cog

The applicant is required to analyze initial conditions to determine mini-flow protection, and then determine the affect of mitigative actions in order to determine the correct availability and flow path.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	17 NRC #13 Bank Question 8016

Development References

OP-CN-PSS-RN (Nuclear Service Water LP), Section 19.2
AP/0/5500/030, Case II, Step 6, and Encl. 8, Steps 4 & 13

Student References Provided

APE062 AA2.01 - Loss of Nuclear Service Water
Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: (CFR: 43.5 / 45.13)
Location of a leak in the SWS ....

Remarks/Status

APE065 AK3.04 - Loss of Instrument Air
Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: (CFR 41.5,41.10 / 45.6 / 45.13)
Cross-over to backup air supplies ....

Given the following Unit 1 initial conditions:

- The crew has entered AP/0/A/5500/022 (Loss of Instrument Air) following a complete loss of VI
- Both Reactors have been tripped

Subsequently:

- Unit 1 has entered EP/1/A/5000/FR-H.1 (Loss of Heat Sink)
- Feed and Bleed criteria has been met

In order to establish Feed and Bleed, motive force will be available to _____(1)_____ Pressurizer PORVs supplied by _____(2)_____ .

Which ONE of the following correctly completes the statement above?

A. 1. two
2. cold leg accumulators

B. 1. two
2. individual air accumulators

C. 1. three
2. cold leg accumulators

D. 1. three
2. individual air accumulators

General Discussion

Although 3 PZR PORVs are available, two receive backup air supply from the "A" and "B" Cold Leg Accumulator Nitrogen volume.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because auxiliary feedwater flow control valves are provided with individual air accumulators to allow operation upon a loss of instrument air.

Answer C Discussion

Part 1 is plausible because three PZR PORVs are installed and operated with instrument air. It is reasonable to believe that opening all three would provide a better bleed path and, therefore, require a backup supply of air for motive force.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because three PZR PORVs are installed and operated with instrument air. It is reasonable to believe that opening all three would provide a better bleed path and, therefore, require a backup supply of air for motive force.

Part 2 is plausible because auxiliary feedwater flow control valves are provided with individual air accumulators to allow operation upon a loss of instrument air.

Basis for meeting the KA

An accumulator is a form of backup air supply; the "crossover" aspect is met in the fact that the cold leg accumulators are aligned to provide a backup function when normal air supply pressure is lost.

The KA is matched because the applicant is presented with an evolution involving a loss of Instrument Air, and then tested on why it is needed. The "reason for" is implied in the requirement to know how many PZR PORVs are supplied with a backup air supply and thus how many are required to be operated under emergency conditions.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

CLA LP Sect. 6.1
CA LP Sect. 7.3
VI LP Sect. 4.1

Student References Provided

APE065 AK3.04 - Loss of Instrument Air

Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: (CFR 41.5,41.10 / 45.6 / 45.13)

Cross-over to backup air supplies ....

Remarks/Status

APE077 AA1.03 - Generator Voltage and Electric Grid Disturbances
Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 41.10 / 45.5, 45.7, and 45.8)
Voltage regulator controls...

Given the following Unit 1 initial conditions:

- The unit is at 100% RTP with Main Generator power factor at 0.99 lagging
- Unit 1 Voltage Regulator is in "Manual" per Engineering's request
- The “READY” status light for the Voltage Regulator Manual/Auto switch is DARK

Subsequently:

- Generator Voltage and MVARs begin fluctuating
- The CRS enters AP/1/A/5500/037 (Generator Voltage and Electric Grid Disturbances), Case I (Abnormal Generator or Grid Voltage)
    - The CRS has directed the OATC to operate the Voltage Regulator to maintain Generator MVARS within the Generator Capability Curve

With the Voltage Regulator in "Manual", operation of the Voltage Adjust RAISE/LOWER pushbutton _____(1)_____ adjust Reactive Power.

Placing the Voltage Regulator Manual/Auto switch in “Auto” (with the “READY” status light dark) _____(2)_____ transfer the Voltage Regulator to automatic control.

Which ONE of the following correctly completes the statements above?

A. 1. will
   2. will

B. 1. will
   2. will NOT

C. 1. will NOT
   2. will

D. 1. will NOT
   2. will NOT

General Discussion

In MANUAL Mode:
The Voltage Adjust button will still function. The setpoint of the field current is adjusted by means of the RAISE/LOWER commands. In no-load operation, the adjustment changes the generator voltage; in operation under load this adjusts the reactive power.

The "READY" light means that the follower circuit is close enough to provide a "bumpless" transfer, however the circuit will transfer even with the light not illuminated if the Manual/Auto switch is operated.

Answer A Discussion

CORRECT: See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible if applicant concludes that since a "bumpless" transfer is not guaranteed, the voltage regulator will NOT transfer.

Answer C Discussion

Part 1 is plausible if the applicant reasons that the Voltage Adjust pushbutton will not function in this condition or reasons that only generator voltage would be changed by this adjustment.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible if the applicant reasons that the Voltage Adjust pushbutton will not function in this condition or reasons that only generator voltage would be changed by this adjustment.

Part 2 is plausible if applicant concludes that since a "bumpless" transfer is not guaranteed, the voltage regulator will NOT transfer.

Basis for meeting the KA

The applicant is required to demonstrate the ability to operate the main generator voltage regulator controls.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6419

Development References

OP-CN-GEN-EGB Lesson Plan, Sect. 4.3.4
AP/37, Step 3

Student References Provided

APE077 AA1.03 - Generator Voltage and Electric Grid Disturbances
Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 41.10 / 45.5, 45.7, and 45.8)
Voltage regulator controls..

Remarks/Status

PLEASE REVIEW

ALTHOUGH THIS QUESTION HAS BEEN PREVIOUSLY REVIEWED AND APPROVED A MINOR EDITORIAL CHANGE HAS BEEN MADE BASED ON VALIDATOR FEEDBACK

The part 2 question was revised to move the clarifying information in parenthesis from the end of the sentence to just prior to the question blank

WE04 EA2.2 - LOCA Outside Containment
Ability to determine and interpret the following as they apply to the (LOCA Outside Containment)
(CFR: 43.5 / 45.13)
Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

Given the following conditions on Unit 1:

- A LOCA outside containment has occurred
- The crew suspects the leak is located in the 1B ND header
- EP/1/A/5000/ECA-1.2 (LOCA Outside Containment) has been entered
- Crew is performing step 2 to attempt to identify and isolate the leak
- Subcooling is -5°F and stable

In accordance with ECA-1.2:

The crew _____(1)_____ permitted to isolate 1B ND header prior to 1A ND header.

The crew _____(2)_____ evaluate RVLIS level to verify the leak is isolated.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. will

B. 1. is NOT
   2. will

C. 1. is
   2. will NOT

D. 1. is NOT
   2. will NOT

General Discussion

ECA-1.2 (LOCA Outside Containment) will first direct the operators to isolate the A ND header and then the B ND header from the NC system as this is the most likely source of the leak. The procedure will next direct isolation of the NI system. Following each attempt, a check of NC pressure, RVLIS level, and Pzr level is performed to verify success of leak isolation. ECA-1.2 used to only check for NC system pressure increasing to determine if the leak was isolated. RVLIS level and Pzr level were added due to the fact that if the NC system is saturated, NC system pressure may not be a good diagnostic of leak isolation (see NOTE prior to step 2 of ECA-1.2).

Answer A Discussion

Part 1 is plausible because ECA-1.2 does contain guidance to independantly isolate various trains and systems for leak identification. It is reasonable to believe that operators would have the oportunity to choose the order of isolation. However, ECA-1.2 specifically prescribe an order of leak checks. Additionally plausible because OMP 1-7 contains guidance which allows operators to take required action to isolate a known leak path.

Part 2 is correct.

Answer B Discussion

CORRECT - See explanation above.

Answer C Discussion

Part 1 is plausible because ECA-1.2 does contain guidance to independantly isolate various trains and systems for leak identification. It is reasonable to believe that operators would have the oportunity to choose the order of isolation. However, ECA-1.2 specifically prescribe an order of leak checks. Additionally plausible because OMP 1-7 contains guidance which allows operators to take required action to isolate a known leak path.

Part 2 is plausible because NC pressure used to be the only parameter evaluated for leak isolation in ECA-1.2.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because NC pressure used to be the only parameter evaluated for leak isolation in ECA-1.2.

Basis for meeting the KA

The applicant is required to interpret given conditions and determine appropriate adherence to the applicable procedure when given a LOCA Outside Containment.

Basis for Hi Cog

The applicant is required to compare information provided to that recalled from memory in order to determine the correct method of the leak isolation decision.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	2019 NRC #16 Modified

Development References

EP/1/A/5000/ECA-1.2, Step 2

Student References Provided

WE04 EA2.2 - LOCA Outside Containment

Ability to determine and interpret the following as they apply to the (LOCA Outside Containment)

(CFR: 43.5 / 45.13)

Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

Remarks/Status

WE05 EK2.2 - Loss of Secondary Heat Sink

Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following:
(CFR: 41.7 / 45.7)

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.

Given the following Unit 1 conditions:

- A Safety Injection due to Hi Containment Pressure has occurred
- Containment pressure peaked at 2.7 psig and is now slowly lowering
- The crew has implemented EP/1/A/5000/FR-H.1 (Response to Loss of Secondary Heat Sink)
- All attempts to restore CA flow have been unsuccessful

In accordance with FR-H.1:

The NEXT source of feed water attempted for restoration of flow to the S/Gs is through the CM/CF system using _____(1)_____.

The crew will be required to establish bleed and feed when W/R level in at least 3 S/Gs is less than a MAXIMUM level of _____(2)_____.

Which ONE of the following correctly completes the statements above?

A. 1. either Main Feed Water pump
   2. 24%

B. 1. either Main Feed Water pump
   2. 36%

C. 1. Hotwell and Booster pumps
   2. 24%

D. 1. Hotwell and Booster pumps
   2. 36%

General Discussion

If CA flow is not available, FR-H.1 will first attempt to place MFPs in service followed by a depressurization and attempts to feed from the Hotwell and Booster pumps.

Bleed and Feed initiation criteria is 3 S/Gs less than 24% or 36% (ACC). Adverse Containment is 3 psig in containment.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because this would be the correct answer if adverse containment condition numbers were in effect.

Answer C Discussion

Part 1 is plausible because the Hotwell and Booster pumps could be used if S/Gs were first depressurized and are an option specified in FR-H.1 (following attempts to place MFPs in service).

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because the Hotwell and Booster pumps could be used if S/Gs were first depressurized and are an option specified in FR-H.1 (following attempts to place MFPs in service).

Part 2 is plausible because this would be the correct answer if adverse containment condition numbers were in effect.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of heat removal system prioritization and initiation setpoints upon a loss of secondary heat sink.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 8221

Development References
OP-CN-EP-FPH (FR-H Lesson Plan), Section 3.2 EP/1/A/5000/FR-H.1 (Response to Loss of Secondary Heat Sink), Enclosure 1, Step 1

Student References Provided

WE05 EK2.2 - Loss of Secondary Heat Sink

Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following:
(CFR: 41.7 / 45.7)

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.

Remarks/Status

WE11 EK3.4 - Loss of Emergency Coolant Recirculation

Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation)
(CFR: 41.5 / 41.10, 45.6, 45.13)

RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Given the following Unit 1 conditions:

- EP/1/A/5000/ECA-1.1 (Loss of Emergency Coolant Recirculation) is in progress
- NC Pressure is 1700 psig
- FWST level is trending down

_____(1)_____ are currently providing injection flow into the NC System.

Per ECA-1.1, as the FWST level lowers less than _____(2)_____ the operator will secure these pumps.

Which ONE of the following correctly completes the statements above?

A. 1. NV AND NI Pumps
   2. 20%

B. 1. NV AND NI Pumps
   2. 5%

C. 1. ONLY NV Pumps
   2. 20%

D. 1. ONLY NV Pumps
   2. 5%

General Discussion

NI pumps begin to discharge to the NC system at approx. 1520 psig.

Per Enclosure 1 if ECA-1.1, all pumps NV and NI pumps taking suction from the FWST are secured at 5% level.

Answer A Discussion

Part 1 is plausible if the applicant is unaware of the discharge pressure of the NI pumps.

Part 2 is plausible because this is the setpoint at which the ND pumps are re-aligned to the containment sump.

Answer B Discussion

Part 1 is plausible if the applicant is unaware of the discharge pressure of the NI pumps.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because this is the setpoint at which the ND pumps are re-aligned to the containment sump.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

For a given Loss of Emergency Coolant Recirculation, the applicant is required to demonstrate knowledge of the reponses and actions necessary to be performed in order to ensure procedures and limitations of the facility license are adhered to .

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6558

Development References

OP-CN-ECCS-NI Lesson Plan,
Sect. 7.1
EP/1/A/5000/ECA-1.1, Encl. 1, Step 2
EP/1/A/5000/E-1, Encl. 1, Step 6

Student References Provided

WE11 EK3.4 - Loss of Emergency Coolant Recirculation

Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation)
(CFR: 41.5 / 41.10, 45.6, 45.13)

RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

Remarks/Status

EARLY REVIEW QUESTION

APE036 AK1.03 - Fuel Handling Incidents
Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents : CFR 41.8 / 41.10 / 45.3)
Indications of approaching criticality ....

Given the following Unit 1 initial conditions:

- Unit is in Mode 6 performing core loading
- Unit 1 TRN A SMM BORON DILUTION INTLKS switch is in the “Enable” position
- Unit 1 TRN B SMM BORON DILUTION INTLKS switch is in the “Defeat” position
- The VCT Outlet Valve Interlock Keyswitch is in the “Normal” position

Subsequently:

- The OATC has just reset Shutdown Margin Monitor setpoints on 1A and 1B trains
- An error has resulted in mispositioning of several fuel assemblies
    - This mispositioning has resulted in a critical array and rising count rate

1AD-2 E(F)/2 “Train A(B) Shutdown Margin Alarm” will actuate once count rate rises by a MINIMUM factor of _____(1)_____ .

Following alarm actuation, 1A and 1B NV Pump suction will be aligned to the _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. two
   2. VCT

B. 1. two
   2. FWST

C. 1. four
   2. VCT

D. 1. four
   2. FWST

General Discussion

The Boron Dilution Mitigation System will alarm and generate system actuation upon reaching a count rate of two times background. This actuation will secure reactor makeup water pumps and align NV pump suction to the FWST (borated water source). Although the system consists of two trains, only one train actuation is required to generate a complete alignment to the FWST and secure both Reactor Makeup Water Pumps.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible if the applicant reasons that a complete transfer to the FWST will not occur if one train of BDMS is defeated.

Answer B Discussion

CORRECT. See discussion above.

Answer C Discussion

Part 1 is plausible because this was the correct actuation setpoint in the recent past.

Part 2 is plausible if the applicant reasons that a complete transfer to the FWST will not occur if one train of BDMS is defeated.

Answer D Discussion

Part 1 is plausible because this was the correct actuation setpoint in the recent past.

Part 2 is correct.

Basis for meeting the KA

The applicant is required to demonstrate knowlede of the operational implications related to a fuel handling incident along with indications of approaching criticality.

Basis for Hi Cog

The applicant is required compare information provided with that recalled from memory in order to determine correct system operation.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
ENC LP, Sect. 3, 10.1

Student References Provided

APE036 AK1.03 - Fuel Handling Incidents

Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents : CFR 41.8 / 41.10 / 45.3)

Indications of approaching criticality ....

Remarks/Status
EARLY REVIEW QUESTION

APE037 AA1.07 - Steam Generator (S/G) Tube Leak
Ability to operate and / or monitor the following as they apply to the Steam Generator Tube Leak: (CFR 41.7 / 45.5 / 45.6)
CVCS letdown flow indicator ....

Given the following Unit 1 initial conditions:

- Unit is at 100% RTP
- Letdown flow is 85 gpm aligned through the 1NV-10A (Letdn Orif 1B Otlt Cont Isol)

Subsequently:

- 1A S/G develops a tube leak
- The crew has entered AP/1/A/5500/010 (Reactor Coolant Leak), Case I (Steam Generator Tube Leak)
- 1NV-294 (NV Pmps A&B Disch Flow Ctrl) has been fully opened
    - Pressurizer level continues to lower at 0.1% / minute
- The CRS desires letdown flow reduction to 45 gpm

Prior to this event, 1NV-849 (Letdn Flow Var Orif Ctrl) SLIM station was in _____(1)_____ .

Per AP/10, letdown flow reduction will be accomplished via the _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. manual
   2. 45 gpm orifice

B. 1. manual
   2. variable orifice

C. 1. automatic
   2. 45 gpm orifice

D. 1. automatic
   2. variable orifice

General Discussion

Variable flow orifice letdown control valve is the one SLIM controlled valve which remains in manual control at all time.

AP/10 provides options for using either the 45 gpm orifice or the variable orifice for letdown flow reduction. However, if the 45 gpm orifice is isolated (which it is for normal ops), the procedure provides guidance to use the variable orifice.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because this would accomplish the intended function, and guidance is provided in AP/10. However, AP/10 direct use of the variable orifice if this flowpath is currently isolated which it would be under these conditions.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because virtually all control room SLIM stations remain in auto with exception of NV-849.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because virtually all control room SLIM stations remain in auto with exception of NV-849.

Part 2 is plausible because this would accomplish the intended function, and guidance is provided in AP/10. However, AP/10 direct use of the variable orifice if this flowpath is currently isolated which it would be under these conditions.

Basis for meeting the KA

Given conditions related to a SGTL, the applicant is required to demonstrate the ability to operate/monitor the variable orifice letdown flow control and indication.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References	Student References Provided
NV LP Sect. 2.1.3 AP/10, Case I, Step 2 RNO	

APE037 AA1.07 - Steam Generator (S/G) Tube Leak

Ability to operate and / or monitor the following as they apply to the Steam Generator Tube Leak: (CFR 41.7 / 45.5 / 45.6)

CVCS letdown flow indicator ....

Remarks/Status

APE051 AK3.01 - Loss of Condenser Vacuum

Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: (CFR 41.5,41.10 / 45.6 / 45.13)

Loss of steam dump capability upon loss of condenser vacuum ....

Concerning the Main Condensers:

In order to prevent over pressurization, Condenser Steam Dump operation requires a MINIMUM vacuum of _____(1)_____ which is sensed in the _____(2)_____ Main Condensers.

Which ONE of the following correctly completes the statements above?

A. 1. 15” Hg
   2. “A” OR “B”

B. 1. 16.9” Hg
   2. “A” OR “B”

C. 1. 15” Hg
   2. “A” AND “B”

D. 1. 16.9” Hg
   2. “A” AND “B”

General Discussion

In order to prevent Main Condenser overpressurization, a permissive (C-9) is required to be met in order to allow condenser steam dump operation. This permissive requires a minimum of 15" Hg vacuum in two (A & B) of the three condenser sections along with condenser cooling water flow sensed by 1 of 4 condenser cooling water pump breakers closed.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible if the applicant believes that vacuum in only one condenser is acceptable to receive steam dumps. Logic requirements of most plant permissives do not require complete agreement (i.e. most are 2/4, 2/3, or 1/2).

Answer B Discussion

Part 1 is plausible because this value represents the Main Feedwater Pump low vacuum trip setpoint.

Part 2 is plausible if the applicant believes that vacuum in only one condenser is acceptable to receive steam dumps. Logic requirements of most plant permissives do not require complete agreement (i.e. most are 2/4, 2/3, or 1/2).

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is plausible because this value represents the Main Feedwater Pump low vacuum trip setpoint.

Part 2 is correct.

Basis for meeting the KA

The knowledge of why steam dump capability is lost upon low condenser vacuum is below the threshold of plausibility. The applicant is required to demonstrate knowledge of how the reason (overpressurization) is prevented.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	17 Audit #60 Bank Question 6563

Development References
IDE LP Sect. 3.1.1

Student References Provided

APE051 AK3.01 - Loss of Condenser Vacuum

Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: (CFR 41.5,41.10 / 45.6 / 45.13)

Loss of steam dump capability upon loss of condenser vacuum ....

Remarks/Status

APE059 AA2.03 - Accidental Liquid Radioactive-Waste Release

Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: (CFR: 43.5 / 45.13)

Failure modes, their symptoms, and the causes of misleading indications on a radioactive-liquid monitor ....

Given the following conditions:

- A planned Liquid Waste Release of Waste Monitor Tank (WMT) A was initiated at 1110

The following timeline of events then occurs:

1120 1RAD-1, C/5 (EMF-49 LIQUID WASTE DISCH HI RAD) alarms
1130 The release is manually re-initiated without re-sampling
1145 1RAD-1, C/5 (EMF-49 LIQUID WASTE DISCH HI RAD) alarms
1155 The release is manually re-initiated without re-sampling
1215 1RAD-1, C/5 (EMF-49 LIQUID WASTE DISCH HI RAD) alarms

The release _____(1)_____ be manually re-initiated, without re-sampling, per OP/0/B/6500/113 (Operations Liquid Waste Release).

The release was automatically isolated by closure of _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. can
2. 1WL-X28

B. 1. can
2. 1WL-124

C. 1. can NOT
2. 1WL-X28

D. 1. can NOT
2. 1WL-124

General Discussion

The Operating Procedure contains a Limit and Precaution that allows 3 initiations of a radioactive liquid waste release without re-sampling, if it is interrupted by EMF-49 high rad trips. An EMF-49 Hi Rad alarm will close 1WL-124 to terminate the release.

Answer A Discussion

Part 1 is plausible if the applicant confuses the guidance given in the operating procedure for Operations Liquid Waste Release. It specifies that 3 initiations of a radioactive liquid waste release without re-sampling, are permitted, if it is interrupted by EMF-49 high rad trips. However, a important point is that this includes the original initiation as one of the three. It is plausible that the stem conditions could be interpreted incorrectly, if the applicant notes that only TWO re-initiations have occurred, and ONE more is allowed. This is incorrect, since THREE total initiations have occurred, and no more are allowed.

Part 2 is plausible because 1WL-X28 does receive an auto isolation signal from an associated rad monitor.. However, this valve will isolate a discharge from the Monitor Tank Building (vs Auxiliary Building).

Answer B Discussion

Part 1 is plausible if the applicant confuses the guidance given in the operating procedure for Operations Liquid Waste Release. It specifies that 3 initiations of a radioactive liquid waste release without re-sampling, are permitted, if it is interrupted by EMF-49 high rad trips. However, a important point is that this includes the original initiation as one of the three. It is plausible that the stem conditions could be interpreted incorrectly, if the applicant notes that only TWO re-initiations have occurred, and ONE more is allowed. This is incorrect, since THREE total initiations have occurred, and no more are allowed.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because 1WL-X28 does receive an auto isolation signal from an associated rad monitor.. However, this valve will isolate a discharge from the Monitor Tank Building (vs Auxiliary Building).

Answer D Discussion

CORRECT - See discussion above.

Basis for meeting the KA

The applicant is required to determine and interpret the cause of a misleading indication (EMF spike) on the radioactive monitor used for liquid waste release.

Basis for Hi Cog

This is a high cognitive level question because the applicant must analyze conditions, apply system knowledge, apply knowledge of procedural requirements, and determine a required action, and a valve designed to close.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 7259

Development References	Student References Provided
WL LP, Sect. 28.1 OP/0/B/6500/113 L&P	

APE059 AA2.03 - Accidental Liquid Radioactive-Waste Release

Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: (CFR: 43.5 / 45.13)

Failure modes, their symptoms, and the causes of misleading indications on a radioactive-liquid monitor ....

Remarks/Status

WE14 EK3.2 - High Containment Pressure
Knowledge of the reasons for the following responses as they apply to the (High Containment Pressure)
(CFR: 41.5 / 41.10, 45.6, 45.13)
Normal, abnormal and emergency operating procedures associated with (High Containment Pressure).

Given the following Unit 1 initial conditions:

- The unit is at 100% RTP
- Lower Containment Ventilation Units (LCVUs) 1A, 1B, 1D are operating in LOW speed
- Upper Containment Ventilation Unit (UCVU) 1A is operating in NORM

Subsequently:

- Containment humidity begins rising
- Containment temperature and pressure are slowly rising
- Containment radiation levels are normal
- The crew enters AP/1/A/5500/028 (Secondary Steam Leak)

Per AP/28, 1C LCVU will be started in _____(1)_____ speed and placed in _____(2)_____ cooling.

Which ONE of the following correctly completes the statement above?

A. 1. LOW
   2. MAX

B. 1. LOW
   2. NORM

C. 1. HIGH
   2. MAX

D. 1. HIGH
   2. NORM

General Discussion

With the conditions given, AP/28 will direct operaators to:
- Start all LCVUs in low speed
- Start all UCVUs
- Place all UCVUs and LCVUs in MAX cooling

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Plausible to believe that with slowly rising containment pressure additonal units started in normal will provide mitigation.

Answer C Discussion

Plausible to believe that this procedure will direct all available actions to mitigate rising containment pressure to be taken.

Answer D Discussion

Plausible to believe that increasing fan speed to high while maintaining normal cooling water flow would mitigate a slowly rising containment pressure trend.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of reasons for responses as applied to an abnormal procedure entered due to high containment pressure. The reasons for response is inherant to the knowledge of the actions required by the procedure.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	13 NRC #40

Development References
AP/28, Step 9 RNO

Student References Provided

WE14 EK3.2 - High Containment Pressure

Knowledge of the reasons for the following responses as they apply to the (High Containment Pressure)
(CFR: 41.5 / 41.10, 45.6, 45.13)

Normal, abnormal and emergency operating procedures associated with (High Containment Pressure).

Remarks/Status

EPE074 EK2.09 - Inadequate Core Cooling

Knowledge of the interrelations between the Inadequate Core Cooling and the following : (CFR 41.7 / 45.7)

Controllers and positioners ....

Given the following Unit 1 conditions:

- A Loss of Off-Site Power (LOOP) has occurred
- Due to multiple equipment failures, the crew has implemented EP/1/A/5000/FR-C.1 (Response To Inadequate Core Cooling)
- ECCS steam pressure “BLOCK” pushbuttons have been depressed
- Operators are preparing to depressurize intact steam generators to 140 PSIG

Based on the conditions above:

The CRS will direct the operating crew to establish a _____(1)_____ cooldown rate.

The cooldown and depressurization will be performed using the _____(2)_____.

Which ONE of the following correctly completes the statements above?

A. 1. 100 °F/hour
   2. steam dumps

B. 1. 100 °F/hour
   2. S/G PORVs

C. 1. maximum
   2. steam dumps

D. 1. maximum
   2. S/G PORVs

General Discussion

Per FR-C.1 Step 17d, Dump steam from intact S/G(s) to condenser at maximum rate while attempting to avoid a Main Steam Isolation.
With a loss of offsite power, no RC pumps will be in operation so the condenser is not available for dumping steam to.
Step 16.d RNO directs dumping steam from all intact S/Gs at maximum rate using the S/G PORVs.

Answer A Discussion

Part 1 is plausible since 100 degrees per hour is the TS cooldown limit and cooldown rate required in most of the EP procedures (FR-C.2, ES-1.2, etc.).

Part 2 is plausible since steam dumps are the preferred method and would be used if a LOOP had not occurred.

Answer B Discussion

Part 1 is plausible since 100 degrees per hour is the TS cooldown limit and cooldown rate required in most of the EP procedures (FR-C.2, ES-1.2, etc.).

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible since steam dumps are the preferred method and would be used if a LOOP had not occurred.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

With a given Inadequate Core Cooling condition, the applicant is required to demonstrate knowledge of the available controller / positioner used to conduct a
cooldown and the associated procedural requirement for cooldown rate.

Basis for Hi Cog

This question is high cog because it requires more than one mental step. The applicant is required to analyze the conditions in the stem (LOOP) and determine what affect this will have on plant operations required to mitigate this event. Then recall from memory the correct cooldown rate.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 7261

Development References

FR-C.1
FR-C.2

Student References Provided

EPE074 EK2.09 - Inadequate Core Cooling
Knowledge of the interrelations between the Inadequate Core Cooling and the following : (CFR 41.7 / 45.7)
Controllers and positioners ....

Remarks/Status

WE02 EA1.3 - SI Termination
Ability to operate and / or monitor the following as they apply to the (SI Termination)
(CFR: 41.7 / 45.5 / 45.6)
Desired operating results during abnormal and emergency situations.

Given the following Unit 1 conditions:

- A medium break LOCA has occurred
- EP/1/A/5000/E-1 (Loss of Reactor Coolant or Secondary Coolant) has been implemented and the crew is evaluating Safety Injection termination criteria
- The BOP reports the following data:
  - NC pressure is 1200 PSIG and STABLE
  - Containment pressure is 3.3 PSIG and trending down
  - NC subcooling is 2°F
  - Pressurizer level is 17% and STABLE

Based on S/I termination criteria of E-1:

NC Pressure requirement _____(1)_____ met.

Pressurizer level requirement _____(2)_____ met.

Which ONE of the following correctly completes the statement above?

A. 1. is
2. is

B. 1. is NOT
2. is

C. 1. is
2. is NOT

D. 1. is NOT
2. is NOT

General Discussion

SI termination criteria are:
Subcooling >0
One S/G >11% (29% ACC) or total feed flow >450 gpm
NC pressure stable or increasing
PZR level > 11% (30% ACC)

Required values in parenthesis apply to ACC condition (greater than 3 psig containment pressure). These values do apply.

For the conditions specified RCS pressure requirement is met because stable (or increasing) pressure is required.
PZR level requirement is not met due to ACC conditions in containment.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because this would be the correct answer if ACC conditions were not established.

Answer B Discussion

Part 1 is plausible because it is reasonable to believe that increasing pressure would be required to terminate Safety Injection. With all possible injections paths in operation, the applicant may reason that termination would result in uncontrollable pressure once SI pumps/paths are secured.

Part 2 is plausible because this would be the correct answer if ACC conditions were not established.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is plausible because it is reasonable to believe that increasing pressure would be required to terminate Safety Injection. With all possible injections paths in operation, the applicant may reason that termination would result in uncontrollable pressure once SI pumps/paths are secured.

Part 2 is plausible because this would be the correct answer if ACC conditions were not established.

Basis for meeting the KA

The applicant is required to determine if monitored operating parameters meet those desired for SI termination during an emergency situation.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step.

First, it requires the applicant to recall from memory all of the Safety Injection termination criteria (including adverse numbers).

Next, the applicant must evaluate the conditions given against the recalled knowledge to determine if the Safety Injection termination criteria are met.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
E-1 Step 9

Student References Provided

WE02 EA1.3 - SI Termination
Ability to operate and / or monitor the following as they apply to the (SI Termination)
(CFR: 41.7 / 45.5 / 45.6)
Desired operating results during abnormal and emergency situations.

Remarks/Status

WE15 EA2.2 - Containment Flooding
Ability to determine and interpret the following as they apply to the (Containment Flooding)
(CFR: 43.5 / 45.13)
Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

Given the following Unit 1 initial conditions:

- The Unit is at 100% RTP

Subsequently:

1100 A LOCA occurs
1215 Containment sump level is 13 feet and slowly rising

If containment sump level is rising at a constant rate of 0.25 feet per minute, EP/1/A/5000/FR-Z.2 (Response to Containment Flooding) entry will be REQUIRED at _____(1)_____ .

Valves, inside containment, NOT qualified for submergence will be isolated by an _____(2)_____ signal.

Which ONE of the following correctly completes the statements above?

A. 1. 1225
2. S_T

B. 1. 1245
2. S_T

C. 1. 1225
2. S_P

D. 1. 1245
2. S_P

General Discussion

A containment sump level of 15.5 ft requires an Orange Path entry into FR-Z.2. A level of 13 ft could be reached if FWST was in sump and ice melted, which would be expected after ~ one hour. Containment Sump level instrumentation span is from .5 feet to 20.5 feet. Valves that are not qualified for submergence due to inadequate operator seals get a close signal on an St and therefore have time to re-position before they are submerged.

Answer A Discussion

CORRECT: Orange path for Containment sump level is 15.5 feet which will require 10 minutes to reach at .25 ft/min and valves not qualified for submergence will close on an St signal.

Answer B Discussion

Part 1 is plausible because 20,5 feet is top of scale for Containment Sump level instrumentation. Off scale high would indicate flooding and if actual setpoint for orange path is not known, student could remember and choose top of scale. At .25 feet per minute, it would take 30 minutes to increase sump level to 20.5 feet.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because an Sp signal will close some containment isolation valves and is always expected to be receive following a large break LOCA.

Answer D Discussion

Part 1 is plausible because 20,5 feet is top of scale for Containment Sump level instrumentation. Off scale high would indicate flooding and if actual setpoint for orange path is not known, student could remember and choose top of scale. At .25 feet per minute, it would take 30 minutes to increase sump level to 20.5 feet.

Part 2 is plausible because an Sp signal will close some containment isolation valves and is always expected to be receive following a large break LOCA.

Basis for meeting the KA

The question deals with containment flooding and monitoring of containment sump level. The required knowledges are the entry conditions for containment flooding CSF procedure (WE15) and evaluates an operating characteristic of the facility (valves that are not qualified for submergence receive a closed signal on an St signal)

Basis for Hi Cog

Comprehension because the student must know the level that requires entry into FR-Z2 and calculate the time to reach that level based on the input rate.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 7263

Development References

Lesson Plan OP-CN-CNT-CNT, Sect. 2.7
EP/1/A/5000/F-0, Containment Status Tree

Student References Provided

WE15 EA2.2 - Containment Flooding

Ability to determine and interpret the following as they apply to the (Containment Flooding)
(CFR: 43.5 / 45.13)

Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

Remarks/Status

WE03 2.1.20 - LOCA Cooldown and Depressurization
WE03 GENERIC
Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

Given the following Unit 1 conditions:

- A LOCA has occurred
- Containment pressure peaked at 2.8 PSIG, and is now 2.2 PSIG and slowly lowering
- Crew has entered EP/1/A/5000/ES-1.2 (Post LOCA Cooldown and Depressurization) and is performing the initial cooldown

LOOP DATA

		LOOP A	LOOP B	LOOP C	LOOP D
CURRENT T-COLD, BEST	(DEG F)	546.0	546.4	546.1	546.1
ADMINSTRATIVE LIMIT	(DEG F)	479.8	480.2	479.8	479.8
TECH SPEC LIMIT	(DEG F)	459.8	460.2	459.8	459.8
T-COLD CURRENT MINUS 1 HR T-COLD MAXIMUM	(DEG F)	-12.8	-13.6	-12.3	-12.1
15-MIN RATE	(DEG F/HR)	-38	-38	-36	-36
5-MIN RATE	(DEG F/HR)	-131	-134	-128	-127
1-MIN RATE	(DEG F/HR)	-113	-107	-111	-102

In accordance with ES-1.2:

The **INITIAL** cooldown will be started using the _____(1)_____.

With rates established, per the graphic above, the cooldown _____(2)_____ continue at this time.

Which ONE of the following correctly completes the statements above?

A. 1. S/G PORVs
   2. can NOT

B. 1. S/G PORVs
   2. can

C. 1. Condenser Steam Dumps
   2. can NOT

D. 1. Condenser Steam Dumps

General Discussion

Containment pressure peaking at a value > 3 PSIG causes a Main Steam Isolation Signal to be initiated which will subsequently close all Main Steam Isolation Valves. These valves being closed prevents the steam dumps from being immediately available to start the initial cooldown in ES-1.2. However, since containment pressure did not exceed 3 psig MSIVs will remain open and steam dumps will be used for cooldown.

Cooldown rate in ES-1.2 is as close as possible without exceeding 100°F per hour. Exceeding 100°F/hr rate on the 1-MIN & 5-MIN above is allowed as long as the T-Cold current minus 1 Hr T-Cold maximum does NOT exceed 100°F.

Answer A Discussion

Part 1 is plausible because S/G PORVs would be used if containment pressure exceeded 3 psig resulting in main steam isolation.

Part 2 is plausible because these values do exceed a 100°F/hr rate, but do not exceed the limit of 100°F cooldown in an hour.

Answer B Discussion

Part 1 is plausible because S/G PORVs would be used if containment pressure exceeded 3 psig resulting in main steam isolation.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because these values do exceed a 100°F/hr rate, but do not exceed the limit of 100°F cooldown in an hour.

Answer D Discussion

CORRECT - See discussion above.

Basis for meeting the KA

The K/A is matched because the applicants are tested on their ability to interpret the correct operation necessary to successfully execute procedure steps of Post LOCA Cooldown and Depressurization.

Basis for Hi Cog

This question is higher cognitive due to requiring analyzing conditions in the stem to determine that a Main Steam Isolation has occurred and comparing that to system knowledge of the Main Steam system to determine the correct answer.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	19 NRC #43 MODIFIED Bank Question 8346

Development References

EP/1/A/5000/ES-1.2 rev 36 step 10
EP/1/A/5000/ES-0.3 rev 14 step 4
EP/1/A/5000/ES-3.1 rev 20 step 7
OP-CN-STM-SM rev 102b page 21

Student References Provided

WE03 2.1.20 - LOCA Cooldown and Depressurization

WE03 GENERIC

Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)

Remarks/Status

SYS003 A3.01 - Reactor Coolant Pump System (RCPS)
Ability to monitor automatic operation of the RCPS, including: (CFR: 41.7 / 45.5)
Seal injection flow ....

Given the following Unit 1 conditions:

- The Unit is at 100% RTP
- Total charging flow is currently 90 gpm
- 1NV-294 (NV Pmps A&B Disch Flow Ctrl) is in MANUAL
- 1NV-309 (Seal Water Injection Flow) is in AUTO

Assuming stable plant conditions, as 1NV-294 is throttled CLOSED, 1NV-309 will throttle in the _____(1)_____ direction in order to maintain _____(2)_____ seal injection flow.

Which ONE of the following correctly completes the statements above?

A. 1. OPEN
   2. 32 gpm

B. 1. CLOSED
   2. 32 gpm

C. 1. OPEN
   2. 40 gpm

D. 1. CLOSED
   2. 40 gpm

General Discussion

RCP Seal Injection Control is accomplished via a backpressure control valve. More Charging water will flow to the RCP seals as the control valve throttles closed and vice versa. Therefore, assuming stable plant conditions, as charging flow is decreased, 1NV-309 will throttle closed to allow more less flow to the RCS while maintaining the same seal injection flow.

NV-309 is automatically set to maintain 32 gpm seal injection flow.

Answer A Discussion

Part 1 is plausible if the applicant reverses the logic of the backpressure control valve.

Part 2 is correct.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible if the applicant reverses the logic of the backpressure control valve.

Part 2 is plausible because 40 gpm is the TS value for maximum seal injection flow with the charging flow control valve fully open.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because 40 gpm is the TS value for maximum seal injection flow with the charging flow control valve fully open.

Basis for meeting the KA

The applicant is required to demonstrate the ability to monitor automatic operation of the seal injection flow control valve.

Basis for Hi Cog

This question requires more than one mental step:
1. The applicant must compare provided information with that recalled from memory to determine the effect, upon seal injection flow, of increasing charging.
2. The applicant must recall (from memory) operation of the Pzr level master as related to manual operation of the charging flow control valve and apply this information to given conditions.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	17 NRC #28 Bank Question 8031

Development References

OP-CN-PS-NV (Chemical and Volume Control System LP), Sect. 2.3.9
T.S. 3.5.5

Student References Provided

SYS003 A3.01 - Reactor Coolant Pump System (RCPS)
Ability to monitor automatic operation of the RCPS, including: (CFR: 41.7 / 45.5)
Seal injection flow ....

Remarks/Status

SYS004 K2.06 - Chemical and Volume Control System
Knowledge of bus power supplies to the following: (CFR: 41.7)
Control instrumentation ....

Concerning operation of the Unit 1 Volume Control Tank (VCT):

A loss of power to 1LT-5761 (VCT Level CH 1) _____(1)_____ result in a DCS Alternate Action.

A loss of 1ERPA will result in a loss of _____(2)_____ makeup capability to the VCT.

Consider each statement separately

Which ONE of the following correctly completes the statements above?

A. 1. will
   2. auto ONLY

B. 1. will
   2. auto AND manual

C. 1. will NOT
   2. auto ONLY

D. 1. will NOT
   2. auto AND manual

General Discussion

Most DCS control systems have 3 or 4 inputs and require a loss of at least 2 channels in order to initiate alternate action. VCT level is an exception. It is fed by only 2 channels. Therefore, one bad channel will result in alternate action.

A loss of 1ERPA will result in a loss of Auto AND Manual makeup capability to the VCT.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because it is partially true. However, a loss of 1ERPA also results in a loss of manual makeup capability.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because all other DCS control systems require a loss of at least two channels in order to initiate alternate action.

Part 2 is plausible because it is partially true. However, a loss of 1ERPA also results in a loss of manual makeup capability.

Answer D Discussion

Part 1 is plausible because all other DCS control systems require a loss of at least two channels in order to initiate alternate action.

Part 2 is correct.

Basis for meeting the KA

Applicant is required to demonstrate knowledge of a system response related to a loss of power to CVCS control instrumentation.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
NV LP Sect. 2.2.2 & 4.2

Student References Provided

SYS004 K2.06 - Chemical and Volume Control System
Knowledge of bus power supplies to the following: (CFR: 41.7)
Control instrumentation ....

Remarks/Status

SYS005 K6.03 - Residual Heat Removal System (RHRS)
Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: (CFR: 41.7 / 45.7)
RHR heat exchanger ....

Given the following Unit 2 initial conditions:

- Unit is in Mode 4
- 2A ND train in service in RHR Mode
- 2B ND train remains in Injection Mode

Subsequently:

- Instrument Air is lost to 2ND-26 (ND Hx 2A Outlet Ctrl)

Assuming no operator acton:

ND system flow _____(1)_____ automatically adjust to compensate for the change caused by this malfunction.

2A ND Heat Exchanger outlet temperature will _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. will
2. rise

B. 1. will NOT
2. rise

C. 1. will
2. lower

D. 1. will NOT
2. lower

General Discussion

The loss of instrument air to 2ND-26 will cause the valve to fail open resulting in more flow through the heat exchanger and a lower outlet temperature. The bypass valve (2ND-27) will automatically throttle closed in order to maintain flow at setpoint.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because many air operated valves fail closed upon loss of air supply. This failure mode would result in a higher heat exchanger outlet temperature.

Answer B Discussion

Part 1 plausible if the applicant does not recall that 2ND-27 automatically controls system flow or confuses 2ND-27 and 2ND-26 (which is only manually controlled).

Part 2 is plausible because many air operated valves fail closed upon loss of air supply. This failure mode would result in a higher heat exchanger outlet temperature.

Answer C Discussion

CORRECT: See explanation above.

Answer D Discussion

Part 1 plausible if the applicant does not recall that 2ND-27 automatically controls system flow or confuses 2ND-27 and 2ND-26 (which is only manually controlled).

Part 2 is correct.

Basis for meeting the KA

KA is matched because the candidate must understand how flow is controlled through the ND System Heat Exchangers during shutdown cooling mode and how failures associated with the flow/temperature control valves will effect overall system parameters.

Basis for Hi Cog

This is a higher cognitive level question because the candidate must analyze the effect of the loss of VI to 2ND-26 and once the candidate determines that the valve fails open determine how this malfunction will be mitigated.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	16 Audit #31 Bank Question 6435

Development References	Student References Provided
ND LP Section 2.8	

SYS005 K6.03 - Residual Heat Removal System (RHRS)

Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: (CFR: 41.7 / 45.7)

RHR heat exchanger ....

Remarks/Status

SYS006 A1.15 - Emergency Core Cooling System (ECCS)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: (CFR: 41.5 / 45.5)

RWST Level and temperature ....

Given the following Unit 2 conditions:

- Unit is at 75% RTP
- Several banks of FWST heaters have failed "ON"
- Current FWST temperature is 91°F
- FWST temperature is rising at a rate of 0.5°F/min

Based on the conditions above, FWST temperature will reach the T.S. 3.5.4 (Refueling Water Storage Tank (RWST)) limit in a MINIMUM of _____(1)_____ minutes.

Normally, Group 1 FWST heaters cycle automatically to maintain FWST temperature greater than T.S. 3.5.4 MINIMUM of _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. 10
   2. 65°F

B. 1. 10
   2. 70°F

C. 1. 18
   2. 65°F

D. 1. 18
   2. 70°F

General Discussion

The following parameters are associated with the FWST:

- Minimum Temperature - 70°F
- Maximum Temperature - 100°F

At approximately 75°F, 40 KW of heaters are energized.
At approximately 72°F, the remaining 80 KW of heaters are energized.
At 72F, if selected to Auto, all heaters will be energized and "FWST Emerg Lo Temp" annunciator alarms.
All Heaters are de-energized at approximately 80°F.
High temperature cutoff at 200°F at the heater. This detector is in the heater element.
Each group of heaters has a High temperature cutoff reset switch provided to reset.

Answer A Discussion

Part 1 is plausible because Refueling Water Line Temp provide an OAC alarm at 96 degrees.

Part 2 is plausible because this value represents the minimum Boric Acid Tank solution temperature.

Answer B Discussion

Part 1 is plausible because Refueling Water Line Temp provide an OAC alarm at 96 degrees.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because this value represents the minimum Boric Acid Tank solution temperature.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

The applicant is required to monitor changes in parameters to prevent exceeding limits (Tech Spec) associated with RWST level and temperature.

Basis for Hi Cog

This question is high cog because the applicant must recall from memory the FWST maximum and minimum temperature limit of T.S 3.5.4, Then calculate, based on the given trend, the time required to exceed that limit.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 7504

Development References

TS 3.5.4
SLC 16.9-12
FW LP, Sect. 2.6

Student References Provided

SYS006 A1.15 - Emergency Core Cooling System (ECCS)
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: (CFR: 41.5 / 45.5)
RWST Level and temperature ....

Remarks/Status

SYS007 K3.01 - Pressurizer Relief Tank/Quench Tank System (PRTS)
Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: (CFR: 41.7 / 45.6)
Containment ....

Given the Unit 1 conditions:

- A load rejection resulted in a reactor trip from 100% RTP
- Following the trip, a Pressurizer Safety valve opens, and will NOT reseat
- The PRT rupture disks function as designed
- Containment pressure is 0.1 psig and rising at 0.03 psig every 5 minutes
- Lower Containment temperature is 110°F and rising at 2°F every 5 minutes

Assuming these conditions remain constant,

Plant conditions will FIRST require entry into LCO _____(1)_____ .

30 minutes from now, conditions for entry into _____(2)_____ will be met.

Which ONE of the following correctly completes the statements above?

LEGEND:
LCO 3.6.4 (Containment Pressure)
LCO 3.6.5 (Containment Air Temperature)

A. 1. 3.6.4
2. 3.6.4 ONLY

B. 1. 3.6.4
2. 3.6.4 AND 3.6.5

C. 1. 3.6.5
2. 3.6.5 ONLY

D. 1. 3.6.5
2. 3.6.5 AND 3.6.4

General Discussion

Although the Containment pressure administrative limit is 0.25 psig, the actual 3.6.4 LCO limit is 0.3 psig. With given conditions, Containment pressure would reach 0.3 psig is approximately 33.6 minutes.

The Lower Containment temperature limit of LCO 3.6.5 is 120 F. With given conditions, this limit would be reached in 25 minutes.

Answer A Discussion

Part 1 is plausible if the applicant confuses the Containment pressure administrative limit with the TS LCO limit.

Part 2 is plausible if the applicant confuses the Containment pressure administrative limit with the TS LCO limit and also confuses the setpoint of the Containment temperature LCO limit.

Answer B Discussion

Part 1 is plausible if the applicant confuses the Containment pressure administrative limit with the TS LCO limit.

Part 2 is plausible if the applicant confuses the Containment pressure administrative limit with the TS LCO limit but correctly applies the Containment temperature limit.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is plausible if the applicant confuses the Containment pressure administrative limit with the TS LCO limit.

Part 2 is plausible if the applicant confuses the Containment pressure administrative limit with the TS LCO limit but correctly applies the Containment temperature limit.

Basis for meeting the KA

The KA is matched because the operator must demonstrate Knowledge of the effect that a loss the PRTS (Rupture Disk Releases) will have on the Containment (i.e. TS Limits are challenged).

Basis for Hi Cog

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information and then use the information to predict an outcome, in order to answer the question correctly.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References

LCO 3.6.4
LCO 3.6.5
VQ LP Sect. 1.5.2
ARP 1AD-19 C/9

Student References Provided

SYS007 K3.01 - Pressurizer Relief Tank/Quench Tank System (PRTS)

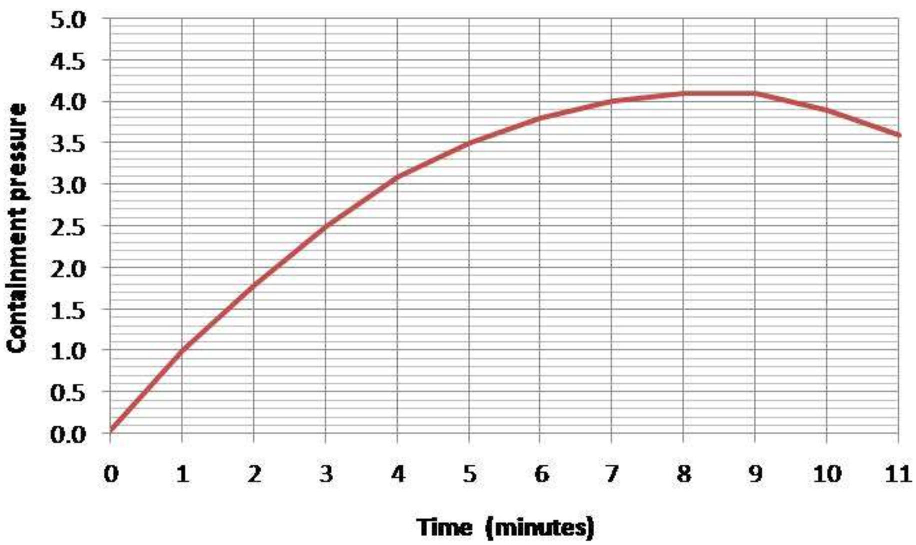
Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: (CFR: 41.7 / 45.6)

Containment ....

Remarks/Status

SYS007 K4.01 - Pressurizer Relief Tank/Quench Tank System (PRTS)
Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Quench tank cooling ....

A LOCA has occurred on Unit 1 at Time = 0 minutes. Given the following containment pressure trend:



At Time = 3 minutes, the PRT _____(1)_____ be cooled using the NCDT Heat Exchanger.

At Time = 3 minutes, the PRT _____(2)_____ be cooled using spray flow from the RMWST.

Which ONE of the following correctly completes the statements above?

A. 1. can
   2. can

B. 1. can
   2. can NOT

C. 1. can NOT
   2. can

D. 1. can NOT
   2. can NOT

General Discussion

The following PRT- related valves will close on a Phase A (ST) Isolation Signal:

1(2)NC-56B (RMW Pump Disch Cont Isol)
1(2)NC-53B (N2 To PRT Cont Isol)
1(2)NC-54A (N2 To PRT Cont Isol)

KC water to the NCDT heat exchanger will also be isolated on a Phase A. With both Reactor Makeup Water and KC to the NCDT Heat Exchanger, there is no available method to cool the PRT with the Phase A signal present.

Answer A Discussion

Part 1 is plausible because the NCDT Heat Exchanger is in containment and, therefore, not isolated following a LOCA. However, cooling water flow will be isolated upon a Phase A containment Isolation signal.

Part 2 is plausible if the applicant confuses the two containment isolation signals (Phase A and Phase B) and reasons that the RMWST containment isolation valve closes at Phase B containment isolation setpoint of 3.0 psig.

Answer B Discussion

Part 1 is plausible because the NCDT Heat Exchanger is in containment and, therefore, not isolated following a LOCA. However, cooling water flow will be isolated upon a Phase A containment Isolation signal.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2is plausible if the applicant confuses the two containment isolation signals (Phase A and Phase B) and reasons that the RMWST containment isolation valve closes at Phase B containment isolation setpoint of 3.0 psig.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the a design feature which provides for (isolates) quench tank cooling of the PRT.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 1334

Development References
PRT LP Sect. 5.6

Student References Provided

SYS007 K4.01 - Pressurizer Relief Tank/Quench Tank System (PRTS)
Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Quench tank cooling ....

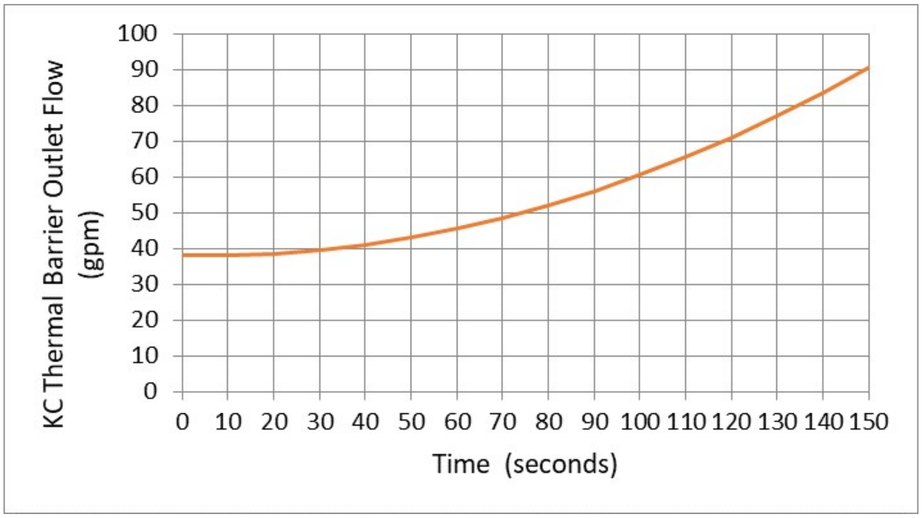
Remarks/Status

SYS008 K4.02 - Component Cooling Water System (CCWS)
Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Operation of the surge tank, including the associated valves and controls ..

Unit 1 is in Mode 3 when the following alarm is received:

- 1AD-6, E/1 (NCP A Thermal Barrier KC Outlet Hi/Lo Flow)

The flow trend is given below:



(1) At what time on the above graph did 1KC-394A (NC Pump 1A Therm Bar Otlt) automatically close?

(2) If the NCP 1A thermal barrier cannot be isolated from the KC System by any means, how is over pressurization of the KC surge tanks prevented?

A. 1. 100 seconds
   2. The KC surge tanks vent line is large enough to prevent over pressurization.

B. 1. 130 seconds
   2. The KC surge tanks vent line is large enough to prevent over pressurization.

C. 1. 100 seconds
   2. The KC surge tanks relief valve to the KC drain sump is large enough to prevent over pressurization.

D. 1. 130 seconds
   2. The KC surge tanks relief valve to the KC drain sump is large enough to prevent over pressurization.

General Discussion

The thermal barrier KC outlet valve will auto close at >/= 60 gpm for 30 seconds (time delay) Overpressurization of surge tanks is prevented by vent piping design size Correct flow and time delay required.

Answer A Discussion

Part 1 is plausible if the applicant is not aware of the 30 second delay and believes flow will isolate at 60 gpm.

Part 2 is correct.

Answer B Discussion

CORRECT: See discussion above.

Answer C Discussion

Part 1 is plausible if the applicant is not aware of the 30 second delay and believes flow will isolate at 60 gpm.

Part 2 is plausible because overflow is directed to the KC drain sump and a relief valve is a common tank overpressure device

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because overflow is directed to the KC drain sump and a relief valve is a common tank overpressure device

Basis for meeting the KA

KA is matched because the applicant must demonstrate knowledge of CCW surge tank and design features which provide overpressure protection in the event of leakage into the system.

Basis for Hi Cog

This is a higher cognitive level question because the applicant must compare data provided in graph form to that recalled from memory and perform a calculation to determine the correct answer.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	14 Audit #35

Development References	Student References Provided
OP-CN-PSS-KC LP Sect. 2.7	

SYS008 K4.02 - Component Cooling Water System (CCWS)

Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)

Operation of the surge tank, including the associated valves and controls ..

Remarks/Status

SYS010 K2.01 - Pressurizer Pressure Control System (PZR PCS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
PZR heaters ....

Given the following Unit 1 conditions:

- The crew has entered AP/1/A/500/017 (Loss of Control Room) due to a Security Event
- Transfer to the SSF has been completed

Based on the conditions above, NC System Pressure control will be accomplished via use of PZR Heater Group _____(1)_____ which will be powered from _____(2)_____.

Which ONE of the following correctly completes the statement above?

A. 1. A
   2. 1LXH

B. 1. A
   2. SMXG

C. 1. D
   2. 1LXH

D. 1. D
   2. SMXG

General Discussion

The backup heaters are either on or off - they cannot have their output varied like the C heaters. They can be manually energized but, after a Blackout condition, that is delayed 12 minutes. Groups A, B, D can be controlled from MCB and Groups A & B can be controlled from the Auxiliary Shutdown Panels (ASP). The one bank in Group D that is powered from the SSF can be controlled there too.

Answer A Discussion

Part 1 is plausible because this would be the correct answer if control was transferred to the Aux Shutdown Panels.

Part 2 is plausible because this is a "B" train power supply and powers PZR Group B heaters.

Answer B Discussion

Part 1 is plausible because this would be the correct answer if control was transferred to the Aux Shutdown Panels.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because this is a "B" train power supply and powers PZR Group B heaters.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

The KA is matched because the applicant must demonstrate knowledge of PZR heater bus power supplies.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

IPE LP, Backup Heaters

Student References Provided

SYS010 K2.01 - Pressurizer Pressure Control System (PZR PCS)
Knowledge of bus power supplies to the following: (CFR: 41.7)
PZR heaters ....

Remarks/Status

SYS010 K6.01 - Pressurizer Pressure Control System (PZR PCS)
Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: (CFR: 41.7 / 45.7)
Pressure detection systems ....

Given the following Unit 1 conditions:

- Unit is at 100% RTP
- A slight cooldown of the NC system causes the "C" PZR heaters to be full "on"
- A malfunction of two PZR pressure transmitters causes an Alternate Action to occur on the Pressurizer Pressure Control System

Assuming NO operator actions:

The PZR Pressure Master will be in _____(1)_____ control AND “C” Heaters _____(2)_____ be energized.

Which ONE of the following correctly completes the statement above?

A. 1. Manual
2. will

B. 1. Manual
2. will NOT

C. 1. Automatic
2. will

D. 1. Automatic
2. will NOT

General Discussion

Alternate Action in Selected PZR Pressure-1 results in the following:
PZR Pressure Master controller goes to manual with last good value (C heater operation and spray valve position do not change)
All automatic PORV actuation is blocked
Blocks high and low pressure control deviation alarms
Blocks low pressure backup heaters on at 25 psig < setpoint (nominal 2235 psig)

Alternate Action in Selected PZR Pressure-2 results in the following:
All automatic PORV actuation is blocked
Blocks the high pressure alarm at 2310 psig

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because most control functions are blocked (i.e. PORV and Backup heater).

Answer C Discussion

Part 1 is plausible because the Distributed Control System is capable of automatic operation with multiple suspect inputs (by selection of quality and validation of channels). The applicant may be unaware that Alternate Action removes the Automatic function.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because the Distributed Control System is capable of automatic operation with multiple suspect inputs (by selection of quality and validation of channels). The applicant may be unaware that Alternate Action removes the Automatic function.

Part 2 is plausible because most control functions are blocked (i.e. PORV and Backup heater).

Basis for meeting the KA

The applicant is required to demonstrate knowledge of effect a given malfunction of PZR pressure detectors will have upon the Pressurizer Pressure Control System.

Basis for Hi Cog

The applicant is required to compare information provided with that recalled from memory in order to determine correct system alignment.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6438

Development References
OP-CN-PS-IPE (Pressurizer Pressure Control Lesson Plan), Section 6.4

Student References Provided

SYS010 K6.01 - Pressurizer Pressure Control System (PZR PCS)
Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: (CFR: 41.7 / 45.7)
Pressure detection systems ....

Remarks/Status

SYS012 A4.04 - Reactor Protection System (RPS)
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
Bistable, trips, reset and test switches ....

Given the following Unit 1 conditions:

- Reactor startup is in progress
- The permissive P-6 status light on 1SI-18 has just LIT
- Reactor power is rising

In accordance with PT/0/A/4150/019 (1/M Approach to Criticality), the operator will manually block the _____(1)_____ high flux reactor trip. Following this, the reactor trip setpoint for high flux is _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. Source Range
2. 10%

B. 1. Source Range
2. 25%

C. 1. Intermediate Range
2. 10%

D. 1. Intermediate Range
2. 25%

General Discussion

The referenced procedure directs that when the P-6 light is lit the operator will block the SR high flux trip. With the Source Range trip blocked, the trip setpoint now shift to Intermediate Range at 25%.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because 10% is a more conservative setpoint since reactor power is well below 10% when P-6 status (10-5 cps) is reached.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible if the applicant confuses P-6 and P-10 bistable. Intermediate range high flux trip will be blocked following P-10 status.

Part 2 is plausible because 10% is a more conservative setpoint since reactor power is well below 10% when P-6 status (10-5 cps) is reached.

Answer D Discussion

Part 1 is plausible if the applicant confuses P-6 and P-10 bistable. Intermediate range high flux trip will be blocked following P-10 status.

Part 2 is correct.

Basis for meeting the KA

The applicant is required to demonstrate the ability to manually operate an RPS reset based on indication of a bistable.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	12 NRC #37

Development References

PT/0/A/4150/019, Encl. 13.4, Step 13.4.1
IPX LP

Student References Provided

SYS012 A4.04 - Reactor Protection System (RPS)
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
Bistable, trips, reset and test switches ....

Remarks/Status

SYS013 K1.13 - Engineered Safety Features Actuation System (ESFAS)
Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
HVAC ....

Given the following initial conditions:

- Both units are at 100% RTP

Subsequently:

- An inadvertent 1A Train Safety Injection occurs

As a result of this event and assuming no operator actions:

The 1B Aux Building Unfiltered Exhaust Fan (ABUFXF) _____(1)_____ secured.

The 2A Aux Building Unfiltered Exhaust Fan (ABUFXF) _____(2)_____ secured.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. is

B. 1. is
   2. is NOT

C. 1. is NOT
   2. is

D. 1. is NOT
   2. is NOT

General Discussion

Upon receipt of a Safety Injection signal to initiate LOCA operation the following train related actions take place in the VA System (i.e. a Unit 1 Train A Safety Injection will perform the below actions for Unit 1 A Train and Unit 2 A Train Components.):

-Filtered Exhaust Fans continue operating or start if they are shut down
-Filter Units operate in the filtered mode of operation
-Unfiltered Exhaust Fans trips which cause the associated Supply Fans to trip

The following actions also occur on a Safety Injection signal, but these signals are unitrelated:

-Unit and train related ASP supply units are tripped off and then restarted.
-Trips Unit Related Filter Room Exhaust Fans.

For this question, SI initiation effects upon the Untiltered Exhaust Fans are train related rather than unit related. The 2A fan would be secured but the 1B fan would continue to operate.

Answer A Discussion

This answer choice is plausible because it is reasonable to believe all unfiltered fans would secure for any SI initiation since the Auxiliary Building is open and shared between both units.

Answer B Discussion

This answer choice is plausible because it is reasonable to believe that VA system actions would be unit related since some responses are unit related (i.e. ASP supply units and Filter Room exhaust fans).

Answer C Discussion

CORRECT - See explanation above.

Answer D Discussion

This answer choice is plausible because it is reasonable to believe that VA system actions would occur only to the train and unit related to the actuation.

Basis for meeting the KA

The applicant is jrequired to demonstrate knowledge of cause effect relationship via Auxiliary Building Ventilation (HVAC) operation following an ESFAS actuation.

Basis for Hi Cog

The applicant is required to compare information provided to that recalled from memory in order to correctly answer correctly

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
VA LP Sect. 10.1

Student References Provided

SYS013 K1.13 - Engineered Safety Features Actuation System (ESFAS)

Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

HVAC ....

Remarks/Status

SYS022 A4.04 - Containment Cooling System (CCS)
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
Valves in the CCS ....

Concerning operation of the Containment Ventilation Units:

Under normal conditions, Containment Ventilation Units are cooled by the _____(1)_____ system.

Containment Ventilation cooling water supply is isolated by a _____(2)_____ signal.

Which ONE of the following correctly completes the statements above?

A. 1. YV
   2. Phase A

B. 1. YV
   2. Phase B

C. 1. RN
   2. Phase A

D. 1. RN
   2. Phase B

General Discussion

The YV system provides normal cooling to containment components. However, these components can also be cooled by the Nuclear Service Water system upon a loss of YV. The shared flowpath is isolated on a Phase B signal.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because many containment isolations occur on Phase A signal.

Answer B Discussion

CORRECT: See explanation above.

Answer C Discussion

Part 1 is plausible because RN does provide Containment Ventilation cooling. However, it is not the normal cooling medium.

Part 2 is plausible because many containment isolations occur on Phase A signal.

Answer D Discussion

Part 1 is plausible because RN does provide Containment Ventilation cooling. However, it is not the normal cooling medium.

Part 2 is correct.

Basis for meeting the KA

The applicant must demonstrate knowledge of monitoring CCS valves and which are providing cooling under varying conditions. The applicant must also demonstrate knowledge of when CCS valves isolate cooling water flow.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6639

Development References
OP-CN-CNT-VV (Containment Ventilation System Lesson Plan), Section 4.1 and 4.2 OP-CN-PSS-RN (Nuclear Service Water Lesson Plan), Section 8.4

Student References Provided

SYS022 A4.04 - Containment Cooling System (CCS)

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

Valves in the CCS ....

Remarks/Status

SYS025 A1.03 - Ice Condenser System

Ability to predict and/or monitor changes in parameters associated with operating the ice condenser system controls including: (CFR: 41.5 / 45.5)

Glycol flow to ice condenser air handling units ....

Given the following Unit 1 conditions:

- Operators are performing Aux Safeguards testing
- A spurious automatic signal caused 1NF-233B (Containment Return Isolation) to inadvertently close during the testing

1NF-233B was closed by an inadvertent _____(1)_____ signal.

The glycol expansion tank _____(2)_____ overflow inside containment.

Which ONE of the following correctly completes the statements above?

A. 1. S_T
   2. will NOT

B. 1. S_P
   2. will NOT

C. 1. S_T
   2. will

D. 1. S_P
   2. will

General Discussion

1NF-233B receives a Phase A Containment Isolation Signal (St) to close.
There is a bypass valve around 1NF-233B, however it is designed to bleed pressure back into containment following a containment isolation signal.
With 1NF-233B only closed, the glycol pumped into containment will fill the glycol expansion tank until it overflows to the containment floor and equipment sump.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because there is a bypass line around 1NF-233B, however it bleeds excess glycol pressure back into containment following a containment isolation.

Answer B Discussion

Part 1 is plausible because 1NF-233B does receive a containment isolation signal to close. It is however, an St signal and not an Sp signal that closes it.

Part 2 is plausible because there is a bypass line around 1NF-233B, however it bleeds excess glycol pressure back into containment following a containment isolation.

Answer C Discussion

CORRECT - See discussion above.

Answer D Discussion

Part 1 is plausible because 1NF-233B does receive a containment isolation signal to close. It is however, an St signal and not an Sp signal that closes it.

Part 2 is correct.

Basis for meeting the KA

The applicant is required to determine which control function isolates glycol flow to the ice condenser air handling units and describe the changes in parameters, within containment, following glycol isolation.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6347

Development References	Student References Provided
OP-CN-CNT-NF pages 22, 26, & 32	

SYS025 A1.03 - Ice Condenser System
Ability to predict and/or monitor changes in parameters associated with operating the ice condenser system controls including: (CFR: 41.5 / 45.5)
Glycol flow to ice condenser air handling units ....

Remarks/Status

SYS025 K5.01 - Ice Condenser System

Knowledge of operational implications of the following concepts as they apply to the ice condenser system: (CFR: 41.5 / 45.7)
Relationships between pressure and temperature ....

Given the following Unit 1 conditions:

- Unit is in Mode 4
- It has been determined that eight Ice Condenser Intermediate Deck doors will not open due to excessive ice buildup

Based on the conditions listed above, peak pressure following a LOCA will be reached _____(1)_____ than normal.

Tech Spec 3.6.13 (Ice Condenser Doors) _____(2)_____ applicable.

Which ONE of the following correctly completes the statements above?

A. 1. sooner
   2. is

B. 1. later
   2. is

C. 1. sooner
   2. is NOT

D. 1. later
   2. is NOT

General Discussion

With 8 ice condenser doors blocked from opening with ice buildup, the overall capability of the ice condenser is reduced, and peak containment pressure following an accident would occur sooner than normal.

TS 3.6.13 is applicable in Modes 1 - 4.

Answer A Discussion

CORRECT - See discussion above.

Answer B Discussion

Part 1 is plausible if the applicant believes that with ice buildup on the intermediate deck doors, it would increase the capability of the ice condenser (more ice) and would therefore result in peak containment pressure being reached later than normal.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because it is reasonable to believe the lower RCS temperatures of Mode 4 would not require function of the ice condenser.

Answer D Discussion

Part 1 is plausible if the applicant believes that with ice buildup on the intermediate deck doors, it would increase the capability of the ice condenser (more ice) and would therefore result in peak containment pressure being reached later than normal.

Part 2 is plausible because it is reasonable to believe the lower RCS temperatures of Mode 4 would not require function of the ice condenser.

Basis for meeting the KA

The K/A is matched because applicants are tested on their knowledge of operational implications of ice buildup blocking ice condenser intermediate deck doors, causing a reduction in the cooling capability and resulting in reaching peak containment pressure sooner than normal following an accident.

Basis for Hi Cog

Question is higher cognitive due to requiring the applicants to analyze conditions in the stem, determine the effect on cooling capability and eventually peak containment pressure following an accident. This requires more than one mental step to accomplish.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	19 NRC #41 Modified Bank Question 8344

Development References

OP-CN-CNT-CNT, Sect. 2.8
TS 3.6.13

Student References Provided

SYS025 K5.01 - Ice Condenser System

Knowledge of operational implications of the following concepts as they apply to the ice condenser system: (CFR: 41.5 / 45.7)

Relationships between pressure and temperature ....

Remarks/Status

SYS026 A2.07 - Containment Spray System (CSS)

Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding), or sump level below cutoff (interlock) limit

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Given the following Unit 1 conditions:

- The crew is performing EP/1/A/5000/ES-1.3 (Transfer To Cold Leg Recirculation)
- Containment Spray (NS) has been aligned for recirculation
- Containment pressure is 3.2 PSIG and rising slowly

ES-1.3 will require starting _____(1)_____ NS pump(s).

Based on the conditions above, if a loss of NS flow occurs, a Containment ORANGE path _____(2)_____ occur.

Which ONE of the following correctly completes the statements above?

A. 1. both
   2. will

B. 1. ONLY one
   2. will

C. 1. both
   2. will NOT

D. 1. ONLY one
   2. will NOT

General Discussion

Only one train of NS is aligned to recirc mode in ES-1.3 to conserve FWST inventory. One of the first steps when aligning asks if the opposite train pump is running. If it is, you are directed to the next step of the procedure. Direction is provide to start only one pump in this procedure.

Per the note in ES-1.3, at step 7 RNO An invalid SPDS orange path may briefly exist between opening NS suction valve from sump and starting NS pump. FR-Z.1 should not be entered unless NS pump fails to start.. This is also validated by the containment status tree. When the valve alignment is made a loss of NS flow will result in an orange path condition.

Answer A Discussion

Part 1 is plausible because both trains of pumps on all other safety systems (ND, NI, NV) are started if available, when aligning for cold leg recirc .

Part 2 is correct.

Answer B Discussion

CORRECT: See explanation above.

Answer C Discussion

Part 1 is plausible because both trains of pumps on all other safety systems (ND, NI, NV) are started if available, when aligning for cold leg recirc .

Part 2 is plausible because had containment pressure in the stem been less than 3 psig an orange path would not be generated.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because had containment pressure in the stem been less than 3 psig an orange path would not be generated.

Basis for meeting the KA

Basis for Hi Cog

This question is high cog because the applicant is required to analyze the conditions in the stem, determine what effect each of these conditions will have on plant operational alignment and recall what procedural guidance is given to control this occurrence.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 7213

Development References

EP/1/A/5000/F-0, Pg. 9
EP/1A/500/ES-1.3, Encl. 2

Student References Provided

SYS026 A2.07 - Containment Spray System (CSS)

Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding), or sump level below cutoff (interlock) limit

Remarks/Status

SYS026 K3.02 - Containment Spray System (CSS)
Knowledge of the effect that a loss or malfunction of the CSS will have on the following: (CFR: 41.7 / 45.6)
Recirculation spray system ....

Given the following Unit 1 timeline:

0800 Reactor trip and Safety Injection initiated due to large break LOCA
0845 Crew enters EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation)

- Containment pressure is currently 3.0 psig and **rising** at 0.8 psig / min
- Crew is unable to align either Containment Spray pump for recirculation per Enclosure 2 (Aligning NS for Recirculation)

In accordance with EP/1/A/5000/FR-Z.1 (Response to High Containment Pressure), the earliest time that the crew will align ND Spray is __________ .

Which ONE of the following correctly completes the statement above?

A. 0845

B. 0850

C. 0900

D. 0930

General Discussion

In order to align ND for Aux Containment Spray, FR-Z.1 requires Containment pressure greater than 15 psig and elapsed time since reactor trip to greater than 50 minutes.

The elapsed time requirement will be met at 0850 but 15 psig will not be reached until 0900.

Answer A Discussion

Plausible if the applicant is unaware of the 50 minute elapsed time requirement and reasons containment pressure must be greater than the Containment Hi-Hi limit of 3.0 psig.

Answer B Discussion

Plausible because this represents the elapsed time since trip requirement. The applicant may be unaware of the containment pressure requirement of 15 psig or reason this requirement is actually the Containment Hi-Hi limit of 3.0 psig.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Plausible becaue this represents the 90 minute elapsed time requirement for establishment of Feed and Bleed with a single train of Safety Injection system flow.

Basis for meeting the KA

Given a loss of the CSS, the applicant is required to demonstrate knowledge of the effect on containment status and mitigation actions.

Basis for Hi Cog

This question requires more than one mental step. The applicant must analyze the given information, compare to knowledge recalled from memory, and then make a determination in order to correctly answer the question.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	NEW	

Development References
FR-Z.1, Step 11 FR-H.1

Student References Provided

SYS026 K3.02 - Containment Spray System (CSS)

Knowledge of the effect that a loss or malfunction of the CSS will have on the following: (CFR: 41.7 / 45.6)

Recirculation spray system ....

Remarks/Status

SYS039 A1.06 - Main and Reheat Steam System (MRSS)
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: (CFR: 41.5 / 45.5)
Main steam pressure ....

Given the following Unit 1 initial conditions:

- The Unit is at 63% RTP following a refueling outage
- AP/1/A/5500/028 (Secondary Steam Leak) has been entered following the discovery of a leak on the Unit 1 Main Turbine Crossover line

Subsequently:

- The Unit 1 Main Turbine is tripped to isolate the leak

At this time, _____(1)_____ steam dumps will operate to control main steam pressure at approximately _____(2)_____ psig.

Which ONE of the following correctly completes the statement above?

A. 1. ONLY condenser
   2. 1085

B. 1. ONLY condenser
   2. 1115

C. 1. condenser AND atmospheric
   2. 1085

D. 1. condenser AND atmospheric
   2. 1115

General Discussion

Based on given conditions, the Load Rejection Controller will be controlling operation of the Steam Dumps. This controller operates dumps to achieve a 3 degree F deadband in order to allow for control rod insertion. This deadband will correspond to a RCS temperature of 560 degrees (approx. 1130 psia or 1115 psig).

Atmospheric Steam dumps are only available to be operated by the Load Rejection Controller, which is currently in service. Atmospheric dumps also require a C-7B (Loss of Load Interlock) signal which is instated following a 30% step change in load.

Answer A Discussion

Part 1 is plausible because this would be the correct answer if initial reactor power was greater than 69% since atmospheric steam dumps only operate on load rejection and not on reactor trip.

Part 2 is plausible because this steam pressure corresponds to the reference temperature (557 F) value for the turbine being tripped. The applicant fails to apply the 3 degree deadband.

Answer B Discussion

Part 1 is plausible because this would be the correct answer if initial reactor power was greater than 69% since atmospheric steam dumps only operate on load rejection and not on reactor trip.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because this steam pressure corresponds to the reference temperature (557 F) value for the turbine being tripped. The applicant fails to apply the 3 degree deadband.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The applicant is required to predict changes in main steam pressure associated with operation of MRSS controls.

Basis for Hi Cog

The applicant must calculate reference temperature and also calculate a power change percentage in order to correctly answer this question.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	17 NRC #45 Modified Bank Question 8048

Development References
OP-CN-STM-IDE (Steam Dump Control LP), Sect. 2.2 & Pg 40

Student References Provided

SYS039 A1.06 - Main and Reheat Steam System (MRSS)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: (CFR: 41.5 / 45.5)

Main steam pressure ....

Remarks/Status

SYS039 A2.03 - Main and Reheat Steam System (MRSS)

Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Indications and alarms for main steam and area radiation monitors (during SGTR) ....

Given the following Unit 1 conditions:

- Following a refueling outage, the operating crew began a power escalation
- Due to chemistry concerns, the unit has been placed in hold at 65%
- The unit has been in hold for 3 hours
- A tube leak has developed in 1A S/G

In accordance with CSD-CP-CNS-0020 (CNS Primary to Secondary Leak Rate Monitoring Program), leak rate monitoring will be based on readings obtained from __________ .

Which ONE of the following correctly completes the statement above?

COMPONENT LEGEND:
1EMF-29 (Steam Line 1A)
1EMF-33 (Condenser Air Ejector Exhaust)
1EMF-71 (S/G A Leakage)

A. 1EMF-29 ONLY

B. 1EMF-71 ONLY

C. 1EMF-33 AND 1EMF-29

D. 1EMF-33 AND 1EMF-71

General Discussion

In accordance with CSD-CP-CNS-0020, leak rate monitoring is specified as follows:

Modes 2, 3, & 4: Tritium concentration from secondary grab samples

5% - 40%: Condenser off-gas (EMF-33)

40% - 95%: N-16 monitors (EMFs 71-74) and off-gas (EMF-33)

95% - 100%: Off-gas (EMF-33) PREFERRED and N-16 (EMFs 71-74)

Answer A Discussion

This EMF is used as reference to enter abnormal procedure for SGTL. Plausible if the applicant is not aware of the required monitors.

Answer B Discussion

This is a listed leak rate monitor. Plausible if the applicant is not aware that two monitoring points are specified at this power level.

Answer C Discussion

EMF-33 is a listed leak rate monitor and EMF-29 (steam line area monitor) is listed in entry criteria for S/G tube leak AP and will provide leakage indication. Plausible if the applicant is not aware of the two specified monitoring points at this power level.

Answer D Discussion

CORRECT. See information above.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of site procedures to determine radiation monitor monitoring requirements when presented with a S/G tube leak.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
CSD-CP-CNS-0020, Pg. 19, Section 2

Student References Provided

SYS039 A2.03 - Main and Reheat Steam System (MRSS)

Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Indications and alarms for main steam and area radiation monitors (during SGTR) ....

Remarks/Status

SYS059 K4.08 - Main Feedwater (MFW) System
Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Feedwater regulatory valve operation (on basis of steam flow, feed flow mismatch) ....

Given the following Unit 1 conditions:

- The Unit is at 12% RTP and rising

DCS will maintain S/G Level Control for each S/G in the LO Power mode until a MAXIMUM Selected CF Flow of _____(1)_____ is reached on each S/G.

With DCS in the LO Power mode, the CF Control Valves will start to OPEN when the CF Control Bypass Valves demand signal reaches a MINIMUM of _____(2)_____.

Which ONE of the following correctly completes the statements above?

A. 1. 17%
   2. 55%

B. 1. 17%
   2. 65%

C. 1. 20%
   2. 55%

D. 1. 20%
   2. 65%

General Discussion

As the plant increases power, CF flow will increase. When an individual S/G's Selected CF flow reaches 20%, DCS will swap to the HI Power mode of operation for that individual S/G's Level Control.

DCS provides programming to the CF Control Valves and the CF Control Bypass Valves. The plant is started up using the CF Control Bypass Valves with DCS in the LO Power mode of operation. The CF Control Valves Start to open when the CF Control Bypass Valve demand signal reaches 55%. The CF Control Bypass Valves close a proportional amount, to maintain Steam Generator Level stable

Answer A Discussion

Part 1 is plausible because this is the correct number for DCS selecting the LO power mode on a unit down power.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because this is the correct number for DCS selecting the LO power mode on a unit down power.

Part 2 is plausible because this value corresponds to the power level required to initiate the Unit 2 single MFP trip level program control circuit.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because this value corresponds to the power level required to initiate the Unit 2 single MFP trip level program control circuit.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the design features of DCS that provide for operation of the Feed Control and Feed Control Bypass valves during Lo and Hi power modes.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 7217

Development References
OP-CN-CF-IFE, Sect. 4.5, 4.6, & 9

Student References Provided

SYS059 K4.08 - Main Feedwater (MFW) System

Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)

Feedwater regulatory valve operation (on basis of steam flow, feed flow mismatch) ....

Remarks/Status

SYS061 K5.02 - Auxiliary / Emergency Feedwater (AFW) System
Knowledge of the operational implications of the following concepts as the apply to the AFW: (CFR: 41.5 / 45.7)
Decay heat sources and magnitude ....

Given the following Unit 1 conditions:

- The unit was at 100% RTP when a reactor trip occurred
- Reactor Trip Breaker 1B failed to open
- All S/G Narrow Range levels are OFF-Scale LOW

In order to meet secondary heat sink requirements CA flow must be greater than a MINIMUM value of _____(1)_____ GPM

With regard to core age, more decay heat will be generated following a reactor trip at the _____(2)_____ of core life.

Which ONE of the following correctly completes the statements above?

A. 1. 1000
   2. beginning

B. 1. 1000
   2. end

C. 1. 450
   2. beginning

D. 1. 450
   2. end

General Discussion

Secondary Heat sink on Unit 1 requires narrow range level of 11% or auxiliary feed flow rate of 450 gpm to compensate for decay heat. However, this would be insufficient if reactor operations were to continue. FR-S.1 requires Aux Feed flow of 1000 gpm if the reactor is not shutdown.

More decay heat is produced at the end of core life vs. beginning.

Answer A Discussion

Part 1 is plausible because this would be the required auxiliary feedwater flow required by FR-S.1. Applicant may confuse failure of reactor trip breaker with FR-S.1 entry.

Part 2 is plausible because more excess reactivity is present prior to cycle fuel burn.

Answer B Discussion

Part 1 is plausible because this would be the required auxiliary feedwater flow required by FR-S.1. Applicant may confuse failure of reactor trip breaker with FR-S.1 entry.

Part 2 is correct.

Answer C Discussion

Part 1 correct.

Part 2 is plausible because more excess reactivity is present prior to cycle fuel burn.

Answer D Discussion

CORRECT. See explanation above

Basis for meeting the KA

The applicant is required to demonstrate knowledge of decay heat magnitude and auxiliary feedwater operational implications.

Basis for Hi Cog

The question is at the memory cognitive level because the operator candidate must recall bits of information regarding decay hear magnitude and requirements of AFW flow during FR-S.1.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	Bank Question 8451

Development References
FR-S.1, Step 13 RNO

Student References Provided

SYS061 K5.02 - Auxiliary / Emergency Feedwater (AFW) System

Knowledge of the operational implications of the following concepts as the apply to the AFW: (CFR: 41.5 / 45.7)

Decay heat sources and magnitude ....

Remarks/Status

SYS062 K2.01 - AC Electrical Distribution System
Knowledge of bus power supplies to the following : (CFR: 41.7)
Major system loads ....

Given the following Unit 1 conditions:

- Unit is at 100% RTP
- 1B Transformer Loss of Cooler Power results in a Zone B Lockout

The 1B NCP supply breaker is located on the _____(1)_____ side of 1TB Switchgear.

Following the Zone B Lockout, the 1B NCP _____(2)_____ continue to operate.

Which ONE of the following correctly completes the statements above?

A. 1. long
   2. will

B. 1. long
   2. will NOT

C. 1. short
   2. will

D. 1. short
   2. will NOT

General Discussion

All NCPs are powered from the short side of their associated swichgear.

The "7KV Bus 1TB Mode Select" switch will normally be in Auto at power. An auto transfer will be initiated by a zone lockout and a fast transfer will occur. The incoming breaker opens before the tie breaker closes, but the transfer occurs within a few cycles with no loss of load.

Answer A Discussion

First part is plausible because other large motor loads are powered from the long side (CBPs, HWPs, RC pumps).

Second part is correct.

Answer B Discussion

First part is plausible because other large motor loads are powered from the long side (CBPs, HWPs, RC pumps).

Second part is plausible if the applicant does not understand the "Fast Transfer" feature associated with 7KV switchgear. Additionally, this feature opens the normal incoming breaker before closing the tie breaker which may lead applicant to believe loads are lost.

Answer C Discussion

CORRECT: See explanation above.

Answer D Discussion

The first part is correct.

Second part is plausible if the applicant does not understand the "Fast Transfer" feature associated with 7KV switchgear. Additionally, this feature opens the normal incoming breaker before closing the tie breaker which may lead applicant to believe loads are lost.

Basis for meeting the KA

The KA is matched because the applicant must have knowledge of the power supplies to the NC pumps (major system load) and demonstrate knowledge of the affect of power transfer.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. The applicant must first recall from memory the requirements for a fast bus and slow bus transfer. The applicant must then apply the recalled memory to the given conditions and analyze each answer based on that analysis to determine the correct response.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6523

Development References

EPB LP Section 3.2 & 11
EPA LP Section 4.2

Student References Provided

SYS062 K2.01 - AC Electrical Distribution System
Knowledge of bus power supplies to the following : (CFR: 41.7)
Major system loads ....

Remarks/Status

SYS063 A3.01 - DC Electrical Distribution System

Ability to monitor automatic operation of the DC electrical system, including: (CFR: 41.7 / 45.5)

Meters, annunciators, dials, recorders, and indicating lights ....

Given the following Unit 1 conditions:

- 1KXIB has experienced a complete loss of DC input voltage
- Stable power is restored two (2) minutes later

Based on the conditions above,

An indication used to determine that the backup power supply has been aligned is the _____(1)_____ light LIT.

When 1KXIB loss of voltage condition clears, the normal power supply _____(2)_____ be automatically realigned.

Which ONE of the following correctly completes the statements above?

A. 1. 1KXMB "In Sync"
   2. will NOT

B. 1. 1KXMB "In Sync"
   2. will

C. 1. 1KXAB "Alternate AC Source Supplying Load"
   2. will NOT

D. 1. 1KXAB "Alternate AC Source Supplying Load"
   2. will

General Discussion

The operating procedure specifies that the "Alternate AC Source Supplying Load" light is to be verified ON as part of the Shutdown and Return to Service procedure. System design is such that once the transfer switch has auto swapped to an alternate source, a 60 second relay is activated. If the transfer was due to inverter overcurrent, inverter failure or inverter undervoltage, the switch will automatically transfer back to the inverter when the conditions have cleared and remained stable for 60 seconds.

Answer A Discussion

Part 1 is plausible since the In Sync light indicates that both sources are energized and in sync, but doesn’t indicate which is aligned to supply the load.

Part 2 is plausible since a manual swap back to inverter supplying load is possible and proceduralized but not needed in this case, since the automatic transfer will occur first.

Answer B Discussion

Part 1 is plausible since the In Sync light indicates that both sources are energized and in sync, but doesn’t indicate which is aligned to supply the load.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible since a manual swap back to inverter supplying load is possible and proceduralized but not needed in this case, since the automatic transfer will occur first.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

The applicant is required to determine DC Electrical system operation and monitor automatic operation via use of an indicating light.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 7221

Development References	Student References Provided
OP-CN-EL-EPF, Sect. 2.8 OP/1/A/B/6350/009, Encl. 4.17, Pg. 5	

SYS063 A3.01 - DC Electrical Distribution System

Ability to monitor automatic operation of the DC electrical system, including: (CFR: 41.7 / 45.5)

Meters, annunciators, dials, recorders, and indicating lights ....

Remarks/Status

SYS064 A2.15 - Emergency Diesel Generator (ED/G) System

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Water buildup in cylinders ....

Concerning operation of the Emergency Diesel Generators:

In accordance with OP/1/A/6350/002 (Diesel Generator Operation), an inspection for water accumulation in the 1A D/G is performed by opening _____(1)_____ . This inspection is performed _____(2)_____ D/G operational testing.

Which ONE of the following correctly completes the statements above?

A. 1. Cylinder indicator cocks 1L – 8L and 1R – 8R
   2. prior to

B. 1. Cylinder indicator cocks 1L – 8L and 1R – 8R
   2. following

C. 1. 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain)
   2. prior to

D. 1. 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain)
   2. following

General Discussion

Prior to periodic D/G testing the engine is manually barred at least two revolutions and then rolled with air to ensure smooth rotation. Prior to these evolutions cylinder indicator cocks are opened and then inspected for liquid expulsion following engine rotations.

Following engine operation a vent line is aligned to drain any accumulated oil in the crankcase.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because inspections are performed following engine operation (i.e. draining liquid accumulation from the engine exhaust silencer and oil accumulation from the crank case).

Answer C Discussion

Part 1 is plausible if an applicant reasons that water would accumulate in the crankcase of the engine.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible if an applicant reasons that water would accumulate in the crankcase of the engine.

Part 2 is plausible because inspections are performed following engine operation (i.e. draining liquid accumulation from the engine exhaust silencer and oil accumulation from the crank case).

Basis for meeting the KA

The applicant is required to demonstrate the ability to use procedures to mitigate the impact of water buildup in D/G cylinders.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
OP/1/A/6350/002, Encl. 4.140 & 4.25

Student References Provided

SYS064 A2.15 - Emergency Diesel Generator (ED/G) System

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Water buildup in cylinders ....

Remarks/Status

SYS073 2.1.25 - Process Radiation Monitoring (PRM) System
SYS073 GENERIC
Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12)

Given the following Unit 2 initial conditions:

- Unit was operating at 100% RTP
- A containment air release (VQ) was in progress

Subsequently:

- A LOCA occurs
- "B" Train safety injection failed to actuate and was performed manually when it was recognized by the crew
- The following indications are noted for:
  - Containment pressure
  - 2EMF-36 (Unit Vent Gas Monitor)
  - 2EMF-39 (Containment Gas Monitor)
  - E/S Load Sequencers status lights

Time	0200	0201	0202	0203
Containment pressure (psig)	0.5	1.1	1.4	1.8
2EMF-36 Trip 2 Light	LIT	LIT	LIT	LIT
2EMF-39 Trip 2 Light	DARK	LIT	LIT	LIT
E/S LOAD SEQ ACTUATED TRAIN "A" status light	DARK	DARK	LIT	LIT
E/S LOAD SEQ ACTUATED TRAIN "B" status light	DARK	DARK	DARK	LIT

Based on the above indications and conditions, what is the earliest time that an operator can be assured that 2VQ-3B (VQ Fan Suct From Cont Isol) has received a close signal?

A. 0200

B. 0201

C. 0202

D. 0203

General Discussion

From ECCS lesson plan:

Sh shuts down and isolates VP and isolates VQ Containment Isolation valves. Four Signals can actuate an Sh:

1. Manual Phase A (St): Train related
2. Manual Phase B (Sp): VX Initiate, Cont Vent Isol (Phase B, VX Initiate, Cont Vent Isol) will actuate train A (B) Sh. This is a single pushbutton that actuates three functions. Train related.
3. Safety Injection Signal: Train A (B) Ss will actuate Train A (B) Sh. Train related
4. EMF 38, or 39 L, TRIP 2: High Containment Particulate or Gas will actuate BOTH Trains of SH.

Answer A Discussion

Plausible if the applicant confuses EMF-36 with EMF-38 and reasons that this actuation would generate an Sh signal.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

This actuation will generate an isolation signal for this system (VQ) but only for the "A" train. Plausible if the applicant is unaware of the separate ventilation isolation signal (Sh) actuated from EMF-39 and also reasons that a Phase A signal from either train will generate a complete containment isolation signal.

Answer D Discussion

This actuation will generate an isolation signal for this system (VQ) and for the applicable train. Plausible if the applicant is unaware of the separate ventilation isolation signal (Sh) actuated from EMF-39 and also correctly reasons that a Phase A signal from the A ES load sequencer will not generate a "B" train containment isolation signal.

Basis for meeting the KA

The applicant is required to interpret the supplied table in order to determine automatic actions related to process radiation monitoring systems.

Basis for Hi Cog

The applicant must analyze specific data related to an individual time and then compare to knowledge recalled from memory.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	Bank Question 453

Development References	Student References Provided
Lesson OP-CN-ECCS-ISE, Sect. 5.4	

SYS073 2.1.25 - Process Radiation Monitoring (PRM) System
SYS073 GENERIC
Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12)

Remarks/Status

EARLY REVIEW QUESTION

SYS076 K1.19 - Service Water System (SWS)

Knowledge of the physical connections and/or cause- effect relationships between the SWS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

SWS emergency heat loads ....

Given the following Unit 1 conditions:

- A Large Break LOCA has occurred
- The crew is performing EP/1/A/5000/ES-1.3 (Transfer To Cold Leg Recirculation)
- The BOP is instructed to align NS for recirc per Enclosure 2 (Aligning NS for Recirculation)
- Containment pressure is currently 2 PSIG and stable

Based on the above conditions, once Enclosure 2 is complete, RN flow through the NS Heat Exchanger _____(1)_____ be aligned.

If RN flow is aligned through the NS Heat Exchanger, a CAUTION in ES-1.3 states that RN flow shall not exceed a MAXIMUM of _____(2)_____ to prevent damage to the Heat Exchanger tubes.

Which ONE of the following correctly completes the statements above?

A. 1. will
   2. 5700 GPM

B. 1. will
   2. 4650 GPM

C. 1. will NOT
   2. 5700 GPM

D. 1. will NOT
   2. 4650 GPM

General Discussion

ES-1.3 Enclosure 2 will only start an NS pump and align RN flow through the NS heat exchanger if containment pressure was > 3 PSIG when entering the Enclosure. In the stem of the question containment pressure is 2 PSIG and stable, therefore the NS pump will not be started and RN flow through the NS heat exchanger will not be aligned. Enclosure 2 will only align the NS pump suction to the containment sump and then operators are directed to return to procedure and step in effect.

If RN flow is aligned through an NS heat exchanger, a CAUTION in Encl. 2 states that "Exceeding 4650 GPM RN flow through an NS Hx will cause damage to the Hx tubes.

5700 GPM is the flow associated with KC pump single pump runout.

Answer A Discussion

Part 1 is plausible because Enclosure 2 will start an NS pump and align RN flow through the NS Hx once the NS spray header isolations are opened and containment pressure is > 1 PSIG. The NS spray header isolations, however are only opened if containment pressure exceeds 3 PSIG.

Part 2 is plausible because this is a flowrate that does not need to be exceeded to protect a safety related component (KC pump single pump runout), but is not associated with the NS Hx.

Answer B Discussion

Part 1 is plausible because Enclosure 2 will start an NS pump and align RN flow through the NS Hx once the NS spray header isolations are opened and containment pressure is > 1 PSIG. The NS spray header isolations, however are only opened if containment pressure exceeds 3 PSIG.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because this is a flowrate that does not need to be exceeded to protect a safety related component (KC pump single pump runout), but is not associated with the NS Hx.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

The KA is matched because applicants are tested on their knowledge of when RN flow is aligned to the NS heat exchanger (physical connection) and the cause/effect relationship of what the maximum RN flow established to this Hx to prevent damage to the Hx tubes.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 7224

Development References

OP-CN-PSS-RN, Sect. 4.5
EP/1/A/5000/ES-1.3, Caution prior to step 5.k, Step 5.k

Student References Provided

SYS076 K1.19 - Service Water System (SWS)

Knowledge of the physical connections and/or cause- effect relationships between the SWS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

SWS emergency heat loads ....

Remarks/Status

SYS078 K3.03 - Instrument Air System (IAS)
Knowledge of the effect that a loss or malfunction of the IAS will have on the following: (CFR: 41.7 / 45.6)
Cross-tied units ....

Given the following conditions:

- Unit 1 is in Mode 5
- Unit 2 is at 100% RTP
- 1A train of ND is in service
- Both units enter AP/0/A/5500/022 (Loss of Instrument Air) following a VI pipe rupture

Per AP/22:

Unit 1 will stabilize NC temperature by _____(1)_____ .

Unit 2 Reactor Trip will be required at a MAXIMUM lowering VI pressure of _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. throttling 1NI-173A (ND Hdr 1A To Cold Legs C&D)
   2. 55 psig

B. 1. placing 1B ND in service
   2. 55 psig

C. 1. throttling 1NI-173A (ND Hdr 1A To Cold Legs C&D)
   2. 76 psig

D. 1. placing 1B ND in service
   2. 76 psig

General Discussion

Per AP/22 in both units enclosures:
1. IF AT ANY TIME VI pressure is less than
55 PSIG AND decreasing, THEN:
a. Trip reactor.
IF ND System is in RHR mode, THEN
control ND flow and cooldown rate for
the operating ND train(s) as follows:
 Train A:
1) Place the "PWR DISCON FOR
2NI-173A" switch in "THROT"
position.

76 psig is the setpoint for VS opening to supply VI on low pressure
The HX outlet valves fail open and the bypasses fail closed on loss of VI. This would result in more cooling so 173 would have to be throttled further closed to maintain temperature. Starting the idle train of ND would be logical if they believe that the hx outlet and bypass valves work opposite of reality.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is plausible if the applicant reasons that the opposite train will not be affected by the loss of instrument air.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because the VS to VI isolation valve opens at this setpoint.

Answer D Discussion

Part 1 is plausible if the applicant reasons that the opposite train will not be affected by the loss of instrument air.

Part 2 is plausible because the VS to VI isolation valve opens at this setpoint.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the effect of a loss of instrument air will have upon both units, each of which is in a different mode of operation.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 7225

Development References
AP/0/A/5500/022, Encl 3, Step 1 & 2 RNO a

Student References Provided

SYS078 K3.03 - Instrument Air System (IAS)
Knowledge of the effect that a loss or malfunction of the IAS will have on the following: (CFR: 41.7 / 45.6)
Cross-tied units ....

Remarks/Status

SYS103 A3.01 - Containment System
Ability to monitor automatic operation of the containment system, including: (CFR: 41.7 / 45.5)
Containment isolation ....

Given the following Unit 1 conditions:

- Unit is at 100% RTP
- A steam break occurred on the Main Steam Equalization Header
- Train 1B Safety Injection failed to automatically actuate

Assuming no operator action:

Phase A (St) isolation has been initiated on train _____(1)_____.

Phase B (Sp) isolation _____(2)_____ been initiated.

Which one of the following correctly completes the statements above?

A. 1. 1A ONLY
   2. has

B. 1. 1A ONLY
   2. has NOT

C. 1. 1A AND 1B
   2. has

D. 1. 1A AND 1B
   2. has NOT

General Discussion

Phase A is train related and initiated manually or from an Ss signal.

Phase B is also train related and initiated manually or a Hi-Hi Containment signal. A Phase B initiation will also initiate a Main Steam Isolation. However, a Main Steam isolation will NOT initiate Phase B. Since the steam leak is outside containment, a Hi-Hi containment pressure signal will not be present.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because if the applicant confuses the relationship of Phase B and Main Steam Isolation signals. The given conditions have most likely resulted in an automatic main steam isolation. This isolation would also be initiated from a Phase B signal but not the reverse.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible if the appliant is unaware that Phase A signals are strictly train related and only automatically generated from Ss (vs low PZR pressure). Other Ss signals are cross train (i.e. Reactor Trip Breakers).

Part 2 is plausible because if the applicant confuses the relationship of Phase B and Main Steam Isolation signals. The given conditions have most likely resulted in an automatic main steam isolation. This isolation would also be initiated from a Phase B signal but not the reverse.

Answer D Discussion

Part 1 is plausible if the appliant is unaware that Phase A signals are strictly train related and only automatically generated from Ss (vs low PZR pressure). Other Ss signals are cross train (i.e. Reactor Trip Breakers).

Part 2 is correct.

Basis for meeting the KA

The applicant is required to demonstrate the ability to monitor automatic operation of containment isolation signals.

Basis for Hi Cog

This question requires more than one mental step. The applicant must apply given information to that recalled from memory and then determine order of isolation signals and relationships.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	Bank Question 4785

Development References
ISE LP, Sect. 5.2 & 5.3

Student References Provided

SYS103 A3.01 - Containment System

Ability to monitor automatic operation of the containment system, including: (CFR: 41.7 / 45.5)

Containment isolation ....

Remarks/Status

SYS103 2.4.50 - Containment System

SYS103 GENERIC

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

Given the following Unit 1 conditions:

- A rapid downpower is in progress due to a secondary steam leak inside Containment
- 1A, 1B, and 1D Lower Containment Vent Units (LCVU) are in operation
- Current Unit 1 Containment pressure is 0.58 psig and rising slowly

Assuming no operator action,

1RN-473 (LCVU A Full Flow Valve) _____(1)_____ currently open.

1A LCVU _____(2)_____ operating in “Hi Speed”.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. is

B. 1. is
   2. is NOT

C. 1. is NOT
   2. is

D. 1. is NOT
   2. is NOT

General Discussion

Per the Containment Ventilation LP:

A BYPASS valve or FULL FLOW valve is installed in parallel with the normal cooling water flow control valve. This valve will automatically open for each of the following conditions:
1. An LCVU is selected to HIGH speed.
2. Containment pressure rises to greater than or equal to 0.5 psig.

Per Annunciator Response Procedure for 1AD-19, B/12 will direct LCVUs be placed in High Speed as required. This is not an automatic response.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because of the title of the associated annunciator "Lower Cont Press 0.5 psig Initiate Hi Speed". Also, plausible because High Speed fan operation automatically opens the associated Full Flow valve.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible if the applicant confuses the operation of this valve with LCVU fan speed selection.

Part 2 is plausible because of the title of the associated annunciator "Lower Cont Press 0.5 psig Initiate Hi Speed". Also, plausible because High Speed fan operation automatically opens the associated Full Flow valve.

Answer D Discussion

Part 1 is plausible if the applicant confuses the operation of this valve with LCVU fan speed selection.

Part 2 is correct.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the actions contained in the annunciator response manual for a system alarm (hi containment pressure).

Basis for Hi Cog

The applicant must compare provided conditions with a setpoint, recalled from memory, and apply that determination to a specific set of conditions.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 7152

Development References

OP-CN-CNT-VV (Containment Ventilatin System LP), Pg. 12, 30,

Student References Provided

SYS103 2.4.50 - Containment System
SYS103 GENERIC
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

Remarks/Status

SYS001 K2.02 - Control Rod Drive System
Knowledge of bus power supplies to the following: (CFR: 41.7)
One-line diagram of power supply to trip breakers ....

Concerning Reactor Trip Breaker 1B:

The Undervoltage coil receives power auctioneered from _____(1)_____ and 1ERPD and is _____(2)_____ to actuate.

Which ONE of the following correctly completes the statement above?

A. 1. 1ERPB
   2. energized

B. 1. 1ERPB
   2. de-energized

C. 1. 1ERPC
   2. energized

D. 1. 1ERPC
   2. de-energized

General Discussion

Although ERPC is a Train A related power supply, it powers "B" train SSPS equipment. The undervoltage coil is de-energized to actuate a reactor trip.

Answer A Discussion

Part 1 is plausible because 1ERPB is a Train "B" related power supply. However, SSPS power supplies are arranged in a alternate method (ERPA & ERPB supply A and ERPC/ERPD supply B).

Part 1 is plausible because the shunt trip is energized to actuate.

Answer B Discussion

Part 1 is plausible because 1ERPB is a Train "B" related power supply. However, SSPS power supplies are arranged in a alternate method (ERPA & ERPB supply A and ERPC/ERPD supply B).

Part 1 is correct.

Answer C Discussion

Part 1 is correct.

Part 1 is plausible because the shunt trip is energized to actuate.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The applicant s required to demonstrate knowledge of reactor trip related power supplies.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

IPX LP Sect. 3.1

Student References Provided

SYS001 K2.02 - Control Rod Drive System

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

One-line diagram of power supply to trip breakers ....

Remarks/Status

SYS011 K4.07 - Pressurizer Level Control System (PZR LCS)
Knowledge of PZR LCS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)
Cold-calibrated channel ....

Concerning the Pressurizer Cold Cal Channel:

The Pressurizer level “Cold Calibrated” Channel is calibrated for _____(1)_____. This channel _____(2)_____ a required safety related indication.

Which ONE of the following correctly completes the statements above?

A. 1. 100°F
   2. is

B. 1. 100°F
   2. is NOT

C. 1. 120°F
   2. is

D. 1. 120°F
   2. is NOT

General Discussion

The Cold Cal channel is calibrated at 100ºF and 14.7 psia (~atmospheric pressure).

The cold cal channel is used during plant startup, shutdown, and refueling for indication and is considered a non-safety related indication.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because all other PZR level indications are required and only 1 channel of cold cal exists. It stands to reason that this channel would be required because there is no redundancy (at a lower temperature calibration).

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible as this is the upper limit for containment temperature.

Part 2 is plausible because all other PZR level indications are required and only 1 channel of cold cal exists. It stands to reason that this channel would be required because there is no redundancy (at a lower temperature calibration).

Answer D Discussion

Part 1 is plausible as this is the upper limit for containment temperature.

Part 2 is correct.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the Cold Calibrated channel design (calibration setpoint and safety significance).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

ILE LP Sect. 2.1

Student References Provided

SYS011 K4.07 - Pressurizer Level Control System (PZR LCS)

Knowledge of PZR LCS design feature(s) and/or interlock(s) which provide for the following: (CFR: 41.7)

Cold-calibrated channel ....

Remarks/Status

SYS015 K1.02 - Nuclear Instrumentation System (NIS)

Knowledge of the physical connections and/or cause-effect relationships between the NIS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Vital ac systems ....

Given the following Unit 1 initial conditions:

- Unit is at 12% RTP following startup
- Required actions for being greater than P-10 have been taken

Subsequently:

- 1ERPA de-energizes

As a result of this failure:

Power Range detector N-41 will lose _____(1)_____.

The crew will FIRST enter _____(2)_____.

Which ONE of the following correctly completes the statements above?

A. 1. control power ONLY
2. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)

B. 1. control power ONLY
2. AP/1/A/5500/016 (Malfunction of Nuclear Instrumentation)

C. 1. control and instrument power
2. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)

D. 1. control and instrument power
2. AP/1/A/5500/016 (Malfunction of Nuclear Instrumentation)

General Discussion

ERPA supplies instrument and control power to NIs.

If power is less than 10%, and ERPA or ERPB lose power, then the unit will trip due to loss of 1 IR channel. Therefor E-0 entry would be correct.

If power is greater than 10%, a reactor trip will not occur and entry into AP/16 is appropriate.

Answer A Discussion

Part 1 is plausible because instrument and control power are independent supplies. However, both are supplied from the same vital panelboard.

Part 2 is plausible because this would be true if reactor power were < 10% due to the reactor tripping on a loss of I/R channel N35.

Answer B Discussion

Part 1 is plausible because instrument and control power are independent supplies. However, both are supplied from the same vital panelboard.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because this would be true if reactor power were < 10% due to the reactor tripping on a loss of I/R channel N35.

Answer D Discussion

CORRECT - See discussion above.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of a power supply associated with the Reactor Protection system associated NI channel and the design feature associated with loss of power to one channel not causing a reactor trip above 10% power.

Basis for Hi Cog

This question is a higher cognitive question because the applicant must be able to analyze plant conditions to determine the status of the IR high flux Rx trip and apply the given power failure in order to determine the impact this failure will have.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 8252

Development References

ENB lesson plan

Student References Provided

SYS015 K1.02 - Nuclear Instrumentation System (NIS)

Knowledge of the physical connections and/or cause-effect relationships between the NIS and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Vital ac systems ....

Remarks/Status

B

ILT 21 CNS RO NRC Examination **QUESTION 59**

SYS016 2.2.39 - Non-Nuclear Instrumentation System (NNIS)
SYS016 GENERIC
Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13)

Given the following Unit 1 conditions:

- A Shift Maintenance Technician performing Reactor Building rounds reports that the Ice Condenser Inlet Door Positioning Monitor System panel has no indicating lights lit

Which one of the following is the minimum action required to maintain compliance with SLC 16.6-3 Inlet Door Position Monitoring System?

A. Immediately verify the ice bed temperature is less than or equal to 27 °F

B. Immediately verify the Ice Bed Temperature Monitoring System is Functional

C. Within 1 hour verify the ice bed temperature is less than or equal to 27 °F

D. Within 1 hour verify the Ice Bed Temperature Monitoring System is Functional

---

General Discussion

Ice Bed Temperature monitoring must be verified functional immediately per SLC16.6-3

Answer A Discussion

Verifying ice bed temperature is a requirement but it is 4 hours, not 1 or immediately.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Verifying ice bed temperature is a requirement but it is 4 hours, not 1 or immediately.

Answer D Discussion

Wrong time , correct action

Basis for meeting the KA

The K/A is matched because the applicant must demonstrate knowledge of less than or equal to one hour T.S for a non-nuclear instrumentation system (ice condenser door monitoring instrumentation).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6573

Development References

SLC 16.6-3

Student References Provided

SYS016 2.2.39 - Non-Nuclear Instrumentation System (NNIS)

SYS016 GENERIC

Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13)

Remarks/Status

SYS028 K5.03 - Hydrogen Recombiner and Purge Control System (HRPS)
Knowledge of the operational implications of the following concepts as they apply to the HRPS: (CFR: 41.5 / 45.7)
Sources of hydrogen within containment ....

Concerning hydrogen production during a design basis large break LOCA:

1. Which one of the following is a larger contributor to post LOCA hydrogen buildup in containment?

2. Per OP/1/A/6450/010 (Containment Hydrogen Control Systems), what minimum containment hydrogen concentration requires Technical Support Center (TSC) approval prior to placing the Hydrogen Recombiners in service?

A. 1. Zirc-Water reaction in core region
2. 4%

B. 1. Zirc-Water reaction in core region
2. 6%

C. 1. Dissolved hydrogen in the NC System
2. 4%

D. 1. Dissolved hydrogen in the NC System
2. 6%

General Discussion

In the conditions provided, the major source of hydrogen production in a post LOCA inadequate core cooling scenario is Zirc-water reaction. This reaction is self-sustaining at temperatures above 2600 °F and can result in flammable or explosive concentrations of hydrogen. Hydrogen is added to the NC system during normal operation and would be released to the containment atmosphere during a LOCA but the volume would be insignificant compared to that which would be produced from the metal water reaction.
Per the operating procedure, placing hydrogen Recombiners in service, if containment H2 concentration is >= 6%, the crew is directed to consult with the TSC prior to taking this action

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because 4% is the flammability limit of H2 so it would be reasonable for the applicant to consider that placing a recombiner in service could represent a risk of initiating a H2 burn and therefore seeking further consideration from the TSC would be appropriate.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because H2 coming out of solution would be a contributor of H2, the applicant may not understand the significance of Zirc-water compared to other post accident contributors.

Part 2 is plausible because 4% is the flammability limit of H2 so it would be reasonable for the applicant to consider that placing a recombiner in service could represent a risk of initiating a H2 burn and therefore seeking further consideration from the TSC would be appropriate.

Answer D Discussion

Part 1 is plausible because H2 coming out of solution would be a contributor of H2, the applicant may not understand the significance of Zirc-water compared to other post accident contributors.

Part 2 is correct.

Basis for meeting the KA

KA is matched because question tests the primary source of hydrogen within containment in a post accident environment. And tests knowledge of the maximum H2 concentration that allows Recombiners to be placed in service without TSC approval (operational implication).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 1761

Development References

AM lesson
OP/1/A/6450/10

Student References Provided

SYS028 K5.03 - Hydrogen Recombiner and Purge Control System (HRPS)

Knowledge of the operational implications of the following concepts as they apply to the HRPS: (CFR: 41.5 / 45.7)

Sources of hydrogen within containment ....

Remarks/Status

SYS045 A2.12 - Main Turbine Generator (MT/G) System

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 : (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Control rod insertion limits exceeded (stabilize secondary) ....

Given the following Unit 1 timeline:

1000

- The Unit has experienced a runback, from 100% power, following a trip of 1A CFPT

1003

- Main Turbine target load has been reached
- 1AD-2 A/9 (Control Rod Bank Lo Limit) illuminates

1005

- 1AD-2 B/9 (Control Rod Bank Lo-Lo Limit) illuminates
- Steam Dumps have closed
- Temperature Error meter indicates (+) 1.8° F

Entry into the Action Statement of Tech Spec 3.1.6 (Control Bank Insertion Limits) is FIRST required at _____(1)_____ .

In accordance with the conditions provided at **1005,** OMP 1-7 (Emergency / Abnormal Procedure Implementation Guidelines) _____(2)_____ state that control rods should be placed in MANUAL.

Which ONE of the following correctly completes the statements above?

A. 1. 1003
   2. does

B. 1. 1003
   2. does NOT

C. 1. 1005
   2. does

D. 1. 1005
   2. does NOT

General Discussion

OMP 1-7 Attachment 11.1 (General Statements of Philosophy) states the following:
"Following a load rejection/turbine runback, the control room crew should place control rods in manual once the steam dumps have closed. This will stop auto rod insertion with Tavg ~ 3°F higher than Tref. With the negative reactivity from the Xenon transient still causing Tavg to decrease, this should allow the crew the time to have a focus brief on borating the NC system and restoring rods above insertion limits (if necessary) without Tavg decreasing excessively below Tref."

The Control Rod Bank Lo Limit annunciator alarms when control rods are within 10 steps of rod insertion limits via DCS calculation. The Control Rod Bank Lo-Lo Limit annunciator alarms when control rods have reached rod insertion limits.

Answer A Discussion

Part 1 is plausible because the Control Rod Bank Lo Limit annunciator has alarmed. However, this is only a warning to inform the operator that control rods are within 10 steps of insertion limits.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because the Control Rod Bank Lo Limit annunciator has alarmed. However, this is only a warning to inform the operator that control rods are within 10 steps of insertion limits.

Part 2 is plausible because AP/03 contains guidance to insert control rods as necessary to maintain T-avg within 1 degree of T-Ref.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because AP/03 contains guidance to insert control rods as necessary to maintain T-avg within 1 degree of T-Ref.

Basis for meeting the KA

The applicant is required predict the impact of a violation of control rod insertion limits during a main generator runback and mitigate the consequences via use of Tech Spec required actions and plant management directive.

Basis for Hi Cog

This question requires more than one mental step:
1. The applicant must compare provided information with that recalled from memory in order to determine the listed temperature error is less than that specified in OMP 1-7.
2. The applicant must separately analyze all listed distractors, as they are valid procedural action for given conditions, and then select the correct answer.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	17 NRC #70 Bank Question 8272

Development References

OMP 1-7 (Emergency/Abnormal Procedure Implementation Guidelines), Attachment 11.1 Sect. 14
UFSAR, Sect. 7.7.1.3.3
AP/1/A/5500/003, Step 2RNO & 13RNOb.3

Student References Provided

SYS045 A2.12 - Main Turbine Generator (MT/G) System

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 : (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Control rod insertion limits exceeded (stabilize secondary) ....

Remarks/Status

SYS055 K3.01 - Condenser Air Removal System (CARS)
Knowledge of the effect that a loss or malfunction of the CARS will have on the following: (CFR: 41.7 / 45.6)
Main condenser ....

Given the following Unit 1 conditions:

- The Unit 1 is at 100% RTP
- Main Condenser Vacuum is lowering due to CSAE malfunction
- The crew has entered AP/1/A/5500/023 (Loss of Condenser Vacuum)

In accordance with AP/23:

A turbine load reduction _____(1)_____ be effective.

As vacuum lowers, the OATC will trip the reactor once it is imminent that vacuum, in any main condenser, will reach a MAXIMUM value of _____(2)_____.

Which ONE of the following correctly completes the statements above?

A. 1. will
2. 24.3 in Hg

B. 1. will
2. 22 in Hg

C. 1. will NOT
2. 24.3 in Hg

D. 1. will NOT
2. 22 in Hg

General Discussion

A note in AP/23 (prior to step 4) states that reducing turbine load is only effective when low vacuum is caused by reduced condenser circulating water cooling.

AP/23 Foldout page provides direction to trip the reactor if reactor power is >69% and condenser vacuum is trending to 22 in Hg in any condenser section imminent.

Answer A Discussion

Part 1 is plausible because this is a strategy utilized within AP/23 and an applicant may reason that this would be effective.

Part 2 is plausible because the AP/23 foldout page guidance directs turbine trip at 24.3 "Hg but only if turbine load is less than 360 MWs.

Answer B Discussion

Part 1 is plausible because this is a strategy utilized within AP/23 and an applicant may reason that this would be effective.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because the AP/23 foldout page guidance directs turbine trip at 24.3 "Hg but only if turbine load is less than 360 MWs.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

When presented a malfunction of the CARS, the applicant must demonstrate knowledge of the effect (i.e. procedural trip setpoint to protect the condenser along with effect of turbine load reduction combined with CARS malfunction).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

AP/23 Note prior to Step 4
Enclosure 1

Student References Provided

SYS055 K3.01 - Condenser Air Removal System (CARS)
Knowledge of the effect that a loss or malfunction of the CARS will have on the following: (CFR: 41.7 / 45.6)
Main condenser ....

Remarks/Status

SYS072 A1.01 - Area Radiation Monitoring (ARM) System

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: (CFR: 41.5 / 45.5)

Radiation levels ....

Concerning area monitors 1EMF-18 and 1EMF-19 (Reactor Coolant Filter A and B):

1EMF-18 and 1EMF-19_____(1)_____ required by Tech Specs, and _____(2)_____ listed as symptoms for entry into AP/1/A/5500/018 (High Activity in Reactor Coolant).

Which ONE of the following correctly completes the statement above?

A. 1. are
   2. are

B. 1. are
   2. are NOT

C. 1. are NOT
   2. are

D. 1. are NOT
   2. are NOT

General Discussion

EMF18 and 19 are listed in "Symptoms" section of AP/18. Section 5.6 of the lesson plan for EMF explains that there are NO SLC or TS requirements associated with these radiation monitors

Answer A Discussion

Part 1 is plausible because other area monitors (i.e. EMF-15), other RCS process monitors (i.e. EMF-48) and most other process monitors are required to be functional per TS / SLC.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because other area monitors (i.e. EMF-15), other RCS process monitors (i.e. EMF-48) and most other process monitors are required to be functional per TS / SLC.

Part 2 is plausible if the applicant believes that only EMF-48 (RCS Sample line) process EMF is required for AP/18 entry.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible if the applicant believes that only EMF-48 (RCS Sample line) process EMF is required for AP/18 entry.

Basis for meeting the KA

Monitoring rising radiation levels on area radiation monitors (EMF-18 & 19), the applicant is required to predict procedure and Tech Spec entry requirements.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	13 Audit #24

Development References	Student References Provided
AP/18 EMF LP, Sect. 5.5	

SYS072 A1.01 - Area Radiation Monitoring (ARM) System

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: (CFR: 41.5 / 45.5)

Radiation levels ....

Remarks/Status

EARLY REVIEW QUESTION

SYS075 A4.01 - Circulating Water System

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

Emergency/essential SWS pumps ....

Given the following initial conditions:

- Units 1 & 2 are at 100% RTP
- 1A RN pump is in service

Subsequently:

- The following Unit 1 annunciators are lit
  - 1AD-12 B/2 “RN PIT A Screen Hi D/P”
  - 1AD-12 B/1 “RN Pump Intake Pit A Level – LO”
  - 1AD-12 E/2 “RN Pit A Swap to SNSWP”
- The crew has entered AP/0/A/5500/020 (Loss of Nuclear Service Water), Case II (Loss of RN Pit Level)

Enclosure 2 (RN Valve Alignment for RN Swap to SNSWP) will direct the BOP to ensure _____(1)_____ is closed .

Assuming no operator action, 1B RN Pump _____(2)_____ in service.

Which ONE of the following correctly completes the statements above?

A. 1. 1RN-47A (RN Supply X-Over Isol)
   2. is

B. 1. 1RN-47A (RN Supply X-Over Isol)
   2. is NOT

C. 1. 1RN-48B (RN Supply X-Over Isol)
   2. is

D. 1. 1RN-48B (RN Supply X-Over Isol)
   2. is NOT

General Discussion

As a result of an Emergency Low Pit Level actuation signal, opposite train crossover isolation valves will close automatically. AP/20 contains steps to ensure proper alignment of these valves.

Low Pit level in either pit will start both trains of NSW pumps. Therefore, low A pit level will start B train pumps.

Answer A Discussion

Part 1 is plausible because the low pit level has been experienced in "A" train. However, the associated signal repositions the opposite train valve.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because the low pit level has been experienced in "A" train. However, the associated signal repositions the opposite train valve.

Part 2 is plausible because low pit level was actuated on "A" train.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because low pit level was actuated on "A" train.

Basis for meeting the KA

The applicant is required to monitor essential SWS pump operation in the control room by demonstrating knowledge of opposite train pump response following low pit level actuation.

Basis for Hi Cog

This question requires more than one mental step. The applicant must analyze the given information, compare to knowledge recalled from memory, and then make a determination in order to correctly answer the question.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	MODIFIED	Bank Question 6615

Development References

OP-CN-PSS-RN (Nuclear Service Water Lesson Plan), Section 4.4 & 12.2
AP/0/A/5500/020 (Loss of Nuclear Service Water), Case 2 (Loss of RN Pit Level), step 7, and Encl 2 (RN Valve Alignment for RN Swap to SNSWP), Step 4

Student References Provided

SYS075 A4.01 - Circulating Water System
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
Emergency/essential SWS pumps ....

Remarks/Status

SYS086 K6.04 - Fire Protection System (FPS)
Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the : (CFR: 41.7 / 45.7)
Fire, smoke, and heat detectors ....

Given the following Unit 1 conditions:

- A sprinkler head located at Unit 1 Main Turbine Bearing #8 is open due to failure of the heat sensitive element

As a result of this malfunction, RF discharge flow to the Main Turbine Bearing #8 _____(1)_____ initiate.

The purpose of this deluge system is to engulf equipment with a large quantity of _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. will
   2. water

B. 1. will
   2. foam extinguishing agent

C. 1. will NOT
   2. water

D. 1. will NOT
   2. foam extinguishing agent

General Discussion

The turbine bearing deluge system is a dry piping distribution system. In order to prevent inadvertent initiation two events must occur to initiate flow. Heat sensitive elements in sprinkler heads must be destroyed to allow flow, and the mulsifyre valve must be manually actuated.

The purpose of this system is to deliver large quantities of fire protection water to totally engulf equipment.

Answer A Discussion

Part 1 is plausible because this would be the correct answer for a wet piping system (i.e. transformers, main turbine lube oil tanks).

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because this would be the correct answer for a wet piping system (i.e. transformers, main turbine lube oil tanks).

Part 2 is plausible because a foam extinguishing agent is the best method of extinguishing an oil based fire (main turbine bearings).

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because a foam extinguishing agent is the best method of extinguishing an oil based fire (main turbine bearings).

Basis for meeting the KA

Given a malfunction of the fire protection system, the applicant is required to determine effect on associated heat detectors.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
RFY LP Sections K.3, and L

Student References Provided

SYS086 K6.04 - Fire Protection System (FPS)

Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the : (CFR: 41.7 / 45.7)

Fire, smoke, and heat detectors ....

Remarks/Status

GEN2.1 2.1.15 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Given the following conditions:

- A clarification related to EAL classification needs to be communicated to all licensed SROs via Standing Instruction and has been prepared by a member of the Emergency Planning group

In accordance with AD-OP-ALL-0111 (Operations Communications):

This Standing Instruction _____(1)_____ be approved by another member of the Emergency Planning group..

Non-impacted operators (i.e. AOs) _____(2)_____ be exempted from documented review of this Standing Instruction

Which ONE of the following correctly completes the statements above?

A. 1. can
   2. can

B. 1. can NOT
   2. can

C. 1. can
   2. can NOT

D. 1. can NOT
   2. can NOT

ILT 21 CNS RO NRC Examination **QUESTION 66** **B**

General Discussion

Per AD-OP-ALL-0111

Standing Instructions are written and prepared by any knowledgeable person and are approved and signed by Operations management (SM or higher)

If the Standing Instruction only impacts certain individuals in Operations, the SM can allow an exemption for that Standing Instruction review (e.g., a Standing Instruction that discusses EAL classification may not be applicable to AO's and the SM can waive their requirement to review).

.

Answer A Discussion

Part 1 is plausible because any knowledgeable person can prepare a Standing Instruction.

Part 2 is correct.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because any knowledgeable person can prepare a Standing Instruction.

Part 2 is plausible because other informational communication methods contain no allowance for exemption (i.e. Operational Supplemental Information Package).

Answer D Discussion

Part 1 is correct..

Part 2 is plausible because other informational communication methods contain no allowance for exemption (i.e. Operational Supplemental Information Package).

Basis for meeting the KA

The applicant is required to demonstrate knowledge of administrative requirements related to Operations communications (Standing Instructions).

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References

AD-OP-ALL-0111 Sections 5.6.2.c & 5.6.7

Student References Provided

GEN2.1 2.1.15 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Remarks/Status

GEN2.1 2.1.37 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of procedures, guidelines, or limitations associated with reactivity management. (CFR: 41.1 / 43.6 / 45.6)

Given the following conditions:

- Unit 1 has suffered a loss of Main Feed Pump runback from 100% RTP
  - Control rods failed to automatically insert on the runback
- Unit 2 is currently raising power to 100% RTP following Control Valve Movement Testing
  - An ILT student is manipulating control rods under the instruction of the OATC

In accordance with AD-OP-ALL-0203 (Reactivity Management):

An additional Reactor Operator _____(1)_____ required to peer check the Unit 1 OATC manually operating failed control rods.

An additional Reactor Operator _____(2)_____ required to peer check control rod manipulations performed by the ILT student.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. is

B. 1. is
   2. is NOT

C. 1. is NOT
   2. is

D. 1. is NOT
   2 is NOT

General Discussion

Although normally required, peer checks are not required for reactivity manipulations during performance of Abnormal and Emergency procedures. The Unit 1 conditions represents an abnormal conditions which would require entry into the load rejection AP. Therefore, peer check for manual operation of failed control rod function would not be required.

Trainees and supervising reactor operators are considered to be one operator. Therefore, a second licensed operator is required to perform peer checks for reactivity manipulations performed by a trainee/RO combination.

Answer A Discussion

Part 1 is plausible because all other control rod manipulations require a peer check. Also plausible because the question does not directly state that abnormal procedure guidance is in effect. Applicant must deduce plant conditions to determine AP entry and then apply reactivity management guidance.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because all other control rod manipulations require a peer check. Also plausible because the question does not directly state that abnormal procedure guidance is in effect. Applicant must deduce plant conditions to determine AP entry and then apply reactivity management guidance.

Part 2 is plausible because the trainee is being monitored by a licensed reactor operator which could be reasoned to be the peer checker.

Answer C Discussion

CORRECT: See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because the trainee is being monitored by a licensed reactor operator which could be reasoned to be the peer checker.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the guidelines associated with reactivity management procedures.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
AD-OP-ALL-0203 Sect. 4.4.6.2 5.4.7.7.c

Student References Provided

GEN2.1 2.1.37 - GENERIC - Conduct of Operations

Conduct of Operations

Knowledge of procedures, guidelines, or limitations associated with reactivity management. (CFR: 41.1 / 43.6 / 45.6)

Remarks/Status

GEN2.2 2.2.1 - GENERIC - Equipment Control
Equipment Control
Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. (CFR: 41.5 / 41.10 / 43.5 / 43.6 / 45.1)

Given the following Unit 1 conditions:

- A Unit startup is in progress in accordance with OP/1/A/6100/001 (Controlling Procedure for Unit Startup)
- Auxiliary Steam (AS) from Unit 2 is being used for turbine warming
- NC system pressure is 2235 psig
- Steam dumps are controlling NC Tavg at 557ºF
- The crew is preparing to restore AS to a normal alignment by closing 1AS-4, (Main Steam to AS HDR CTRL Bypass)

Operation of 1AS-4 is performed _____(1)_____ the Control Room.

In accordance with AD-OP-ALL-0203, (Reactivity Management), the Unit startup will require a dedicated _____(2)_____ with no concurrent duties.

Which ONE of the following correctly completes the statements above?

A. 1. outside
   2. RO AND SRO

B. 1. inside
   2. RO AND SRO

C. 1. outside
   2. RO ONLY

D. 1. inside
   2. RO ONLY

General Discussion

1AS-4 is a manually operated valve in the turbine building and the operation of this valve, as described in the stem, is designated as a reactivity management per the Operating Procedure for Unit Startup, Enclosure 4.1, Step 3.141. Per AD-OP-ALL-0203, an SRO and an RO shall be dedicated to the reactor startup, with no concurrent responsibilities.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is plausible because 1AS-2 (Main Steam to Aux Stm) is operated by the control operator.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible if applicant misapplies guidance for a Reactivity Management Activity (R2) which only requires a dedicated RO with no concurrent duties, and CRS oversight (but not required to be dedicated). The stem conditions are for an R1 activity, which requires an SRO and an RO.

Answer D Discussion

Part 1 is plausible because 1AS-2 (Main Steam to Aux Stm) is operated by the control operator.

Part 2 is plausible if applicant misapplies guidance for a Reactivity Management Activity (R2) which only requires a dedicated RO with no concurrent duties, and CRS oversight (but not required to be dedicated). The stem conditions are for an R1 activity, which requires an SRO and an RO.

Basis for meeting the KA

While performing pre-startup activities, the applicant is required to demonstrate knowledge of the control locations for specific plant equipment and associated administrative controls and requirements for management of the reactivity evolution.

Basis for Hi Cog

At first look, may appear to be simple recall/recognition of the definition of reactivity management from a procedure. But, in this case, the applicant must analyze the given conditions and recognize that a Unit startup is categorized as a Reactivity Management evolution that is an R1 (though they do not need to know this number), and that it is the "highest" level of reactivity management requirements, to arrive at the correct answer.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6965

Development References	Student References Provided
OP/1/A/6100/001 (Controlling Procedure for Unit Startup), Encl. 4.18, steps 3.1.6, and 3.2.5	

GEN2.2 2.2.1 - GENERIC - Equipment Control

Equipment Control

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. (CFR: 41.5 / 41.10 / 43.5 / 43.6 / 45.1)

Remarks/Status

GEN2.2 2.2.22 - GENERIC - Equipment Control
Equipment Control
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Concerning Tech Spec 2.1.1 (Reactor Core SLs):

The peak centerline fuel temperature shall be maintained less than _____(1)_____ .

This limit _____(2)_____ change over core life.

Which ONE of the following correctly completes the statements above?

A. 1. 2200°F
   2. does

B. 1. 2200°F
   2. does NOT

C. 1. 5080°F
   2. does

D. 1. 5080°F
   2. does NOT

General Discussion

Per T.S. 2.1.1, the peak centerline temperature shall be maintained <5080 deg F, decreasing 58 deg ever 10,000MWd/mtU of fuel burnup.

Answer A Discussion

Part 1 is plausible because this is the value listed in the Fuel Design Criteria.

Part 2 part is correct.

Answer B Discussion

Part 1 is plausible because this is the value listed in the Fuel Design Criteria.

Part 2 is plausible because it is reasonable to believe that this value does NOT change since it is a "safety limit", and the other safety limits do not have a similar change over core life.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is correc.t

Part 2 is plausible because it is reasonable to believe that this value does NOT change since it is a "safety limit", and the other safety limits do not have a similar change over core life.

Basis for meeting the KA

The KA is matched because the applicant must demonstrate knowledge of the a Tech Spec Safety Limit.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	14 NRC #70

Development References

T.S. 2.1.2 (RCS Pressure SL)
T.S.B 3.5.2

Student References Provided

GEN2.2 2.2.22 - GENERIC - Equipment Control
Equipment Control
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Remarks/Status

GEN2.2 2.2.43 - GENERIC - Equipment Control
Equipment Control
Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 2 conditions:

- It has been discovered that an OAC alarm is providing false indication due to a failed input

In accordance with AD-OP-ALL-1000 (Conduct of Operations):

The deleted OAC alarm _____(1)_____ required to be logged in eSOMS.

An audit of deleted computer alarms is required ____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. weekly

B. 1. is
   2. monthly

C. 1. is NOT
   2. weekly

D. 1. is NOT
   2. monthly

General Discussion

Per AD-OP-ALL-1000:

Ensure deleted OAC computer alarms are logged in the eSOMS narrative log, unless being deleted per an approved procedure that will restore them at the conclusion of the procedure.

A weekly audit of deleted computer alarms will be performed to ensure that the issue is captured in an active Work Order/Work Request with appropriate priority assigned.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because an audit of control board work request stickers is required to be performed monthly.

Answer C Discussion

Part 1 is plausible because those OAC alarms deleted by procedure do not require logging.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because those OAC alarms deleted by procedure do not require logging.

Part 2 is plausible because an audit of control board work request stickers is required to be performed monthly.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the process used to track an inoperable computer alarm.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	NEW	

Development References
AD-OP-ALL-1000, Sect. 5.5.8.1 & 5.5.8.3.f(3)

Student References Provided

GEN2.2 2.2.43 - GENERIC - Equipment Control

Equipment Control

Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13)

Remarks/Status

GEN2.3 2.3.4 - GENERIC - Radiation Control
Radiation Control
Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)

Given the following:

- A General Emergency has been declared
- A mission to protect valuable property is required
- The operator who volunteers for the mission has received radiation exposure (TEDE) of 1649 mRem this year, all of which was received at Duke Energy Sites

The limit associated with this emergency exposure is _____(1)_____ and it _____(2)_____ account for the worker's current occupational exposure.

Which ONE of the following correctly completes the statement above?

A. 1. 5 Rem
   2. does

B. 1. 5 Rem
   2. does NOT

C. 1. 10 Rem
   2 does

D. 1 10 Rem
   2 does NOT

General Discussion

According to AD-EP-ALL-0205 Section 5.1 Emergency Exposure Determination, the NOTES states "Emergency exposure limits are exclusive of current occupational exposure." In Table 1, Emergency Exposure Limits, the Activity "Protecting valuable property when lower dose is not practicable." has a TEDE Limit (Rem) of 10 Rem.

Answer A Discussion

Part 1 is plausible because AD-EP-ALL-0205 clearly states that 5 Rem is the limit for all activities during an emergency.

Part 2 is plausible the candidate may incorrectly recall how emergency exposures are accounted for. This guidance is an exception to normal accounting of occupational exposure.

Answer B Discussion

Part 1 is plausible because AD-EP-ALL-0205 clearly states that 5 Rem is the limit for all activities during an emergency.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible the candidate may incorrectly recall how emergency exposures are accounted for. This guidance is an exception to normal accounting of occupational exposure.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The KA is matched because the operator candidate must have knowledge of the emergency radiation exposure limits in order to answer the question correctly.

Basis for Hi Cog

The question is at the memory cognitive level, because the operator candidate must recall bits of informatin (emergency exposure limits and rules) in order to answer the question correctly.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 8776

Development References	Student References Provided
AD-EP-ALL-0205, Sect. 5.1	

GEN2.3 2.3.4 - GENERIC - Radiation Control

Radiation Control

Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)

Remarks/Status

GEN2.3 2.3.5 - GENERIC - Radiation Control
Radiation Control
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.11 / 41.12 / 43.4 / 45.9)

Given the following Unit 1 conditions:

- An Operator is performing a valve lineup in the Unit 1 Auxiliary Building
- While working in the area, the Operator receives a Dose Rate alarm on his Electronic Dosimeter (ED)
- After a few seconds, the Dose Rate alarm automatically clears
- The possibility of a Dose Rate alarm was NOT discussed during the RP brief

In accordance with PD-RP-ALL-0001 (Radiation Worker Responsibilities):

the Operator _____(1)_____ .

if the Operator receives a Dose alarm, the alarm _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. must stop work, exit the area, and notify RP immediately
   2. will not clear until the ED is reset

B. 1. must stop work, exit the area, and notify RP immediately
   2. will automatically clear after 10 seconds

C. 1. may continue to work unless two additional dose rate alarms are received
   2. will not clear until the ED is reset

D. 1. may continue to work unless two additional dose rate alarms are received
   2. will automatically clear after 10 seconds

General Discussion

For Dose Rate alarms, the Electronic Dosimeter (ED) alarm will automatically clear when the dose rate decrease to less than 80% of the Dose Rate alarm setpoint.

If a Dose Rate alarm is received while transiting to a work area, the individual may continue following the travel path provided the alarm clears prior to arriving at the work area.

If a Dose Rate alarm is anticipated (i.e. it has been discussed during the RP brief prior to beginning work) the individual may continue work. However, if a third anticipated Dose Rate alarm is received, the worker must exit the area and notify RP.

For any unexpected Dose Rate alarm, the worker is expected to stop work, exit the area, and notify RP immediately.

For ED Dose alarms, the alarm WILL NOT clear until the ED is reset.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because a Dose Rate alarm will automatically clear. The 10 seconds is plausible because ED set points can be viewed any time after logging on to EDC by pressing and holding the Dose/Dose Rate toggle switch on the ED for 10 seconds.

Answer C Discussion

Part 1 is plausible because the alarm immediately clears and for an anticipated alarm this would be the correct action.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because the alarm immediately clears and for an anticipated alarm this would be the correct action.

Part 2 is plausible because a Dose Rate alarm will automatically clear. The 10 seconds is plausible because ED set points can be viewed any time after logging on to EDC by pressing and holding the Dose/Dose Rate toggle switch on the ED for 10 seconds.

Basis for meeting the KA

The applicant must demonstrate knowledge of the use Electronic Dosimeters (i.e. personnel monitoring equipment) and the actions required to respond to an ED alarm.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6575

Development References
PD-RP-ALL-0001 (Radiation Worker Responsibilities) Sect. 5.3.5.2.d(2) & 5.3.5.h(3)

Student References Provided

GEN2.3 2.3.5 - GENERIC - Radiation Control

Radiation Control

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.11 / 41.12 / 43.4 / 45.9)

Remarks/Status

GEN2.4 2.4.29 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Knowledge of the emergency plan. (CFR: 41.10 / 43.5 / 45.11)

During an emergency event:

The MINIMUM level of emergency classification that ALWAYS requires activation of the OSC, TSC, and EOF is a _____(1)_____.

The MINIMUM level of emergency classification that ALWAYS requires an evacuation of all non-essential personnel from the site is a _____(2)_____.

Which ONE of the following correctly completes the statements above?

A. 1. Alert
   2. Site Area Emergency

B. 1. Alert
   2. General Emergency

C. 1. Unusual Event
   2. Site Area Emergency

D. 1. Unusual Event
   2. General Emergency

General Discussion

Per the Station Emergency Plan Lesson Plan:
If an Alert or Higher is declared, activate the ERO.
o OSC, TSC, EOF activation
o ERO can be activated for unusual event at EC discretion

Site Evacuation may be initated based on Emergency Coordinator discretion but is required to be initiated for a Site Area Emergency or above.

Answer A Discussion

CORRECT. See discussion above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because this is the correct answer under previous EP plan guidance.

Answer C Discussion

Part 1 is plausible this is the lowest threshold of the E plan and it is reasonable to believe that any entry into the E plan would warrant activation of ERO.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because the OSC is responsible for communications during an emergency with other onsite groups in the OSC.

Part 2 is plausible because this is the correct answer under previous E plan guidance.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the E plan, including activation and generic evacuation requirements.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	MODIFIED	Bank Question 6377

Development References	Student References Provided
OP-CN-EP-SEP	

GEN2.4 2.4.29 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Knowledge of the emergency plan. (CFR: 41.10 / 43.5 / 45.11)

Remarks/Status

GEN2.4 2.4.3 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Ability to identify post-accident instrumentation. (CFR: 41.6 / 45.4)

Of the four (4) nuclear instruments listed in F-0, (Critical Safety Function Status Trees), for assessing the "Subcriticality" safety function, which ONE is a Post-Accident Monitoring (PAM) instrument required by LCO 3.3.3, "PAM (Post-Accident Monitoring) Instrumentation?"

A. Source Range

B. Intermediate Range

C. Power Range

D. Wide Range

General Discussion

Neutron Flux (Wide Range) is the only one of the four nuclear instruments in the question which is required by LCO 3.3.3, as listed in Table 3.3.3-1, Post Accident Monitoring Instrumentation. All four of the listed nuclear instruments are in F-0 under "Subcriticality".

Answer A Discussion

Plausible, since the applicant could misapply the knowledge that this instrument does appear in a procedure used during an accident.

Answer B Discussion

Plausible, since the applicant could misapply the knowledge that this instrument does appear in a procedure used during an accident.

Answer C Discussion

Plausible, since the applicant could misapply the knowledge that this instrument does appear in a procedure used during an accident.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

This question is a straightforward approach to the K/A by testing the applicant's ability to identify which instrument is required by the PAM Tech. Spec.

Basis for Hi Cog

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Memory	BANK	Bank Question 6970

Development References

T.S. 3.3.3 (Post Accident Monitoring Instrumentation), Table 3.3.3-1
EP/1/A/5000/F-0 (Critical Safety Function Status Trees), Page 3

Student References Provided

GEN2.4 2.4.3 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Ability to identify post-accident instrumentation. (CFR: 41.6 / 45.4)

Remarks/Status

GEN2.4 2.4.9 - GENERIC - Emergency Procedures / Plan

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. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 initial conditions:

- Unit is cooling down for a refueling outage
- NC Thots are at 365ºF
- NC pressure is 400 psig
- All CLAs have been isolated

Subsequently:

- NC pressure and PZR level are steadily lowering
- Containment pressure is rising
- The crew enters AP/1/A/5500/027, (Shutdown LOCA) and maximizes charging and isolates letdown
- Pressurizer level and pressure continue to lower

In accordance with AP/27, the cooldown rate is required to be LESS THAN _____(1)_____ in one hour.

In accordance with OP/1/A/6200/004, (Residual Heat Removal System), ND can be placed in RHR Mode once NC Thots are less than a MAXIMUM temperature of_____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. 100ºF
   2. 350ºF

B. 1. 80ºF
   2. 350ºF

C. 1. 100ºF
   2. 300ºF

D. 1. 80ºF
   2. 300ºF

General Discussion

Per the AP/27, the cooldown rate allowed is 100ºF per hour, and RHR cannot be placed in service until Thots are less than 350ºF.
.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is plausible because AP/10 contains guidance to maintain cooldown less than 80° when placing ND in service.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because this represents the temperature at which LTOP is placed in service.

Answer D Discussion

Part 1 is plausible because AP/10 contains guidance to maintain cooldown less than 80° when placing ND in service.

Part 2 is plausible because this represents the temperature at which LTOP is placed in service.

Basis for meeting the KA

The KA is matched because the question involves a LOCA initiating from a shutdown condition. It is a Large Break LOCA because PZR pressure and level continue to lower, even after the crew maximizes charging and isolates letdown. The applicant is tested on mitigation strategy aspects such as when the low head injection system can be operated in RHR mode, and what the maximum allowed cooldown rate is.

Basis for Hi Cog

The applicant must recognize that emergency conditions exist for placing RHR in service, and then apply the correct guidance (operating procedure vs. an abnormal procedure). Then, recall of the guidance in the AP is required. This is more than one mental step, and therefore, is a high cog level question.

Basis for SRO only

Job Level	Cognitive Level	QuestionType	Question Source
RO	Comprehension	BANK	Bank Question 6576

Development References

AP/1/A/5500/027, Step 32
OP/1/A/6200/004, Encl. 4.10, Step 2.3

Student References Provided

GEN2.4 2.4.9 - GENERIC - Emergency Procedures / Plan

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. (CFR: 41.10 / 43.5 / 45.13)

Remarks/Status

APE027 2.2.22 - Pressurizer Pressure Control System (PZR PCS) Malfunction
APE027 GENERIC
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Given the following Unit 2 conditions:

- Unit is at 100% RTP
- A DCS malfunction causes the Selected Pressurizer Pressure 1 input to the Pressurizer Master Controller to fail **HIGH**
- In accordance with AP/2/A/5500/011 (Pressurizer Pressure Anomalies) the BOP has manually closed 2NC-34A (PZR PORV) and 2NC-27 / 2NC-29 (Pressurizer Spray Valves)
- The Pressurizer Pressure Master Controller is in MANUAL

In accordance with **Tech Spec 3.4.11** (Pressurizer Power Operated Relief Valves):

2NC-34A is required to operable in MODES _____(1)_____

2NC-34A _____(2)_____ currently OPERABLE.

Which ONE of the following correctly completes the statements above?

A. 1. 1 - 3
   2. is

B. 1. 1 - 3
   2. is NOT

C. 1. 1 - 4
   2. is

D. 1. 1 - 4
   2. is NOT

General Discussion

In accordance with Tech Spec 3.4.11, the Pressurizer PORVs are required to be OPERABLE in MODES 1, 2, and 3.

In accordance with Tech Spec 3.4.11 Bases, the Pressurizer PORVs must be capable of being manually opened to relieve pressure. The actions taken by the crew have removed the automatic opening capability of 2NC-34A. However, since it can still be opened manually, it remains OPERABLE.

Answer A Discussion

CORRECT: See explanation above.

Answer B Discussion

Part 1 is plausible because Pressurizer PORVs support the LTOP system in MODE 4. Therefore, the applicant could conclude the Pressurizer PORVs are required to be OPERABLE in MODES 1-4. While this is technically true, it is covered by two separate Technical Specifications.

Part 2 is plausible if the applicant concludes that the Pressurizer PORV must be capable of opening automatically to perform its intended function. This is reasonable since automatic operation of the Pressurizer PORVs is discussed in the Tech Spec 3.4.11 basis as part of the PORVs ability to prevent a Reactor Trip on high pressure during continuous rod withdrawal events. However automatic operation is not required to perform its intended Tech Spec function. Additionally, TS 3.4.12 requires automatic PORV operation for LTOP.

Answer C Discussion

Part 1 is plausible because Pressurizer PORVs support the LTOP system in MODE 4. Therefore, the applicant could conclude the Pressurizer PORVs are required to be OPERABLE in MODES 1-4. While this is technically true, it is covered by two separate Technical Specifications.

Part 2 is correct.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible if the applicant concludes that the Pressurizer PORV must be capable of opening automatically to perform its intended function. This is reasonable since automatic operation of the Pressurizer PORVs is discussed in the Tech Spec 3.4.11 basis as part of the PORVs ability to prevent a Reactor Trip on high pressure during continuous rod withdrawal events. However automatic operation is not required to perform its intended Tech Spec function. Additionally, TS 3.4.12 requires automatic PORV operation for LTOP.

Basis for meeting the KA

The K/A is matched because the applicant is required to demonstrate knowledge of LCOs as applied to a PZR PCS malfunction.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, the PZR PORV automatic function is not required to establish operability per TS 3.4.11 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	Bank Question 7584

Development References	Student References Provided
T.S. 3.4.11 T.S. 3.4.12	

APE027 2.2.22 - Pressurizer Pressure Control System (PZR PCS) Malfunction
APE027 GENERIC
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Remarks/Status

EPE038 EA2.17 - Steam Generator Tube Rupture (SGTR)
Ability to determine or interpret the following as they apply to a SGTR : (CFR 43.5 / 45.13)
RCP restart criteria ....

Given the following Unit 2 initial conditions:

- A SGTR has occurred on 2C S/G
- Prior to initiating NC system cooldown, all NC pumps were stopped due to loss of subcooling

Subsequently:

- NC system cooldown and depressurization is COMPLETE
- NC subcooling is 25°F
- The crew has reached the step in EP/2/A/5000/E-3 (Steam Generator Tube Rupture) to check NCP status

In accordance with the EOP background document for E-3, the principal reason for restarting an NC pump at this point in E-3 is _____(1)_____ during the subsequent recovery.

In accordance with E-3, Enclosure 3 (NC Pump Start), the reason for checking Reactor Vessel Upper Range level prior to the NC pump start is to ensure sufficient inventory to _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. reduce S/G tube thermal stress
   2. accommodate void collapse

B. 1. provide normal Pressurizer spray flow
   2. accommodate void collapse

C. 1. reduce S/G tube thermal stress
   2. provide adequate NC pump NPSH

D. 1. provide normal Pressurizer spray flow
   2. provide adequate NC pump NPSH

General Discussion

In accordance with the background document, NC pump operation is preferred during recovery from a SGTR to provide normal PZR Spray and homogenous fluid temperatures and boron concentrations.

In E-3, NC pumps may not be restarted unless starting requirements are met.

Answer A Discussion

Part 1 is plausible because S/G tube thermal stress is one of the factors considered when determining whether a faulted S/G should be used during recovery actions in E-3.

Part 2 is correct.

Answer B Discussion

CORRECT: See explanation above.

Answer C Discussion

Part 1 is plausible because S/G tube thermal stress is one of the factors considered when determining whether a faulted S/G should be used during recovery actions in E-3.

Part 2 is plausible because NC pumps were stopped earlier due to a loss of subcooling. I would therefore be plausible for applicants to conclude that the check of Reactor Vessel Upper Range level would be one of the starting requirements for the NC pumps to ensure adequate NPSH.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because NC pumps were stopped earlier due to a loss of subcooling. I would therefore be plausible for applicants to conclude that the check of Reactor Vessel Upper Range level would be one of the starting requirements for the NC pumps to ensure adequate NPSH.

Basis for meeting the KA

The KA is matched because the applicant must have knowledge of the NCP restart criteria contained in E-3 (SG Tube Rupture).

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge.

2) The question can NOT be answered by knowing immediate operator actions.

3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.

4) The question can NOT be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.

5) The question does involve assessment of plant conditions and procedure knowledge vs knowledge of the procedure's overall mitigative strategy (procedure basis knowledge for starting a RCP along with start requirements). As such the question is SRO-level knowledge.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	Bank Question 5771

Development References	Student References Provided
EBG/1/5000/E-3 (E-3 Background Document), Page 4,72, & 75	

EPE038 EA2.17 - Steam Generator Tube Rupture (SGTR)

Ability to determine or interpret the following as they apply to a SGTR : (CFR 43.5 / 45.13)

RCP restart criteria ....

Remarks/Status

EPE055 EA2.03 - Loss of Offsite and Onsite Power (Station Blackout)
Ability to determine or interpret the following as they apply to a Station Blackout : (CFR 43.5 / 45.13)
Actions necessary to restore power ....

Given the following Unit 1 conditions:

- The crew has entered EP/1/A/5000/ECA-0.0 (Loss of All AC Power) following a loss of offsite power
- Neither Diesel Generator started manually
- 1B D/G started upon Safety Injection initiation

Following the start of 1B D/G, the CRS will transition to _____(1)_____.

AP/07 Enclosure 5 (Aligning Alternate Power to 1ETA) _____(2)_____ be used for 1ETA alternate power alignment.

Which ONE of the following correctly completes the statements above?

A. 1. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)
   2. will NOT

B. 1. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)
   2. will

C. 1. EP/1/A/5000/ES-1.1 (Safety Injection Termination)
   2. will NOT

D. 1. EP/1/A/5000/ES-1.1 (Safety Injection Termination)
   2. will

General Discussion

ECA-0.0 will direct the crew to attempt to manually start D/Gs. If unsuccessful, SI is initiated in an attempt to restore power. If either method restores power to one train, an exit to procedure in progress is directed. E-0 is the procedure in effect.

Since Safety Injection has been initiated in ECA-0.0, E-0 will direct transition to ES-1.1 in order to complete SI termination.

If power were not restored prior to completion of step 8 (D/G manual start and manual SI initiation) then the crew will remain in ECA-0.0 until exit to ECA-0.1 (SI not required) or ECA-0.2 (SI required).

Once E-0 is entered, direction is provided to refer to AP/07 for essential power restoration. Although ECA-0.0 also contains guidance for power restoration, this guidance will be referenced following the step which directs exit from the procedure.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because direction for alternate power alignement to essential busses is contained in other procedures (i.e. ECA-0.0, and operating procedures). The applicant must know this enclosure is directed from E-0 for a dead essential bus.

Answer B Discussion

CORRECT - See explanation above.

Part 1 is plausible because this would be the correct flowpath if SI were not successful in restoring power to one train of essential power and SI were in progress.

Part 2 is correct.

This pairing is plausible in that AP/07 is referenced from other EPs for power restoration (i.e. E-0 directs AP/07 performance).

Answer C Discussion

Part 1 is plausible because this procedure will be used to secure from the unnecessary safety injection initated in ECA-0.0. However, this transition will occur after completion of applicable steps of E-0.

Part 2 is plausible because direction for alternate power alignement to essential busses is contained in other procedures (i.e. ECA-0.0, and operating procedures). The applicant must know this enclosure is directed from E-0 for a dead essential bus.

Answer D Discussion

Part 1 is plausible because this procedure will be used to secure from the unnecessary safety injection initated in ECA-0.0. However, this transition will occur after completion of applicable steps of E-0.

Part 2 is correct.

Basis for meeting the KA

Given a Loss of Offsite and Onsite Power, the applicant must determine necessary actions (and direct appropriate procedure flowpath) in order to restore power.

Basis for Hi Cog

Question is a higher cognitive level due to requiring the applicants to perform more than one mental step to answer the question. The applicants must compare information provided in the question to that recalled from memory in order to determine procedure flowpath.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge.

2) The question can NOT be answered by knowing immediate operator actions.

3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.

4) The question can NOT be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.

5) The question does require:

- Assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed (assessment of plant conditions to determine proper AC power recovery).

- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps (knowledge of appendix and/or procedure for alternate power alignment).

- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References
EP/1/A/5000/ECA-0.0, Steps 8, 19 EP/1/A/5000/E-0 Step 4, 38

Student References Provided

EPE055 EA2.03 - Loss of Offsite and Onsite Power (Station Blackout)

Ability to determine or interpret the following as they apply to a Station Blackout : (CFR 43.5 / 45.13)

Actions necessary to restore power ....

Remarks/Status
EARLY REVIEW QUESTION

APE057 2.2.44 - Loss of Vital AC Electrical Instrument Bus
APE057 GENERIC
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)

Given the following Unit 1 conditions:

- Unit is at 100% RTP
- 1ERPA has de-energized due to an inverter failure
- The crew is preparing to re-energize 1ERPA via 1VRD per AP/1/A/5500/029 (Loss of Vital or Aux Control Power)

The power supply transfer required to re-align 1ERPA to 1VRD will be performed via a(n) _____(1)_____ .

Once 1ERPA is aligned to 1VRD, the crew _____(2)_____ exit the action statement of Tech Spec 3.8.7 (Inverters – Operating).

Which ONE of the following correctly completes the statements above?

A. 1. manual bypass switch
   2. will

B. 1. manual bypass switch
   2. will NOT

C. 1. automatic transfer switch
   2. will

D. 1. automatic transfer switch
   2. will NOT

General Discussion

Manual Bypass Switches allow transfer from inverter power to Alternate Power without a loss of power to AC Buses.

Per T.S. 3.8.7 Basis for LCO Action A.1:
Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its voltage regulated transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.
If the channel-related inoperable inverter is replaced by its train's swing inverter, the 24 hour limit does not apply (unless the swing inverter is also inoperable).

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because VRD is an acceptable power supply to meet the operability requirements per T.S. 3.8.8 (Inverters - Shutdown).

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because the auxiliary control power system utilizes an automatic transfer switch.

Part 2 is plausible because VRD is an acceptable power supply to meet the operability requirements per T.S. 3.8.8 (Inverters - Shutdown).

Answer D Discussion

Part 1 is plausible because the auxiliary control power system utilizes an automatic transfer switch.

Part 2 is correct.

Basis for meeting the KA

With a given loss of vital AC, the applicant must demonstrate knowledge of the effect of a directive upon system status (operability).

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. The applicant must determine vital AC alignment in accordance with provided information as compared to information recalled from memory and then apply TS basis information.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, regulated power does not satisfy inverter operability requirements per TS 3.8.7 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	MODIFIED	Bank Question 8292

Development References

T.S. 3.8.7 Basis
T.S. 3.8.8 Basis
AP/29, Encl. 10, Step 7.k
EPL Lesson Plan, Page 20

Student References Provided

APE057 2.2.44 - Loss of Vital AC Electrical Instrument Bus
APE057 GENERIC

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)

Remarks/Status

APE058 2.4.41 - Loss of DC Power
APE058 GENERIC
Knowledge of the emergency action level thresholds and classifications. (CFR: 41.10 / 43.5 / 45.11)

Given the following timeline:

1000

- Unit 1 is at 100% RTP
- Unit 2 is currently at 40% RTP and rising following refueling outage
    - 2ETA and 2ETB remain aligned to Unit 1
- Unit 1 suffers a Loss of Offsite Power (LOOP)
- 1A D/G is the ONLY D/G in operation

1230

- Crew disconnects Unit 2 Vital Batteries (per applicable procedure) due to decay of DC Bus voltage

As a result of these events:

At time 1015, the correct EAL classification will be _____(1)_____.

At time 1245, the correct EAL classification will be _____(2)_____.

Which ONE of the following correctly completes the statements above?

REFERENCE PROVIDED

A. 1. ALERT
   2. SITE AREA EMERGENCY

B. 1. ALERT
   2. GENERAL EMERGENCY

C. 1. SITE AREA EMERGENCY
   2. SITE AREA EMERGENCY

D. 1. SITE AREA EMERGENCY
   2. GENERAL EMERGENCY

General Discussion

At time 1000, Unit 1 has suffered a LOOP and has one power source available (1B D/G). Although Unit 2 did not lose normal incoming offsite power it has also suffered a LOOP due to the alignment of essential busses to Unit 1 power. Therefore, Unit 2 has experienced a Loss of all AC. The correct classification would be a Site Area Emergency (SS1.1) This classification would be applicable 15 minutes following the LOOP.

At time 1230, all Unit 2 Vital DC voltage is lost. This results in a SAE. However, combined with a loss of all AC (for 15 minutes), the event is upgraded to General Emergency.

Answer A Discussion

Part 1 is plausible because this would be the correct classification for Unit 1 (only). The stem of the question does not state that the LOOP affects Unit 2. This has to be deciphered via operational knowledge.

Part 2 is plausible because this would be the correct classification if Unit 2 were not in a Loss of all AC condition. The condition of Unit 2 power supply is not directly stated in the stem of the question.

Answer B Discussion

Part 1 is plausible because this would be the correct classification for Unit 1 (only). The stem of the question does not state that the LOOP affects Unit 2. This has to be deciphered via operational knowledge.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because this would be the correct classification if Unit 2 were not in a Loss of all AC condition. The condition of Unit 2 power supply is not directly stated in the stem of the question.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

The applicant must demonstrate knowledge of EAL thresholds related to a Loss of DC power.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. First, the applicant must analyze the conditions given and compare those conditions to the EAL Wall Charts to determine the correct classification.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge.

2) The question can NOT be answered by knowing immediate operator actions.

3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.

4) The question can NOT be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.

5) The question does require:
- Assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed (assessment of plant conditions to determine EAL classification).

Additionally, this question can be classified as an SRO Plant Specific Example. This question requires additional knowledge required for the higher license level and is unique to the SRO/SM position. At CNS it is the responsibility of the SRO to classify the event in the event that an emergency is declared. Per Lesson Plan OP-CN-EP-SEP, Emergency Plan, Objective #2, the SRO is trained to: "When given a set of plant conditions and access to reference materials, correctly classify an event using AD-EP-ALL-0101 (Emergency Classification) and CSD-EP-CNS-0101-02 (EAL Wallcharts). This is identified as an SRO only learning objective.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

D

ILT 21 CNS SRO NRC Examination

QUESTION 80

Development References	Student References Provided
EAL Wall Chart	EAL Wall Charts

APE058 2.4.41 - Loss of DC Power
APE058 GENERIC
Knowledge of the emergency action level thresholds and classifications. (CFR: 41.10 / 43.5 / 45.11)

Remarks/Status
EARLY REVIEW QUESTION

ILT 21 CNS SRO NRC Examination **QUESTION 81** **C**

WE11 EA2.1 - Loss of Emergency Coolant Recirculation
Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation)
(CFR: 43.5 / 45.13)
Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Given the following Unit 1 conditions:

- A LOCA has occurred
- The crew has implemented EP/1/A/5000/ECA-1.1 (Loss of Emergency Coolant Recirc)
- Containment pressure is currently 3.2 PSIG

In accordance with ECA-1.1, while attempting to establish recirculation flow the crew will cool the core by _____(1)_____.

While performing ECA-1.1, if a RED Path occurs on Core Cooling the crew _____(2)_____ transition to FR-C.1 (Response To Inadequate Core Cooling).

Which ONE of the following correctly completes the statements above?

A. 1. aligning the NC system for Bleed and Feed
   2. will

B. 1. aligning the NC system for Bleed and Feed
   2. will NOT

C. 1. dumping steam from intact Steam Generators
   2. will

D. 1. dumping steam from intact Steam Generators
   2. will NOT

General Discussion

In accordance with ECA-1.1, NC system cooling is provided by dumping steam from intact S/Gs to the Main Condenser (if it is available) or to atmosphere. For the example given, a Main Steam Line Isolation has occurred. Therefore, the MSI signal must be reset before dumping steam to cooldown the NC system is available. However, the MSI signal can be reset even with Containment pressure greater than 3.0 PSIG.

If a RED Path occurs on Core Cooling while in ECA-1.1, the crew WILL transition to FR-C.1 (Response to Inadequate Core Cooling). The reason for this is that the crew transitioned to ECA-1.1 from ES-1.3 (Transfer to Cold Leg Recirc) when they were unable to establish Cold Leg Recirc. And, all transitions from ES-1.3 to ECA-1.1 have a step prior to the transition to commence monitoring CSFSTs.

Answer A Discussion

Part 1 is plausible because NC system Feed and Bleed is used in other procedures in the EOP network (i.e. ECA-0.0, FR-H.1). Additionally, because a Main Steam Isolation has occurred, the applicant may conclude that dumping steam is not an option. Especially since Containment pressure is still above 3.0 PSIG (MSI setpoint).

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because NC system Feed and Bleed is used in other procedures in the EOP network (i.e. ECA-0.0, FR-H.1). Additionally, because a Main Steam Isolation has occurred, the applicant may conclude that dumping steam is not an option. Especially since Containment pressure is still above 3.0 PSIG (MSI setpoint).

Part 2 is plausible if the applicant does not recall the step in ES-1.3 prior to the transition to ECA-1.1 which directs the crew to commence monitoring CSFSTs. Also plausible since some Emergency procedures (ES-1.3 Transfer to Cold Leg Recirc) require monitoring CSF status trees for information only.

Answer C Discussion

CORRECT: See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible if the applicant does not recall the step in ES-1.3 prior to the transition to ECA-1.1 which directs the crew to commence monitoring CSFSTs. Also plausible since some Emergency procedures (ES-1.3 Transfer to Cold Leg Recirc) require monitoring CSF status trees for information only.

Basis for meeting the KA

The K/A is match because the applicant is presented with conditions were ECA-1.1 (Loss of Emergency Coolant Recirc) has been implemented and they must determine based on given conditions if transition to FR-C.1 is allowed should the RED Path conditions for Core Cooling be met.

Basis for Hi Cog

This is a higher cognitive level question because it requires more than one mental step. First the applicant must recall from memory the methods of cooling the NC system allowed by ECA-1.1. Next , the applicant must evaluate the conditions given to determine how the crew entered ECA-1.1 to determine if CSFST monitoring has been implemented.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge.

2) The question can NOT be answered by knowing immediate operator actions.

3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.

4) The question can NOT be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure. As it relates to the first part of the question, knowledge of the fact that initiating a cooldown to cold shutdown is part of the overall mitigating strategy of ECA-1.1, knowledge of the specifics of how that cooling is accomplished is not part of the mitigating strategy. In the second part of the question the requisite knowledge is related to procedure selection/transition and is therefore not part of the purpose, sequence of events, or overall mitigative strategy of the procedure.

5) The question requires the applicant to have sufficiently detailed knowledge of the associated procedure to allow them to make a correct decision regarding procedure transition. The applicant must know that prior to all transitions from ES-1.1 that lead to ECA-1.1, there is a step which directs implementation of the Critical Safety Function Status Trees (CSFSTs). This level of detailed procedure knowledge related to procedure transitions is not expected of ROs. Consequently, the knowledge required to answer this question is SRO level knowledge.

ILT 21 CNS SRO NRC Examination **QUESTION 81** **C**

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	Bank Question 7287

Development References
ES-1.3 (Transfer to Cold Leg Recirc) ECA-1.1 (Loss of Emergency Coolant Recirc)

Student References Provided

WE11 EA2.1 - Loss of Emergency Coolant Recirculation

Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation)
(CFR: 43.5 / 45.13)

Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Remarks/Status

APE005 2.2.12 - Inoperable/Stuck Control Rod
APE005 GENERIC
Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)

Unit 1 is operating at 98% power. PT/1/A/4600/001 (RCCA Movement Test) is in progress. As Control Bank D was being moved, one control rod in Control Bank D slipped to 120 steps withdrawn and stopped. Below is an incore thermocouple map one minute later.

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1						576		572		576					
2			546		599		611		603		605				
3				605				624				BAD		569	
4			597		602				609				617		
5		602				625				616				614	
6	563		611				611				626		623		568
7		602		606				617				620			
8	561				612		634		572				626		
9		619				611				613				614	
10	548		610				606				613				574
11				627				BAD				610		614	
12			612		599				608				622		
13				620		619				576		626		562	
14			540		628		613		608		614				
15						BAD		622		580					

Rod _____(1)_____ is misaligned.

In order to continue Mode 1 operation, Tech. Spec. 3.1.4 (Rod Group Alignment Limits) will require a power reduction, SDM verification, and completion of _____(2)_____.

Which ONE of the following correctly completes the statements above?

A. 1. D-12
   2. $F^N_{\Delta H}(X,Y)$ surveillance ONLY

B. 1. M-4
   2. $F^N_{\Delta H}(X,Y)$ surveillance ONLY

C. 1. D-12
   2. $F^N_{\Delta H}(X,Y)$ AND $F_Q(X,Y,Z)$ surveillances

D. 1. M-4

General Discussion

Based on the map provided, ROD M4 is the indicated rod that slipped to 120 swd (temperature is depressed in this area and higher in opposite side of core). Tech Specs requires Fq and Fdelta h be monitored within 72 hours of having a misaligned rod if we chose to continue operation below 75%.

Answer A Discussion

Part 1is plausible that rod D-12 is the affected rod (located in control bank D), but should see by map evaluation that D4 and M12 have similar indications and are at similar locations in the core, and therefore show that D12 is not misaligned.

Part 2 is plausible because the applicant may reason that local power density will not exceed design limits following the required Tech Spec power reduction. Therefore, the Heat Flux Hot Channel Factor calculation would not be required.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because the applicant may reason that local power density will not exceed design limits following the required Tech Spec power reduction. Therefore, the Heat Flux Hot Channel Factor calculation would not be required.

Answer C Discussion

Part 1 is plausible that rod D-12 is the affected rod (located in control bank D), but should see by map evaluation that D4 and M12 have similar indications and are at similar locations in the core, and therefore show that D12 is not misaligned.

Part 2 is correct.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

Question requires interpretation of incore TC map for stuck control rod discovered per a surveillance procedure.

Basis for Hi Cog

The applicant must evaluate core map to determine which rod is misaligned based on temperature profile and apply Tech Spec knowledge.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021, Rev. 11, ES-401, Attachment 2 (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, Fq and Fdeltah are both required to be monitored per TS 3.1.4 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	Bank Question 7288

Development References
TS 3.1.4 (Rod Group Alignment Limits), T.S.B. 3.1.4 (Rod Group Alignment Limits), Applicable Safety Analysis

Student References Provided

APE005 2.2.12 - Inoperable/Stuck Control Rod

APE005 GENERIC

Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)

Remarks/Status

APE028 AA2.11 - Pressurizer (PZR) Level Control Malfunction

Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: (CFR: 43.5 / 45.13)

Leak in PZR ....

Given the following Unit 1 conditions:

- Unit is at 100% RTP
- A leak has developed on the Pressurizer Level Channel II reference leg
- The CRS has entered the action statement of Tech Spec 3.3.1 (RTS Instrumentation)
- Pressurizer Level Channels indicate as follows:
  - Channel 1 – 55.0%
  - Channel 2 – 96.6%
  - Channel 3 – 55.2%

The action statement of Tech Spec 3.4.9 (Pressurizer) _____(1)_____ required to be entered.

In accordance with Tech Spec 3.3.1 Bases, three channels of pressurizer level (vs. four) are required because _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. of the slow rate of charging that is available

B. 1. is
   2. pressurizer level does NOT provide a backup signal to any other reactor trips

C. 1. is NOT
   2. of the slow rate of charging that is available

D. 1. is NOT
   2. pressurizer level does NOT provide a backup signal to any other reactor trips

---

General Discussion

Although TS 3.4.9 does require water level to be maintained less than 92% in Modes 1-3, the requirement refers to actual level vs. indicated due to malfunction.

Tech Spec 3.3.1 Bases states "A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the valve to lift before reactor high pressure trip."

Answer A Discussion

Part 1 is plausible because TS 3.4.9 requires PZR level to be maintained less than 92%.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because TS 3.4.9 requires PZR level to be maintained less than 92%.

Part 2 is plausible because this wording is similar to that listed in the TS 3.3.1 bases. However, the bases states that PZR level does provide a backup function to the PZR high pressure trip.

Answer C Discussion

CORRECT - See discussion above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because this wording is similar to that listed in the TS 3.3.1 bases. However, the bases states that PZR level does provide a backup function to the PZR high pressure trip.

Basis for meeting the KA

With a given PZR leak and resulting level control malfunction, the applicant is required to determine and interpret resulting actions (TS entry) and bases.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021, Rev. 11, ES-401, Attachment 2 (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, the reason for installing three channels of PZR level instrumentation per TS 3.3.1 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	MODIFIED	14 NRC #88

Development References
TS 3.4.9 TS 3.3.1 Bases, Sect. 9

Student References Provided
PT/1/A/4600/002A rev 234 page 11

APE028 AA2.11 - Pressurizer (PZR) Level Control Malfunction

Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: (CFR: 43.5 / 45.13)

Leak in PZR ....

Remarks/Status

APE033 2.4.46 - Loss of Intermediate Range Nuclear Instrumentation
APE033 GENERIC
Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Given the following Unit 1 initial conditions:

- The crew has begun a Reactor startup
- Intermediate Range Channel N-35 begins to operate erratically
- Reactor Power is stabilized at 10^{-6} % RTP
- AP/1/A/5500/016 (Malfunction of Nuclear Instrumentation System) Case III (Intermediate Range Malfunction) is entered

Subsequently:

- Power remains stable at 10^{-6} % RTP
- AP/16 actions are complete

The "1/N-35A I/R CHANNEL 1 TRIP BYPASS" status light on 1SI-19 _____(1)_____ LIT.

Per Tech Spec 3.3.1, (RTS Instrumentation), the startup to MODE 1 _____(2)_____ continue.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. may

B. 1. is
   2. may NOT

C. 1. is NOT
   2. may

D. 1. is NOT
   2. may NOT

General Discussion

AP/16 Case III directs the crew to place the affected I/R channel level trip bypass switch to the BYPASS position and verify that channels trip bypass status light is lit.

The applicant must compare the given power level to the P-6 (10-5% power) setpoint and realize that the P-6 interlock has not yet been reached.

Per TS 3.3.1, if power is greater than P-6 with one Intermediate Range instrument inoperable then power may be raised into Mode 1. However, if less than P-6 and one IR instrument becomes inoperable then that channel must be restored to operable prior to increasing power > P-6.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because this would be the correct answer if power level was greater than the P-6 setpoint (10 -5%).

Answer B Discussion

CORRECT - See explanation above.

Answer C Discussion

Part 1 is plausible because this would be true if the loss of this I/R channel was due to a loss of control power (i.e. can NOT be bypassed).

Part 2 is plausible because this would be the correct answer if power level was greater than the P-6 setpoint (10 -5%).

Answer D Discussion

Part 1 is plausible because this would be true if the loss of this I/R channel was due to a loss of control power (i.e. can NOT be bypassed).

Part 2 is correct.

Basis for meeting the KA

When given a loss of Intermediate Range Nuclear Instrumentation, the applicant is required to demonstrate knowledge of alarms consistent with plant conditions and knowledge of mitigative actions.

Basis for Hi Cog

Question is higher cognitive because the applicant is required to analyze conditions in the stem and apply to knowledge of Tech Specs and the operation of the ENB system to determine the correct answer. This requires more than one mental step to accomplish.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021, Rev. 11, ES-401, Attachment 2 (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, the TS 3.3.1 Basis requirements for one IR channel inoperable below P-6 interlock setpoint.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	MODIFIED	19 NRC #85 Modified

Development References	Student References Provided
AP/1/A/5000/016, Case III, Step 4 & 5 TSB 3.3.1, Condition H.1	

APE033 2.4.46 - Loss of Intermediate Range Nuclear Instrumentation

APE033 GENERIC

Ability to verify that the alarms are consistent with the plant conditions. (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Remarks/Status

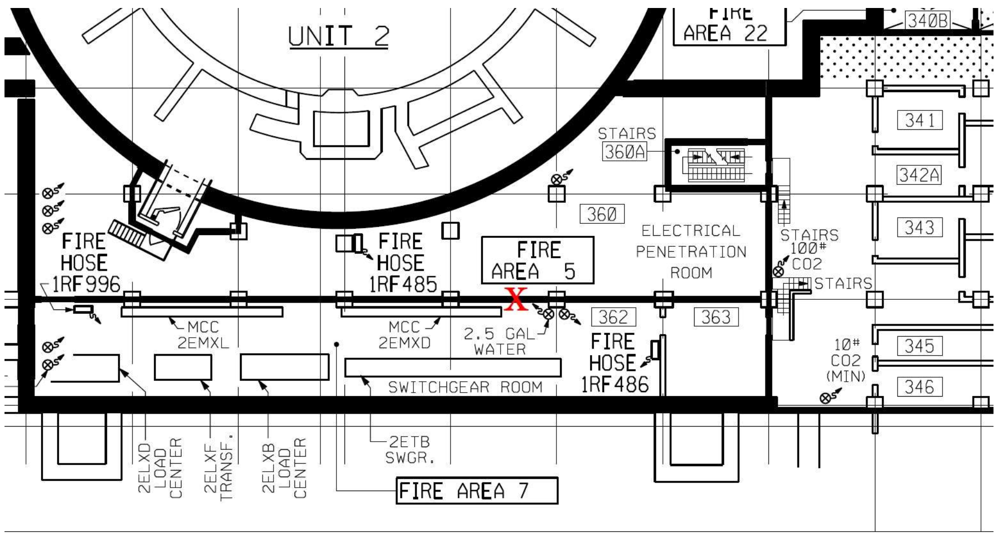
EARLY REVIEW QUESTION

D

APE067 AA2.15 - Plant Fire On Site
Ability to determine and interpret the following as they apply to the Plant Fire on Site: (CFR: 43.5 / 45.13)
Requirements for establishing a fire watch ....

Given the following Unit 2 conditions:

- An Auxiliary Operator has reported a hole in the wall represented by the "X" in the drawing below



In accordance with SLC 16.9-5 (Fire Rated Assemblies):

A High Safety Significant (HSS) Fire Area _____(1)_____ impacted.

A Fire Watch will be established per Condition _____(2)_____ .

Which ONE of the following correctly completes the statements above?

REFERENCE PROVIDED

A. 1. is
2. A

B. 1. is
2. C

C. 1. is NOT
2. A

D. 1. is NOT
2. C

General Discussion

In accordance with Table 16.9-5-3, the Unit 2 ETB Switchgear room and Unit 2 560' Elevation Electrical Penetration rooms are not HSS Fire areas, although the 2ETA switchgear and Unit 2 577' Elevation Electrical Penetrations rooms are HSS. Per Table 16.9-5-4, each Fire Area is assigned an assured shutdown method. This representation does not apply to the SLC condition to be entered upon fire area degradation. A comparison must be performed for each side of the barrier via the chart provided in the Remedial Action section of the SLC. The listed Fire Areas of 5 and 7 each correspond to Assured Shutdown Train/Method "A". Once compared, this requires entry into Condition C of the SLC.

Answer A Discussion

Part 1 is plausible because the "A" train essential switchgear room and electrical penetration room will impact a HSS Fire Area.

Part 2 is plausible if the applicant believes an HSS Fire Area is impacted. Additionally plausible because the listed Fire Areas of 5 and 7 correspond to an Assured Shutdown Train/Method of "A". This could be misconstrued to required entry into condition A of the SLC.

Answer B Discussion

Part 1 is plausible because the "A" train essential switchgear room and electrical penetration room will impact a HSS Fire Area.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because the listed Fire Areas of 5 and 7 correspond to an Assured Shutdown Train/Method of "A". This could be misconstrued to required entry into condition A of the SLC.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The applicant is required to determine proper fire watch requirements after interpreting fire mitigation strategy guidelines of the applicable SLC.

Basis for Hi Cog

This question requires more than one mental step. The applicant must determine applicable fire areas and then reference a chart for comparison to determine the correct answer.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires application of required actions in accordance with rules of application requirements.

Additionally, this question meets the following criteria for an SRO only question as described in NUREG-1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(1) (Conditions and Limitations in the Facility License): Administration of fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

Development References
SLC 16.9-5

Student References Provided
SLC 16.9-5

APE067 AA2.15 - Plant Fire On Site

Ability to determine and interpret the following as they apply to the Plant Fire on Site: (CFR: 43.5 / 45.13)

Requirements for establishing a fire watch ....

Remarks/Status

SYS004 2.2.22 - Chemical and Volume Control System
SYS004 GENERIC
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Given the following Unit 1 conditions:

- The Unit is stable at 100% RTP
- Operators are performing PT/1/A/4150/001 A (NC Pump Seal Injection Flow Verification)
- 1NV-294 is fully open
- Seal Injection flow is 42 gpm and stable

Based on the conditions above, the Action Statement of Tech Spec 3.5.5 (Seal Injection Flow) _____(1)_____ required to be entered.

In accordance with the Bases of Tech Spec 3.5.5, the Seal Injection Flow Limit _____(2)_____ based on the safety analysis assumptions for minimum ECCS Injection flow.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. is

B. 1. is NOT
   2. is

C. 1. is
   2. is NOT

D. 1. is NOT
   2. is NOT

General Discussion

Tech Spec 3.5.5 (Seal Injection Flow) LCO states: Reactor coolant pump seal injection flow shall be < or equal to 40 gpm with centrifugal charging pump operating and the charging flow control valve full open.

Tech Spec 3.5.5 bases states:
The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS). The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions
that are required because RCP seal injection flow is not isolated during SI.

Seal Return isolation valves receive an St containment isolation signal to close following an SI.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is plausible if the applicant believes that the LCO for seal injection flow is to maintain seal flow > 32 gpm, which is the requirement in several procedures associated with the NV system.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible if the applicant believes that seal injection flow is isolated following a safety injection (seal return flow is isolated following an SI).

Answer D Discussion

Part 1 is plausible if the applicant believes that the LCO for seal injection flow is to maintain seal flow > 32 gpm, which is the requirement in several procedures associated with the NV system.

Part 2 is plausible if the applicant believes that seal injection flow is isolated following a safety injection (seal return flow is isolated following an SI).

Basis for meeting the KA

The applicant is required to evaluate performance testing of the seal injection system and make an operational judgement (operability of T.S. 3.5.5) based on that performance.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, Seal Injection Flow limit assumptions listed in TS 3.5.5 Basis.

This question requires knowledge of TS required actions as detailed in the applicable Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	17 NRC #86 Bank Question 7183

Development References	Student References Provided
T.S. 3.5.5 and bases	

SYS004 2.2.22 - Chemical and Volume Control System
SYS004 GENERIC
Knowledge of limiting conditions for operations and safety limits. (CFR: 41.5 / 43.2 / 45.2)

Remarks/Status

SYS013 A2.04 - Engineered Safety Features Actuation System (ESFAS)
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; (CFR: 41.5 / 43.5 / 45.3 / 45.13)
Loss of instrument bus ....

Given the following Unit 2 conditions:

- The Unit is at 100% RTP
- A loss of 2ERPD has occurred
- The crew has entered AP/2/A/5500/029 (Loss of Vital or Aux Control Power)
- NO Tech Spec actions have been addressed

The current Containment Pressure channel logic for the remaining Containment Pressure channels which will cause a **Phase B** actuation is _____(1)_____ .

In accordance with Tech Spec 3.3.2 (ESFAS Instrumentation), when the failed channel is removed from service, I&E will place the Containment Pressure **Hi-Hi** Bistable in _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. 2/3
   2. Trip

B. 1. 2/3
   2. Bypass

C. 1. 1/3
   2. Trip

D. 1. 1/3
   2. Bypass

General Discussion

The normal logic for Phase B actuation based on Hi-Hi Containment Pressure is 2/4 channels. With a loss of ERPD, the associated channel 4 bistable would not be in a tripped condition because Phase B is required to be energized to actuate. Therefore, two of the remaining three channels would be required to actuate.

In accordance with Tech Spec 3.3.2 (ESFAS Instrumentation), the Containment Pressure Hi-Hi Bistable will be placed in BYPASS when the channel is removed from service. The basis for this is that an inadvertent signal on any one of the remaining Containment Pressure Hi-Hi Bistables would result in an inadvertent Phase B isolation.

Answer A Discussion

Part 1 is correct.

Part 2 is plausible because when most channels are removed from service the bistables associated with the failed channel are placed in the tripped position. Even with Containment Pressure the Hi Pressure Bistable is placed in TRIP while the Hi-Hi Pressure Bistable is placed in BYPASS.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

Part 1 is plausible because this would be the correct answer if this instrument actuated upon a loss of power.

Part 2 is plausible because when most channels are removed from service the bistables associated with the failed channel are placed in the tripped position. Even with Containment Pressure the Hi Pressure Bistable is placed in TRIP while the Hi-Hi Pressure Bistable is placed in BYPASS.

Answer D Discussion

Part 1 is plausible because this would be the correct answer if this instrument actuated upon a loss of power.

Part 2 is correct.

Basis for meeting the KA

Given a loss of a vital instrument bus, the applicant is required to predict the impact of this malfunction on an ESFAS system and use procedures (TS) to mitigate the consequences.

Basis for Hi Cog

The applicant must recall from memory the logic for Phase B actuation and then analyze given conditions to determine the effect upon this logic. The applicant must then recall from memory the requirement for removing a Containment Pressure channel (Hi vs. Hi-Hi) from service.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, Hi-Hi Containment pressure channel bistables are placed in bypass to prevent inadvertant signal actuation per TS 3.3.2 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	17 NRC #87 Bank Question 7184

Development References
T.S.B. 3.3.2 (ESFAS), Actions - Condition E T.S. 3.3.2 (ESFAS), Table 3.3.2-1 OP-CN-ECCS-ISE (ESFAS LP), Sect. 5.3 ISE lesson plan Obj. 12

Student References Provided

SYS013 A2.04 - Engineered Safety Features Actuation System (ESFAS)

ILT 21 CNS SRO NRC Examination **QUESTION 87** **B**

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; (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Loss of instrument bus ....

Remarks/Status

SYS059 A2.11 - Main Feedwater (MFW) System

Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Failure of feedwater control system ....

Given the following Unit 1 conditions:

- Unit is at 50% RTP
- A failure of the IFE System (Steam Generator Water Level and Feedwater Pump Speed Control System) results in SG narrow range levels rising to 87%

In order to mitigate the above condition, the CRS will enter _____(1)_____ .

In accordance with T.S. 3.3.2 (ESFAS Instrumentation) Bases, feedwater isolation coincident with low Tavg is a function of the _____(2)_____ interlock.

Consider each statement separately

Which ONE of the following correctly completes the statements above?

A. 1. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)
   2. P-14

B. 1. AP/1/A/5500/002 (Turbine Generator Trip)
   2. P-14

C. 1. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)
   2. P-4

D. 1. AP/1/A/5500/002 (Turbine Generator Trip)
   2. P-4

General Discussion

87% SG level is above the P-14 setpoint for Unit 1 and will cause a Turbine trip Feed Pump trip and Main Feed Water Isolation. With RTP greater than 5% AP-06 immediate actions will require tripping Unit 1 RX and entering E0.

Per TS 3.3.2 Basis, Both interlock P-14 and P-4 have functions to provide a main feedwater isolation signal, but only P-4 provides a main feedwater isolation coinicdent with a lo Tavg signal.

Answer A Discussion

Part 1 Correct

Part 2 Plausible because P-14 does initiate a MFW isolation just not coincident with a lo Tavg signal.

Answer B Discussion

Part 1 is plausible because with Unit 1 less than 69% RTP AP 02 entry criteria have been met.

Part 2 Plausible because P-14 does initiate a MFW isolation just not coincident with a lo Tavg signal.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is plausible with Unit 1 less than 69% RTP AP 02 entry criteria have been met.

Part 2 is correct

Basis for meeting the KA

The applicant is required to predict the impact of the IFE system failure and choose the correct procedure to mitigate the consquence. The applicant is also required to demonstrate knowledge of procedure (Tech Specs) related to mitigation.

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, the specified backup indication used to validate Wide Range S/G level per TS 3.3.3 Basis.

This question meets the following criteria for an SRO only question as described in NUREG 1021, Rev. 11, ES-401, Attachment 2 (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs
2) This question can NOT be answered by knowing information listed "above-the-line".
3) This question can NOT be answered by knowing the TS Safety Limits.
4) This question requires the applicant to have knowledge of the Tech Spec Basis.

This question requires knowledge of TS required actions as detailed in the applicable Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

QUESTION 88

C

Development References
TSB 3.3.3 Sect. 15 & 21 IFE LP Sect. 4.4.2

Student References Provided

SYS059 A2.11 - Main Feedwater (MFW) System

Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Failure of feedwater control system ....

Remarks/Status

SYS061 2.4.30 - Auxiliary / Emergency Feedwater (AFW) System
SYS061 GENERIC
Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)

Given the following Unit 2 conditions:

- A loss of all Feedwater has occurred
- The crew has implemented FR-H.1 (Response To Loss Of Secondary Heat Sink)

Based on the conditions above, this Shift Manager will classify this event as a/an _____(1)_____ .

Per AD-EP-ALL-0111 (Control Room Activation of the ERO), **initial** NRC notification will be performed via transmittal of _____(2)_____ .

Which ONE of the following correctly completes the statements above?

REFERENCE PROVIDED

A. 1. Alert
2. Emergency Notification Form (ENF)

B. 1. Alert
2. NRC Form 361 (Reactor Plant Event Notification Worksheet)

C. 1. Site Area Emergency
2. Emergency Notification Form (ENF)

D. 1. Site Area Emergency
2. NRC Form 361 (Reactor Plant Event Notification Worksheet)

General Discussion

This event will be classified as a Site Area Emergency based on the Potential Loss of both NCS Barrier AND Fuel Clad Barrier from the RED PATH condition on Heat Sink which required entry into FR-H.1.

In accordance with AD-EP-ALL-0111, the Emergency Coordinator will direct the Offsite Communicator to notify the NRC via the approved ENF.

Answer A Discussion

Part 1 is plausible if the applicant fails to recognize a loss of Heat Sink results in a partial loss of RCS barrier and Fuel Clad barrier, or concludes the a potential loss of Fuel Clad barrier requires a loss of Heat Sink AND Core Cooling.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible if the applicant fails to recognize a loss of Heat Sink results in a partial loss of RCS barrier and Fuel Clad barrier, or concludes the a potential loss of Fuel Clad barrier requires a loss of Heat Sink AND Core Cooling.

Part 2 is plausible because Form 361 is the normal required notification tool for NRC notifications.

Answer C Discussion

CORRECT: See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because Form 361 is the normal required notification tool for NRC notifications.

Basis for meeting the KA

With a given loss of auxiliary feedwater, the applicant is required to demonstrate knowledge of an event classification related to system status and determine reporting method to an external agency (NRC).

Basis for Hi Cog

The applicant is required to analyze provided information and then compare with that recalled from memory in order to correctly answer this question.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge.

2) The question can NOT be answered by knowing immediate operator actions.

3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.

4) The question can NOT be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.

5) The question does require:
- Assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed (assessment of plant conditions to determine EAL classification).

Additionally, this question can be classified as an SRO Plant Specific Example. This question requires additional knowledge required for the higher license level and is unique to the SRO/SM position. At CNS it is the responsibility of the SRO to classify the event in the event that an emergency is declared. Per Lesson Plan OP-CN-EP-SEP, Emergency Plan, Objective #2, the SRO is trained to: "When given a set of plant conditions and access to reference materials, correctly classify an event using AD-EP-ALL-0101 (Emergency Classification) and CSD-EP-CNS-0101-02 (EAL Wallcharts). This is identified as an SRO only learning objective.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

ILT 21 CNS SRO NRC Examination

QUESTION 89 C

Development References
Wall Chart AD-EP-ALL-0111, Attach. 1, Step 1.6, Attach. 3, Step 1.2.3

Student References Provided
EAL Wall Chart

SYS061 2.4.30 - Auxiliary / Emergency Feedwater (AFW) System

SYS061 GENERIC

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)

Remarks/Status

SYS064 A2.06 - Emergency Diesel Generator (ED/G) System

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Operating unloaded, lightly loaded, and highly loaded time limit ....

Concerning operation of the Emergency Diesel Generator:

In accordance with OP/1/A/6350/002 (Diesel Generator Operation), the reason for not operating a D/G unloaded for a long period of time _____(1)_____ to prevent sludge buildup in the engine.

In accordance with TS 3.8.1 (AC Sources - Operating) Bases, the maximum loading limit for each D/G is _____(2)_____ KW for up to 2 hours in any 24 hour period.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. 7700

B. 1. is
   2. 8750

C. 1. is NOT
   2. 7700

D. 1. is NOT
   2. 8750

General Discussion

Per OP/1/A/6350/002 "Do NOT operate the D/G at no load or light loads for long periods of time to prevent buildup of carbon and sludge in the engine."

Per procedure caution "The "Turbo Oil Sol Vlv" shall NOT be open for more than 5 minutes to prevent oil overflowing from the turbocharger bearing to the exhaust manifold potentially causing a turbocharger fire."

Per TS 3.8.1 Bases "The continuous service rating of each DG is 7000 kW with 10% overload permissible for up to 2 hours in any 24 hour period"

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because this corresponds to a value of continuous service rating plus 25% (vs. 10%) for a limited period of time. 25% is reasonable as it is used in other applications (i.e. Aux Feedwater Turbine Driven Pump overspeed setpoint).

Answer C Discussion

Part 1 is plausible because the applicant may confuse complications caused by unloaded operation with that caused by excessive turbo oil solenoid use (i.e. oil in exhaust).

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because the applicant may confuse complications caused by unloaded operation with that caused by excessive turbo oil solenoid use (i.e. oil in exhaust).

Part 2 is plausible because this corresponds to a value of continuous service rating plus 25% (vs. 10%) for a limited period of time. 25% is reasonable as it is used in other applications (i.e. Aux Feedwater Turbine Driven Pump overspeed setpoint).

Basis for meeting the KA

The applicant is required to predict the impacts of unloaded D/G operation and use procedure (Tech Spec) to control highly loaded limit time.

Basis for Hi Cog

This question required a calculation in order to determine maximum D/G service rating for limited time.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, the limited continuous service rating specified per TS 3.8.1 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	NEW	

Development References
OP/1/A/6350/002, L&P 2.19 Encl. 4.10, Caution prior to step 3.11 CA LP Sect. 7.2 TSB 3.8.1, Background

Student References Provided

SYS064 A2.06 - Emergency Diesel Generator (ED/G) System

Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use

procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.13)
Operating unloaded, lightly loaded, and highly loaded time limit ....

Remarks/Status
EARLY REVIEW QUESTION

SYS002 A2.04 - Reactor Coolant System (RCS)

Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Loss of heat sinks ....

Given the following Unit 1 initial conditions:

- Unit was at 100% RTP
- The #1 CAPT is out-of-service for maintenance

Subsequently:

- The unit is manually tripped when both CF pumps trip
- 1A and 1B CA pumps fail to automatically start and cannot be started manually
- The crew has implemented EP/1/A/5000/FR-H.1 (Response to Loss of Secondary Heat Sink)
- SG WR level is 40% in all SGs

In accordance with FR-H.1:

NC system Feed and Bleed must be initiated within a MAXIMUM of _____(1)_____ minutes after reaching Feed and Bleed initiation criteria.

After Feed and Bleed is initiated, efforts to restore feedwater flow to the S/Gs _____(2)_____ be terminated.

Which ONE of the following correctly completes the statements above?

A. 1. 8
   2. MAY

B. 1. 4
   2. MAY

C. 1. 8
   2. MAY NOT

D. 1. 4
   2. MAY NOT

General Discussion

In accordance with FR-H.1 Background Document:

"Feed and bleed should be initiated as quickly as possible after meeting criteria in FR-H.1, but must be initiated within 4 minutes of meeting criteria. Note that feed and bleed may need to be initiated within 8 minutes after reactor trips on low-low SG level, since the criteria will be quickly met. "

In accordance with FR-H.1, efforts to restore feedwater flow to the S/Gs must continue until at least 450 GPM flow to one S/G is established (from CF or CA) or NR level in at least one intact S/G is greater than 11%.

Answer A Discussion

Part 1 is plausible because the basis document states that feed and bleed may need to be initiated within 8 minutes after the reactor trips on low-low SG level.

Part 2 is plausible if the applicant concludes that once some form of NC system heat removal is established that efforts to restore an adequate heat sink (via the S/Gs) may be terminated.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible if the applicant concludes that once some form of NC system heat removal is established that efforts to restore an adequate heat sink (via the S/Gs) may be terminated.

Answer C Discussion

Part 1 is plausible because the basis document states that feed and bleed may need to be initiated within 8 minutes after the reactor trips on low-low SG level.

Part 2 is correct.

Answer D Discussion

CORRECT. See explanation above.

Basis for meeting the KA

The K/A is matched because a loss of heat sink has occurred and the applicant must have knowledge of the impact on the RCS (i.e. how long they have after criteria has been met to initiate Feed and Bleed of the NC system). Also, they demonstrate the ability to use procedures to correct, control, and mitigate by having knowledge of when (procedurally) effort to restore feedwater can be terminated and by knowing the procedural time requirement (from the basis document) for initiating feed and bleed.

Basis for Hi Cog

This questions requires more than one mental step. The applicant must first determine the cause of the Reactor Trip and requirements for FR-H.1 entry (per the stem) and then apply that information to procedure basis information recalled from memory.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge.

2) The question can NOT be answered by knowing immediate operator actions.

3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.

4) The question can NOT be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.

5) The question does require:

Assessment of plant conditions and procedure knowledge vs knowledge of the procedure's overall mitigative strategy (knowledge of procedural time requirement for initiation of Feed and Bleed along with procedure direction once F & B has been initiated.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	Bank Question 6583

Development References	Student References Provided
FR-H.1 Step 38 FR-H.1 Background Document, Step 24	

SYS002 A2.04 - Reactor Coolant System (RCS)

Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Loss of heat sinks ....

Remarks/Status

SYS014 A2.03 - Rod Position Indication System (RPIS)
Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations : (CFR: 41.5 / 43.5 / 45.3 / 45.13)
Dropped rod ....

Given the following Unit 1 conditions:

- The crew is performing PT/0/A/4150/019 (1/M Approach to Criticality)
- All shutdown banks are fully withdrawn
- Control Bank A rods are fully withdrawn
- Control Bank B rods are being withdrawn

Subsequently:

- Control Bank B rod K14 drops
- DRPI indication for rod K14:
  - Rod position indication is '0' with RB indication below the rod
- Demand position counter for Bank B Groups 1 & 2 indicates 114 steps

In accordance with T.S. 3.1.7 (Rod Position Indication) Bases, LCO 3.1.7 _____(1)_____ met.

In accordance with PT/0/A/4150/019, the CRS will direct insertion of _____(2)____.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. all control banks ONLY

B. 1. is NOT
   2. all control banks ONLY

C. 1. is
   2. all control banks AND shutdown banks

D. 1. is NOT
   2. all control banks AND shutdown banks

General Discussion

Tech Spec 3.1.7 bases states that one DRPI system and one Bank Demand Position Indication system shall be operable for each control rod. For the control rod position indicators to be OPERABLE, the SR of the LCO must be met and the following:
a. The DRPI system indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4 (Rod Group Alignment Limits)
b. For the DRPI system either Data A or Data B is OPERABLE for each rod; and
c. The Bank Demand Indication system has been calibrated either in the fully inserted position or to the DRPI system.
Conditions in the stem state that Control Bank 'D' rod M14 has fully inserted. Therefore, DRPI indication will not agree within 12 steps of demand counter position for this control rod.

Per the Limits and Precautions of PT/0/A/4150/019:
IF a control rod fails to withdraw or a single rod is dropped during approach to criticality perform one of the following:
-IF malfunction is in Control Bank, insert all Control Banks.
-IF malfunction is in Shutdown Bank, insert all Control and Shutdown Banks.
This guidance is more conservative than that given in AP/1(2)/5500/015, Rod Control Malfunction, and therefore shall take precedence. {PIP C-06-4287}

Answer A Discussion

Part 1 is plausible because DRPI system has suffered no equipment failure. It is within reason to believe only TS 3.1.4 (Rod Group Alignment Limits) would be affected by this malfunction.

Part 2 is correct.

Answer B Discussion

CORRECT - See discussion above.

Answer C Discussion

Part 1 is plausible because DRPI system has suffered no equipment failure. It is within reason to believe only TS 3.1.4 (Rod Group Alignment Limits) would be affected by this malfunction.

Part 2 is plausible because this would be the correct answer if shutdown bank rod dropped into the core.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because this would be the correct answer if shutdown bank rod dropped into the core.

Basis for meeting the KA

Given a Dropped Control Rod, the applicant is required to predict the impact upon the RPIS (TS application) and then demonstrate knowledge of the procedure use to mitigate the consequence.

Basis for Hi Cog

This question is higher cognitive level due to requiring more than one mental step to answer. The applicants have to interpret indications given in the stem of the question, then apply knowledge of Tech Spec 3.1.7 bases to determine the correct answer.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, the detailed requirements for rod position operability including rod group alignment limits per TS 3.1.7 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	MODIFIED	2016 NRC #76

B

QUESTION 92

Development References
PT/0/A/4150/019 L&P 6.4 TSB 3.1.7

Student References Provided

SYS014 A2.03 - Rod Position Indication System (RPIS)

Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations : (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Dropped rod ....

Remarks/Status

SYS017 2.1.7 - In-Core Temperature Monitor (ITM) System
SYS017 GENERIC
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)

Given the following Unit 1 initial conditions:

- Unit is at 100% RTP
- At **0900** a Large Break LOCA occurs
- At **0910** the Shift Manager declares an Alert
- Efforts to establish adequate Safety Injection flow to the core are unsuccessful
- The crew has secured NCPs due to loss of subcooling

Subsequently:

- Core Exit Thermocouples and RVLIS level indications are as follows for the times listed

Time	CETC	RVLIS
0930	650° - Rising	50% - lowering
0945	720° - Rising	46% - lowering
1000	770° - Rising	41% - lowering
1015	790° - Rising	38% - lowering

The Shift Manager re-evaluated EALs at **0945** and is evaluating again at **1010**.

At time **0945**, EAL upgrade criteria _____(1)_____ met.

At time **1015**, the Shift Manager _____(2)_____ declare a General Emergency.

Which ONE of the following correctly completes the statements above?

REFERENCE PROVIDED

A. 1. is
2. will

B. 1. is
2. will NOT

C. 1. is NOT
2. will

D. 1. is NOT
2. will NOT

General Discussion

An Alert (FA1.1) will be declared following the LOCA due to Loss of NCS Barrier - Unisolable leak requiring SI actuation.

At 0945 Core Cooling Orange Path conditions are met - Core Exit Thermocouples greater than 700° and RVLIS level greater than 41%. This condition results in a Potential Loss of the Fuel Clad Barrier and necessitates an upgrade to Site Area Emergency (FS1.1).

At 1000 Core Cooling Red Path conditions are met - Core Exit Thermocouples greater than 700° and RVLIS level NOT greater than 41%. This condition results in a Loss of the Fuel Clad Barrier (vs Potential Loss) which does not change the EAL classification. However, this condition also results in a Potential Loss of the Containment Barrier (after 15 minutes of ineffective restoration).

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible if the applicant notes the change in Fuel Clad barrier, due to Core Cooling Red Path enry, but fails to recognize this condition also affects the Containment Barrier.

Answer C Discussion

Part 1 is plausible if the applicant does not realize that the conditions listed result in a Core Cooling Orange path condition or that this condition causes a Potential Loss of the Fuel Clad Barrier.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible if the applicant does not realize that the conditions listed result in a Core Cooling Orange path condition or that this condition causes a Potential Loss of the Fuel Clad Barrier.

Part 2 is plausible if the applicant notes the change in Fuel Clad barrier, due to Core Cooling Red Path enry, but fails to recognize this condition also affects the Containment Barrier.

Basis for meeting the KA

The applicant is required to evaluate plant performance via In-Core Temperature indications and make an operational judgement (EAL classification) based on core conditions (reactor behavior).

Basis for Hi Cog

The applicant is required to compare listed conditions to items recalled from memory and make operational determinations.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge.

2) The question can NOT be answered by knowing immediate operator actions.

3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.

4) The question can NOT be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.

5) The question does require:
- Assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed (assessment of plant conditions to determine EAL classification).

Additionally, this question can be classified as an SRO Plant Specific Example. This question requires additional knowledge required for the higher license level and is unique to the SRO/SM position. At CNS it is the responsibility of the SRO to classify the event in the event that an emergency is declared. Per Lesson Plan OP-CN-EP-SEP, Emergency Plan, Objective #2, the SRO is trained to: "When given a set of plant conditions and access to reference materials, correctly classify an event using AD-EP-ALL-0101 (Emergency Classification) and CSD-EP-CNS-0101-02 (EAL Wallcharts). This is identified as an SRO only learning objective.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	NEW	

ILT 21 CNS SRO NRC Examination **QUESTION 93** **A**

Development References
EP/1/A/5000/F-0, Core Cooling CSD-EP-CNS-0101-02 Wall Chart

Student References Provided
EAL Wallcharts

SYS017 2.1.7 - In-Core Temperature Monitor (ITM) System
SYS017 GENERIC
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)

Remarks/Status
EARLY REVIEW QUESTION

GEN2.1 2.1.1 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of conduct of operations requirements. (CFR: 41.10 / 45.13)

In accordance with AD-OP-ALL-1001, (Conduct of Abnormal Operations), regarding 10 CFR 50.54 (x) Deviations:

the SRO approving the Deviation _____(1)_____ required to obtain concurrence from a second SRO prior to taking the action.

the NRC Operations Center must be notified within a maximum time of _____(2)_____ after taking the action.

Which ONE of the following correctly completes the statements above?

A. 1. is
   2. 1 HOUR

B. 1. is
   2. 15 minutes

C. 1. is NOT
   2. 1 HOUR

D. 1. is NOT
   2. 15 minutes

General Discussion

Per AD-OP-ALL-1001:

f the TSC is NOT activated, then a SRO must approve a Deviation prior to taking action. If time allows, then concurrence from a second SRO should be obtained prior to taking action.

Notify the NRC per 10 CFR 50.72 requirements due to invoking 50.54(x).
a.Notify the NRC Operations Center using the Emergency Notification System (ENS).
b.When time permits, then notify the NRC Operations Center prior to taking the action.
c.Otherwise, notify the NRC Operations Center as soon as possible and in all cases, within 1 hour after taking the action.

Answer A Discussion

Part 1 is plausible because it is reasonable to believe that a decision of this magnitude would required concurrence. Other declarations do require concurrence (i.e. EAL and PARs). Additionally, AD-OP-ALL-1001 states that concurrence should be obtained, if time allows.

Part 2 is correct.

Answer B Discussion

Part 1 is plausible because it is reasonable to believe that a decision of this magnitude would required concurrence. Other declarations do require concurrence (i.e. EAL and PARs). Additionally, AD-OP-ALL-1001 states that concurrence should be obtained, if time allows.

Part 2 is plausible because other offsite notifications are required within 15 minutes (i.e. EAL notifications). Reasonable to believe that a decision of this magnitude would require the least notification time.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Part 1 is correct.

Part 2 is plausible because other offsite notifications are required within 15 minutes (i.e. EAL notifications). Reasonable to believe that a decision of this magnitude would require the least notification time.

Basis for meeting the KA

The KA is matched because the operator candidate must know information obtained in a conduct of operations procedure, in this case the requirements associated with 10CFR50.54(x) deviations.

Basis for Hi Cog

Basis for SRO only

The question is SRO-ONLY because it requires the operator to possess knowledge of the process by which to take a Deviation from the license conditions, invoking 10CFR50.54(x) which is an SRO function.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	Bank Question 8797

Development References
AD-OP-ALL-1001, Sect. 5.2.7

Student References Provided

GEN2.1 2.1.1 - GENERIC - Conduct of Operations
Conduct of Operations
Knowledge of conduct of operations requirements. (CFR: 41.10 / 45.13)

Remarks/Status

GEN2.1 2.1.25 - GENERIC - Conduct of Operations
Conduct of Operations
Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12)

Given the following Unit 1 conditions:

- Preparation to set the reactor vessel head following **core reload** in progress
- NC system boron concentration is 2705 PPM
- The following surveillances are being performed:
  - PT/1/A/4600/017 (Surveillance Requirements For Unit 1 Shutdown)
  - PT/1/A/4600/002 F (Mode 6 Periodic Surveillance Items)

The surveillance for NC system boron concentration performed during PT/1/A/4600/017 (SR 3.9.1.1) ensures that k_{eff} during MODE 6 remains less than a MAXIMUM of _____(1)_____.

The MINIMUM Boric Acid Tank level required to meet the surveillance requirements of PT/1/A/4600/002 F (TR 16.9-11-4) is _____(2)_____.

Which ONE of the following correctly completes the statements above?

REFERENCE PROVIDED

A. 1. 0.95
   2. 9.8%

B. 1. 0.95
   2. 14.9%

C. 1. 0.98
   2. 9.8%

D. 1. 0.98
   2. 14.9%

General Discussion

In accordance with TS 3.9.1 (Boron Concentration), in MODE 6 boron concentration shall be maintained within the limits specified in the COLR (Core Operating Limits Report). In accordance with the Basis Document for TS 3.9.1 the basis for maintaining boron concentration within limits is to ensure that Keff remains less than 0.95 to ensure that a recriticality does not occur while in MODE 6 (i.e. during fuel movement).

In accordance with SLC 16.9.14, Borated Water Sources (Shutdown) for the BAT to be considered OPERABLE it must meet the minimum volume requirements specified in the COLR. There are two conditions that would require different BAT levels. If the unit was in MODE 6 at the end of a cycle after 455 EFPD and the core had not yet been off loaded, Figure 6 of the COLR would be used to determine the minimum volume and the required minimum level would be 9.8%. However, for the conditions given, since core reload is in progress (i.e. the full cycle core is no longer loaded), the requirements of COLR 2.17 would apply (i.e. 14.9%).

Answer A Discussion

Part 1 is correct

Part 2 is plausible because this would be the correct answer if the core had not yet been off loaded.

Answer B Discussion

CORRECT: See explanation above.

Answer C Discussion

Part 1 is plausible because < .99 is the maximum Keff for Cold Shutdown.

Part 2 is plausible because this would be the correct answer if the core had not yet been off loaded.

Answer D Discussion

Part 1 is plausible because < .99 is the maximum Keff for Cold Shutdown.

Part 2 is correct.

Basis for meeting the KA

The applicant is required to demonstrate the ability to interpret reference materials including making a determination between a table and graph in order to correctly determine minimum boric acid tank volume.

Basis for Hi Cog

This question is higher cognitive level because it requires more than one mental step. The first part of the question requires the applicant to recall from memory the basis for NC system boron surveillance requirement during refueling. The second part requires the applicant to analyze plant conditions and determine which BAT level limit from the COLR applies to the given conditions.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, basis for the NC system boron concentration listed in TS surveillance 3.9.1.1 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	Bank Question 8286

Development References
TS 3.9.1 (Boron Concentration) Basis Core Operating Limits Report (COLR) - 2.17 and Figure 6

Student References Provided
COLR - Pages 28-30

GEN2.1 2.1.25 - GENERIC - Conduct of Operations

Conduct of Operations

Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12)

Remarks/Status

D

GEN2.2 2.2.13 - GENERIC - Equipment Control
Equipment Control
Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)

In accordance with AD-OP-ALL-200 (Clearance and Tagging):

a previously licensed SRO _____(1)_____ sign as clearance APPROVER.

if a clearance is designated as an Exceptional Clearance, at a minimum, it is required to be approved by an Operations _____(2)_____ and a Work Group Supervisor.

Which ONE of the following correctly completes the statements above?

A. 1. can
2. Shift Supervisor

B. 1. can
2. Shift Manager

C. 1. can NOT
2. Shift Supervisor

D. 1. can NOT
2. Shift Manager

General Discussion

Per AD-OP-ALL-0200 (Clearance and Tagging):
Clearance approval requires an active/current SRO license.
Clearance review can be performed by a licensed operator or a previous licensed operator at the station.
All clearances shall be evaluated to determine if they are exceptional per Attachment 3, Exceptional Clearances.
Exceptional clearances shall be approved by the following:
1) A Shift Manager (SM) or higher level Operations Manager
2) A Work Group Supervisor (or higher level manager of the work group)

Answer A Discussion

Part 1 is plausible because previously licensed operators are allowed to sign for clearance reviewer in accordance with AD-OP-ALL-0200.

Part 2 is plausible because in accordance with AD-OP-ALL-0200, clearance approval requires an active/current SRO license.

Answer B Discussion

Part 1 is plausible because previously licensed operators are allowed to sign for clearance reviewer in accordance with AD-OP-ALL-0200.

Part 2 is correct.

Answer C Discussion

Part 1 is correct.

Part 2 is plausible because in accordance with AD-OP-ALL-0200, clearance approval requires an active/current SRO license.

Answer D Discussion

CORRECT: See explanation above.

Basis for meeting the KA

The K/A is matched because the applicant is required to have knowledge of the clearance and tagging procedures (AD-OP-ALL-0200).

Basis for Hi Cog

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG-1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" linked to 10CFR55.43(b)(3) (Facility licensee procedures required to obtain authority for design and operating changes in the facility):
* Requires knowledge of authority for operating changes (Clearance and Tagging) in the facility.
Both questions are functions that are only performed by an active licensed SRO or higher level of Operations management.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	Bank Question 7900

Development References
AD-OP-ALL-0200, Clearnace and Tagging, Sect. 4.6, 5.6

Student References Provided

GEN2.2 2.2.13 - GENERIC - Equipment Control
Equipment Control
Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)

Remarks/Status

GEN2.2 2.2.18 - GENERIC - Equipment Control
Equipment Control
Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.
(CFR: 41.10 / 43.5 / 45.13)

In accordance with AD-WC-ALL-0420 (Shutdown Risk Management) and AD-WC-ALL-0340 (Outage Schedule Development and Revision Process):

defense in Depth (DID) sheets are **first** REQUIRED for risk management once _____(1)_____ is reached during the shutdown.

the Plant Condition Mode Change (PCMC) reports _____(2)_____ intended to track **operability** of structures, systems, or components (SSC) required by Technical Specifications (TS) or Selected License Commitments (SLC).

Which ONE of the following correctly completes the statements above?

A. 1. Mode 3
 2. are NOT

B. 1. Mode 3
 2. are

C. 1. Mode 4
 2. are NOT

D. 1. Mode 4
 2. are

General Discussion

Defense In Depth sheets are first initiated once Mode 4 is reached during the shutdown for risk management per AD-WC-ALL-0420.

PCMC (Plant Condition Mode Change) reports are used to track work associated with Tech Spec or SLC required systems or components.

Answer A Discussion

Part 1 is plausible because Mode 3 is considered a shutdown condition. Therefore the requirements for using DID sheets for risk management would be plausible. Additionally, DID sheets are required in Mode 3 for BWR plants.

Part 2 is CORRECT.

Answer B Discussion

Part 1 is plausible because Mode 3 is considered a shutdown condition. Therefore the requirements for using DID sheets for risk management would be plausible. Additionally, DID sheets are required in Mode 3 for BWR plants.

Part 2 is plausible because PCMC reports do track work associated with Tech Spec or SLC required systems or components and operability of these systems would be dependent on that work being completed.

Answer C Discussion

CORRECT - See discussion above.

Answer D Discussion

Part 1 is CORRECT.

Part 2 is plausible because PCMC reports do track work associated with Tech Spec or SLC required systems or components and operability of these systems would be dependent on that work being completed.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of the process for managing work activities during shutdown operation, including when shutdown risk management is required by use of DID sheets, and the use of PCMC reports to track work associated with Tech Spec or SLC required systems or components.

Basis for Hi Cog

Basis for SRO only

This question is not tied to 10 CFR 55.43(b) but does meet the justification for plant specific exemptions due to the question being linked to a learning objective that is specifically labeled in the lesson plan as being SRO only. See lesson plan OP-CN-TA-SO objective 12 which states "STATE the purpose of the Plant Condition Mode Change List (SRO Only).

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Memory	BANK	Bank Question 6398

Development References

AD-WC-ALL-0420, Att 4, Sect. 1.2
AD-WC-ALL-0340, Sect. 5.4.4
OP-CN-TA-SO, Sect. 2.10.2

Student References Provided

GEN2.2 2.2.18 - GENERIC - Equipment Control

Equipment Control

Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc. (CFR: 41.10 / 43.5 / 45.13)

Remarks/Status

GEN2.3 2.3.11 - GENERIC - Radiation Control
Radiation Control
Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

Given the following conditions:

- A release has been initiated from the Monitor Tank Building
- 0EMF-57 has entered Trip 2 condition
- The CRS is informed that the recirc pump associated with the tank being released has experienced a sheared shaft
- The time at which the recirc pump failed is **UNKNOWN**

The CRS _____(1)_____ use the guidance of OP/0/B/6500/060 (Discharge of an AMT to the Environment) which allows the release to be re-initiated a maximum of two more times.

Per SLC 16.11-4 (Liquid Radwaste Treatment System), the Liquid Radwaste Treatment System shall be demonstrated Functional by meeting _____(2)_____ .

Which ONE of the following correctly completes the statements above?

A. 1. should
   2. SLC 16.11-1 (Liquid Effluents) AND 16.11-3 (Dose)

B. 1. should NOT
   2. SLC 16.11-1 (Liquid Effluents) AND 16.11-3 (Dose)

C. 1. should
   2. SLC 16.11-1 (Liquid Effluents) ONLY

D. 1. should NOT
   2. SLC 16.11-1 (Liquid Effluents) ONLY

General Discussion

Although guidance of the listed OP allows a release to be initiated twice following an EMF spike, the release should not be continued after discovering that recirculation is not occurring. Per a note contained in SLC 16.11-4, SLCs 16.11-1 and 16.11-3 are used in demonstrating the Liquid Radwaste Treatment System functional.

Answer A Discussion

First part is plausible is the applicant uses only the guidance of the listed OP. Second part is correct.

Answer B Discussion

CORRECT. See explanation above.

Answer C Discussion

First part is plausible is the applicant uses only the guidance of the listed OP. Second part is plausible if the applicant assumes that the dose specification does not apply to treatment systems.

Answer D Discussion

First part is correct. Second part is plausible if the applicant assumes that the dose specification does not apply to treatment systems.

Basis for meeting the KA

Question tests the impact of improper tank recirculation during a Liquid Radwaste Release. The "procedure and controlling, mitigating, or correcting" aspect is met by application of a Selected Licensee Commitment which controls / directs functionality assessment of the Liquid Radwaste Treatment System.

Basis for Hi Cog

The applicant must analyze a situation and apply operational knowledge to determine the proper course of action.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(4) (Radiation Hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions):

It involves the process for a liquid release approval.

Additionally, this question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, requirement to demonstrate functionality per SLC 16.11-4 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	Bank Question 6988

Development References

SLC 16.11-4 (Liquid Radwaste Treatment System),
OP/0/B/6500/060 (Discharge of an AMT to the Environment), L&P 2.18

Student References Provided

GEN2.3 2.3.11 - GENERIC - Radiation Control
Radiation Control
Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10)

Remarks/Status

B

GEN2.3 2.3.12 - GENERIC - Radiation Control

Radiation Control

Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

Given the following Unit 1 conditions:

- Unit is in Mode 3 following a refueling outage
- The status of the Personnel Air Locks (PAL) is as follows:
  - Upper Airlock Inner Door Operable
  - Upper Airlock Outer Door Operable
  - Lower Airlock Inner Door Inoperable
  - Lower Airlock Outer Door Operable
- Repairs required are on the barrel (airlock side of the inner door)

The guidance for Containment entry, to repair the Lower Airlock Door, is contained in _____(1)_____ .

The Lower Airlock Outer Door _____(2)_____ be opened to make the repair.

Which ONE of the following correctly completes the statements above?

A. 1. Tech. Spec. 3.6.2, (Containment Air Locks)
   2. may

B. 1. Tech. Spec. 3.6.2, (Containment Air Locks)
   2. may NOT

C. 1. Site Directive 3.1.2, (Access to Reactor Building and Areas Having High Pressure Steam Relief Devices)
   2. may

D. 1. Site Directive 3.1.2, (Access to Reactor Building and Areas Having High Pressure Steam Relief Devices)
   2. may NOT

General Discussion

TS 3.6.2 Bases (Containment Air Locks), contains a Note that allows entry and exit to perform repairs on air lock components. The conditions in the stem involve repairs on the barrel side (inside the airlock) of an INNER door. In this case, the OUTER door is opened, and the barrel side of the INNER door is accessed for making the repairs. Per the Tech. Spec. basis, there is a period of time (when the OUTER door is opened for access) that the containment boundary may not be intact. However, it is a short period of time, and there is a low probability of an event which could pressurize containment, and therefore is reasonable, per the Bases document.

Answer A Discussion

CORRECT. See explanation above.

Answer B Discussion

Part 1 is correct.

Part 2 is plausible because opening the Lower Airlock Outer Door would temporarily violate the containment boundary. The applicant could reason that entry through this door would not be allowed and access from Upper Containment would be required.

Answer C Discussion

Part 1 is plausible because Site Directive 3.1.2 contains guidance for Containment entries under varying conditions (different modes) and could be reasoned to contain specific guidance for airlock door repairs.

Part 2 is correct.

Answer D Discussion

Part 1 is plausible because Site Directive 3.1.2 contains guidance for Containment entries under varying conditions (different modes) and could be reasoned to contain specific guidance for airlock door repairs.

Part 2 is plausible because opening the Lower Airlock Outer Door would temporarily violate the containment boundary. The applicant could reason that entry through this door would not be allowed and access from Upper Containment would be required.

Basis for meeting the KA

The applicant must apply knowledge of containment entry requirements to analyze a set of conditions pertaining to a required repair on one of the Personnel Air Locks. Therefore, the K/A is matched.

Basis for Hi Cog

This is a higher cognitive level question because it involves a level of analysis of the given situation, applying knowledge of containment entry requirements, including Technical Specifications for Containment Air Locks, and predicting the method which should be used to accomplish a needed repair on one of the airlock doors.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(2) (Tech Specs):

1) This question can NOT be answered by knowing less than 1 hour Tech Specs.

2) This question can NOT be answered by knowing information listed "above-the-line".

3) This question can NOT be answered by knowing the TS Safety Limits.

4) This question requires the applicant to have knowledge of the Tech Spec Basis. Specifically, the question requires knowledge of the guidance for airlock entry and repair provided in TS 3.6.2 Basis.

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	Bank Question 4349

Development References
Site Directive 3.1.2 (Access to Reactor Building and Areas Having High Pressure, Steam Relief Devices Section 5.1.5 T.S.B 3.6.2 (Containment Air Locks), Actions

Student References Provided

GEN2.3 2.3.12 - GENERIC - Radiation Control

Radiation Control

Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

Remarks/Status

GEN2.4 2.4.5 - GENERIC - Emergency Procedures / Plan
Emergency Procedures / Plan
Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions. (CFR: 41.10 / 43.5 / 45.13)

Given the following Unit 1 conditions:

- Unit is responding to a main steam line break inside containment
- The operators completed E-0 (Reactor Trip and Safety Injection) and transitioned to EP/1/A/5000/E-2 (Faulted Steam Generator Isolation)
- A RED PATH on Containment Integrity occurred and the operators transitioned to EP/1/A/5000/FR-Z.1 (Response to High Containment Pressure) at Step 8 of E-2
- A RED PATH on NC Integrity occurred and the operators transitioned to EP/1/A/5000/FR-P.1 (Response to Imminent Pressurized Thermal Shock Condition) from Step 4 of FR-Z.1

Following a completion of FR-P.1 and a report from the STA that all CSFs are now GREEN, what is the correct procedure transition?

A. Return to E-2 Step 1 and continue.

B. Return to E-2 step 8 and continue.

C. Return to FR-Z.1 Step 4 and complete the procedure, then return to E-2 step 8.

D. Enter EP/1/A/5000/ES-0.0 (Rediagnosis) and use guidance for procedure entry.

General Discussion

Per OMP 1-7, once a procedure is entered due to a valid red or orange condition, that procedure shall be performed to completion unless preempted by some higher priority condition. It is expected that the actions in the procedure will clear the red or orange condition before all the operator actions are complete. However, these procedures shall be performed to the point of the defined transition to a specific procedure. At this point, any lower priority red or orange paths currently indicating or previously started but not completed shall be addressed.

Answer A Discussion

Plausible because all CSFs are green. Applicant could reason that a return to Step 1 is appropriate based on this.

Answer B Discussion

Plausible since the condition causing procedure entry has cleared. This would be correct if FR-Z.1 had been completed.

Answer C Discussion

CORRECT. See explanation above.

Answer D Discussion

Plausible because ES-0.0 can be used at any time, but entering this procedure would neglect addressing the CSF, as required.

Basis for meeting the KA

The applicant is required to demonstrate knowledge of order of entry and completion requirements of EOPs.

Basis for Hi Cog

Requires analysis of several conditions involving flowpaths of EOP usage and Success Paths. This analysis involves knowledge of CSF priorities, and of administrative requirements for EOP usage.

Basis for SRO only

This question meets the following criteria for an SRO only question as described in NUREG 1021 Rev. 11, ES-401 Attachment 2 "Clarification Guidance for SRO-only Questions" for screening questions linked to 10CFR55.43(b)(5) (Assessment and selection of procedures):

1) The question can NOT be answered solely by knowing systems knowledge.

2) The question can NOT be answered by knowing immediate operator actions.

3) The question can NOT be answered solely by knowing entry conditions for AOP or direct entry conditions for EOPs.

4) The question can NOT be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of the procedure.

5) The question does require:

- Assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed (assessment of plant conditions and knowledge of administrative guidance to determine procedure sequence and particular procedure steps for implementation).

Job Level	Cognitive Level	QuestionType	Question Source
SRO	Comprehension	BANK	Bank Question 6995

Development References
•OMP 1-7 (Emergency/Abnormal Procedure Implementation Guidelines), Section 7.3 E

Student References Provided

GEN2.4 2.4.5 - GENERIC - Emergency Procedures / Plan

Emergency Procedures / Plan

Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions. (CFR: 41.10 / 43.5 / 45.13)

Remarks/Status

Question 1

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. **Verify adequate shutdown margin as follows:**	
__ a. DRPI indication - AVAILABLE.	a. Verify adequate shutdown margin as follows:
	1) Emergency borate 10,000 gallons of 7000 PPM boron solution as follows:
Part 1	__ a) OPEN 1NV-236B (Boric Acid To NV Pumps Suct).
DRPI Data B power supply failure does not constitute DRPI unavailability	__ b) **IF** 1NV-236B will not open, **THEN** dispatch operator to open 1NV-236B (Boric Acid To NV Pumps Suct) (AB-550, HH-JJ, 53-54, Rm 234).
	c) **WHEN** 1NV-236B open, **THEN** perform the following:
	__ (1) Start boric acid transfer pumps.
	__ (2) Calculate required injection time based on boric acid flowrate.
	__ (3) **WHEN** required boric acid injected, **THEN** secure emergency boration.
	__ 2) Notify Reactor Group Duty Engineer to perform analysis to determine required shutdown margin.
	__ 3) **GO TO** Step 8.c.

Question 1

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. (Continued)	
__ d. All NC T-Colds - GREATER THAN 545°F.	d. Perform the following:
Part 2	__ 1) Determine lowest T-Cold.
	__ 2) Determine core burnup in effective full power days (EFPD) (OAC Point P1457 or from Reactor Operators logbook).
	__ 3) Verify lowest current T-Cold greater than or equal to allowable limit at present burnup. **REFER TO** Unit 1 ROD Book, Section 2.6.
	__ 4) **IF** lowest T-Cold less than allowable limit, **THEN** immediately add 40 gallons of 7000 PPM boron solution at greater than or equal to 30 GPM for each degree lowest T-Cold below limit of Unit 1 ROD Book, Section 2.6.
	__ 5) **GO TO** Step 9.
__ e. **IF AT ANY TIME** NC T-Colds trend down to less than or equal to 545°F, **THEN** perform Step 8.d.	

Question 3

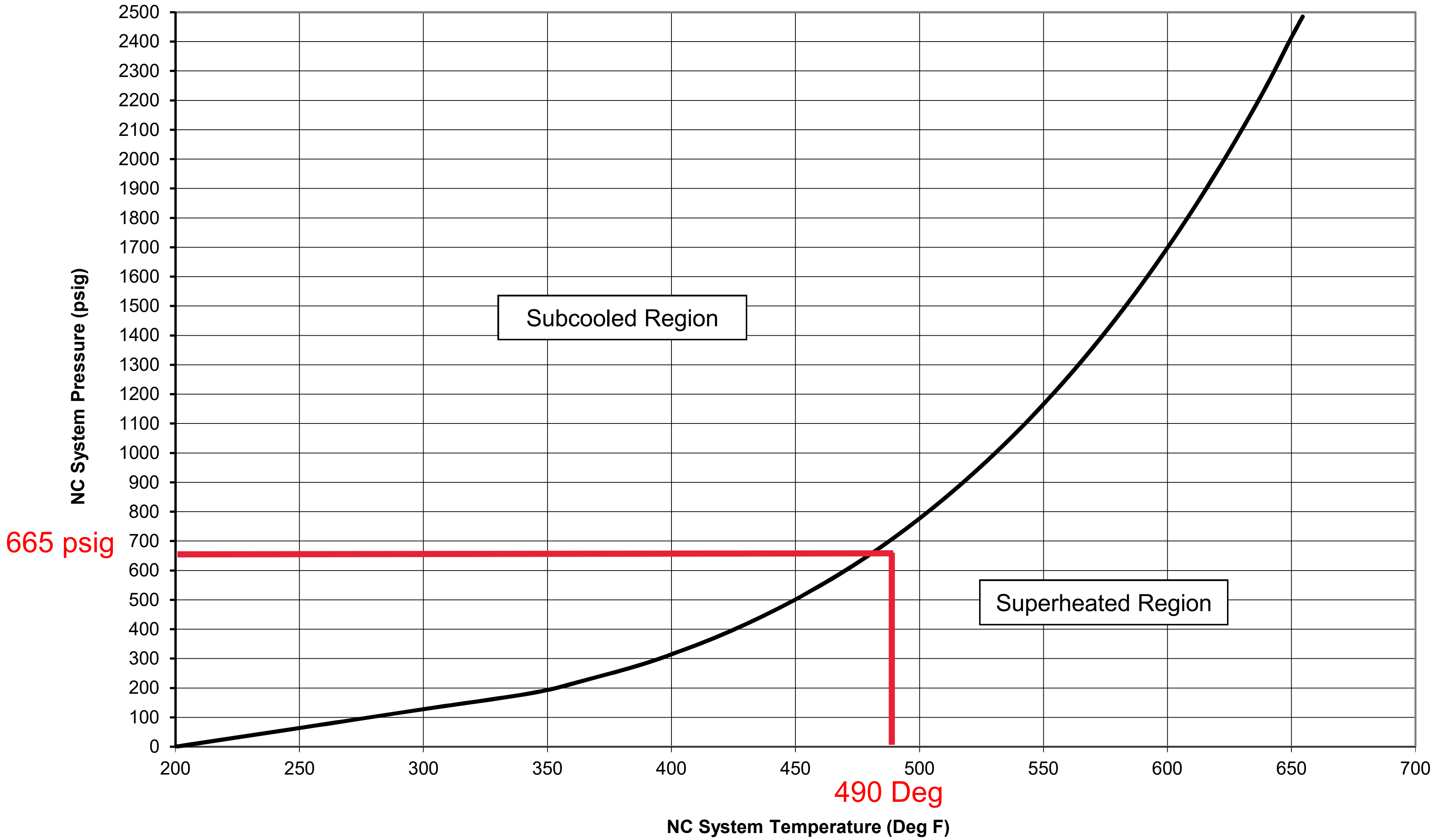


Figure 57 - Reactor Coolant Saturation Curve, Wide Range

- Temperature of each thermocouple
- Average of the 5 highest thermocouple temperatures used for indication and subcooling calculation
- Identifies the five highest thermocouples by core location
- Indicates the maximum, minimum and average T/C temperature for each core quadrant
- Average of all T/C temperatures used only for RVLIS density calculations for the quadrant

2.10 Subcooling Margin Monitor

Objective 3F

The Subcooling Margin monitor calculates and displays the subcooling margin of the average of the 5 highest T/Cs and two loop wide range T_h and provides alarms for approaching and loss of subcooling.

The monitor calculates subcooling using the equation:

Subcooling $= T_{sat} - T_{measured}$

The T_{sat} used has a 20°F conservatism factored into it.

This comparison is made for:

- T_{sat} - T/C 5 highest = subcooling based on the average of 5 highest core exit T/Cs)
- $T_{sat} - T_{hot}$ = subcooling based on loop wide range T_{hot}

The monitor determines T_{sat} by using wide range loop pressure and then adjusting the value for instrument error (i.e the 20°F conservatism mentioned above) associated with the pressure and temperature instruments. The results is a saturation curve similar to the Data Book Curve which the operator uses if the instrument is inoperable.

Inputs to the subcooling monitor are:

- Wide range loop pressure (loop "B" for Train A CCM and loop "C" for Train B CCM) from the Distributed Control System.
- Average of the 5 highest T/Cs from the core exit thermocouple monitor calculation.
- Wide range T_{hot} (Train A CCM uses loops "C" and "D" while Train B CCM uses loops "A" and "B") from the Distributed Control System.

The subcooling monitor has a range of -35°F to +200°F. A <u>positive subcooling margin</u> indicates the coolant is below the saturation temperature (subcooling of the coolant by the number of °F indicated). A <u>negative subcooling margin</u> indicates the coolant is above the saturation temperature (superheated). The numerical value indicates the

Question 3

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
__ 2. **Verify NC subcooling based on core exit T/Cs - GREATER THAN 0°F.**	**IF any NV OR NI pump on, THEN perform the following:** __ a. Ensure all NC pumps - OFF. __ b. Maintain seal injection flow.
3. **Verify main steamlines intact:** __ • All S/G pressures - STABLE **OR** TRENDING UP __ • All S/Gs - PRESSURIZED.	**IF pressure in any S/G trending down in uncontrolled manner OR depressurized, THEN perform the following:** a. **IF** any faulted S/G(s) feedlines **OR** steamlines not isolated, **THEN** perform the following: __ 1) **IF** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation) has been performed for affected S/G, **THEN GO TO** Step 3, RNO b. __ 2) **GO TO** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation). b. **IF** affected S/G(s) faulted outside containment, **THEN** request RP to perform the following: __ 1) Monitor area of steam fault for radiation. __ 2) Notify Control Room of any abnormal radiation conditions.

Question 4

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Verify NC pressure - GREATER THAN 285 PSIG.**	__ **IF ND flow to C-Legs greater than 675 GPM, THEN RETURN TO procedure and step in effect.**
__ 2. **Monitor Enclosure 1 (Foldout Page).**	
__ 3. **Verify all NC T-Colds - STABLE OR TRENDING UP.**	**Attempt to stop NC System cooldown as follows:** __ a. Ensure all S/G PORVs - CLOSED. __ b. **IF** any S/G PORV cannot be closed, **THEN** CLOSE its isolation valve. __ c. Ensure all steam dump valves - CLOSED. __ d. **IF** ND in RHR mode, **THEN** stop any cooldown from ND System. e. Identify faulted S/G(s) as follows: __ • Any S/G pressure - TRENDING DOWN IN UNCONTROLLED MANNER OR __ • Any S/G - DEPRESSURIZED. (RNO continued on next page)

Question 4

STEP 1: Verify NC pressure - GREATER THAN 285 PSIG.

PURPOSE:

To determine if the entry into FR-P.1 was due to a large-break LOCA.

APPLICABLE ERG BASIS:

For transients where NC System pressure is less than the ND pump shutoff head and flow from the ND pumps has been verified, the operator should return to the procedure and step in effect since these symptoms are indicative of a large-break LOCA. In this instance, the actions in FR-P.1 should not be performed since pressurized thermal shock is not a serious concern for a large-break LOCA.

A large-break LOCA will depressurize the NC System to containment pressure in a matter of minutes. NC pumps will be tripped due to low pressure and natural circulation will stop as fluid levels in the S/G tubes drop below the U-bends. A safety injection signal will be actuated by low pressurizer pressure. Large volumes of water will be injected into the loop cold legs, leading directly to the severe cooling of the vessel downcomer. Most of the water will come from the FWST, which is at a minimum of 70°F. Another source of water injected into the cold legs comes from pressurized accumulators located inside containment. Depending on containment design and power history, ambient temperature could range from 70°F to 120°F. This relatively cold safety injection water produces significant thermal shock in the downcomer region, but for the large-break LOCA, pressure is not a concern. Most of the water initially in the NC System, and most of the colder water added by injection, will flow out the break. This break flow mixes together in the containment sump to result in a somewhat higher temperature for the sump water than that purely from injection. During the recirculation phase of recovery, this warmer fluid is pumped back into the NC System, somewhat offsetting the initial thermal shock.

As discussed in the Pressurized Thermal Shock (PTS) Training Program documentation, temperature stress alone is normally not sufficient to threaten the integrity of the pressure vessel. The PTS scenario also assumes that the NC System is pressurized and a critical flaw which is the correct size, shape, and orientation exists at the inner surface of the vessel wall. In addition, the warm pre stressing effect, which results in an apparent increase in fracture toughness of the vessel material, is applicable to this situation since repressurization of the NC System is virtually impossible following a large-break LOCA. Any material flaw would be prevented from reinitiating and growing beyond 75% of the vessel wall thickness. Therefore, for a large-break LOCA, PTS is not a serious concern and procedure FR-P.1 does not need to be implemented.

PLANT SPECIFIC INFORMATION:

Continued On Next Page.

Question 5

Given the following:

- Unit 1 is in Mode 3
- Rod control is capable of rod withdrawal.
- NC loops 1A, 1B, and 1D are in operation.
- The crew has entered AP/1/A/5500/008 (Malfunction of Reactor Coolant Pump)
- 1A NC Pump lower bearing temperature is currently 190ºF and increasing 5ºF per minute

(1) 1A NC Pump lower bearing temperature will reach trip setpoint in ________________ .

(2) What is the specified Completion Time of the action required by TS 3.4.5 (RCS Loops – MODE 3) following the trip of 1A NCP?

A. 1. 7 minutes
 2. immediately

B. 1. 7 minutes
 2. 1 hour

C. 1. 1 minute
 2. immediately

D. 1. 1 minute
 2. 1 hour

Original Question

MODIFIED

Question 5

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

__ 1. **Monitor Enclosure 1 (Foldout Page).**

2. **Verify the following parameters for all NC Pumps:**

- __ #1 seal outlet temperature - LESS THAN 235°F
- __ Lower bearing temperature - LESS THAN 225°F

Correct answer for Part 1

- __ #1 seal delta P - GREATER THAN 200 PSID.

Perform the following:

a. **IF** in Mode 1 or 2, **THEN** perform the following:

1) **IF** all NC pumps affected, **THEN** perform the following:

__ a) Place steam dumps in pressure mode.

__ b) Ensure "STM DUMP CTRL" - SET AT 1090 PSIG STEAM HEADER PRESSURE.

__ 2) Trip reactor.

3) **WHEN** reactor power less than 5%, **THEN** perform the following:

__ a) Trip affected NC pump(s).

__ b) Ensure normal spray valve associated with tripped NC pump(s) - IN MANUAL AND CLOSED.

__ 4) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

__ b. Trip affected NC pump(s).

__ c. Ensure normal spray valve associated with tripped NC pump(s) - IN MANUAL AND CLOSED.

__ d. **GO TO** AP/1/A/5500/004 (Loss of Reactor Coolant Pump).

Question 5

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

__ 1. **Monitor Enclosure 1 (Foldout Page).**

2. **Verify the following parameters for all NC Pumps:**

- __ Stator winding temperature - LESS THAN 311°F
- __ Motor bearing temperature - LESS THAN 195°F.

Distractor for Part 1

Perform the following:

a. **IF** in Mode 1 or 2, **THEN** perform the following:

1) **IF** all NC pumps affected, **THEN** perform the following:

__ a) Place steam dumps in pressure mode.

__ b) Ensure "STM DUMP CTRL" - SET AT 1090 PSIG STEAM HEADER PRESSURE.

__ 2) Trip reactor.

3) **WHEN** reactor power less than 5%, **THEN** perform the following:

__ a) Trip affected NC pump(s).

__ b) Ensure normal spray valve associated with tripped NC pump(s) - IN MANUAL AND CLOSED.

__ 4) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

__ b. Trip affected NC pump(s).

__ c. Ensure normal spray valve associated with tripped NC pump(s) - IN MANUAL AND CLOSED.

__ d. **GO TO** AP/1/A/5500/004 (Loss of Reactor Coolant Pump).

Question 5

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops — MODE 3

LCO 3.4.5 Three RCS loops shall be OPERABLE, and either:

a. Three RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or **Correct answer for Part 2**

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal. **Distractor for Part 2**

--------------------------------NOTE--------------------------------
All reactor coolant pumps may be de-energized for ≤ 1 hour per 8 hour period provided:

a. No operations are permitted that would cause introduction of coolant into the RCS with boron concentration less than required to meet SDM of LCO 3.1.1 and maintain k_{eff} < 0.99; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required RCS loop(s) inoperable.	A.1 Restore required RCS loop(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

(continued)

6. ABNORMAL AND EMERGENCY OPERATION

Note: Refer to the associated AP lesson plan for coverage of the following APs:

- AP/1/A/5500/004, Loss of Reactor Coolant Pump purpose is to verify proper response in the event of a loss of one or more NC pumps.
- AP/1/A/5500/008 Malfunction of Reactor Coolant Pump purpose is to verify proper response in the event of an NC pump malfunction and to identify the appropriate recovery steps. Also, the manual NCP Trip Requirements are listed in AP/1/A/5500/008

6.1 Reactor Coolant Pump Trip Requirements

Objective # 9A, Licensed

No. 1 seal outlet temperature reaches 235°F

BASIS: The 235° F maximum No. 1 seal outlet temperature was established to reduce the possibility of the seal leakoff flashing to steam in the No. 2 seal area. Steam generation in the seal area can result in significant seal damage.

Any pump bearing temperature exceeds 225°F **Correct answer for Part 1**

BASIS: The injection water flow from the bearing traveling to the seal area will rise in temperature from 225° F to approximately 235ºF as a result of conduction of heat through the main flange surfaces and seal friction. At seal temperatures above 235ºF, flashing may occur in the No. 2 seal area. Steam generation in the seal area can result in significant seal damage. In other words, any pump bearing exceeding 225ºF will likely result in No. 1 seal outlet temperature reaching 235ºF.

Any motor bearing temperature exceeds 195°F **Distractor for Part 1**

BASIS: The 195°F shutdown limit for the motor bearing temperature was established to minimize the risk of bearing damage. Temperatures above 195° F can result in damage (melting) of the babbitted surfaces of the bearing shoes. Damaged shoes will adversely affect the operation of the motor and could impair the NCP coastdown that is used for residual core cooling during plant shutdown. Also, operation of damaged bearings will result in higher bearing friction which may result in rapidly rising bearing temperatures which is not predictable or controllable.

Motor winding temp. exceeds 311°F

BASIS: Reduce the possibility of any further degradation of the motor.

2.3.1 Thermal barrier

Objective #3I, All

The thermal barrier assembly is a flanged cylindrical shell that contains the thermal barrier, the thermal barrier heat exchanger, and the pump radial bearing. The assembly has three flanged water connections including supply and return water for water for Component Cooling Water System (KC) and a connection for the seal water injection Chemical and Volume Control System (NV).

The thermal barrier heat exchanger is welded to the bottom of thermal barrier. It is located above the impeller, inside the diffuser-turning vane assembly. The thermal barrier heat exchanger limits heat transfer between hot system NCS water and seal injection water. Component Cooling Water (KC) flows through the tube side of the heat exchanger at approximately 40 gpm. The thermal barrier heat exchanger will cool Reactor Coolant water in the event seal injection flow is lost. The temperature is not expected to exceed 200°F with a minimum KC flow of 35 gpm.

The thermal barrier cans are a series of 6 thin vertical cylinders containing stagnant NCS water that limits the transfer of heat from the Reactor Coolant to the pump internals and seals.

High pressure seal injection water is introduced through a connection on the thermal barrier flange and discharges approximately 8 gpm between the pump radial bearing and the thermal barrier HX. The approximately 8 gpm seal injection splits, about 3 gpm flows up through the radial bearing and NCP seals and about 5 gpm flows down along the shaft through the thermal barrier HX where it acts as a buffer to prevent system water from entering the radial bearing and seal section of the unit. The primary design purpose of the thermal barrier heat exchanger is to provide backup cooling in the event injection flow is lost. When injection flow is lost (during an accident) flow through the Thermal Barrier Heat exchanger is reversed and the temperature of the NC water coming up the shaft is cooled to an acceptable level. **Correct answer for Part 1**

The Thermal Barrier prevents flashing of the depressurized NC system water as it leaks through seals and protects the pump radial bearing and seals. In the event of NCS leakage in the Thermal Barrier, the outlet KC Isol Valves close automatically if hi flow (60 gpm) exists for 30 sec. The 30 sec. time delay prevents the valves from closing on the surge associated with a pump start. If there is a > 60 gpm leak for > 30 seconds it is probably a leak in the heat exchanger.

Component cooling water is provided to the following components in the Auxiliary Building which are not essential to safe plant shutdown following a Design Basis Event:

- H2 recombiners (2)
- Waste gas comp (2)
- Recycle evap. pkg.
- Seal water Hx (1)
- Fuel pool cool Hx (2) - Normal flow to the KF heat exchanger is 1000 to 3000 gpm dependent on the fuel in the pool.
- KF pump motor cooler (2)
- Letdown Hx
- All flow is auto controlled except:
    - Evaporator concentrates Hx's and pump bearing coolers (Manual isolations)
    - KF Hx Manual loader in Control Room
    - KF Pump Motor Cooler

Reactor Building Non-Essential Header

Objective 4c and 7, All

Refer to figure 8 and 9 for Reactor Building Non-Essential Header

The Reactor Building Non-Essential Header supply (1(2)KC-228B, 1(2)KC-230A) and return (1(2)KC-3A, 1(2)KC-18B) isolations are both normally open. They will close on the associated unit/train for any of the following:

- Sp signal **Correct answer for part 2**
- Lo Level in FWST 20% following Ss
- Lo-Lo level in the surge tank (34%).
    - When closed by the KC surge tank level they can be reopened when the appropriate train's level switch is reset, between 40% and 48% KC Surge Tank Level.

The Reactor Building Non-Essential Header supply (1(2)KC-228B, 1(2)KC-230A) and return (1(2)KC-3A, 1(2)KC-18B) isolations are controlled from CR and the train related ASP.

When Unit 1(2) ASP A and B are transferred to "LOCAL", Excess Letdown and NC Pump isolation valves (1(2)KC-305B, 1(2)KC-315B, 1(2)KC-338B, 1(2)KC-424B, 1(2)KC-425A) will open. The KC supply header valves 1(2)KC-228B, 1(2)KC-3A, 1(2)KC-18B and 1(2)KC-230A will also open when Unit 1(2) ASP A and B are transferred to "LOCAL" and the train related pump is running.

Reactor Building Non-Essential Header cools the following:

- NC Pumps
- NCDT
- Excess Letdown Hx's

Component Cooling cools the NC pump Thermal Barrier Hx, and the upper and lower bearing oil coolers of the NC Pumps. The NC Pump thermal barrier and oil cooler flows for both units can be read on gauges on the 543 elevation of Aux building.

The Thermal Barrier Hx has an inlet check valve and the outlet valve that can auto close in the event there are indications of a thermal barrier rupture. All the piping between the check valve inlet and outlet valve and the valves themselves are rated for NCS pressure and temp with a relief set at 2485 psig. Output from the relief valve is directed to the Containment Floor and Equipment Sump. The outlet valve will auto close @ 60 gpm after 30 seconds. (The 30 sec. time delay prevents the valve from closing on surge, during a Pump Start).

This arrangement of an inlet check valve and auto closed outlet valve should isolate any Thermal Barrier HX leak. Flow to the Thermal Barrier is manually throttled to 40 gpm/pump, with a high flow alarm at 60 gpm and a low flow alarm at 35 gpm. Westinghouse performed specific analysis in 2009 to determine the bounding leak rate for the thermal barrier heat exchangers used at Catawba and McGuire. The analysis determined that the bounding leak rate is 10.55 gpm (AR 01540034 Corrective Action #1).

The NC pump upper and lower bearing oil coolers are also cooled by Component cooling. The upper Bearing flow is controlled at 165 gpm/pump by a flow controller located in Aux. building on 543' elevation. There is a high flow alarm at 200 gpm and a low flow alarm at 140 gpm on the pump upper bearing Component Cooling flow.

The NC pump lower bearing oil cooler KC flow is manually adjusted to 6 gpm/pump. There is a low flow alarm at 5 gpm.

The KC Supply Header Flow to NCP's has a Low alarm @ 425 GPM. The Annunciator alarm on AD20/21 requires verification of flowpath and monitoring the motor bearing temperatures. Refer to AP/021 (Loss of KC) if alarm is due to a loss of flow.

On a Unit 1(2) Sp signal the NC Pump Containment Supply and Return isolation valves (1(2)KC-338B, 1(2)KC-424B, and 1(2)KC-425A) Isolations will close.

Question 7

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 10. **Verify KC flow to desired ND heat exchanger - GREATER THAN 5000 GPM.** Correct answer for Part 1	**Perform the following:** a. Ensure desired train valve open: __ • 1KC-56A (KC To ND Hx 1A Sup Isol) OR __ • 1KC-81B (KC To ND Hx 1B Sup Isol). __ b. **IF** KC flow to at least one ND Hx cannot be established, **THEN REFER TO** AP/1/A/5500/021 (Loss of Component Cooling).
__ 11. **Verify at least one RN pump - ON.**	__ **REFER TO AP/0/A/5500/020 (Loss of Nuclear Service Water).**
__ 12. **Verify ND Train 1A - DESIRED TO BE PLACED IN SERVICE.**	__ **GO TO Step 30.**
__ 13. **Verify 1ETA - ENERGIZED.**	**Perform the following:** __ a. Attempt to restore power to affected bus. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power). __ b. Do not continue in this enclosure until power restored to 1ETA.
__ 14. **Verify 1FW-27A (ND Pump 1A Suct From FWST) - CLOSED.**	__ **Align ND Train 1A suction to NC loop. REFER TO Enclosure 23 (Align ND Suction to NC Loop).**

Question 7

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18. **CLOSE the following valves:** __ • 1ND-26 (ND Hx 1A Outlet Ctrl) __ • 1ND-27 (ND Hx 1A Bypass Ctrl). **Correct answer for Part 2**	**Control ND flow rate as follows:** __ a. Place "PWR DISCON FOR 1NI-173A" in "THROT". __ b. CLOSE 1NI-173A (ND Hdr 1A To Cold Legs C&D). __ c. Ensure 1ND-27 (ND Hx 1A Bypass Ctrl) - IN MANUAL. __ d. Raise output for 1ND-26 (ND Hx 1A Outlet Ctrl) to 100%. __ e. Lower output for 1ND-27 (ND Hx 1A Bypass Ctrl) to 0%. __ f. THROTTLE 1NI-173A (ND Hdr 1A To Cold Legs C&D) to maintain desired flowrate in subsequent steps.
__ 19. **Verify Enclosure 3 (ND Suction Header and Pump Casing Venting) - IN PROGRESS.**	__ **<u>GO</u> <u>TO</u> Step 21.**
__ 20. **Do not continue in this enclosure until Enclosure 3 (ND Suction Header and Pump Casing Venting) complete.**	
__ 21. **Ensure D/G 1A load sequencer - RESET.**	

Question 7

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 22. **Start ND Pump 1A.**	**Perform the following:** __ a. **IF** ND Train 1B available - **THEN RETURN TO** Step 10. b. **IF AT ANY TIME** one train of ND cannot be placed in service, **THEN** perform the following: __ 1) Continue attempts to return one train of ND to service. __ 2) **WHEN** any train of ND available, **THEN RETURN TO** Step 1. __ 3) **RETURN TO** step in effect.
__ 23. **Verify 1ND-25A (ND Pump 1A Miniflow) - OPENS.**	**Perform the following:** __ a. OPEN 1ND-25A (ND Pump 1A Miniflow). __ b. **WHEN** flow through ND Train 1A greater than 1400 GPM, **THEN** CLOSE 1ND-25A (ND Pump 1A Miniflow).

Question 7

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
24. **Determine any required ND flow restrictions as follows:**	
__ a. Verify NC System - IN LOOPS NOT FILLED CONDITION.	__ a. **GO TO** Step 25.

CAUTION **Flashing may occur in the high point of the ND pump suction line if the following ND flowrate limits are exceeded.**

__ b. Determine maximum allowable ND flow from table below:

NC Level	Maximum Allowed ND Flowrate
Greater than or equal to 39%	3000 GPM
Greater than or equal to 24%	2000 GPM
Greater than or equal to 15%	1500 GPM
Greater than or equal to 11%	1000 GPM

__ c. **IF AT ANY TIME** NC Level changes, **THEN** repeat Step 24.b.

Distractor for Part 2

Question 7

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

25. **Establish flow to NC System as follows:**

a. Slowly THROTTLE open 1ND-26 (ND Hx 1A Outlet Ctrl) to establish one of the following flowrates while maintaining NC System cooldown rate less than 100°F in an hour:

__ • Less than or equal to maximum flowrate as determined in Step 24

OR

__ • 3000 GPM. Distractor for Part 2

a. Control ND flow rate as follows:

__ 1) Place "PWR DISCON FOR 1NI-173A" in "THROT".

__ 2) CLOSE 1NI-173A (ND Hdr 1A To Cold Legs C&D).

__ 3) Ensure 1ND-27 (ND Hx 1A Bypass Ctrl) - IN MANUAL.

__ 4) Raise output for 1ND-26 (ND Hx 1A Outlet Ctrl) to 100%.

__ 5) Lower output for 1ND-27 (ND Hx 1A Bypass Ctrl) to 0%.

6) THROTTLE 1NI-173A (ND Hdr 1A To Cold Legs C&D) to maintain the following:

• ND flowrate through ND Train 1A less than or equal to one of the following:

__ • Maximum flowrate determined in Step 24

OR

__ • 3500 GPM.

__ • NC System cooldown rate less than 100°F in an hour.

__ 7) **GO TO** Step 26.

Question: 33
(1 point)

Given the following timeline on Unit 1:

1000

- The Unit is at 100% RTP
- 1A2 KC Pump is in service
- 1A KC flow is 5000 gpm and stable

1005

- 1KC-9 (1A2 KC Pump Disch) is inadvertently closed

1007

- 1KC-9 is reopened
- No additional KC pumps have been started

Following closure of 1KC-9, 1KC-C37A (Train A Miniflow Isol) opened as flow decreased to a MAXIMUM value of _____(1)_____.

Assuming no operator action, once 1KC-9 is reopened 1AD-9 F/5 “KC Train A Single Pump Runout” _____(2)_____ alarm.

Which ONE (1) of the following completes the statements above?

A. 1. 3150 gpm
   2. will

B. 1. 1100 gpm
   2. will

C. 1. 3150 gpm
   2. will NOT

D. 1. 1100 gpm
   2. will NOT

Original Question

MODIFIED

F/6

KC TRAIN B SINGLE PUMP RUNOUT

SETPOINT: 5700 gpm increasing and KC Pump 1B1 or 1B2 running.

ORIGIN: 1KCFT5540

PROBABLE CAUSE:
1. **NOT** enough flow restriction in the KC pump flow path
2. Pipe rupture
3. Loss of power at power panelboard 1EKPB Breaker #34

AUTOMATIC ACTIONS: None

IMMEDIATE ACTIONS:
1. Close 1KC-C40B (Trn 1B Miniflow Isol).
2. **IF** closing miniflow valve does **NOT** clear alarm, start an additional B Train KC pump per OP/1/A/6400/005 (Component Cooling System).
3. **IF** an additional B Train KC pump is **NOT** available, **OR** > 10,800 gpm flow is required, i.e. two trains of ND are in service, start additional KC pumps as required to achieve required flow per OP/1/A/6400/005 (Component Cooling System).
4. Verify 1EKPB Breaker #34 is "ON" (AB-560, FF-56, in 1EMXB).
5. **IF** KC flow remains above the runout setpoint, adjust KC to KF as follows:
   - 5.1 **IF** KC aligned to 1A KF Heat Exchanger, lower KC to KF flow by throttling closed 1KC-149 "KF Hx 1A Cool Wtr Otlt" flow controller on 1MC11.
   - 5.2 **IF** KC aligned to 1B KF Heat Exchanger, lower KC to KF flow by throttling closed 1KC-156 "KF Hx 1B Cool Wtr Otlt" flow controller on 1MC11
   - 5.3 Monitor Unit 1 Spent Fuel Pool Temperature to ensure temperature is < 125 F.

SUPPLEMENTARY ACTIONS:
1. Monitor KC flow.
2. Monitor KC discharge pressure.
3. **IF** pipe rupture is found,
   - 3.1 **IF** possible, isolate the rupture.

     OR
   - 3.2 **IF NOT** isolable, secure the running pumps.
4. Refer to AP/1/A/5500/021 (Loss of Component Cooling).
5. Refer to Tech Spec 3.7.7.

REFERENCES:
1. CNEE-0142-01.58
2. CNEE-0142-01.35
3. CNEE-0142-01.60
4. CN-11372

Question 8

Runout flow Annunciator setpoints are 5,700 gpm for one pump and 10,800 for two pumps. Pump runout is prevented by isolating the non-essential headers when the ND heat exchangers are placed in service or by starting additional KC Pumps

The KC pumps minimum Flow of 1100 gpm/pump is assured by Minimum Flow Valves KC-C37A/C40B. The Minimum Flow Valves open at 3150 gpm decreasing and close at 5800 gpm increasing. It is possible for a minimum flow valve to be open when flow is less than 5800 gpm while the single pump runout annunciator is lit (greater than 5700 gpm. Correct answer for Part 2

The KC pumps cool their own motor via the essential header.

KC Hx's

Objective 4b and 5, All

Refer to figure 5 for KC Heat Exchangers and EMF's

The KC Heat Exchangers are designed to provide the required heat transfer for all modes of operation. One heat exchanger is required for normal operation.

The KC Heat Exchanger is a horizontal, straight tube, single pass heat exchanger with RN on the tube side due to the tendency of fouling (Easier to clean), and KC on the shell side. The heat exchangers are located at the discharge of the KC Pumps to ensure KC pressure remains higher than RN pressure for the purpose of preventing in-leakage of RN water. RN flow is temperature controlled when in "KC Temp" mode such that the KC outlet temp is ~ 87ºF.

EMF's

Objective 6b, All

1(2)EMF46A and 46B monitor the KC system in order to detect radioactivity entering the system from the KF system or the NC system and its associated auxiliary systems. These EMF's sample the discharge of the KC heat exchanger to detect any radioactive leaks into the KC System. Sample Flow is taken from the KC supply header. 1(2)EMF46A and B monitor Unit 1(2) KC System and provide an alarm on high radiation. (No auto action associated with the EMF's.) High radiation in the KC System is an indication of KC activation and possible NC in-leakage.

1(2)EMF46A and 46B are not considered to be inoperable just because there is no KC flow through their respective train. The EMFs would be considered inoperable if one of the inlet/outlet KC isolation valves to the EMF were closed, the EMF itself was not functioning properly, or PMs/Calibrations were being done on the EMF which rendered it out of service. The EMF is still considered to be operable even though the KC cross-over valves on a particular train are closed and none of the KC pumps on that train are running. AR 01538057 provides the full operability evaluation associated with the KC System EMFs.

If an intersystem leak developed while a train of KC was isolated (cross-over valves closed), the only radioactive leak path component on the isolated train would be the ND HX. If a leak

Component cooling water is provided to the following components in the Auxiliary Building which are not essential to safe plant shutdown following a Design Basis Event:

- H2 recombiners (2)
- Waste gas comp (2)
- Recycle evap. pkg.
- Seal water Hx (1)
- Fuel pool cool Hx (2) - Normal flow to the KF heat exchanger is 1000 to 3000 gpm dependent on the fuel in the pool.
- KF pump motor cooler (2)
- Letdown Hx
- All flow is auto controlled except:
    - Evaporator concentrates Hx's and pump bearing coolers (Manual isolations)
    - KF Hx Manual loader in Control Room **Correct answer for Part 1**
    - KF Pump Motor Cooler

Reactor Building Non-Essential Header

Objective 4c and 7, All

Refer to figure 8 and 9 for Reactor Building Non-Essential Header

The Reactor Building Non-Essential Header supply (1(2)KC-228B, 1(2)KC-230A) and return (1(2)KC-3A, 1(2)KC-18B) isolations are both normally open. They will close on the associated unit/train for any of the following:

- Sp signal
- Lo Level in FWST 20% following Ss
- Lo-Lo level in the surge tank (34%).
    - When closed by the KC surge tank level they can be reopened when the appropriate train's level switch is reset, between 40% and 48% KC Surge Tank Level.

The Reactor Building Non-Essential Header supply (1(2)KC-228B, 1(2)KC-230A) and return (1(2)KC-3A, 1(2)KC-18B) isolations are controlled from CR and the train related ASP.

When Unit 1(2) ASP A and B are transferred to "LOCAL", Excess Letdown and NC Pump isolation valves (1(2)KC-305B, 1(2)KC-315B, 1(2)KC-338B, 1(2)KC-424B, 1(2)KC-425A) will open. The KC supply header valves 1(2)KC-228B, 1(2)KC-3A, 1(2)KC-18B and 1(2)KC-230A

Objective 2D, Licensed

The red closed indication light on MC-1 for a reactor trip or bypass breaker provides indication of continuity (power) through the shunt trip coil circuit **(Figure 4).** Thus, if the closed indication appears to be burned out, a potential breaker operability condition may exist as the Shunt Trip Coil may be inoperable.

Upon loss of SSPS 48 VDC to the UV coil, the ST relay also de-energizes closing a "b" contact to energize the ST coil (**Figure 4**). The circuitry includes provision to test the UV trip device, the ST device and the ST coil independently.

2.4 Reactor Trip and Bypass Breaker Operation

Objective 3A, All

The output of the logic bay of SSPS provides signals to selected UV and ST coils associated with the reactor trip and bypass trip breakers. The type of reactor trip signal generated determines which breakers and which coils are utilized to trip the reactor. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism. (**Figure 5 and 6**)

Bypass Breaker Interlock

Objective 3A, All

A bypass breaker racked into the "CONN" position and closed will generate a General Warning for that train. Two General Warnings will generate a reactor trip. Thus, if both bypass breakers are racked into "CONN" and closed, all four breakers will trip open due to the reactor trip signal.

A train of SSPS in TEST also generates a General Warning, so if a bypass breaker is racked into "CONN" and closed while the opposite train is in TEST, then a General Warning Reactor Trip will be generated.

UV and ST Coil Signals

Objective 3A, Licensed

Automatic Reactor Trip or Safety Injection Signal

SSPS will generate an automatic reactor trip when any of the reactor trip logic (as discussed later in the lesson plan) is met or an automatic safety injection occurs.

- An SSPS Train A reactor trip signal sends trip signals to the UV coil and ST coil for Reactor Trip Breaker A (RTA) and the UV coil only for Bypass Breaker B, BYB.
- An SSPS Train B reactor trip signal sends trip signals to the UV coil and ST coil for Reactor Trip Breaker B (RTB) and the UV coil only for Bypass Breaker A (BYA).

Manual Reactor Trip or Safety Injection Signal

The Trip/Close pistol-type Reactor Trip switch on the main control board sends a trip signal directly to the train-related reactor trip and bypass breakers' shunt trip coils. Additionally, the manual reactor trip generates the train related automatic trip signal from SSPS. A Manual Safety Injection performs the same actions as the Manual Reactor Trip Handles.

Example: The following will occur if the Train A Manual Reactor Trip handle is turned to "TRIP":

- Reactor Trip Breaker A (RTA) shunt trip coil is energized.
- Bypass Breaker A (BYA) shunt trip coil is energized.
- An automatic trip signal is generated by SSPS A Train and, as a result, the following occurs:
  - Reactor Trip Breaker A (RTA) undervoltage coil is de-energized.
  - Reactor Trip Breaker A (RTA) shunt trip coil is energized.
  - Bypass Breaker B (BYB) undervoltage coil is de-energized.

As discussed above, the UV coils for the train related bypass breakers are supplied from the opposite train of SSPS. Therefore, BYA's UV coil is de-energized upon an automatic trip signal from SSPS Train B and BYB's UV coil is de-energized upon an automatic trip signal from SSPS Train A. This arrangement allows the train of SSPS that is "not-in-test" to trip its normal reactor trip breaker plus the bypass breaker for the other train. The bypass breakers shunt trip coils are only energized from a MANUAL reactor trip or MANUAL SI signal.

UV and ST Coil Signal Summary (Figure 5)

Reactor Trip Breakers:

- The UV coils are de-energized by any reactor trip signal (train related).
- The shunt trip coils are energized by any reactor trip signal (train related).
- The shunt trip coils are energized directly by any Manual trip or Manual SI actuation (train related).

Bypass Reactor Trip Breakers:

- The UV coils are de-energized by any trip signal from the opposite train.
- Shunt coils are energized only by Manual trips (or Manual SI actuations) on the associated train.

Question 9

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Verify the following trips occurred:**	
__ a. Reactor Trip.	a. Dispatch operator to open the following Unit 1 breakers: __ • Reactor trip breakers __ • Reactor trip bypass breakers • The following breakers for CRD M/G sets: __ • "MOTOR" Breaker __ • "GENERATOR" Breaker. **Correct answer for Part 2**
__ b. Turbine Trip.	__ b. Dispatch operator to trip Unit 1 turbine.
8. **Verify reactor subcritical as follows:**	
a. All of the following conditions exist: __ • P/R channels - LESS THAN 5% __ • I/R channels - LESS THAN 5% __ • W/R NEUTRON POWER channels - LESS THAN 5% __ • I/R SUR - NEGATIVE.	a. Perform the following: 1) **WHEN** the following conditions exist: __ • P/R channels - LESS THAN 5% __ • I/R channels - LESS THAN 5% __ • W/R NEUTRON POWER channels - LESS THAN 5% __ • I/R SUR - NEGATIVE, __ **THEN GO TO** Step 16. __ 2) **GO TO** Step 9.
__ b. **GO TO** Step 16.	

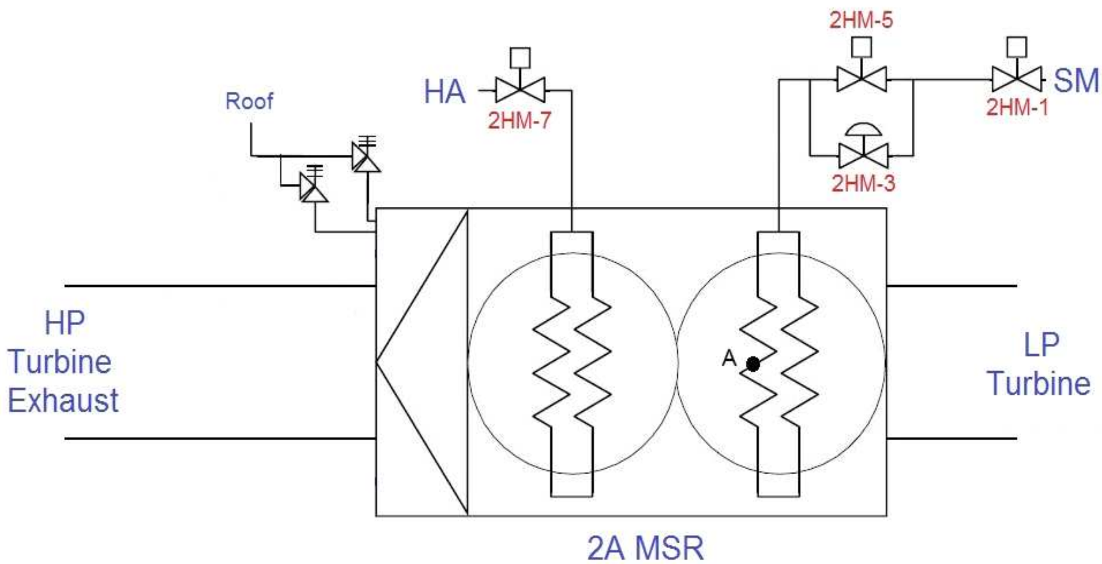
Catawba Nuclear Station

ILT-17 NRC Written Exam CNS RO NRC Examination

Question: 10
(1 point)

Given the following Unit 2 conditions:

- The Unit is at 45% RTP
- The crew has entered AP/2/A/5500/028 (Secondary Steam Leak)



A steam leak at location A, will cause Main Turbine Megawatts to _____(1)_____ .

In order to isolate the leak, AP/28 will direct the crew to _____(2)_____ .

Which ONE of the following completes the statements above?

A. 1. increase
   2. trip the Main Turbine

B. 1. increase
   2. trip the Reactor and close MSIVs

C. 1. decrease
   2. trip the Main Turbine

D. 1. decrease
   2. trip the Reactor and close MSIVs

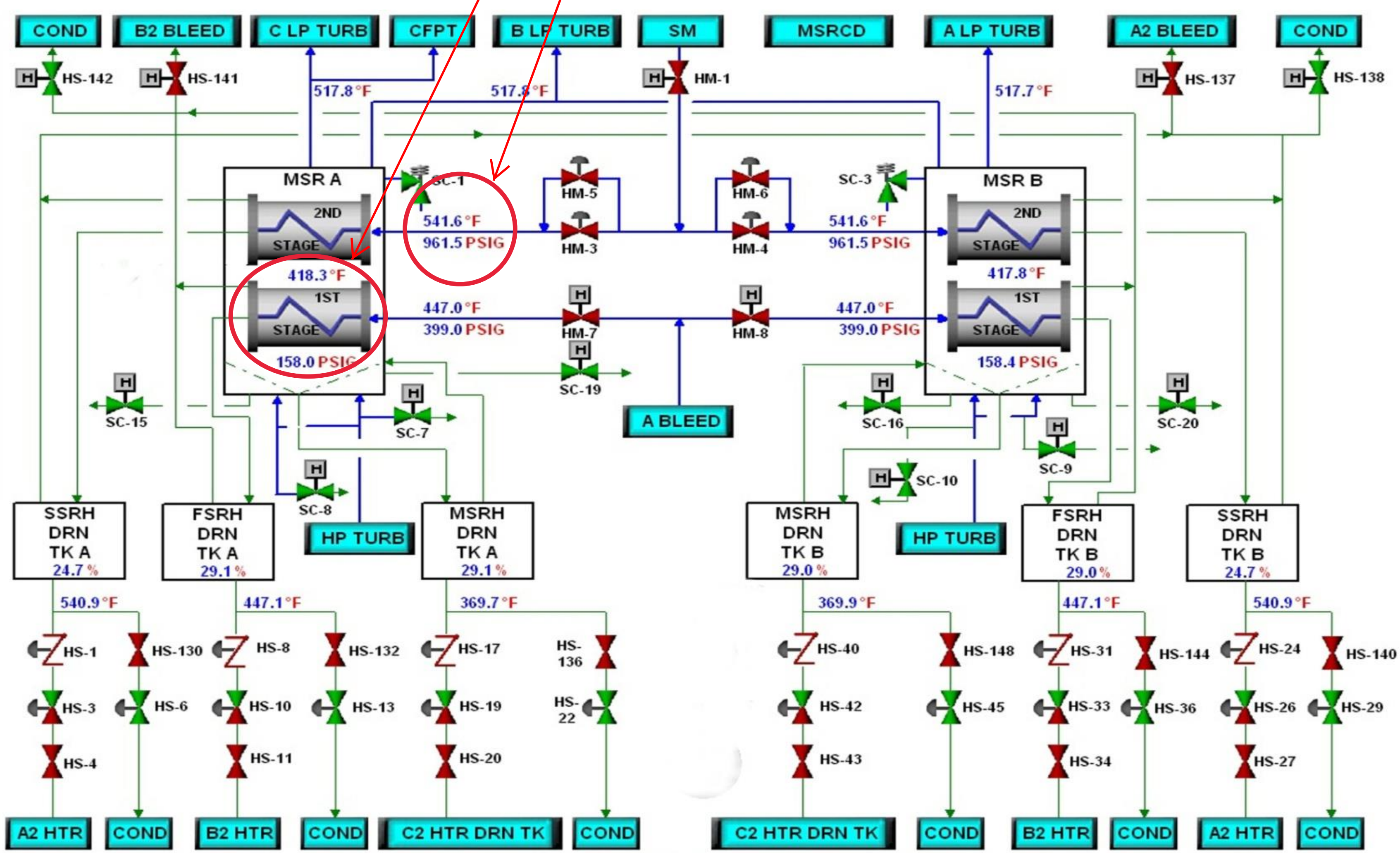
Bank Question

Question 10

Correct answer - higher temperature and pressure results in MW increase

10. FIGURES

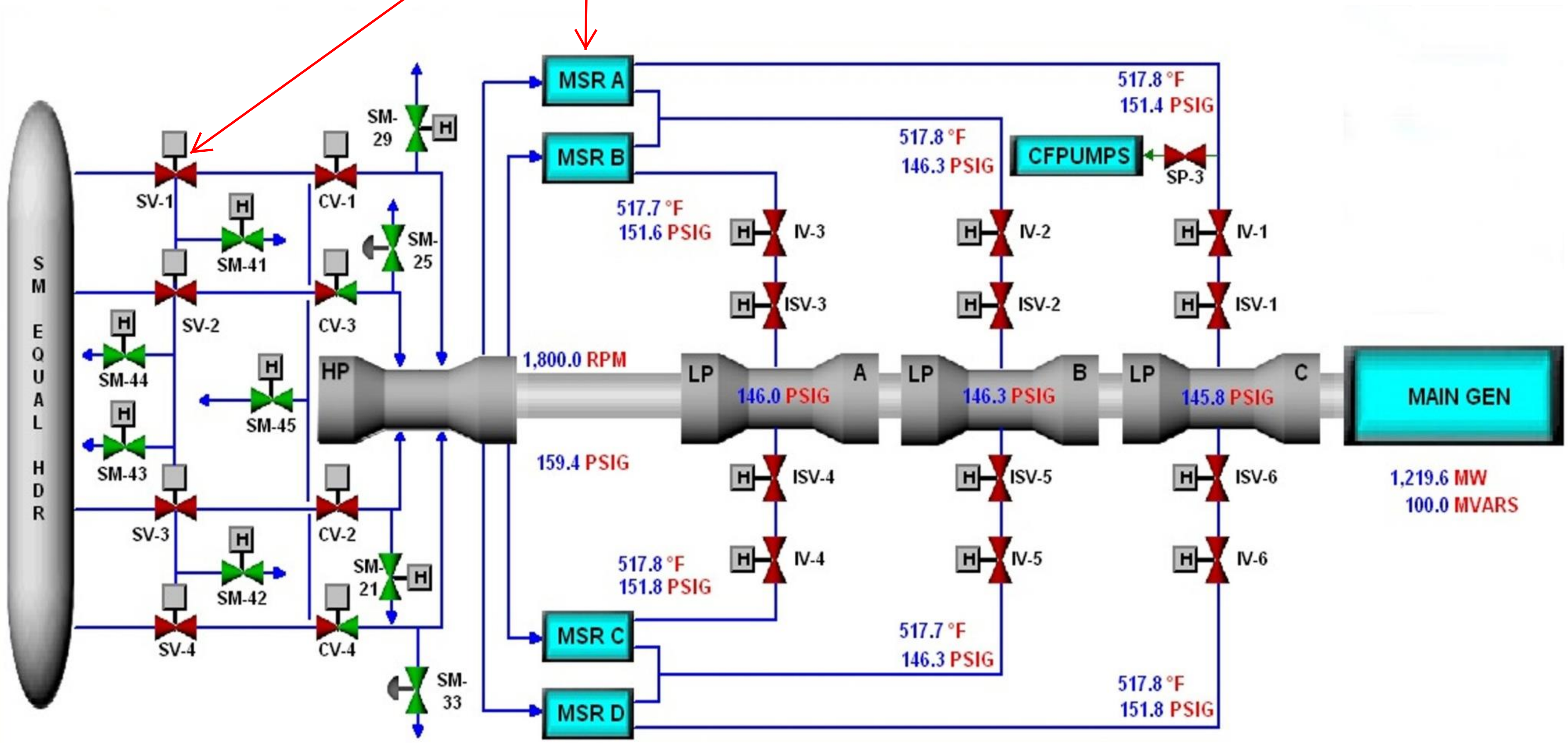
10.1 Figure 1: MSR A and B



Question 10

Distractors A.2 and C.2

10.3Figure 3: Main Turbine



Question 10

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

16. (Continued)

__ e. Verify steam leak - KNOWN TO BE ISOLABLE BY TURBINE TRIP.

e. Perform the following:

__ 1) Trip Unit 1 reactor.

2) CLOSE the following valves:

__ • All MSIVs

__ • All MSIV bypass valves.

__ 3) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Correct Answer

__ f. Verify reactor power - GREATER THAN 69%.

f. Perform the following:

__ 1) Trip turbine.

__ 2) **GO TO** AP/1/A/5500/002 (Turbine Generator Trip).

Distractors A.2 and C.2

__ g. Trip Unit 1 reactor.

__ h. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

i. Initiate Unit 1 shutdown. **REFER TO** one of the following:

__ • OP/1/A/6100/003 (Controlling Procedure For Unit Operation)

OR

__ • OP/1/A/6100/002 (Controlling Procedure For Unit Shutdown)

OR

__ • AP/1/A/5500/009 (Rapid Downpower).

4. Using the current revision of AD-OP-ALL-1001, discuss the requirements for Enclosure 1 foldout page usage.

D. There are two generic types of faults for a S/G - a feed line break or a steam line break with either of the two being inside or outside of containment.

1. A steam line break will be indicated by S/G level decreasing due to steaming out the break as pressure decreases. This will have a relatively large impact on NC temperature and pressure.
2. A feed line break will be indicated by a rapid decrease in S/G Level while pressure is relatively stable until the leak is uncovered and then pressure will rapidly decrease. This will have a relatively small impact on NC temperature and pressure.
3. Discuss the differences observed in the following parameters between a steam line break in containment and a feedwater line break in containment:
   a) Tave - large (>100ºF) rapid decrease for steam line break; smaller (~30ºF) decrease for feed line break.
   b) Tcold - large (>100ºF) rapid decrease for steam line break; smaller (~30ºF) decrease for feed line break.
   c) SG level - decreases to 0% as inventory is steamed for steam line break; rapidly decreases to 0% in about half of the time for a feed line break.
   d) SG Pressure - steady fairly fast decrease to 0 psig as inventory is steamed for steam line break; remains fairly stable initially, until the feed ring is uncovered, and then rapidly decreases to 0 psig for feed line break.
   e) Containment pressure - rapid increase to peak pressure (~6 psig) for steam line break; faster increase (~2x) to a slightly lower peak pressure for feed line break.
   f) Pressurizer level - rapid decrease to 0%, then steady increase after faulted S/G finishes blowing down, for steam line break; much smaller decrease (stays on scale throughout), then steady increase for feed line break.
   g) NC Pressure - rapid decrease to ~600 psig, then increases rapidly after faulted S/G finishes blowing down, for steam line break; much smaller decrease (PZR pressure stays on scale), then increases rapidly after faulted S/G finishes blowing down, for feed line break.
4. Having either of the above occur inside containment will complicate the recovery due to the need to respond to containment pressure increases. Remember that even with "water" dumping on the floor of containment due to a feed line break, the water is at least 440°F and will immediately flash to steam in the containment atmosphere.

Question 11

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 7. **Maintain at least one S/G available for NC System cooldown in subsequent steps.**	
__ 8. **Verify faulted S/G(s) PORV - CLOSED.**	**Perform the following:** __ a. CLOSE faulted S/G(s) PORV. __ b. **IF** S/G PORV cannot be closed, **THEN** CLOSE S/G PORV isolation valve. __ c. **IF** S/G PORV isolation valve cannot be closed, **THEN** dispatch operator to close valve.
__ 9. **Ensure CA System valve control - RESET.**	
10. **Isolate all faulted S/G(s) as follows:** • S/G 1A: __ a. Verify S/G 1A Feedwater Isolation status light (1SI-5) - LIT.	**Correct answer for Part 2** a. Perform the following: 1) Ensure the following valves - CLOSED: __ • 1CF-28 (S/G 1A CF Ctrl) __ • 1CF-30 (S/G 1A CF Byp Ctrl) __ • 1CF-33 (S/G 1A CF Cont Isol) __ • 1CF-90 (S/G 1A CF Cont Isol Byp) __ • 1CA-149 (S/G 1A CF Byp To CA Nozzle) __ • 1CA-185 (S/G 1A CA Nozz Tempering Isol). (RNO continued on next page)

Question 11

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. (Continued)	
__ f. **WHEN** activity results reported, **THEN** notify Station Management to evaluate S/G(s) activity results.	
13. **Verify S/I termination criteria:**	
__ a. NC subcooling based on core exit T/Cs - GREATER THAN 0°F.	__ a. **GO TO** Step 14.
b. Verify secondary heat sink as follows:	__ b. **GO TO** Step 14.
__ • Any intact S/G N/R level - GREATER THAN 11% (29% ACC)	
OR	
__ • Total feed flow to intact S/Gs - GREATER THAN 450 GPM.	
__ c. NC pressure - STABLE OR TRENDING UP.	__ c. **GO TO** Step 14.
__ d. Pzr level - GREATER THAN 11% (30% ACC).	__ d. **GO TO** Step 14.
__ e. **GO TO** EP/1/A/5000/ES-1.1 (Safety Injection Termination).	
__ 14. **GO TO EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant).**	**Distractor Part 2**

END

Catawba Nuclear Station

ILT-17 NRC Written Exam CNS RO NRC Examination

Question: 12
(1 point)

Given the following Unit 2 timeline:

1000
- A Loss of All Offsite Power has occurred
- Both Unit 2 D/Gs started and loaded their associated bus

1005
- While monitoring D/G operating parameters, the Unit 2 BOP notes that D/G 2B "VOLTS" indicates 3925 V

1007
- Voltage is adjusted to within the normal operating range by operation of the "D/G 2B Volt Adjust" controls on 2MC-11

As a result of the 2B D/G voltage adjustment at 1007:

2B D/G "AMPS" _____(1)_____ .

2B D/G "P/F" _____(2)_____ .

Which ONE of the following completes the statements above?

A. 1. decreased
2. remained the same

B. 1. decreased
2. became more lagging

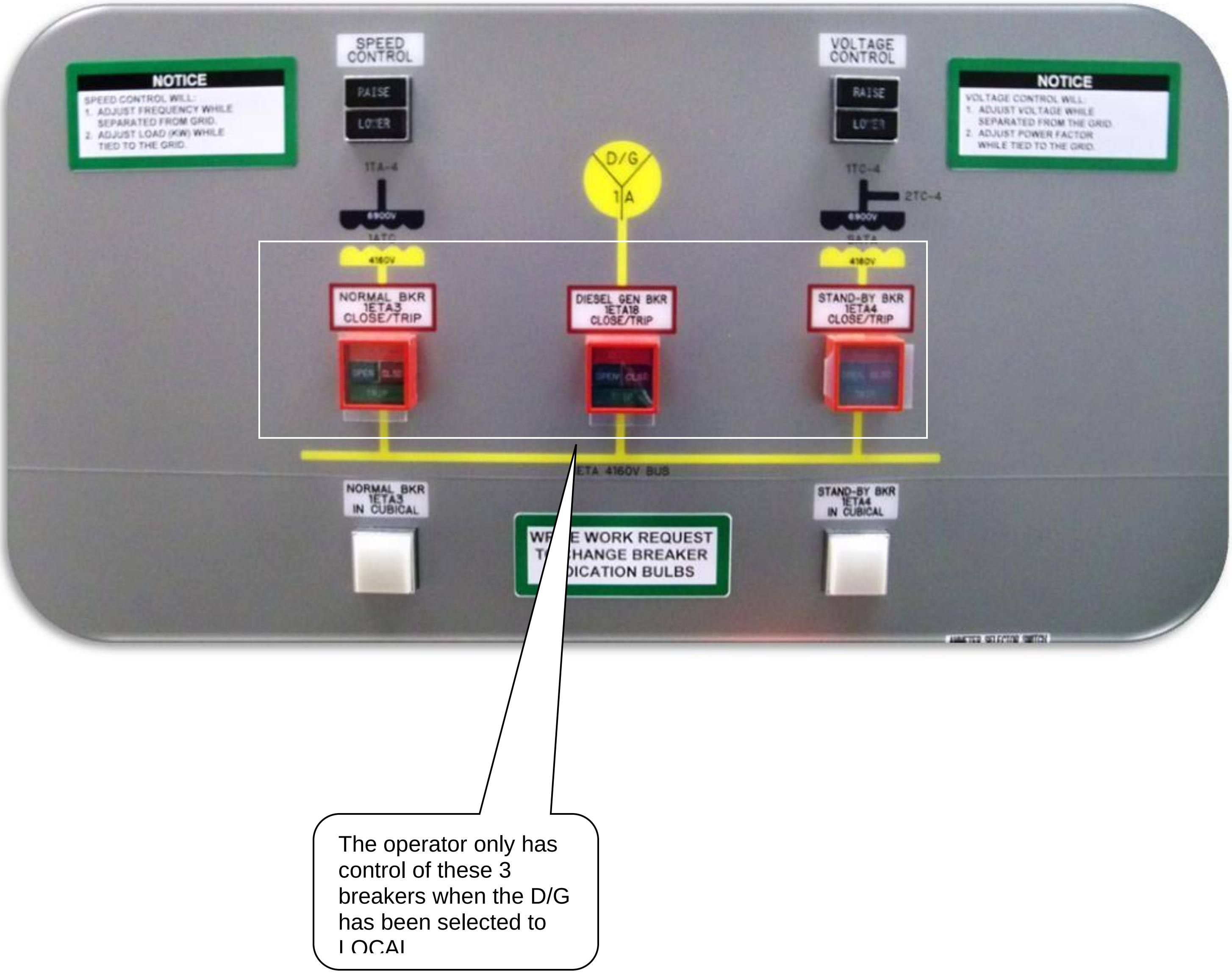
C. 1. increased
2. remained the same

D. 1. increased
2. became more lagging

Original Question

MODIFIED

15.10 FIGURE 10: DGCP BREAKER, SPEED, AND VOLTAGE CONTROLS



- Depress STOP pushbutton on DECPA(B) and remove key from Manual Test Start keyswitch.
- Ensure the keep warm and pre-lube diesel auxiliaries start.
- Have the Control Room Operator place the “D/G CTRL LOCATION” switch in the “CTRL-RM” position on MC-11.

On shutdown following a manual start, the operator must wait at least two (2) minutes prior to attempting another start to allow the diesel to coast down and the pneumatic controls to reset.

6.2 Remote Manual Start, Parallel, Loading, and Shutdown

Objective 11B, Licensed

The remote manual start, parallel, loading, and shutdown is very similar to the sequence described above with the following differences:

- The “D/G CTRL LOCATION” switch on MC11 is maintained in the “CTRL-RM” position.
- Parallel and loading operations are performed using control room controls and indications.
- To start and stop the diesel from the control room, the operator depresses the diesel ON/OFF pushbutton on MC11. (No key required.)

The steps to parallel, load, and shutdown the diesel are the same as described previously for the Local Manual operation with the exception that the control room controls are used.

6.3 Speed and Voltage Control

Objective 11C, All

Distractor Part 2

Parallel Operation (Droop Mode)

When operating in parallel with the grid, both voltage and frequency are a function of the grid. Adjusting voltage only affects Power Factor and KVAR Load of the D/G. (Voltage Control). Adjusting speed only affects KW Load of the D/G. (Speed Control).

Objective 23 and 24, All

If either the normal supply breaker or the backup supply breaker and the diesel generator breaker are closed onto the bus, then the diesel governor will operate in the speed droop mode. The diesel engine governor’s response to changes in load is intentionally slowed by design, allowing other generation sources to pick up new load when the diesel generator is tied to the grid.

A lagging power factor should be maintained at all times. When tied to the grid (Speed Droop Mode), the VOLTAGE CONTROL affects reactive loading or VARs, and the SPEED CONTROL primarily affects real load, or KW. As real load is picked up, reactive load increases as well due to counterforce causing a reduction in voltage from the generator. This causes power factor to move in the leading direction. If a leading power factor is observed, the operator should use the VOLTAGE CONTROL RAISE pushbutton to adjust the power factor to a lagging value.

Isochronous Operation

Objective 25, All

Correct answer for Part 2

When the D/G is operated in isochronous mode, there is no other source of power available to the associated essential bus. The operator controls generator speed, which is frequency, and generator voltage. **The operator has no control of Power Factor in this configuration.** Lowering voltage causes higher current flow to the loads. Voltage can be lowered to the degraded or undervoltage setpoints and cause higher current flow to the loads. Operating equipment at voltage or current different from design may cause damage. Lowering voltage to the undervoltage setpoint will cause the D/G Load Sequencer to automatically actuate in the BLACKOUT mode.

The number of loads on the essential bus controls KW load in the Isochronous mode. To change load more motors, etc. need to be added to or taken off the buss being fed by the Diesel Generator. Adjusting diesel speed affects only generator frequency (Speed Control). The type of loads on the essential bus controls KVAR and Power Factor. Adjusting diesel voltage only affects generator voltage (Voltage Control). During isochronous operation, the diesel governor responds quickly to changes in load to maintain engine speed constant.

Generator response to a Full Load Rejection While in Isochronous Mode

Objectives 21 and 22, All

When the generator experiences a load rejection from a fully loaded conditions, speed will increase until the governor returns speed to its set value. If the generator experiences a load rejection from any condition other than fully loaded, speed will still increase but not to the magnitude it would if fully loaded. In this case, the governor will still return speed to its set value.

6.4 Maintenance Mode Operation

Objective 11D, All

A Maintenance Mode selection is available to provide protection for plant maintenance personnel. When in Maintenance Mode, the diesel will not respond to an automatic or manual start signal and is considered to be out of service. Only the local operator can select the Operational mode, using the mode selector controls on the local engine control panel. However, local selection of maintenance mode cannot be accomplished without a remote permissive signal from the control room. Thus, the local and remote MAINTENANCE MODE pushbuttons are depressed simultaneously to actuate the lockout. The Shutdown Cylinder is extended when the D/G is in Maintenance Mode.

Catawba Nuclear Station

ILT-17 NRC Written Exam CNS RO NRC Examination

Question: 13
(1 point)

Given the following initial conditions:

- 1B RN Pump in service
- 1B1 KC Pump in service
- 2A1 KC Pump in service

Subsequently:

- Both units enter AP/0/A/5500/030 (Plant Flooding), Enclosure 8 (Flooding From RN) following discovery of a large RN leak on the 1A Essential Header
- Per AP/30 guidance, the crew has isolated the 1A RN Essential Header (ONLY)

Based on current conditions:

Cooling water supply _____(1)_____ available to the 1A KD Heat Exchanger.

Mini-Flow protection _____(2)_____ available for the 1B RN Pump.

Which ONE of the following completes the statements above?

A. 1. is
2. is

B. 1. is
2. is NOT

C. 1. is NOT
2. is

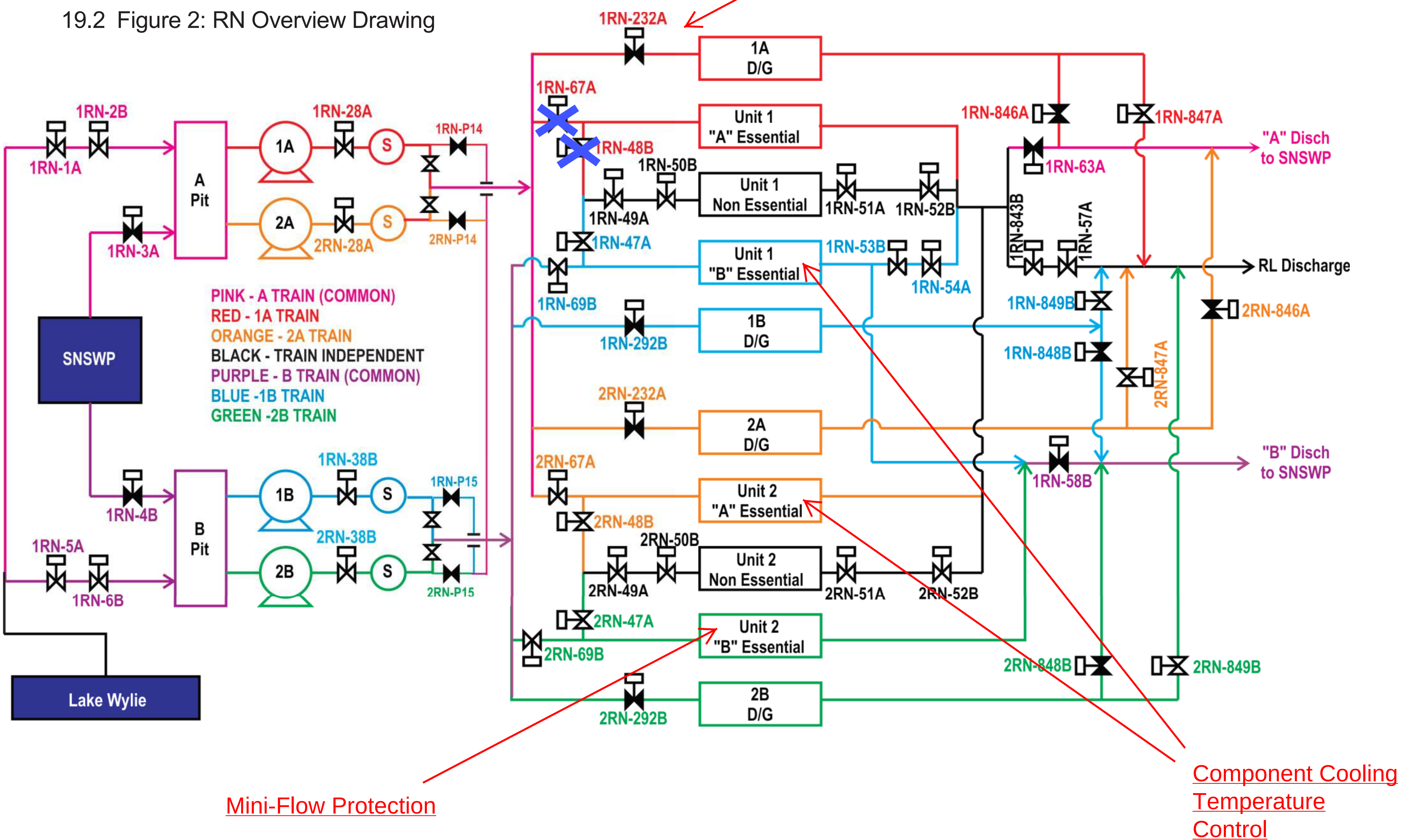
D. 1. is NOT
2. is NOT

Bank Question

Question 13

1A D/G Cooling Water remains available

19.2 Figure 2: RN Overview Drawing



Question 13

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

__ 4. **<u>IF</u> <u>AT</u> <u>ANY</u> <u>TIME</u> reactor trip occurs, <u>THEN</u> continue concurrent use of this procedure to stop flooding.**

__ 5. **Secure any liquid waste release in progress.**

__ 6. **<u>WHEN</u> source or location of flooding identified, <u>THEN</u> <u>GO</u> <u>TO</u> appropriate enclosure from table below:**

Source	Enclosure
RF/RY	Enclosure 3 (RF/RY Flooding)
Unit 1 CA Pump Room	Enclosure 6 (Flooding In Unit 1 CA Pump Room)
Unit 2 CA Pump Room	Enclosure 7 (Flooding In Unit 2 CA Pump Room)
RN	Enclosure 8 (Flooding From RN)
Any D/G Room	Enclosure 8 (Flooding From RN)
External source	Enclosure 11 (Flooding from External Sources)

7. **<u>WHEN</u> flooding under control, <u>THEN</u> perform the following:**

__ a. Initiate action to locally isolate as close to piping break as possible.

__ b. Evaluate potential of returning portions of isolated systems back to service.

__ c. Ensure configuration control maintained for equipment placed in an out of normal alignment by this AP.

Question 13

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

CAUTION **If RN flooding is caused by a sabotage event, RN leak must be isolated or RN pumps tripped within 35 minutes to prevent flooding of CA pump room.**

NOTE Normal RN System flowpath is diagramed for information in Enclosure 17 (RN Normal Alignment).

__ 3. **Verify location of flooding from RN - IDENTIFIED.**

Perform the following:

__ a. **IF AT ANY TIME** RN flooding jeopardizes essential plant equipment **AND** immediate isolation required, **THEN** observe Caution prior to Step 17 and **GO TO** Step 17.

__ b. Do **NOT** continue until location of flooding from RN identified.

__ 4. **GO TO appropriate step from table below to isolate flooding:**

Location	Step
Any KC Hx	Step 5
Any NS Hx	Step 6
A YC Chiller	Step 7
B YC Chiller	Step 8
Other Isolable Component or Branch Line	Step 9
Any D/G Room	Step 10
Unit 1 Non Essential Hdr	Step 11
Unit 2 Non Essential Hdr	Step 12
Unit 1 A Train Essential Hdr	Step 13
Unit 1 B Train Essential Hdr	Step 14
Unit 2 A Train Essential Hdr	Step 15
Unit 2 B Train Essential Hdr	Step 16
Unisolable Leak Requiring Isolation of Entire A Train	Step 17
Unisolable Leak Requiring Isolation of Entire B Train	Step 18

Question 13

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE The following steps isolate Unit 1 A Train RN. 1A KC and A YC cooling will be lost.	
13. **<u>IF</u> leak on Unit 1 RN A Train Essential Header, <u>THEN</u> perform the following:**	
a. **<u>IF</u>** time and manpower permit, **<u>THEN</u>** perform the following:	
__ • Verify A YC Chiller - SECURED	__ • Align VC/YC Trains as required. **<u>REFER TO</u>** Enclosure 15 (Shifting Operating VC/YC Trains).
__ • Verify 1A Train KC - SECURED.	__ • Align KC Trains as required. **<u>REFER TO</u>** OP/1/A/6400/005 (Component Cooling Water System).
b. Verify the following valves - OPEN: __ • 2RN-47A (RN Supply X-Over Isol) __ • 2RN-48B (RN Supply X-Over Isol).	__ b. Ensure one RN pump on each train - ON.

Question 13

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. (Continued)	
c. CLOSE the following valves:	
__ 1) 1RN-67A (RN Hdr 1A Supply Isol).	__ 1) Dispatch Operator to close 1RN-67A (1A RN Header Supply Isol) (AB-589, QQ-54, Rm 400).
	NOTE The following steps also isolate the Unit 1 RN Non-Essential Header.
__ 2) 1RN-48B (RN Supply X-Over Isol).	2) Perform the following: __ a) CLOSE 1RN-47A (RN Supply X-Over Isol). b) **IF** 1RN-48B **AND** 1RN-47A cannot be closed from control board, **THEN** dispatch Operator to CLOSE the following: __ • 1RN-48B (RN Supply X-Over Isol) (AB-585, PP-QQ, 53, Rm 400) OR __ • 1RN-47A (RN Supply X-Over Isol) (AB-585, PP-QQ, 53-54, Rm 400).

pressure bleeds down from any leakage that is present, unless a solenoid valve has been actuated to vent air off the actuator.

Air operated components which are required to operate to mitigate the consequences of a Design Basis Event (components which cannot be failed to and left in a safe position) are required to have a qualified backup gas supply. This assured backup gas supply will allow the component to fulfill its safety function if the Instrument Air System pressure to these components is lost.

Overall Plant Response to a Complete Loss of Instrument Air:

- Air will be lost to the following components:
  - Main feedwater control
  - Steam dump control
  - S/G PORVs in AUTO
  - Pzr PORVs and Pzr sprays
  - Main Steam Isolations
- Automatic Reactor trip will occur on “S/G LO-LO Level” due to CF control valves failed closed.
- Steam dumps will not operate. S/G PORVs will not open in “AUTO”. Heat removal will be via S/G safeties.
- NC system pressure may increase beyond the Pressurizer PORV setpoint. Nitrogen may have to be aligned to the Pzr PORVs.

Licensed Only

Loss of VI to Pressurizer (Pzr) PORVs **Distractor Part 1**

Pressure regulating valves supply Pzr PORVs 1(2)NC-34A, 1(2)NC-32B, and 1(2)NC-36B

Nitrogen can be supplied to 1(2)NC-34A and 1(2)NC-32B from the nitrogen supply from Cold Leg Accumulators A and B. **Correct Answer Parts 1 and 2**

The probabilistic risk assessment states the need to have two motive forces for Pressurizer PORVs in case NCS feed and bleed is needed, so the nitrogen supply was added. If no nitrogen is available and a major accident occurs in conjunction with a loss of VI to containment, then there may be no means to control the heat buildup in the reactor and consequently core damage may occur.

Aux Feedwater (CA) System Response to a Loss of Instrument Air **Distractor Part 2**

With a Loss of Instrument Air, the flow control valves for all S/G’s will no longer fail open. A Mod has been completed on both units which added air receiver tanks to the

CA flow control valves with enough air that will allow closure of these valves for 60 minutes after a loss of VI. This will also preclude S/G overfill on a SGTR with a loss of AC Power and a subsequent loss of VI.

The CA Flow Control Valve air accumulators provide a safety related (Unit 1) and non-safety (Unit 2) volume of compressed air to keep these valves closed, following a SGTR, until operator action makes this function unnecessary.

VI Supply to SM Isolation Valves (MSIV) Compressed Air Accumulators and Control Panels

For MSIVs 1/2SM-1, 1/2SM-3, 1/2SM-5 and 1/2SM-7, a safety-related compressed air accumulator and control panel (supplied by the VI System) is located near each valve. The accumulator provides the MSIVs with air-assist-to-close function. This air-assist-to-close function provides additional closing force margin beyond the closing force of the mechanical actuator springs. A safety-related check valve is located in the accumulator air supply piping to keep the accumulator pressurized so the air-assist-to-close function will be available during a depressurization or loss of the VI System.

4.2 Loss of Normal Cooling Water Supply

Objective 7C, All

If KR cooling has been lost, the Fire Protection (RF) System must be manually aligned to “E” and “F” VI compressors.

If aligned, RF cools the compressor aftercooler, oil cooler and intercooler.

Main Fire Pump “B” or “C” should be manually started prior to aligning RF to “E” and “F” VI Compressors to minimize water hammer resulting from Fire Pump auto start on low RF header pressure.

Discharge is manually aligned to the Low Pressure Service Water System (RL).

4.3 Station Blackout

Objective 7D, LICENSED

On a Station Blackout, KR cooling will be lost, and the Fire Protection (RF) System must be manually aligned to “E” and “F” VI compressors.

Both the “E” and “F” VI compressors receive power from the Blackout Load Centers - “E” VI Compressor is fed from Unit 1 blackout power, while “F” VI Compressor is fed from Unit 2 blackout power. E & F VI Compressor breakers will receive a restart permissive signal from the Sequencer with load group 13.

The CENTAC computers are powered from the same power source as the compressors themselves and will receive power when the compressor’s power source is aligned.

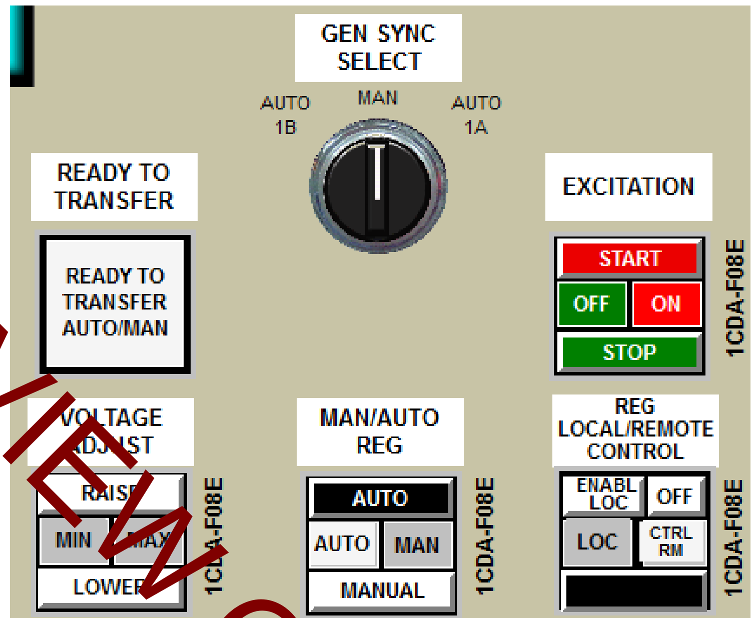
Question 15

power is supplied to the exciter which sends power thru the rectifiers to the main generator field which also starts producing power.

Permissives for excitation START include no Unitrol 5000 faults and no excitation trip inputs from external generator protective relays. When STOP button is depressed, the gate firing will be shutoff and the contactors opened, IF the generator is disconnected from the grid.

Voltage Adjust RAISE/LOWER Pushbutton

A momentary pushbutton switch, with MIN/MAX lights, that is used for adjusting voltage while in either manual or automatic control mode. There are separate limit settings for the manual mode and the auto mode of operation. **Distractor C.1 & D.1**



In the auto mode, the RAISE/LOWER buttons change the terminal voltage. In no-load operation, changing this setpoint adjusts generator voltage by changing Exciter output. In parallel operation, changing this setpoint adjusts reactive power and voltage. When the preset MIN/MAX limit is reached, the appropriate light is turned on. OAC points will alarm for the respective limit. Any further RAISE/LOWER commands beyond the limit are prevented. **Part 1 Correct Answer**

In the manual mode, the RAISE/LOWER buttons also change the exciter field current. When not connected to the power grid, the adjustment changes generator voltage; with the generator loaded the adjustment changes reactive power and voltage.

In manual mode, only an under excitation limiter (prevents magnetic slipping of the generator) and a Volts/Hertz Limiter (prevents magnetic saturation) are available. Also the RAISE/LOWER commands are not always prevented by a limiter, as in auto mode requiring the Operator to ensure the operating limits of the rotor and generator (according to the Generator Capability Curve) are not exceeded.

In emergency manual mode, the regulator maintains the firing angle of the bridge; the RAISE/LOWER buttons change the bridge firing angle which changes exciter field current.

In the manual and emergency manual modes, the operator will need to monitor the terminal voltage and make adjustments as necessary to maintain terminal voltage.

Operator Fundamental: MONITORING: **CRS/RO**
Ensure effective plant monitoring by operators (**CRS**)
Monitor plant parameters at a frequency based on their importance. **(RO)**
Increase the frequency of monitoring key parameters during a transient. **(RO)**

MANUAL/AUTO Regulator Pushbutton

This switch selects manual or automatic mode of voltage regulator control. AUTO is in the top position and MAN on the bottom. There are two indicating lights, AUTO on the left and MAN on the right. In AUTO, generator voltage is regulated so that a constant voltage is produced at the generator terminals. In MANUAL, exciter field current is regulated and generator voltage must be monitored by the Operator.

With a fluctuating generator load, the generator excitation (exciter field current setpoint) must be adjusted manually so that the generator voltage remains constant. It is possible to switch between operating modes at any time, because the inactive regulator always automatically follows the active regulator.

If a fault is detected in the AUTO mode (switch to manual mode), it is not possible to switch back to AUTO mode until the fault has been rectified. The switch from AUTO to MANUAL is prevented if there is a fault in the MANUAL mode.

The generator can operate in AUTO mode within permitted operating ranges, which lie outside of the permitted (and set) operating ranges for MANUAL mode. In these cases, the MANUAL regulator can no longer follow the AUTO regulator.

Regulator LOCAL/REMOTE Control Pushbutton

A single maintained pushbutton switch, with dual indicating lights, that prevents/allows control of the AVR from the LCP. This is a permissive switch to allow control at the LCP.

This is NOT a selector switch for remote or local control despite the name. The selection function can ONLY be performed at the LCP. The left indicating light (White) indicates control is from the Control Room and the right indicating light (White) indicates control is from the LCP.

READY TO TRANSFER Auto/Manual Light

This indication displays the status of the regulator that is not active. The READY TO TRANSFER light means that the follower circuit is close enough to provide a "bumpless" transfer, however the circuit will transfer even with the light not illuminated if the Manual/Auto switch is operated.

Part 2 Correct Answer

Question: 16
(1 point)

Given the following conditions on Unit 1:

- A LOCA outside containment has occurred
- EP/1/A/5000/ECA-1.2 (LOCA Outside Containment) has been entered
- Crew is performing step 2 to attempt to identify and isolate the leak
- Subcooling is -5°F and stable

In accordance with ECA-1.2:

The crew will FIRST attempt to isolate the leak by isolating the _____(1)_____ system from the NC system.

The crew _____(2)_____ evaluate RVLIS level to verify the leak is isolated.

Which ONE (1) of the following completes the statements above?

A. 1. ND
   2. will

B. 1. NI
   2. will

C. 1. ND
   2. will NOT

D. 1. NI
   2. will NOT

Original Question

MODIFIED

Question 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE If NC System is saturated, NC System pressure may not be a good diagnostic of leak isolation. NC System pressure will respond very slowly after leak is isolated, or may continue to drop if cooldown is in progress, until subcooling is restored.	
2. **Attempt to identify and isolate break as follows:**	
a. Isolate ND header 1A to cold legs as follows:	
__ 1) Place "PWR DISCON FOR 1NI-173A" in "ENABLE".	
__ 2) CLOSE 1NI-173A (ND Hdr 1A To Cold Legs C&D).	
3) Evaluate the following to determine if NC System leak isolated: __ • NC System pressure __ • RVLIS __ • Pzr level.	
__ 4) Verify NC System leak - ISOLATED.	4) Perform the following: __ a) OPEN 1NI-173A. __ b) Place "PWR DISCON FOR 1NI-173A" in "DISCON". __ c) **GO TO** Step 2.b.
__ 5) **GO TO** Step 3.	

Question 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	
b. Isolate ND header 1B to cold legs as follows:	
__ 1) Place "PWR DISCON FOR 1NI-178B" in "ENABLE".	
__ 2) CLOSE 1NI-178B (ND Hdr 1B To Cold Legs A&B).	
3) Evaluate the following to determine if NC System leak isolated:	
__ • NC System pressure	
__ • RVLIS	
__ • Pzr level.	
__ 4) Verify NC System leak - ISOLATED.	4) Perform the following: __ a) OPEN 1NI-178B. __ b) Place "PWR DISCON FOR 1NI-178B" in "DISCON". __ c) **GO TO** Step 2.c.
__ 5) **GO TO** Step 3.	

Question 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify NC System feed and bleed required as follows:**	
__ a. W/R level in at least 3 S/Gs - LESS THAN 24% (36% ACC).	a. Perform the following: __ 1) Monitor feed and bleed initiation criteria. **REFER TO** Enclosure 1 (Foldout Page). __ 2) **WHEN** criteria satisfied, **THEN GO TO** Step 21. __ 3) **GO TO** Step 6.
Correct Answer Part 2 (< 3.0 psig)	
Distractor Part 2 ACC Conditions (> 3.0 psig)	
__ b. **GO TO** Step 21.	
__ 6. **Ensure S/G BB and NM valves closed. REFER TO Enclosure 9 (S/G BB and NM Valve Checklist).**	
7. **Attempt to establish CA flow to at least one S/G as follows:**	
__ a. Verify 1AD-8, B/1 "UST LO LEVEL" - DARK.	a. Perform the following: __ 1) **REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater). __ 2) **GO TO** Step 7.c.

Question 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. **Attempt to establish CF flow to at least one S/G as follows:**	
__ a. Verify CM System - IN SERVICE.	__ a. **IF** CM System cannot be placed in service, **THEN** observe Note prior to Step 19 and **GO TO** Step 19.
b. Place the following valves in manual and closed: __ • All CF control valves __ • All CF bypass control valves.	
c. Ensure at least one of the following valves - OPEN: __ • 1CF-10 (1A CF Pump Disch Isol) (TB1-579, 1E-21) __ • 1CF-17 (1B CF Pump Disch Isol) (TB1-579, 1E-20).	__ c. Observe Note prior to Step 19 and **GO TO** Step 19.
__ d. Verify at least one CF pump - AVAILABLE TO BE STARTED. Main Feed Pump	__ d. **IF** both CF pumps known to be incapable of starting, **THEN GO TO** Step 14.
e. Verify following feedwater pump recirc valves - FULLY OPEN: __ • 1CF-6 (CF Pump 1A Recirc Ctrl) __ • 1CF-13 (CF Pump 1B Recirc Ctrl).	__ e. OPEN valve(s).

Question 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. **Attempt to establish feed flow from CM as follows:**	**Hotwell and Booster Pumps**

NOTE
- If feed and bleed has not yet been initiated, it is preferable to depressurize two S/Gs in the next step in order to:
  - Leave two S/G levels above feed and bleed criteria
  - Minimize NC System cooldown.
- If feed and bleed has been initiated, it is preferable to depressurize just one S/G in the following steps.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ a. Depressurize at least one S/G to less than 500 PSIG in the following steps.	
b. Verify condenser available as follows:	__ b. **GO TO** Step 17.i RNO.
__ • "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT	
__ • MSIV on S/Gs to be depressurized - OPEN.	
__ c. Place "STM DUMP CTRL" slim station in manual.	
__ d. Ensure steam dumps in pressure mode.	
__ e. **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK".	

7. ABNORMAL OPERATIONS

7.1 ECCS Operation Overview

Objective 9A, Licensed

A Safety Injection Actuation will occur on any of the following signals:

- **Manual** - 1 of 2 switches
- **High containment pressure** – greater than or equal to 1.2 psig (2/3 channels)
- **Pressurizer low pressure** - less than or equal to 1845 psig (2/4 channels)

(Refer to the ISE lesson plan for more details on Safety Injection signal generation.)

The Centrifugal Charging (NV) Pumps have the highest discharge pressure and are first to be effective in delivering flow to the Reactor Coolant (NC) System. Injection of borated water from the FWST continues and increases with any decrease in NC System pressure. As the NC pressure is reduced, sequentially lower pressure portions of the ECCS become effective in delivering flow. The NI Pumps, having the next highest discharge pressure, provide flow to the core, followed by the Cold Leg Accumulators (CLAs) and the ND Pumps. When the FWST level reaches the low level setpoint, the ND suction is automatically aligned for the recirculation phase of operation. Once FWST level decreases to ~5%, the ECCS is manually aligned into a "piggyback" mode of operation. The ND Pumps take suction from the Containment Recirculation Sump providing suction to the NI and NV Pumps in providing the water supply to either the NC cold legs or hot legs.

Summary of Pressure and Injection flows for all ECCS systems:

Component	NC Pressure for injection to begin	Design Flow	Runout Flow
NV Pumps	>2500 psig	150 gpm at 2500 psig	550 gpm at 600 psig
NI Pumps	1520 psig	400 gpm at 1170 psig	650 gpm at 715 psig
CLAs	580-650 psig	N/A	N/A
ND Pumps	198 psig	3000 gpm at 160 psig	4500 gpm at 117 psig

Part 1 Correct Answer

Question 18

1. **Emergency Coolant Recirc Capability Restoration:**

   - **WHEN** emergency coolant recirc capability restored during this procedure, **THEN** perform the following:

     a. **IF** transfer to Cold Leg Recirc required, **THEN GO TO** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation).

     b. **RETURN TO** procedure and step in effect.

2. **ECCS Suction Source Monitoring Criteria:** **Part 2 Correct Answer**

   - **IF** suction source lost to any ECCS **OR** NS pump, **THEN** stop affected pump.

   - **IF** FWST level lowers to less than 5% (1AD-9, E/8 "FWST LO-LO LEVEL" alarm setpoint), **THEN** stop all pumps taking suction from FWST.

   - **IF** both "CONT. SUMP LEVEL >2.5 ft" annunciators on 1AD-20 and 1AD-21 dark, **THEN** stop all pumps taking suction from Containment Sump.

3. **CA Suction Source Switchover Criterion:**

   - **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

Question 18

5. **1AD-9, C/8 "FWST PRE-LO LEVEL" (27%) LIT Alarm Actions:**

a. Verify at least one of the following annunciators - LIT:

- 1AD-20, B/2 "CONT. SUMP LEVEL >2.5 ft"

OR

- 1AD-21, B/2 "CONT. SUMP LEVEL >2.5 ft".

b. **IF** both alarms dark, **THEN** perform the following for ND pumps with suction aligned to FWST or Containment Sump:

1) Ensure the following - RESET:

- ECCS
- D/G Load Sequencers.

2) Stop ND pumps.

3) **IF** ND pump(s) will not stop, **THEN** perform the following for affected train(s):

- ND Train 1A:
  a) Place "PWR DISCON FOR 1NI-173A" switch in "ENABLE".
  b) CLOSE 1NI-173A (ND Hdr 1A To Cold Legs C&D).
  c) CLOSE 1ND-32A (ND Train 1A Hot Leg Inj Isol).

- ND Train 1B:
  a) Place "PWR DISCON FOR 1NI-178B" switch in "ENABLE".
  b) CLOSE 1NI-178B (ND Hdr 1B To Cold Legs A&B).
  c) CLOSE 1ND-65B (ND Train 1B Hot Leg Inj Isol).

4) Do not restart either ND pump until at least one "CONT. SUMP LEVEL >2.5 ft" annunciator - LIT.

5) Ensure "S/I Reinitiation Criteria" of this foldout page monitored to determine if additional NV or NI SI flow needed.

6. **Cold Leg Recirc Switchover Criterion:**

Distractor A.2 & C.2

- **IF** FWST level lowers to 20% (1AD-9, D/8 "FWST 2/4 LO LEVEL"), **THEN GO TO** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation).

Question 19

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

---NOTE---
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

Correct Answer

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within the limit specified in COLR.	In accordance with the Surveillance Frequency Control Program

Question 19

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by a minimum of four bolts;

b. A minimum of one door in each air lock closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
2. exhausting through an OPERABLE Containment Purge Exhaust System (CPES) HEPA filter and carbon adsorber.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment.

if <72 Hours since shutdown

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

Distractor A

Question 19

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

BDMS are available source range neutron monitors

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required neutron flux monitor inoperable. Distractor C	A.1 Suspend CORE ALTERATIONS. AND A.2 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately Immediately
B. Two required neutron flux monitors inoperable.	B.1 Initiate actions to restore one neutron flux monitor to OPERABLE status. AND B.2 Perform SR 3.9.1.1.	Immediately Once per 12 hours

Question 19

3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) and Coolant Circulation — High Water Level

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation.

---NOTE---

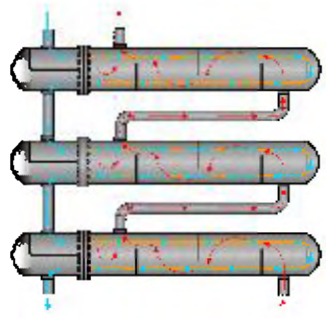
The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause introduction of coolant into the Reactor Coolant System with boron concentration less than required to meet the minimum required boron concentration of LCO 3.9.1.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met. **RHR Loop requirements would NOT be met if water level was <23 ft**	A.1 Suspend operations that would cause introduction of coolant into the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1.	Immediately
	<u>AND</u>	
Distractor D	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	
		(continued)

2.1.2 Regenerative Heat Exchanger



The regenerative heat exchanger, located in containment, is designed to recover heat from letdown by reheating the charging flow. This eliminates the reactivity effects of introducing cold charging flow and reduces the thermal shock on the charging line penetration to the NC system. In addition, the heat exchanger cools the letdown flow prior to depressurization to prevent flashing of water to steam across the letdown orifices.

The heat exchanger is three single-shell, multi-pass Heat Exchangers. The letdown water flows through the shell side of the heat exchanger and charging water flows through the tube-side.

The charging flow, normally at ~2275 psig, enters at 90°F, and exits at 530°F. The letdown flow, normally at 2235 psig, enters at 550ºF and exits at 275ºF.The maximum flow rate through the tube side (charging) of the Regenerative Heat Exchanger is 180 gpm per the manufacturer.

2.1.3 Letdown Orifices and Letdown Variable Orifice Control Valve, NV-849

There are two letdown orifices located in the reactor building. One 75 gpm orifice and one 45 gpm orifice reduce letdown pressure and control the letdown flow. In addition, Letdown Variable Orifice Control Valve NV-849 is an air operated valve located on the letdown line in parallel with the two orifices identified above.

NV-849 may be used by itself or in combination with any of the other two letdown orifices to provide the desired letdown flow. Normal letdown flow is 75 gpm. Maximum design flow through NV-849 is 120 gpm. However, for optimal system performance, the travel stops on NV-849 are set to allow a maximum of 115 gpm through the valve at NCS normal temperature and pressure. This maximum flow ensures that the pressurizer level control system can maintain programmed level without any significant NV charging re-balancing (NCP seal flows, NV-309, and NV-294 positions).

NV-849 is controlled from MC10 using a SLIM station or via soft controls on the DCS graphic. NV-849 is always operated in manual by procedure. **Correct Answer Part 1**

NV-849 is a fail-open air operated valve which is acceptable because both upstream and downstream valves fail closed on loss of air or power.

Question 20

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Monitor Enclosure 1 (Case I Steam Generator Tube Leak Foldout Page).**	
__ 2. **Verify Pzr level - STABLE OR TRENDING UP.**	**Perform the following:** __ a. Maintain charging flow less than 180 GPM. __ b. THROTTLE 1NV-294 (NV Pmps A&B Disch Flow Ctrl) to stabilize Pzr level. __ c. **IF** Pzr level stable **OR** trending up, **THEN GO TO** Step 3. d. **IF** Pzr level continues to trend down, **THEN** reduce or isolate letdown as follows: 1) **IF** desired to reduce letdown flow, **THEN** perform the following: a) **IF** 1NV-10A (Letdn Orif 1B Otlt Cont Isol) open, **THEN** perform the following: __ (1) Control 1NV-148 (Letdn Press Control) to establish letdown pressure between 375 - 400 PSIG. __ (2) THROTTLE 1NV-849 (Letdn Flow Var Orif Ctrl) for 45 GPM letdown flow. __ (3) **WHEN** 45 GPM letdown flow established, **THEN** adjust 1NV-148 (Letdn Press Control) to maintain letdown pressure at 350 PSIG. __ (4) **WHEN** letdown pressure stable at 350 PSIG, **THEN** place 1NV-148 (Letdn Press Control) in auto.
Correct Answer Part 2	

(RNO continued on next page)

Question 20

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. (Continued)

b) **<u>IF</u>** 1NV-13A (Letdn Orif 1A Otlt Cont Isol) open, **<u>THEN</u>** perform the following:

__ (1) Control 1NV-148 (Letdn Press Control) to establish letdown pressure between 150 - 200 PSIG.

Distractor Part 2

__ (2) OPEN 1NV-11A (Letdn Orif 1C Otlt Cont Isol).

__ (3) Adjust 1NV-148 (Letdn Press Control) to establish letdown pressure between 375 - 400 PSIG.

__ (4) CLOSE 1NV-13A (Letdn Orif 1A Otlt Cont Isol).

__ (5) Adjust 1NV-148 (Letdn Press Control) to maintain letdown pressure at 350 PSIG.

__ (6) **<u>WHEN</u>** letdown pressure stable at 350 PSIG, **<u>THEN</u>** place 1NV-148 (Letdn Press Control) in auto.

2) **<u>IF</u>** letdown isolation required, **<u>THEN</u>** ensure the following valves - CLOSED:

__ • 1NV-10A (Letdn Orif 1B Otlt Cont Isol)

__ • 1NV-11A (Letdn Orif 1C Otlt Cont Isol)

__ • 1NV-13A (Letdn Orif 1A Otlt Cont Isol).

__ 3) **<u>IF</u>** Pzr level stable **<u>OR</u>** trending up, **<u>THEN</u> <u>GO TO</u>** Step 3.

(RNO continued on next page)

Question: 60
(1 point)

The C-9 (Condenser Available For Steam Dump) permissive requires:

a minimum vacuum sensed in the _____(1)_____ main condenser(s)

AND

a cooling water supply determined by _____(2)_____.

Which ONE of the following completes the statements above?

A. 1. "A" AND "B"
   2. RC Pump breaker position

B. 1. "A" OR "B"
   2. RC Pump breaker position

C. 1. "A" AND "B"
   2. RC system flowrate

D. 1. "A" OR "B"
   2. RC system flowrate

Original Question

MODIFIED

Question 21

The steam dump system has three modes of operation. The Load Rejection and Plant Trip modes are available when the unit is at high power. The Load Rejection Mode occurs when the turbine load decreases significantly and the reactor does not trip. The Plant Trip Mode occurs when the reactor and turbine trip. The Steam Pressure Mode is primarily used for low power, decay heat removal, and when the T-AVG modes are unavailable.

Each mode of steam dump operation has its respective controller. The Load Rejection controller has a maximum output of 100%. The Plant Trip controller and the Steam Pressure controller have a maximum output of 49%.

NOTE: The maximum output of 100% from the Load Rejection controller is equivalent to 70% steam flow (RTP) for the plant. RTP is Rated Thermal Power and is the total reactor core heat transfer rate to the reactor coolant of 3411MWt.

3. COMPONENT/SYSTEM DESCRIPTION

3.1 Steam Dump Valves

Objective 3A

The IDE accomplishes its purpose by use of five banks of steam dump valves with a total capacity of approximately 70% RTP. The dump valves are divided into two groups, condenser dump valves discharge to the main condenser and the atmospheric dump valves discharge to atmosphere.

3.1.1 Condenser Dumps

The Condenser Dump valves are made up of three banks numbered 1, 2 and 3, with each bank containing three valves with a total capacity of ~35% RTP for all three banks. Actuation of the condenser dump valves requires a combination of several arming signals. **Figures 1 & 2**

Arming Signals

Objective 4A

All actuations of the condenser dump valves require the availability of the main condenser. This availability is assured with the C-9 permissive, Condenser Available For Steam Dump. The requirements for C-9 are:

- 2 out of 2 condenser pressure sensors greater than 15" HgVac on Condensers A and B and
- 1 RC pump breaker closed

Correct Answer

The other requirement for actuation of the condenser dump valves is any one of the following:

- C-7A, " Loss of Load Interlock Condenser Dump", or
- Steam Pressure Mode Selected, or
- Train 'A' P-4 contact

Question 21

MFP TURBINE

1. LP CV → Receives steam from AS up to 85% Power → Supply then swapped to 'A' MSR Outlet → Superheated.
2. HP CV → Steam from SM. Augments LPCV when LP CV is fully open.
3. Turning Gear → Automatically attempts to engage for 5 sec when CFPT speed ↓ to zero.
   a. If the turning Gear fails to engage it must be reset locally.

NOTE: Turning Gear will AUTO disengage < 4 Psig Oil Pressure or excessive speed.

4. Rupture Disk → Prevents over pressurization of MFP Condenser and exhaust Hood.
5. On loss of EPD (125 VDC Control Power) both CFPTs will trip and cannot be RESET.

CFPT Interlocks

1. RX Trip → CFPT Runback to min speed → Slaves go to MANUAL. Can BYPASS with key switch.
2. CF-10/17 (MFP Discharge Valves) → CANNOT be CLOSED if Turbine is RESET.
3. Windmill → Trips all HWPs
4. MFP Recirc Valves → CLOSE on Lo Oil Pressure 4 psig
5. CM-137/140 (MFP Suction Valves) → CANNOT be OPENED with Low Oil Pressure.

CFPT Trips

Distractor Part 2

1. Manual (Control Room and Local Panel)
2. Loss of Power KXPA/RPB, KXPB/RPA
3. Low Vacuum 12.4" Hg abs (16.9 " Hg vac)
4. Lo Lo Bearing Oil Header Pressure 4psig
5. Lo Lo Suction Flow 3000 gpm (20 sec delay)
6. Lo Lo Suction Pressure 275 psig (20 sec delay)
7. Hi Hi Discharge Pressure 1385 psig
8. Hi Hi Doghouse Level 2/3 11 inches
9. Hi Hi S/G Level (P-14) Unit 1 – 83 %, Unit 2 – 77%
10. HP & LP Stop Vals Clsd Defeated 30 sec a/RESET
11. Oil Fire Lockout
12. Primary Overspeed 113 % (6226 rpm)
13. Emergency Overspeed 115 % (6336 rpm)
14. Safety Injection
15. Thrust Bearing Wear 2/2 ± 30 mils
16. Zero Speed 5sec to be ≥ 2 rpm a/RESET and ≥ 0%

Limits and Precautions

1. See the latest revision of OP/1/A/6250/001 for applicable limits and precautions.

**SLC 16.7-14 MFP Trips and Instrumentation

- Requires 3 channels per pump Modes 1/2.
- **One channel of trip of feed pump pressure switch inop – *1 hour* to place in trip OR M3 in 7 hours**
- **More than one channel inop – *1 hour* to initiate actions to go to Mode 3.**

TS 3.3.2 ESFAS (CA Autostart) BASES

- Table 3.3.2-1, Item 6.e: Requires 3 channels of loss of control oil per pump (i.e. CFPT Tripped) Modes 1,2.
- lose one pressure switch, place channel in trip in **1 hour** OR Mode 3 in 7 hours

Question 21

MFP TURBINE

1. LP CV → Receives steam from AS up to 85% Power → Supply then swapped to 'A' MSR Outlet → Superheated.
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CFPT Trips

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15. Thrust Bearing Wear 2/2 ± 30 mils
16. Zero Speed 5sec to be ≥ 2 rpm a/RESET and ≥ 0%

Distractor Part 2

Limits and Precautions

1. See the latest revision of OP/1/A/6250/001 for applicable limits and precautions.

**SLC 16.7-14 MFP Trips and Instrumentation

- Requires 3 channels per pump Modes 1/2.
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- lose one pressure switch, place channel in trip in **1 hour** OR Mode 3 in 7 hours

SUMMARY OF CHANGES

(Short Description of Changes for THIS Revision)

SECTION #	CHANGES
2.7	Corrected document to state that only AUTO feed pump control is locked out when the reactor is tripped.

Question 22

Operations Liquid Waste Release

1. Purpose

To aid the operator in the correct methods of performing steps in Radwaste procedure OP/0/B/6500/015 (Discharging a Monitor Tank to the Environment) and Radiation Protection procedure HP/0/B/1004/004 (Radioactive Liquid Waste Release). Also to aid the operator as to limits and results expected while these procedures are being performed.

2. Limits and Precautions

2.1 Ensure that RN is discharging through at least one RL header.

2.2 Ensure that RN is **NOT** discharging to SNSWP.

2.3 If the pre-set radiation levels are exceeded on EMF-49 or the dilution flow rate drops below the setpoint for 0RLP5080 (RL Discharge Total Flow), 1WL-124 (Waste Monit Tnk Pmps Disch) will trip closed.

2.4 Releases that are interrupted by EMF-49 "HI-RAD" trips may be initiated up to a maximum of three times, including original initiation, without re-sampling per HP/0/B/1004/004 (Radioactive Liquid Waste Release).

2.5 Turbine Building Sump releases are secured if the pre-set levels are exceeded on 1/2EMF-31.

3. Procedure

Refer to Section 4 (Enclosures)

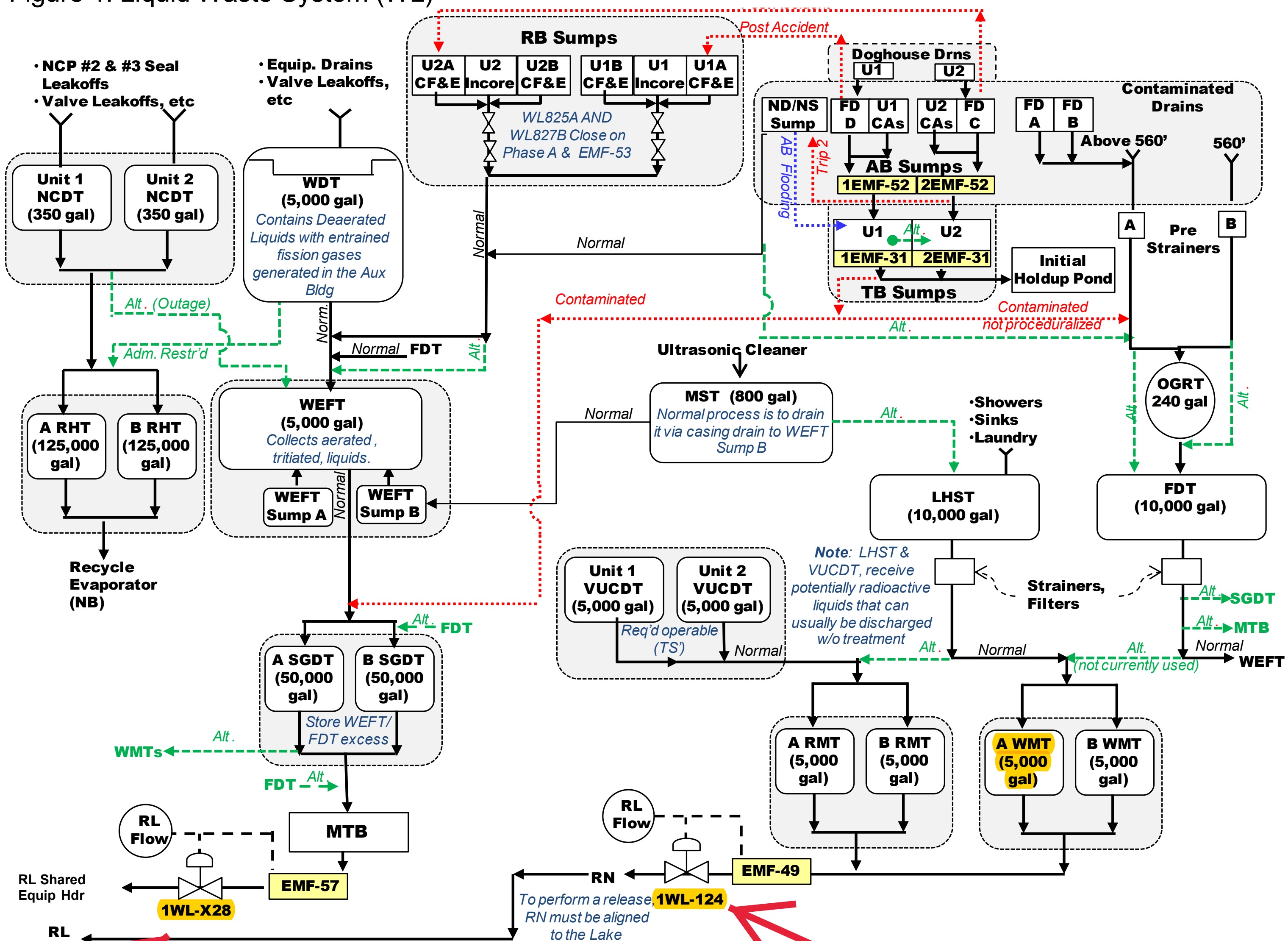
4. Enclosures

4.1 Liquid Waste Release from a Monitor Tank

4.2 Discharging a Contaminated Turbine Building Sump to Holdup Pond

28. FIGURES

28.1 Figure 1: Liquid Waste System (WL)



Distractor Part 2

Correct Answer Part 2

Question 23

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9. **Attempt to identify and isolate leak as follows:**

a. Verify the following conditions - NORMAL:

__ • Containment temperature

__ • Containment pressure

__ • Containment humidity

__ • Containment floor & equipment sump level.

a. Perform the following:

__ 1) Evacuate containment.

2) Perform the following:

__ a) Start all lower containment ventilation units in low speed.

__ b) Start all upper containment ventilation units.

__ c) Place all upper and lower containment ventilation units in "MAX" cooling.

3) **<u>IF</u> <u>AT</u> <u>ANY</u> <u>TIME</u>** containment pressure reaches 1.2 PSIG, **<u>THEN</u>** perform the following:

__ a) Ensure Unit 1 reactor tripped.

__ b) Ensure S/I initiated.

c) CLOSE the following valves:

__ • All MSIVs

__ • All MSIV bypass valves.

__ d) **<u>GO</u> <u>TO</u>** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

__ 4) **<u>GO</u> <u>TO</u>** Step 10.

__ b. Dispatch operators to locate and identify source of steam leak.

Question 24

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16. **<u>WHEN</u> "P-11 PZR S/I BLOCK PERMISSIVE" status light (1SI-18) lit, <u>THEN</u> perform the following:**	
__ a. Depress ECCS steam pressure "BLOCK" pushbuttons.	
__ b. Verify main steam isolation blocked status lights (1SI-13) - LIT.	

<u>NOTE</u>

- Partial uncovering of the S/G tubes is acceptable in the following steps.
- After low steamline pressure main steam isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation.
- OAC graphic SMRATES to monitor S/G pressure rates can be accessed via a hot button in the center of the SM graphic.

C-9 NOT lit due to loss of offsite power

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. **Depressurize all intact S/Gs as follows:**	
a. Verify condenser available as follows:	__ a. **<u>GO TO</u>** Step 17.d RNO.
__ • "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT	
__ • MSIV on intact S/G(s) - OPEN.	
__ b. Verify steam dumps - IN PRESSURE MODE.	b. Place steam dumps in pressure mode as follows:
	__ 1) Place "STM DUMP CTRL" slim station in manual.
	__ 2) Place steam dumps in pressure mode.
__ c. **<u>WHEN</u>** "P-12 LO-LO TAVG" status light (1SI-18) lit, **<u>THEN</u>** place steam dump interlock bypass switches in "BYP INTLK".	

Question 24

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ d. Dump steam to condenser at maximum rate while attempting to avoid Main Steam Isolation. **Distractor Part 2**	__ d. Dump steam from all intact S/G(s) with S/G PORV(s) at maximum rate. **Correct Answer**
__ e. Verify all intact S/G pressures - LESS THAN 140 PSIG.	e. Perform the following: __ 1) **IF** S/G pressure trending down, **THEN RETURN TO** Step 13. __ 2) **GO TO** Step 24.
__ f. Verify at least two NC T-Hots - LESS THAN 370°F.	f. Perform the following: __ 1) **IF** NC T-Hots trending down, **THEN RETURN TO** Step 13. __ 2) **GO TO** Step 24.
__ g. Stop S/G depressurization and maintain S/G pressures stable.	

Question 24

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

15. (Continued)

__ d. Dump steam to condenser while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour.

Distractor Part 1

d. Perform the following:

__ 1) Maintain cooldown rate based on NC T-Colds less than 100°F in an hour while dumping steam in the following steps.

__ 2) Dump steam using intact S/G PORV(s).

3) **IF** any intact S/G PORV cannot be operated from Control Room, **THEN** perform the following:

__ a) Dispatch operator(s) to operate affected S/G(s) PORV. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 10 (Local Operation of S/G PORVs).

__ b) Obtain sound powered phone from storage box on rear wall of Control Room.

__ c) Connect sound powered phone to jack on 1MC-11.

__ d) Monitor sound powered phone for communication from Doghouse(s).

4) **IF** any intact S/G PORV unavailable, **THEN** evaluate using the following to dump steam:

__ • OPEN MSIVs and dump steam to condenser. **REFER TO** Enclosure 4 (Condenser Dump Operation).

__ • Start CA Pump #1.

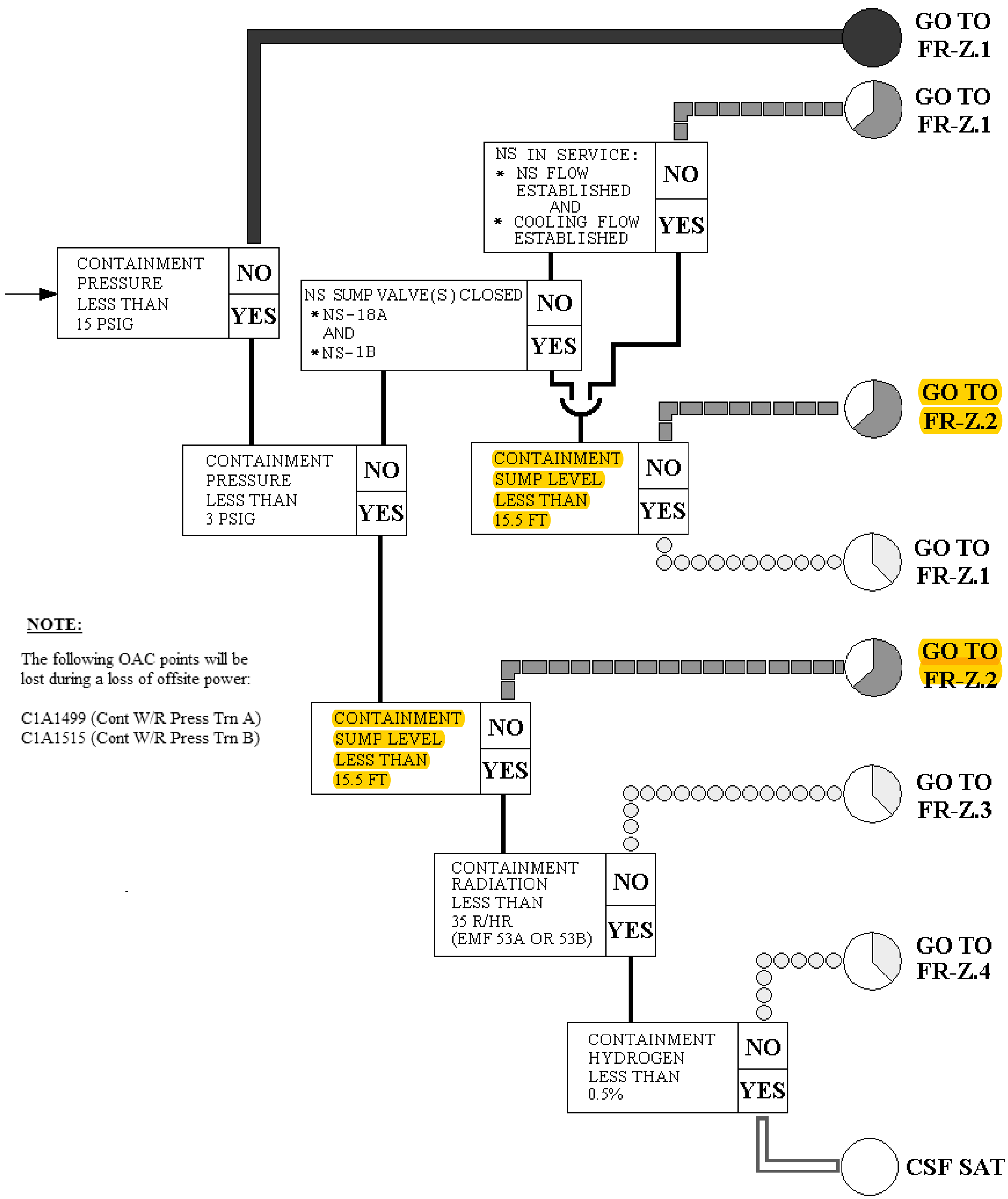
Question 25

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Verify S/I termination criteria:**	
__ a. NC subcooling based on core exit T/Cs - GREATER THAN 0°F.	__ a. **GO TO** Step 9.f.

ACC Conditions apply due to Containment pressure > 3.0 psig

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. Secondary heat sink:	__ b. **GO TO** Step 9.f.
__ • Any intact S/G N/R level - GREATER THAN 11% (29% ACC)	**Distractor Part 1**
OR	
__ • Total feed flow to intact S/Gs - GREATER THAN 450 GPM.	**Correct Answer Part 1**
__ c. NC pressure - STABLE OR TRENDING UP.	__ c. **GO TO** Step 9.f.
__ d. Pzr level - GREATER THAN 11% (30% ACC).	d. Perform the following: __ 1) **IF** NC pressure trending up **AND** normal Pzr spray available, **THEN** attempt to stabilize NC pressure using normal Pzr spray. __ 2) **GO TO** Step 9.f.
__ e. **GO TO** EP/1/A/5000/ES-1.1 (Safety Injection Termination).	**Correct Answer Part 2**
__ f. Monitor S/I termination criteria. **REFER TO** Enclosure 2 (S/I Termination Criteria).	
__ g. **IF AT ANY TIME** S/I termination criteria met while in this procedure, **THEN RETURN TO** Step 9.	**Distractor Part 2**

Question 26



NOTE:

The following OAC points will be lost during a loss of offsite power:

C1A1499 (Cont W/R Press Trn A)
C1A1515 (Cont W/R Press Trn B)

Question 26

Containment boundary penetrations for isolation valves, personnel air locks and equipment hatches are considered in the maximum leak rate calculation (L_a).

The isolation valves or devices are passive or active (automatic). Automatic devices provide two isolation barriers that are closed on a containment isolation signal. Examples of manual devices include de-activated automatic valves, closed manual valves, blind flanges, check valves, or other automatic valves designed to close without operator action.

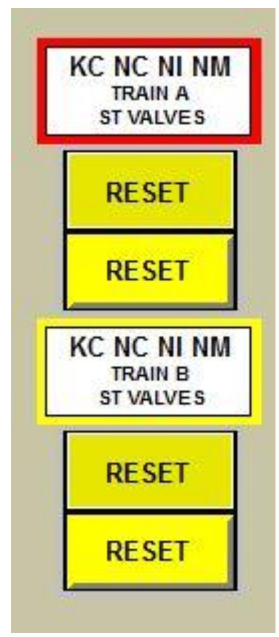
Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses.

Containment Isolation Valves

Containment isolation valves are identified in UFSAR table 6-77 (Containment Isolation Valve Data). UFSAR Table 6-77 lists all containment isolation valves and indicates the appropriate Tech Spec condition for inoperable valves. Most are automatically operated valves. Some are manual valves that are normally locked closed. A few are manually operated and normally open.

Post-Accident Valve Submergence

A list of active valves in containment that are below maximum flood elevation is presented in UFSAR Table 6-96.



Some valve operators were not qualified for submergence. These valves close on Containment Isolation Phase A (S_T) signals. There is sufficient time for them to close before being flooded.

To prevent possible repositioning after flooding, the valves motor controls circuits have been modified. One relay per train will be energized by a Phase A (S_T) signal and mechanically latched in. Normally closed contacts from this relay will be wired between the limit switches and the open motor starter coils of valves of the corresponding train. These contacts will open on S_t and prevent any spurious limit switch operation from repositioning the valves.

These relays have manual reset capability in the control room. (KC, NC, NI, NM S_T)

Correct Answer Part 2

Question: 43
(1 point)

Given the following conditions on Unit 1:

- A LOCA has occurred
- Containment pressure peaked at 3.3 PSIG, and is now 2.5 PSIG and slowly lowering
- Crew has entered EP/1/A/5000/ES-1.2 (Post LOCA Cooldown and Depressurization) and is performing the initial cooldown

LOOP DATA

		LOOP A	LOOP B	LOOP C	LOOP D
CURRENT T-COLD, BEST	(DEG F)	546.0	546.4	546.1	546.1
ADMINSTRATIVE LIMIT	(DEG F)	479.8	480.2	479.8	479.8
TECH SPEC LIMIT	(DEG F)	459.8	460.2	459.8	459.8
T-COLD CURRENT MINUS 1 HR T-COLD MAXIMUM	(DEG F)	-12.8	-13.6	-12.3	-12.1
15-MIN RATE	(DEG F/HR)	-38	-38	-36	-36
5-MIN RATE	(DEG F/HR)	-131	-134	-128	-127
1-MIN RATE	(DEG F/HR)	-113	-107	-111	-102

In accordance with ES-1.2:

The **INITIAL** cooldown will be started using the _____(1)_____.

With rates established, per the graphic above, the cooldown _____(2)_____ continue.

Which ONE (1) of the following completes the statement above?

A. 1. S/G PORVs
   2. can NOT

B. 1. S/G PORVs
   2. can

Original Question

C. 1. Condenser Steam Dumps
   2. can NOT

MODIFIED

D. 1. Condenser Steam Dumps
   2. can

Question 27

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ g. Dump steam to condenser while maintaining cooldown rate based on NC T-Colds as close as possible without exceeding 100°F in an hour.	__ g. **IF** steam cannot be dumped to condenser, **THEN GO TO** Step 10.i.
Correct Answer Part 1	Distractor Part 1
__ h. **GO TO** Step 11.	
__ i. Dump steam from intact S/G PORVs while maintaining cooldown rate based on NC T-Colds as close as possible without exceeding 100°F in an hour.	i. **IF** any intact S/G PORV cannot be operated from Control Room, **THEN** perform the following: __ 1) Dispatch operator(s) to dump steam from S/G(s) PORV. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 10 (Local Operation of S/G PORVs). __ 2) Obtain sound powered phone from storage box on rear wall of Control Room. __ 3) Connect sound powered phone to jack on 1MC-11. __ 4) Monitor sound powered phone for communication from the Doghouse(s).
If Containment pressure exceeded 3 psig condenser dumps would not be available	
__ 11. **Verify NC subcooling based on core exit T/Cs - GREATER THAN 0°F.**	__ **GO TO Step 29.**

Catawba Nuclear Station

ILT-17 NRC Written Exam CNS RO NRC Examination

Question: 28
(1 point)

Given the following Unit 1 conditions:

- The Unit is at 100% RTP
- Total charging flow is currently 90 gpm
- 1NV-294 (NV Pmps A&B Disch Flow Ctrl) is in MANUAL
- 1NV-309 (Seal Water Injection Flow) is in AUTO

Assuming stable plant conditions, as 1NV-294 is throttled OPEN, 1NV-309 will throttle in the ______(1)______ direction.

In order to restore automatic control of the Pressurizer Level Control system ______(2)______ must be placed in AUTO.

Which ONE of the following completes the statements above?

A. 1. OPEN
   2. 1NV-294 ONLY

B. 1. CLOSED
   2. 1NV-294 ONLY

C. 1. OPEN
   2. 1NV-294 AND PZR Level Master

D. 1. CLOSED
   2. 1NV-294 AND PZR Level Master

Original Question

MODIFIED

Question 28

2.3.8 Charging Header Flow Control Valve, NV-294

Charging Header Flow Control Valve NV-294 is an air operated globe valve which controls charging flow on the discharge of the NV pumps based on pressurizer level.

NV-294 is controlled from MC10 by a SLIM station or via DCS Soft Controls. The SLIM station is normally in the "AUTO" position which allows the Pressurizer Level Control System to control valve position. NV-294 will maintain a minimum flow while in AUTO. This minimum flow is set by the operator. Normally it is set for 47 gpm (32 gpm for seal injection flow and 15 gpm for cooling flow through the regenerative heat exchanger). Anytime NV-294 is placed in Manual, the Pressurizer Level Master Controller will also transfer to Manual to prevent windup in the Pressurizer Level Master Controller (discussed in more detail in the ILE Lesson Plan).

On transfer of control to the ASP, control of NV-294 from the control room is disabled by operator action to swap a plug connection.

Upon loss of air or power, NV-294 fails to the open position to ensure charging, so there is no net loss of inventory from the NV System. If control of NV-294 is lost from the control room, a manual valve, NV-295, located just downstream of NV-294, is available to re-establish throttling.

2.3.9 NCP Seal Injection Flow Control Valve, NV-309

NCP Seal Injection Flow Control Valve, NV-309 is an air operated globe valve which maintains backpressure on the charging header to ensure adequate seal water is provided to the NC Pumps. NV-309 is normally maintained in automatic to provide 32 gpm seal injection. NV-309 is located in the charging flow path, so closing this valve diverts more flow to the NCP seals.

Correct Answer

In addition to MC10, this valve may be controlled from ASP A or ASP B. On transfer of control to the ASP, control of NV-309 from the control room is disabled by operator action to swap a plug connection.

Upon a loss of air or power, NV-309 fails to the open position to ensure sufficient charging so there is no net loss of inventory from the NC system. A manual valve (NV-308) can be used to isolate NV-309, and another manual valve (NV-311) in parallel with NV-309 is available to re-establish throttling.

Question 28

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be $\leq$ 40 gpm with centrifugal charging pump operating and the charging flow control valve full open.

Distractor Part 2

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow not within limit.	A.1 Adjust manual seal injection throttle valves to give a flow within limit with centrifugal charging pump operating and the charging flow control valve full open.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	12 hours

Two VCT Level Control signals are used to perform the control functions described above. The following table lists the specific functions of each level control signal.

Control Signal	Associated VCT Level Channel*	Control Function
Selected VCT Level-1	LT-5761	• Auto Makeup start and stop signals • Modulate signal to NV-172 • Swap to FWST on low level** • High Level Alarm • Low Level Alarm
Selected VCT Level - 2	LT-5760	• Provides backup signal to fail open NV-172 on high VCT level • Swap to FWST on low level** • High Level Alarm • Low Level Alarm
*In case of a failure, the associated VCT Level Channel could be swapped to the non-failed channel via DCS controls. **Both Selected VCT Level-1 and -2 are required to swap to the FWST (2/2 logic)		

2.2.2 VCT Level Instrument Failures

The four VCT level indication channels have a common reference and variable leg tap coming off on the VCT (**Figure 3**). Thus, the response of the VCT level instrument failure depends on where the failure occurs. The VCT Level instruments used for control, LT-5760 and LT-5761, have bellows installed on the reference and common legs.

If the difference between LT-5760 and LT-5761 exceeds **3%,** a DCS Trouble Alarm will be generated. When the discrepancy between LT-5760 and LT-5761 reaches **6%,** a DCS Alternate Action will occur. Because only two VCT level channels are used for control, DCS is unable to determine which signal is BAD if either channel fails. Thus, DCS puts both Selected VCT Level -1 and Selected VCT-Level 2 in Alternate Action if either channel fails. (Alternate Actions are discussed in more detail later in the lesson plan.)

Correct Answer Part 1

The impacts of the following VCT level failures were previously analyzed by Engineering:

Failure of VCT Level Common Variable Leg

- Should this common tap become ruptured or isolated, VCT level will be erroneous. If ruptured or leaking, the **level indication for all channels will fail low**. Because the control level channels will agree, an Alternate Action from DCS will not occur.

Loss of ERPA **Correct Answer Part 2**

Auto and Manual Makeup Capability to the VCT is unavailable. If VCT level decreases below 23%, then, by procedure, the NV Pump suction will be manually aligned to the FWST and a downpower will be initiated.

The following valves fail closed if selected to auto:

- NV-181A (B/A Blender Otlt to VCT)
- NV-186A (B/A Blender Otlt to VCT Otlt)
- NV-242A (RMWST to B/A Blender Ctrl)
- NV-238A (B/A to Blendr Ctrl Vlv).

BAT Pumps A and B will not operate if in auto

Reactor Makeup Pumps A and B will not operate

Train A SMM Boron Dilution Interlock will be disabled with switch in "Enable"

Loss of ERPD

Reactor Makeup Pumps A and B will be disabled

Train B SMM Boron Dilution Interlocks will not function as designed.

Loss of CDB

- NV-172A (3-way Divert Vlv to VCT-RHT) will fail to the VCT position.

Loss of KXPA

Automatic VCT makeup capability is lost. (Manual makeup is still available)

Loss of start capability for the following pumps:

- Reactor Makeup Water Pump A
- Reactor Makeup Water Pump B.

Loss of KXPB

NV-849, NV-148, and NV-309 will fail open on a loss of KXPB. In order to control seal water flow, an operator will be dispatched to isolate NV-309 via upstream manual isolation valve NV-308 and manually throttle NV-311 (located in parallel with NV-309) to establish 32 gpm total seal water flow.

Question: 31
(1 point)

Given the following conditions on Unit 2:

- Unit is in Mode 4
- 2A ND train in service in RHR Mode
- 2B ND train remains in Injection Mode

Subsequently:

- Instrument Air is lost to 2ND-26 (ND Hx 2A Outlet Ctrl)

ND system flow _____(1)_____ automatically adjust to compensate for the change caused by this malfunction.

In accordance with OP/2/A/6200/004 (Residual Heat Removal System), 2B ND train _____(2)_____ be realigned to RHR mode.

Which ONE (1) of the following completes the statements above?

A. 1. will
2. can

B. 1. will NOT
2. can

C. 1. will
2. can NOT

D. 1. will NOT
2. can NOT

Original Question

MODIFIED

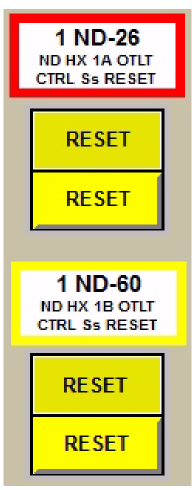
ND Heat Exchanger Outlet Control Valves, ND-26, -60

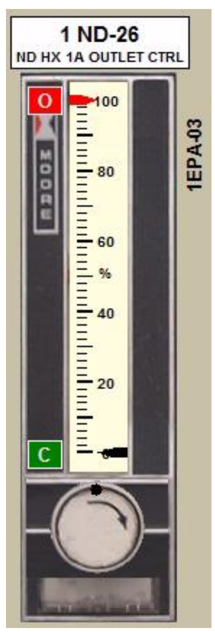
Valves ND-26 and ND-60 are air operated butterfly control valves located downstream of their respective ND Heat Exchanger. These valves may be manually controlled from the Control Room and are used to control the reactor coolant flow through the respective ND Heat Exchanger (ND-26 to A Hx, ND-60 to B Hx). These valves are used to establish the heat removal rate, thereby establishing the NC System cooldown rate. Generally, only one train is in service for cooldown at a time.

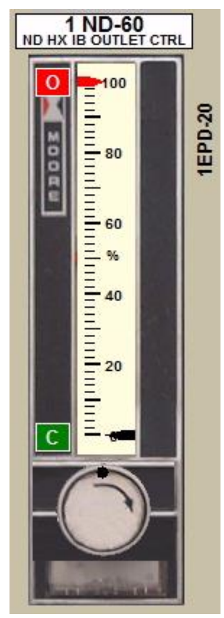
ND-26 and ND-60 automatically open on a Safety Injection signal (S_S) to ensure an ND flow path through the respective ND Heat Exchanger during ECCS operation. Each valve can be controlled manually from the Control Room via manual loaders on control board MC-11. Each valve can also be controlled locally via manual loader.

After receipt of a Safety Injection signal (S_S), the control room operator must depress the ECCS Reset Pushbutton for the appropriate train and push the associated valve Reset Pushbutton to regain control of the valve.

During normal plant operation, ND-26 and ND-60 are in an open position, so that the ND System is aligned for ECCS operation. During normal cooldown, the valves are in a throttled position. Upon loss of instrument air during normal cooldown, the valves again fail open to assure an ND flow path through the ND Heat Exchangers during residual heat removal operation. Should this occur, flow control capability may be re-established by throttling valves NI-173A and NI-178B.







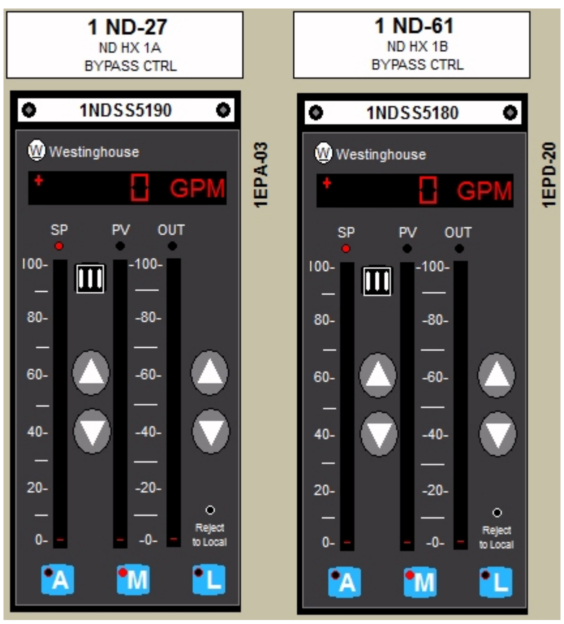
Question 30

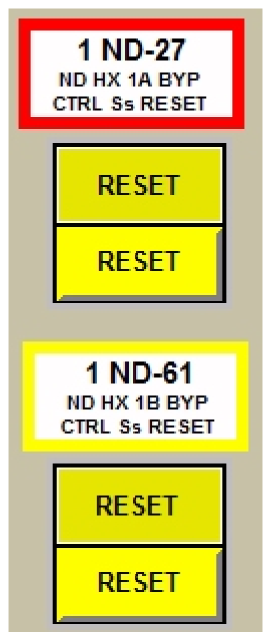
ND Heat Exchanger Bypass Control Valves, ND-27, -61

Valves ND-27 and ND-61 are air operated butterfly control valves located in the ND Heat Exchanger A (B) bypass line. During NC System heatup and cooldown, ND-27 (-61) automatically modulate to maintain the desired total ND Train A (B) flow rate (approximately 3300 gpm), based on the setpoint selected in the Control Room. ND-27 (-61) operates in conjunction with ND Heat Exchanger A (B) flow control valve ND-26 (-60) to establish the NC System cooldown rate by controlling the temperature of the reactor coolant returned to the NC System.

ND-27 (-61) fails closed upon receipt of an S_S signal to prevent bypassing of ND Heat Exchanger A (B) during a Design Basis Event. This instrumentation can be controlled via Manual/Auto (M/A) Stations on control board MC-11 and a soft M/A station on DCS. It can also be controlled following transfer to Local control via a local M/A Station. The Local controller, located outside the ND and NS Heat Exchanger Rooms, is automatically placed in service when the associated ASP is placed in "LOCAL".

If an alternate action occurs on the ND Heat Exchanger outlet flow, the ND-27 (ND-61), M/A Station will transfer to Manual and hold the “last good value.” An alternate action would occur if DCS senses that the ND Heat Exchanger outlet flow signal quality input is BAD.



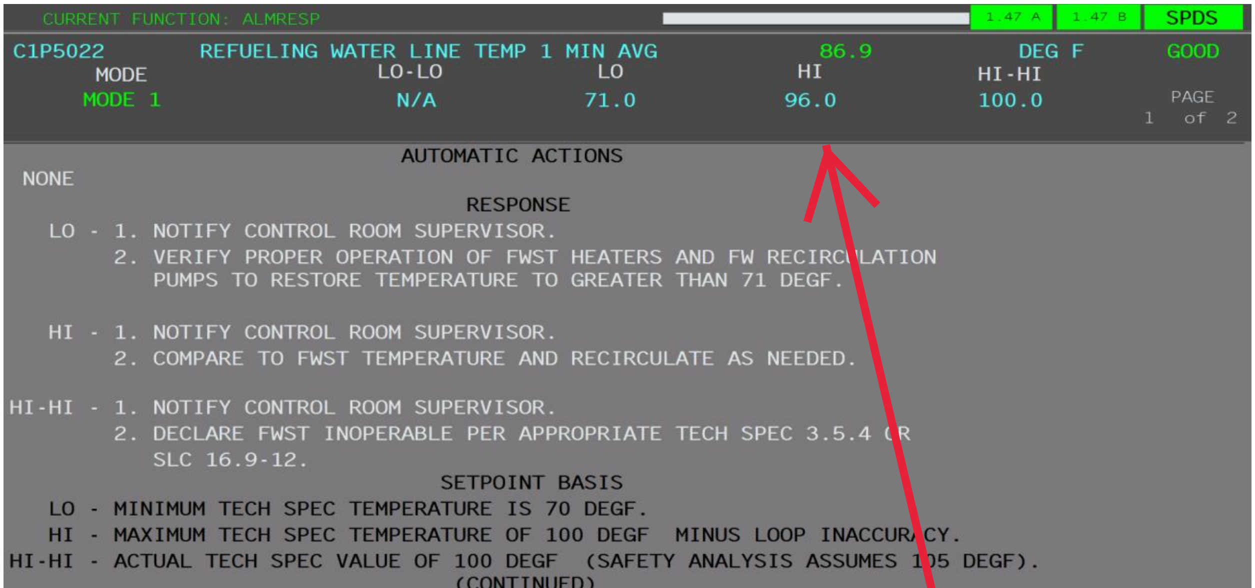


Question 31

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature is $\geq$ 70°F and $\leq$ 100°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is $\geq$ 377,537 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

Question 31


CURRENT FUNCTION: ALMRESP | 1.47 A | 1.47 B | SPDS

C1P5022	REFUELING WATER LINE TEMP 1 MIN AVG		86.9	DEG F	GOOD
MODE	LO-LO	LO	HI	HI-HI	PAGE
MODE 1	N/A	71.0	96.0	100.0	1 of 2

AUTOMATIC ACTIONS

NONE

RESPONSE

LO - 1. NOTIFY CONTROL ROOM SUPERVISOR.
2. VERIFY PROPER OPERATION OF FWST HEATERS AND FW RECIRCULATION PUMPS TO RESTORE TEMPERATURE TO GREATER THAN 71 DEGF.

HI - 1. NOTIFY CONTROL ROOM SUPERVISOR.
2. COMPARE TO FWST TEMPERATURE AND RECIRCULATE AS NEEDED.

HI-HI - 1. NOTIFY CONTROL ROOM SUPERVISOR.
2. DECLARE FWST INOPERABLE PER APPROPRIATE TECH SPEC 3.5.4 OR SLC 16.9-12.

SETPOINT BASIS

LO - MINIMUM TECH SPEC TEMPERATURE IS 70 DEGF.
HI - MAXIMUM TECH SPEC TEMPERATURE OF 100 DEGF MINUS LOOP INACCURACY.
HI-HI - ACTUAL TECH SPEC VALUE OF 100 DEGF (SAFETY ANALYSIS ASSUMES 105 DEGF).

(CONTINUED)

Distractor Part 1

Question 31

16.9 AUXILIARY SYSTEMS

16.9-12 Boration Systems Borated Water Sources - Operating

COMMITMENT The following borated water source(s) shall be FUNCTIONAL as required by SLC 16.9-8:

a. A Boric Acid Tank (BAT) with:

1) A minimum contained borated water volume as specified in the CORE OPERATING LIMITS REPORT (COLR),

2) A minimum boron concentration as specified in the COLR, and

Distractor Part 2

3) A minimum solution temperature of 65°F.

b. The Refueling Water Storage Tank (RWST) with:

1) A minimum contained borated water volume as specified in the COLR or Technical Specification Surveillance Requirement 3.5.4.2, whichever is larger,

2) A minimum boron concentration as specified in the COLR,

3) A minimum solution temperature of 70°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all Reactor Coolant System (RCS) cold leg temperatures > 210°F.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required BAT non-functional.	A.1 Restore the required BAT to FUNCTIONAL status.	72 hours

(continued)

Question 32

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be:

a. $\geq$ 75°F and $\leq$ 100°F for the containment upper compartment, and

Correct Answer b. $\geq$ 100°F and $\leq$ 120°F for the containment lower compartment.

--NOTE--
The minimum containment average air temperature in MODES 2, 3, and 4 may be reduced to 60°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limits.	A.1 Restore containment average air temperature to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Question 32

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

Distractor

LCO 3.6.4 Containment pressure shall be $\geq$ -0.1 psig and $\leq$ +0.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	In accordance with the Surveillance Frequency Control Program

Question 32

PANEL: 1AD-19

C/9

VQ CONTAINMENT PRESSURE ALERT

SETPOINT:
1. ≥ 0.25 psig   Distractor
2. ≤ -.06 psig

ORIGIN:
1. 1VQEM5040 (high pressure signal)
2. 1VQEM5041 (low pressure signal)

PROBABLE CAUSE:

During Modes 1-4:

High Pressure:
1. Plant heatup
2. Steam/air leaks

Low Pressure:
1. Plant cooldown
2. Excess containment cooling capacity in service.

During Modes 5 and 6:

High Pressure:
1. Improper balance of VP supply and exhaust flows.
2. VP has tripped or VP Exhaust Fans have tripped.

Low Pressure:
1. Improper balance of VP supply and exhaust flows.
2. VP Supply Fans have tripped.
3. VF Supply Fans have tripped.

AUTOMATIC ACTIONS: None

IMMEDIATE ACTIONS:

IF in Modes 1-4, perform the following:
1. Verify containment pressure high or low on 1VQP5040 on 1MC5.

IF in Mode 5 **OR** Mode 6 when movement of non-recently irradiated fuel or CORE ALTERATIONS **NOT** in progress, perform the following:

1. Verify containment pressure high or low on 1VQP5040 on 1MC5.
2. **IF** VP is in service, rebalance VP supply and exhaust flows to clear the annunciator per OP/1/A/6450/015 (Containment Purge System).
3. **IF** VP is **NOT** in service, initiate a VQ release or addition per OP/1/A/6450/017 (Containment Air Release and Addition System) as necessary.

CONTINUED ON THE NEXT PAGE

5.6 PRT Response to a Phase A Isolation Signal

Objective 6B

The following PRT- related valves will close on a Phase A (S_T) Isolation Signal:

- 1(2)NC-56B (RMW Pump Disch Cont Isol)
- 1(2)NC-53B (N2 To PRT Cont Isol)
- 1(2)NC-54A (N2 To PRT Cont Isol)

KC water to the NCDT heat exchanger will also be isolated on a Phase A. With both Reactor Makeup Water and KC to the NCDT Heat Exchanger, there is no available method to cool the PRT with the Phase A signal present.

6. LIMITS AND PRECAUTIONS

Objective 7

Reference the limit and precautions for the following procedure:

- OP/1(2)/A/6150/004, Pressurizer Relief Tank

7. OPERATING EXPERIENCE

7.1 Committed OE

None.

7.2 Non-Committed OE

7.2.1 Unexpected PRT Level Response During NCS Venting (AR 01497768)

While performing Unit 2 Reactor Coolant System Venting per OP/2/A/6150/001 Enclosure 4.2, PRT level indication did not increase as expected after 2NC-251B and 2NC-253A were opened. All 4 of the Reactor Head Vent and Reactor Head Vent Block valves were previously opened per the procedure to vent the reactor vessel head. When 2NC-251B and 2NC-253A were opened, PRT Level was monitored by OAC point C2A0879 and level increased as expected. When PRT level increased by at least 1%, only 2NC-251B and 2NC-253A were closed to suspend venting. After ~ 2-3 minutes, 2NC-251B and 2NC-253A were reopened to resume venting the reactor head, PRT level did not increase as expected. Both OAC and control board indications showed the valves to be open. The CRS, OSM, and Primary Group were contacted and a decision was made to cycle Kerotest valves 2NC-251B and 2NC-253A to ensure the disc was separated from the body. Once the affected valves were cycled, PRT level began to increase as expected.

Question 34

The component cooling surge tanks have sufficient capacity to accommodate thermal transients and leakage into or out of the KC System. The vent lines on the surge tanks are maintained open during all operational modes and are sized large enough to prevent excessive vacuum in the tanks should cold water from the KC drain sump be added to the surge tank at its maximum temperature. In addition, this vent line is also sufficient to prevent over pressurization of the tank in the event of a Thermal Barrier Tube rupture, evaluated to be approximately 260 gpm. Surge tank overflow is directed to the component cooling drain sumps via loop seal.

Each units two surge tanks are connected by a separate 8 inch overflow line at approximately the 97% level that allows one surge tank to overflow to the tank on the opposite train. This 8 inch line will equalize variations in surge tank level.

Distractor Part 2

An assured makeup source to KC is provided by the RN system, which can be aligned to the KC pumps suction header via manual valves. Normal makeup is provided by YM demineralized water and is not assured.

Correct Answer Part 2

The Control Room Operators will be alerted of leakage from the system via a KC Surge Tank Lo Level annunciator at 37.3%.

Each surge tank has safety related level instrumentation that automatically isolates the KC non-essential Auxiliary and Reactor Building header isolation valves (train related) upon Lo-Lo level at 34%; achieving train separation. This assures that at least one train of KC will have sufficient NPSH if an out leakage develops when both essential and non-essential trains are aligned together. The Lo-Lo level at 34% uses a 1/1 instrument per tank to cause the train isolations. A separate instrument is used for indication and alarms.

KC Pumps

Objective 4a and 5, All

Refer to figure 4 for KC Pump Layout

Four Component Cooling pumps are provided per unit with each train pair powered from a separate assured power source. Each train pair has 100% flow capacity (50% per pump) and can provide the minimum requirements of both the Essential and Non-essential Header loads.

The KC Pumps are horizontal shaft, centrifugal pumps equipped with mechanical seals to minimize leakage. They can be operated from the Control Room or ASP by two position START/STOP pushbuttons. Normal KC pump discharge pressure is 100 psig. The KC pumps are located in the Auxiliary Building on the 560" level (Unit 1 pumps) and the 577' level (Unit 2 pumps)

Flow through the KC system will be dependent on the components in service. The KC pumps minimum flow rate is 1100 gpm and the maximum is 5700 gpm per pump (manufacturer's recommended minimum flow and pump run-out conditions).

will also open when Unit 1(2) ASP A and B are transferred to “LOCAL” and the train related pump is running.

Reactor Building Non-Essential Header cools the following:

- NC Pumps
- NCDT
- Excess Letdown Hx's

Component Cooling cools the NC pump Thermal Barrier Hx, and the upper and lower bearing oil coolers of the NC Pumps. The NC Pump thermal barrier and oil cooler flows for both units can be read on gauges on the 543 elevation of Aux building. **Correct Answer Part 1**

The Thermal Barrier Hx has an inlet check valve and the outlet valve that can auto close in the event there are indications of a thermal barrier rupture. All the piping between the check valve inlet and outlet valve and the valves themselves are rated for NCS pressure and temp with a relief set at 2485 psig. Output from the relief valve is directed to the Containment Floor and Equipment Sump. The outlet valve will auto close @ 60 gpm after 30 seconds. (The 30 sec. time delay prevents the valve from closing on surge, during a Pump Start).

This arrangement of an inlet check valve and auto closed outlet valve should isolate any Thermal Barrier HX leak. Flow to the Thermal Barrier is manually throttled to 40 gpm/pump, with a high flow alarm at 60 gpm and a low flow alarm at 35 gpm. Westinghouse performed specific analysis in 2009 to determine the bounding leak rate for the thermal barrier heat exchangers used at Catawba and McGuire. The analysis determined that the bounding leak rate is 10.55 gpm (AR 01540034 Corrective Action #1).

The NC pump upper and lower bearing oil coolers are also cooled by Component cooling. The upper Bearing flow is controlled at 165 gpm/pump by a flow controller located in Aux. building on 543' elevation. There is a high flow alarm at 200 gpm and a low flow alarm at 140 gpm on the pump upper bearing Component Cooling flow.

The NC pump lower bearing oil cooler KC flow is manually adjusted to 6 gpm/pump. There is a low flow alarm at 5 gpm.

The KC Supply Header Flow to NCP's has a Low alarm @ 425 GPM. The Annunciator alarm on AD20/21 requires verification of flowpath and monitoring the motor bearing temperatures. Refer to AP/021 (Loss of KC) if alarm is due to a loss of flow.

On a Unit 1(2) Sp signal the NC Pump Containment Supply and Return isolation valves (1(2)KC-338B, 1(2)KC-424B, and 1(2)KC-425A) Isolations will close.

Question 35

Backup Heaters

Objective 2F

The backup heaters are made up of three groups labeled A, B, and D. The A and B Groups each contain 6 banks of heaters worth 416 kW. The D Group contains 7 banks of heaters worth 484 KW. One of the D Group heater banks contains 3 heaters and is powered from SMXG. This Shared Motor Control Center SMXG is normally powered from SLXG, but can also be powered from the SSF D/G.

Correct Answer

These Group D heaters #28, 55, and 56, can be controlled at the SSF. They are 70kW and must be energized within 15 hours of a reactor trip. The heaters are required for SSF functionality which is to maintain NC System pressure control and to ensure that any steam bubble in the pressurizer does not migrate to the reactor vessel.

Manual Heater Control

The backup heaters are either on or off - they cannot have their output varied like the C heaters. They can be manually energized but, after a Blackout condition, that is delayed 12 minutes. Groups A, B, D can be controlled from MCB and Groups A & B can be controlled from the Auxiliary Shutdown Panels (ASP). The one bank in Group D that is powered from the SSF can be controlled there too.

Distractor Part 1

Auto Heater Control

Objective 3A

A low pressure deviation of 25 psi below setpoint (normally 2210 psig) will automatically turn on the heaters.

Upon low-low pressurizer level of 17% the heaters will turn off. This condition blocks control of the backup heaters from the MCB but Groups A & B can still be controlled from the ASP. When the low-low PZR level condition clears the heaters will auto re-energize if a low pressure demand signal is still present.

Pressurizer Operability

Operability of the PZR is predicated on two conditions.

1. The water level must be less than or equal to 92%. This ensures a bubble exists which preserves the steam space needed for adequate pressure control.

Objective 2F, 5

2. Two Groups of Backup heaters, A and B; each with a capacity greater than or equal to 150 KW and capable of being powered from an Emergency Power Supply. Only these two Groups can be powered from an Emergency Power Supply, as they are supplied by 600 VAC Blackout load centers LXI and LXH that are powered from the Blackout 4KV buses FTA and FTB. The 4KV Blackout buses can be powered from ETA and ETB during a Station Blackout.

The failed channel pressure signal will be removed by the median select (MSS) circuitry in the DCS

The IPE DCS circuit will actuate a DCS Trouble Alarm on 1AD-2, F/10

- For a single channel failing low the following annunciators will be received:

  1AD-6, D/8, PZR LO PRESS ALERT

  1AD-6, E/8, PZR LO PRESS SI ALERT

- For a single channel failing high, the annunciator 1AD-6, A/8, PZR HI PRESS ALERT will be received.

- Reactor Trip

  When a reactor trip occurs Tavg is reduced very quickly as the rods fall into the core.

  This decrease in T_{AVG} causes PZR level to decrease due to NCS density increase. As a result of the PZR level decrease, the steam-space to water-space ratio has greatly increased in the PZR which causes PZR pressure to decrease.

  The PZR pressure control system should respond as required to bring PZR pressure back to normal operating pressure over a period of time by energizing PZR heaters.

- Reactor Coolant System Heatup and Pressurization

  During plant startup from Mode 5 to Mode 3 the pressurizer pressure control system is used to raise the NC system pressure to normal operating pressure. First, the PZR heaters are energized. After placing the heaters in service a constant spray flow is established via normal spray from the NC system or alternate spray from either the ND system or the NV system. By establishing spray flow a constant outflow of water is maintained through the PZR surge line.

  The purpose of the constant outflow is to prevent an insurge of water affecting the surge line temperature thus affecting the indicated PZR heatup rate. When NC system pressure reaches 1700 psig the PZR Pressure Master Controller can now be utilized to control system pressure. The PZR Pressure Master may be operated manually with the raise and lower pushbuttons or automatically by adjusting the setpoint. This control of the heaters and sprays allows control of the rate of change of pressurization of the NCS. When normal system pressure of 2235 psig is reached the PZR Pressure Master Controller setpoint is verified correct and the PZR Pressure Master controller is placed in automatic.

6.4 Alternate Actions

Objective 9

Selected Pressurizer Pressure-1 (SPP-1)

- Pressurizer Pressure Master Controller goes to MANUAL with Last Good Value ('C' heater operation and spray valve position do not change)

Question 37

Enclosure 13.4

Control Bank Withdrawal

NOTE: This enclosure is performed by the SRO and RO dedicated to reactor startup.

Initials	Printed Name

NOTE: Step 13.4.1 may be performed out of sequence.

13.4.1 **WHEN** "P-6 S/R BLOCK PERMISSIVE" lamp on 1(2)SI-18 is lit, perform the following:

_____ 13.4.1.1 Block Source Range high flux level trip as follows:

- ☐ "TRN A S/R SELECT" switch to "BLOCK". Correct Answer Part 1
- ☐ "TRN B S/R SELECT" switch to "BLOCK".

_____ 13.4.1.2 Verify following permissive lamps on 1(2)SI-18 are lit:

- ☐ "S/R TRAIN A TRIP BLOCKED"
- ☐ "S/R TRAIN B TRIP BLOCKED"

_____ 13.4.2 Review estimated critical position with Reactor Engineer.

_____ 13.4.3 Verify lowest NC Loop T-Avg is ≥ 551 °F. {S.R. 3.4.2.1}

_____ 13.4.4 Withdraw Control Banks until: (R.M.)

☐ Source Range count rate doubles ____________ N31 ____________ N32

OR

☐ BDMS count rate doubles ____________ BDMS A ____________ BDMS B

OR

☐ Control Bank B reaches fully withdrawn position

OR

☐ Stable startup rate reaches 0.5 DPM.

_____ 13.4.5 Review projected critical position with Reactor Engineer.

Reactor Trip Interlocks

Several reactor trips are only active above or below specific power levels. The following table lists the interlocks that are associated with the Reactor Trips.

INTERLOCK	SETPOINT	FUNCTIONS
P-6	1/2 IR $\geq$ ~10^{-5}% power	• Allows manual block of the Source Range Neutron Flux Rx Trip above P-6 • Automatically enables Source Range Neutron Flux trip below P-6
P-7	P-10 or P-13	Automatically enables the 5 “At Power” Trips above P-7: • NCP Undervoltage • NCP Underfrequency • NCP 2 Loop Loss of Flow • Pressurizer Pressure - Low • Pressurizer Level - High Automatically blocks these trips below P-7.
P-8	2/4 Power Range detectors $\geq$ 48% power	• Automatically enables the Single Loop Reactor Coolant Flow-Low reactor trip above P-8. • Automatically blocks this trip below P-8.
P-9	2/4 Power Range detectors $\geq$ 69% power	• Automatically enables the reactor trip on turbine trip above P-9. • Automatically blocks this trip below P-9.
P-10	2/4 Power Range detectors $\geq$ 10% power	• Input to P-7 (see above) • Automatically provides a backup signal to block the SR Neutron Flux trip above P-10 • Allows manual block of the following above P-10: ○ C-1 rod stop ○ Intermediate Range Neutron Flux Rx Trip ○ Power Range Neutron Flux Low Rx Trip • Automatically reinstates the C-1 rod stop, Intermediate Range Rx Trip, and Power Range Low Rx Trip below P-10.
P-13	1/2 Turbine Impulse Pressure $\geq$10% power	• Input to P-7

9. NORMAL SYSTEM OPERATION

9.1 Normal Operation

Objective 11, ALL

The normal operation mode of System VA shall have the supply air system in operation with the Containment Chilled Water (YV) System and Plant Heating Water (YH) available as required to temper supply air to achieve the desired area temperature. Both units will normally have both Train A and B Filtered Exhaust, Unfiltered Exhaust and Supply in operation.

- All 4 Filtered Exhaust Fans running in bypass mode.
- All 4 Unfiltered Exhaust Fans running
- All 4 Supply Fans running
- Counting Room Supply and Filter Unit running controlled independently
- 4 Radwaste Area AHUs running
- Filter and Demineralizer Room Exhaust Fans running - receive permissive from Auxiliary Bldg Filtered Exhaust Fans.
- Auxiliary Shutdown Panel Air Conditioning Unit's compressor starts if room temp is greater than 68°F.
- Evaporator Room Supply Units placed in "COOL".
- Restricted Instrument Shop Supply Unit runs as necessary.
- PD Pump Room Supply Unit placed in "STOP".
- Doghouse Vent Fans controlled by thermostat in "AUTO".
- UHI Building Vent Fan controlled by room thermostat in "START".
- Fume Hood Exhaust Fans in the Sample Room (3) - permissive from unfiltered exhaust fans (Unit 1 or 2).

10. ABNORMAL OPERATIONS

10.1 Safety Injection Operation

Objective 12A, ALL

Upon receipt of a Safety Injection signal to initiate LOCA operation the following train-related actions take place in the VA System (i.e. a Unit 1 Train A Safety Injection will perform the below actions for Unit 1 A Train and Unit 2 A Train Components.):

- Filtered Exhaust Fans continue operating or start if they are shut down
- Filter Units operate in the filtered mode of operation

- Inlet vanes (vortex dampers) to the Filtered Exhaust Fans will go to a throttled position reducing the filtered exhaust design flow from approximately 30,000 to approximately 6,540 cfm per train.
- Non- safety portion of the Filtered Exhaust are isolated by electro hydraulic isolation dampers that fail to a closed position
- Unfiltered Exhaust Fans trips which cause the associated Supply Fans to trip

The following actions also occur on a Safety Injection signal, but these signals are unit-related:

- Unit and train related ASP supply units are tripped off and then restarted.
- Trips Unit Related Filter Room Exhaust Fans.

The VA System will continue to operate in this mode until the safety injection signal is reset.

The VA System will not go to the LOCA mode of operation if tornado isolation is activated.

The following two options can be used to establish Auxiliary Building ventilation following an S_S signal: Return to Normal Alignment or Post LOCA Operation.

Return to Normal Alignment

The return to normal alignment is directed by EP/ES-1.1 (Safety Injection Termination) and is accomplished by using an enclosure from OP/0/A/6450/003 (Auxiliary Building Ventilation). This process involves a damper reset key switch and two resets for the ASP supply units per train. After this is done, VA may be returned to normal alignment.

Post LOCA Operation

Objective 12B, All

Post LOCA operation is not directed by any EP or AP. This mode should only be used with TSC concurrence (may be used if dose levels in the Auxiliary Building preclude the use of return to normal alignment). Procedure guidance is provided using an enclosure from OP/0/A/6450/003 (Auxiliary Building Ventilation). This process involves moving a connector from one receptacle to another within LOCA Control Panels A and B and placing a selector switch to “ON”.

Activation of the Post LOCA mode of operation will cause the following responses in the VA System:

- Allow the Auxiliary Building unfiltered and supply fans to operate
- Isolation dampers for the non-safety filtered exhaust ducting to the Auxiliary Building will open allowing air to be exhausted from these areas.

All Post LOCA Controls will be disabled upon receipt of a LOCA signal. Additionally, the Post LOCA Controls do not bypass any VA System interlocks (i.e. fan interlocks, smoke detectors, EMF interlocks)

4. SYSTEM INTERRELATIONSHIPS

4.1 Containment Chilled Water System (YV)

Objective 7A, All

YV provides normal cooling to VV components per OP/1/A/6450/020, Containment Chilled Water System. The YV System containment chillers are then in turn cooled by the Low Pressure Service Water System (RL). The YV system is covered in the RN Lesson Plan.

4.1.1 YV design flow to each VV system AHU cooler follows:

- LCVUs – 825 gpm
- UCVUs – 18 gpm
- IIRVU – 10 gpm

4.2 Nuclear Service Water (RN)

Objective 7B, All

RN may be automatically or manually aligned to VV when YV is unavailable. YV will not be available to provide cooling to the UCVUs during a loss of offsite power.

5. NORMAL SYSTEM OPERATION

5.1 Normal VV Cooling Alignment Modes 1 – 4

Objective 8A, All

The normal VV component configuration alignment in Modes 1-4 per OP/1(2)/A/6450/001(Containment Ventilation Systems) is as follows:

- Three (3) LCVUs in “LOW” speed and in MAX COOL
- One (1) PTBF in LOW speed
- Three (3) CRDM vent fans ON
- One (1) IIRVU in NORM
- One (1) or two (2) UCVUs in NORM and associated RAFs in AUTO

15.3 Safety Injection Signal

Objective 16C, All

Upon receipt of a safety injection signal on either unit , all four RN pumps will start and their associated discharge isolation valves and pump motor cooler isolation valves will open. The Unit related KC HXs Control Valves fail open to assure sufficient heat transfer will be available through the KC HX when it is needed during a design basis event. The RN Chemical Addition Dilution Water Crossover Isolation Valves close for train separation. The Auxiliary Building Vent Unit Supply and Return Header Isolation valves close to conserve flow for containment cooling or essential heat loads..

The D/Gs start on a Safety Injection, and the D/G Engine Jacket Water Cooler RN Supply Isolation Valves are interlocked to open whenever a D/G starts and close if the D/G stops (or fails to start)

15.4 Phase B Signal

Objective 16D, All

The following automatic actions occur upon receipt of a Phase B signal. Item 1 occurs upon a Phase B signal from either unit. Items 2-6 occur on the unit which generated the Phase B signal.

1. SNSWP Return Isolation Valves 1RN-58B and 1RN-63A open. These valves open on an SP signal to provide an assured RN discharge flow path if the normal RN discharge flow path to RL is unavailable due to a failed isolation valve.
2. RN Supply Crossover Isolation Valves 1(2)RN-47A and 1(2)RN-48B close. The automatic closure for the unit specific valve isolates RN flow to nonessential components to conserve flow for essential components on that unit. However, if the RN system is aligned in Single Supply Header Operation, 1(2)RN-47A and 1(2)RN-48B remain open and the RN trains remain cross-connected. Therefore, RN trains A and B do not isolate and remain cross-connected. This ensures that RN cooling water flow is available to all four essential headers while the RN system is aligned in Single Supply Header Operation.
3. RN Nonessential Supply Header Isolation Valves 1(2)RN-49A and 1(2)RN-50B close. The automatic closure on the SP signal separates that unit's RN nonessential header from the RN essential headers, and conserves flow for safety related components on the RN essential headers.
4. RN Nonessential Return Header Isolation Valves 1(2)RN-51A and 1(2)RN-52B close to separate that unit's nonessential header from the essential headers, and to conserve flow for safety related components on the essential headers.
5. Upper Containment Isolation Valve 1(2)RN-404B close. This valve is normally open to provide cooling water to containment HVAC units during normal operation. If containment HVAC is lost during normal operation, the resulting increase in containment pressure could generate an "artificial" ESF actuation.

Consequently, valve 1(2)RN-404B is required to remain open until a phase B isolation signal is initiated.

6. Lower Containment Isolation Valves 1(2)RN-437B, 1(2)RN-484A, 1(2)RN-487B close. These valves are normally open to provide cooling water to containment HVAC units during normal operation. If containment HVAC is lost during normal operation, the resulting increase in containment pressure could generate an “artificial” ESF actuation. Additionally, these valves are open to allow cooling water flow to the NC Pump Motor Air Coolers. Without the cooling water flow, motor stator temperatures would increase, possibly resulting in the loss of the NC pumps. Consequently, these valves are required to remain open until a phase B isolation signal is initiated.

15.5 Transfer to ASP

Objective 16E, All

RN is required to operate upon transfer of control to the Auxiliary Shutdown Panels. In order to provide the most assured means of shutdown, automatic actions that occur upon the transfer of controls for RN includes:

- RN Suction(s) and discharge(s) align to SNSWP (Unit 1 ASP's ONLY)
- A/B train discharges split (Unit 1 ASP's ONLY)
- Diesel discharge(s) swap to SNSWP (unit/train related, panel specific valve)
- Essential header supply valves(s) open (unit/train related, panel specific valve)
- Full RN flow is aligned through KC Hx's (unit/train related, panel specific valve)

15.6 SNSWP Ice Melt

Objective 16F, All

Ice formation in the source and intake section is felt to be impossible because the intake structures are well below the surface of Lake Wylie and the SNSWP. In cold weather the RN system suction and discharge can be aligned to the SNSWP to prevent severe ice accumulation on the surface of the pond. Any WL discharge in progress from the Waste Monitoring Tank Building (1WL-124) must be secured prior to making this alignment.

This evolution is accomplished by performing the following:

- Swapping RN returns to the SNSWP
    - Open the RN Header A and B return valves to the SNSWP (1RN-63A &58B)
    - Open the DG 1A, 1B, 2A, 2B HX returns to the SNSWP valves (1(2)RN-846A & 848B
    - Close the DG 1A.1B, 2A, 2B HX returns to Lake Wylie valves. (1(2)RN-847A & 849B)

Glycol Containment Isolation Valves

NF supply containment isolation **NF-228A** is located outside containment. NF-228A is an air operated valve and is controlled from the control room.

NF-228A automatically closes on the following:

- Loss of Instrument Air
- Loss of Power
- Low-Low Expansion Tank Level (A key switch is provided on the local NF control panel to allow the Low-Low Expansion Tank Level interlock to be bypassed.)
- Phase A signal, S_T

NF Return Containment Isolation Valve **NF-233B** is located inside containment. NF-233B is a motor operated valve that automatically closes on a Phase A Containment Isolation (St) signal. A bypass allows glycol that is trapped in the penetration to bleed back into containment in the event of containment isolation.

The outside NF Return Containment Isolation Valve, **NF-234A**, is an air operated valve that automatically closes on the following (same as NF-228A):

- Loss of Instrument Air
- Loss of Power
- Low-Low Expansion Tank Level (A key switch is provided on the local NF control panel to allow the Low-Low Expansion Tank Level interlock to be bypassed.)
- Phase A signal, S_T

Both NF-234A and NF-233B are controlled from the control room as described above.

Some evolutions may require securing glycol flow to containment for a short period. If the NF containment isolation valves are closed and/or the penetration is isolated, action must be taken to:

- Minimize ice bed heatup and/or
- Provide over pressure protection for the isolated penetration

5. AUTOMATIC ACTIONS SUMMARY

Objective 8, Licensed

Containment Isolation Automatic Closures

The following table summarizes the three containment isolation valve actions on the plant conditions shown.

Valves CLOSE on:	Loss of VI	Loss of Power	Low-Low Expansion Tank Level	St Signal
NF-228A Containment Supply Isol	X	X	X	X
NF-233B Containment Return Isol.				X
NF-234A Containment Return Isol.	X	X	X	X

NF Ventilation Chiller Trips

- Low Glycol Flow $\leq$ 2.5 psid
- Low KR Flow
- Low Compressor Oil level $\leq$15 psid
- Low Suction Press
- High Compressor Discharge Pressure

Catawba Nuclear Station

2019 CNS RO NRC Examination

Question: 41
(1 point)

Given the following conditions on Unit 1:

- Unit is at 100% RTP
- It has been determined that eight Ice Condenser Intermediate Deck doors will not open due to excessive ice buildup

Based on the conditions listed above, peak pressure following a Design Basis Accident will be reached _____(1)_____ than normal.

Containment design pressure is _____(2)_____.

Which ONE (1) of the following completes the statements above?

A. 1. sooner
2. 20 psig

B. 1. later
2. 20 psig

C. 1. sooner
2. 15 psig

D. 1. later
2. 15 psig

Original Question

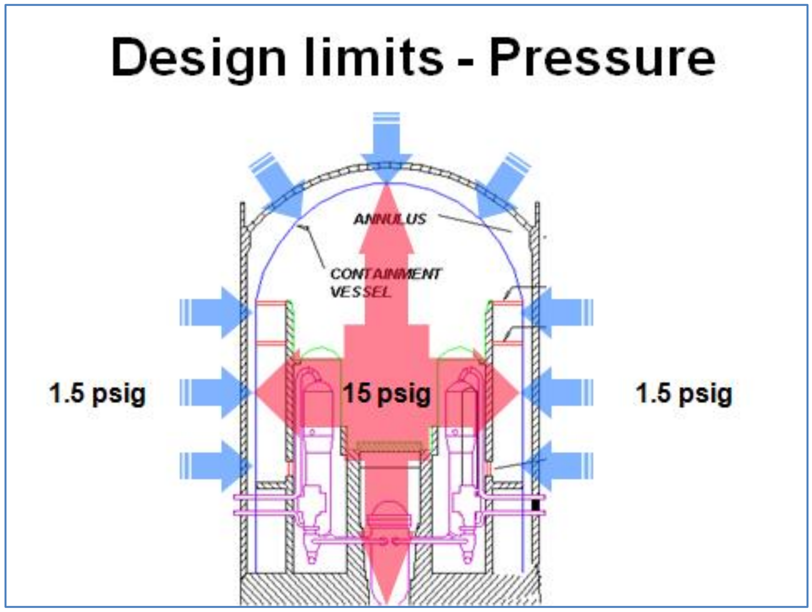
MODIFIED

2.8 Containment Pressure Requirements

Objective 2H and 2I, All

Administratively, containment pressure is maintained between **-0.08 to +0.25 psig**. Primary containment pressure is maintained for operability purposes **≥ -0.1 psig and ≤ +0.3 psig.** Maintaining the primary containment pressure in this range ensures that containment design pressure of **15.0 psig** will not be exceeded under DBA conditions. The worst-case loss of coolant accident (LOCA) generates a larger mass and energy release than the worst-case steam line break (SLB). Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint.

Figure 4: Containment Design Pressure Limits



The maximum negative external pressure, by design, is -1.5 psig. The following five conditions have a potential for resulting in a negative external pressure on the containment:

- Rupture of a hot or high-pressure process pipe in the annulus. (Pressure build up in smaller annulus volume would be at a faster rate than in containment, thus creating a differential pressure across the wall)
- Inadvertent Containment Spray System initiation during normal operation.
- Inadvertent containment air return fan initiation during normal operation.
- Containment purge fan operation with containment purge inlet valves closed.
- Containment air release fan pressure controller failure resulting in fan not shutting off properly.

The containment design of -1.5 psig (negative pressure) is not exceeded in the first four conditions due to either equipment limitations or design features, but may be exceeded in the fifth case. The containment air release fan can pull a negative pressure in containment beyond design limits if allowed to run unchecked. Administrative controls prevent this event from occurring. The operator aid computer (OAC) response to a low-

Question 41

3.6 CONTAINMENT SYSTEMS

3.6.13 Ice Condenser Doors

LCO 3.6.13 The ice condenser lower inlet doors, intermediate deck doors, and top deck doors shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTES---

1. Separate Condition entry is allowed for each ice condenser door.

2. Entry into Condition B is not required due to personnel standing on or opening an intermediate deck or top deck door for short durations to perform required surveillances, minor maintenance such as ice removal, or routine tasks such as system walkdowns.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ice condenser lower inlet doors inoperable due to being physically restrained from opening.	A.1 Restore lower inlet door to OPERABLE status.	1 hour
B. One or more ice condenser doors inoperable for reasons other than Condition A or not closed.	B.1 Verify maximum ice bed temperature is $\leq$ 27°F.	Once per 4 hours
	<u>AND</u>	
	B.2 Restore ice condenser door to OPERABLE status and closed position.	14 days

(continued)

Question 42

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. **Verify at least one of the following annunciators - LIT:**	**Perform the following:**
__ • 1AD-20, B/3 "CONT. SUMP LEVEL >3.3 ft" OR __ • 1AD-21, B/3 "CONT. SUMP LEVEL >3.3 ft".	__ a. **<u>WHEN</u>** at least one "CONT. SUMP LEVEL >3.3 ft" annunciator - LIT, **<u>THEN</u> <u>GO</u> <u>TO</u>** Step 5. __ b. Do not continue in this enclosure until at least one annunciator - LIT.
5. **Align NS train 1A to containment sump as follows:**	
__ a. Verify NS pump 1A - AVAILABLE TO RUN.	__ a. **<u>GO</u> <u>TO</u>** Step 6.
__ b. Verify 1NI-185A (ND Pump 1A Cont Sump Suct) - OPEN.	__ b. **<u>GO</u> <u>TO</u>** Step 6.
__ c. Verify NS pump 1B - OFF. **Enclosure 2 will provide direction to start only one NS pump. If the other pump is in service, guidance will be provided to bypass pump start steps.**	c. **<u>IF</u>** NS pump 1B running **<u>AND</u>** RN established to NS Hx 1B, **<u>THEN</u>** perform the following: __ 1) Ensure 1NS-20A (NS Pump 1A Suct From FWST) - CLOSED. __ 2) Ensure 1NS-18A (NS Pmp A Suct From Cont Sump) - OPEN. __ 3) **<u>GO</u> <u>TO</u>** Step 7.
__ d. OPEN 1NS-29A (NS Spray Hdr 1A Cont Isol).	__ d. **<u>GO</u> <u>TO</u>** Step 6.
__ e. OPEN 1NS-32A (NS Spray Hdr 1A Cont Isol).	__ e. **<u>GO</u> <u>TO</u>** Step 6.

Question 42

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
k. Isolate FWST from NV and NI pumps as follows:	
__ 1) Place "PWR DISCON FOR 1NI-100B" switch in "ENABLE".	
__ 2) CLOSE 1NI-100B (NI Pmps Suct From FWST).	
3) CLOSE the following valves:	
__ • 1NV-252A (NV Pumps Suct From FWST)	
__ • 1NV-253B (NV Pumps Suct From FWST).	
	NOTE An invalid SPDS orange path may briefly exist between opening NS suction valve from sump and starting NS pump. FR-Z.1 should not be entered unless NS pump fails to start.
__ 7. **Verify Enclosure 2 (Aligning NS for Recirculation) - PREVIOUSLY COMPLETED.**	__ **Align NS for recirc. <u>REFER TO</u> Enclosure 2 (Aligning NS for Recirculation).**

Question 43

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
__ 3) CLOSE 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol).	3) Perform the following: __ a) CLOSE 1CA-36 (CA Pump #1 Flow To S/G 1D). __ b) Dispatch operator to close 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol) (DH-584, DD-EE, 43-44, Rm 591).
__ 10. **<u>WHEN</u> NC T-Hots start to increase, <u>THEN</u> dump steam from intact S/G PORVs to stabilize NC T-Hots.**	
11. **Verify conditions allowing alignment of one ND train for Aux Containment Spray as follows:**	
__ a. At least one ND train - ALIGNED AND OPERATING IN COLD LEG RECIRC MODE.	a. Perform the following: __ 1) **<u>WHEN</u>** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation) completed, **<u>THEN</u>** perform Steps 11 and 12. __ 2) **<u>GO TO</u>** Step 13.
__ b. Containment pressure - GREATER THAN 15 PSIG.	__ b. **<u>GO TO</u>** Step 13.

Question 43

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
__ c. Elapsed time since Reactor Trip - GREATER THAN 50 MIN.	c. Perform the following: __ 1) Designate someone to notify Control Room Supervisor when 50 min from Reactor Trip has elapsed. __ 2) **WHEN** time since Reactor Trip greater than 50 min, **THEN** perform Steps 11 and 12. __ 3) **GO TO** Step 13.
d. Verify the following valves - CLOSED: __ • 1NS-43A (ND Pmp 1A To Cont Spray Hdr) __ • 1NS-38B (ND Pmp 1B To Cont Spray Hdr).	__ d. **IF** ND Aux Containment Spray aligned to operating ND train with RN and KC cooling, **THEN GO TO** Step 12.

Question 43

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

24. (Continued)

d. **IF** NI Pump 1B in service, **THEN** ensure the following valves - OPEN:

__ • 1NI-135B (NI Pump 1B Suct)

__ • 1NI-150B (NI Pump 1B C-Leg Inj Isol)

__ • 1NI-162A (NI To C-Legs Inj Hdr Isol)

__ • 1NI-100B (NI Pmps Suct From FWST).

e. **IF** both of the following conditions exist, **THEN** **GO TO** Step 25:

__ • Any NI pump in service **AND** S/I flowpath established

AND

Distractor D

__ • Time between reactor trip and implementation of this procedure - GREATER THAN 90 MINUTES.

(RNO continued on next page)

Question: 45
(1 point)

Given the following Unit 1 initial conditions:

- The Unit is at 25% RTP following a refueling outage.
- AP/1/A/5500/028 (Secondary Steam Leak) has been entered following a discovery of a leak on the Unit 1 Main Turbine Crossover line

Subsequently:

- The Unit 1 Main Turbine is tripped to isolate the leak
- Reactor power is currently 11%

At this time, ______(1)______ steam dumps are operating to control NC temperature at ______(2)______ degrees F.

Which ONE of the following completes the statement above?

A. 1. ONLY condenser
 2. 557

B. 1. ONLY condenser
 2. 560

C. 1. condenser AND atmospheric
 2. 557

D. 1. condenser AND atmospheric
 2. 560

Original Question

MODIFIED

3.1.2 Atmospheric Dumps

The Atmospheric Dump valves are made of two banks numbered 4 and 5, with four valves in Bank 4 and five valves in Bank 5. The total capacity of these two banks is ~35% RTP. The atmospheric valves do not start opening until the condenser dump valves are full open, this requires a minimum opening signal of 49%. The atmospheric dump valves require signal strength of 100% to be full open. Actuation of the atmospheric dump valves requires a combination of several arming signals. **Figure 3**

Arming Signals

Objective 4A

All three of the following are needed in order to open the Atmospheric Steam Dump valves:

- T-AVG Mode selected
- No Train ‘A’ P-4 contact
- C-7B Loss of Load Interlock Atmospheric Dump

3.2 Load Rejection Controller

Objective 3B

The Load Rejection Controller is used during a load rejection event to prevent a large T-AVG increase on a loss of load. The controller is enabled by the Steam Dump Select switch being in the “T-AVG” position and no reactor trip has occurred as sensed by the P-4 Train ‘B’ contact. **Figure 4**

The controller compares Selected T-AVG-2 to Tref and sends a control signal to modulate all banks as necessary. A lead/lag circuit conditions the Selected T-AVG signal. This is to make the steam dumps respond in an anticipatory manner based on the rate of change in the signal. This controller is capable of developing a 100% valve demand signal and therefore all steam dump valves, condenser and atmospheric, could be opened fully. A 3°F deadband exists on the controller to allow rod control insertion to aid in decreasing T-AVG. The controller sends a signal to modulate condenser dump valves open sequentially with some overlap between Banks 1 and 2. When Bank 2 is full open then all valves in Bank 3 start opening (in parallel), then Bank 4, then Bank 5. **Figure 5 & 6**

Loss of Load Interlocks (C-7A & C-7B)

Objective 4B

The Loss of Load interlocks are derived from the three Turbine Impulse Pressure channels using three separate circuits. The current value of impulse pressure is compared to a lagged value (120 second time constant) to determine if there is a difference. This difference value is compared to a fixed negative setpoint of -10% and -30%.

C-7A

The C-7A Loss of Load Interlock Condenser Dump requires greater than a -10% step or -5%/minute ramp difference in 2 out of 3 Turbine Impulse Pressure circuits to allow the condenser dump valves to open. The C-7A signal will open the Arming Solenoid valve to supply air to the condenser dump valves air actuators with C-9 (Condenser available for steam dump) activated. C-7A will arm Banks 1, 2, and 3 by energizing the arming solenoid on the air supply valves.

The C-7A actuation also illuminates Status light, C-7A LOSS OF LOAD INTLK COND DMP VLVS, on MC1. This Status light will stay lit until the solenoid relay is reset. To reset the C-7A interlock the STEAM DUMP SELECT switch is taken to the RESET position. **Figure 4**

The DCS Graphics page for STEAM DUMP has a “C-7A” status light that will go dark as soon as the signal clears.

C-7B

The C-7B Loss of Load Interlock Atmospheric Dump requires greater than a -30% step difference or -15%/minute rate in 2 out of 3 Turbine Impulse Pressure circuits to allow the atmospheric dump valves to open. The C-7B signal will open the Arming Solenoid valve to supply air to the atmospheric dump valves air actuators. C-7B will arm Banks 4 and 5 by energizing the arming solenoid on the air supply valves.

The C-7B actuation also illuminates Status light, C-7B LOSS OF LOAD INTLK ATMOS DMP VLVS, on MC1. This Status light will stay lit until the solenoid relay is reset. To reset the C-7B interlock the STEAM DUMP SELECT switch is taken to the RESET position.

The DCS Graphics page for STEAM DUMP has a “C-7B” status light that will go dark as soon as the signal clears.

3.3 Plant Trip Controller

Objective 3C

The Plant Trip Controller is used to reduce T-AVG to T no-load following a Reactor trip. Tno-load is the equivalent of T-AVG for 0% power or 557°F. The controller is enabled by the STEAM DUMP SELECT switch being in the "T-AVG" position and a Reactor Trip condition has been sensed by the P-4 Train ‘B’ contact. **Figure 4**

The controller compares Selected T-AVG-2 to T no-load and modulates the three banks of the Condenser Dump valves (See **Figure 7**). A lead/lag circuit conditions the Selected T-AVG signal. This circuit initially boosts the magnitude of any change in Selected T-AVG. This is to make the steam dumps respond in an anticipatory manner based on the rate of change in the signal. The controller output is limited to 49% steam dump demand, which is insufficient to open the atmospheric steam dump valves.

Question 44

STEAM PRESSURE CONTROLLER
- Enabled by 'PRESS' selected on Steam Dump Select switch
- Selected Steam Pressure-1 (2nd highest) (SSP-1) used for controller.
- Steam Equalization Header Pressure used if SSP-1 is in Alternate Action for controller.
- Can only actuate the condenser dumps (49% output max)
- Must be in Steam Press Mode in order to use SLIM Station or soft control
- For startup, dumps will control steam pressure at 1092 psig and T-AVG will increase as power increases.

LOAD REJECTION CONTROLLER
- Enable by Steam Dump Select Switch in 'T-AVG' mode AND NO P-4 Train 'B'
- Compares Selected T-AVG-2 (ST-AVG-2) to Tref (Selected Turbine Impulse Pressure- 2 (STIP-2) and sends signal to modulate all banks as necessary (100% output max).
- 3°F deadband exists to allow rod control to actuate first and last

NOTE: 3˚F dead band prevents any Steam dump actuation. I.E. on a full load rejection steam dumps would close with T-AVG at 560˚F if Tref was 557˚F.
- Three channels of Turbine Impulse Pressure are used for developing C-7A and C-7B (2/3 logic for both interlocks - not median select)
- C-7A → 10% Step or 5%/minute ramp
- C-7B → 30% Step or 15%/minute ramp
- To Reset C-7A/B place STEAM DUMP SELECT switch to 'RESET'

560 degrees corresponds to ~ 1130 psia or 115 psig

PLANT TRIP CONTROLLER
- Used to reduce T-AVG to T no-load following a Rx Trip
- Enabled by 'T-AVG' mode AND P-4 Train 'B'
- Compares T-AVG (Sel T-AVG-2) to T no-load and modulates Condenser dumps (49% max output)

P-12 LO-LO T-AVG INTERLOCK
- Provides a block to keep from inadvertently cooling down to <553˚F. Keeps NC above 551˚F → Minimum Temperature for Criticality.
- Set @ 553˚F 2/4 NC loops
- Steam Dump INTLK BYP switches, Train 'A'&'B' →
  - Either switch to 'OFF' blocks steam dump actuation
  - Both switches to 'BYP INTLK"
    - Bypasses P-12 block signal for Bank 1 Dump valves
    - If BOTH switches taken to 'OFF/RESET' – Resets bypass signal

Never isolate more than 3 steam dump valves: Atmospheric Dump or Condenser Dump - any combination.

NOTE: If P-12 clears (3/4 NC loops >553˚F) bypass is automatically RESET

NOTE: At 340˚F switches in SSPS cabinets can be used to allow opening all condenser dump valves for increased C/D capability.

Question 45

- Unit shutdowns
- Major power transients

2. Due to resulting limitations in the various measurement methods, specific methods have been designated for use under various operating conditions as follows:

   a. Modes 2, 3, 4 - Leak rate monitoring based on tritium concentration from secondary system grab samples.

   b. Mode 1 at greater than or equal to 5% and less than or equal to 40% Reactor Power - Leak rate monitoring based on condenser off-gas radiation monitor readings. Suspend power escalation until new data can be input into the OAC calculation (reactor coolant Xe-133 Equivalent activity, condenser off-gas flow rate) if the radiation monitor alarms.

   c. Mode 1 at greater than 40% and less than 95% Reactor Power or stable operation (defined as less than 10% power change in one hour) at any power level greater than 40% Reactor Power - Leak rate monitoring using main steam line N-16 monitors, or condenser off-gas radiation monitor subject to the knowledge that radiation monitor counts will increase as the primary source term activity increases.

      1) If a monitor alarms, then suspend power escalation to allow the system to stabilize and to validate the leak rate with a second method.

   d. Mode 1 Normal Operation (greater than or equal to 95% to 100% Reactor Power, or stable operation (defined as less than 10% power change in one hour) at any power level greater than 40% Reactor Power) - Leak rate monitoring using condenser off-gas radiation monitor, main steam line N-16 monitors, or condenser off-gas grab sampling (listed in order of preference).

3. During startups, reactor coolant Xe-133 Equivalent activity and condenser off-gas flow rate should be obtained and entered into the OAC calculation to minimize alarms due to changing conditions during startup.

   a. Obtain and enter values prior to reaching 5% power and at approximately 90% power.

   b. Updated values can also be entered into the OAC at other times as needed to avoid nuisance alarms as conditions change.

4. During startups and until 48 hours after steady-state power, the alarm set points may be set at a level higher than normal as needed to offset nuisance alarms.

5. The alarm set points shall not exceed the lowest shutdown limit at any time during startups.

4.5 High and Low Power Controllers

Objective 4C

SGWLC operates in two modes of control: Low Power and High Power (dependent on Feedwater Flow). There are separate controllers used for Low and High Power Modes of operation.

Low Power Mode

This portion of SGWLC is used when the steam and feed flow transmitters are unable to provide SGWLC with reliable input for level control. This controller is in use when the Median-Selected CF Flow on the associated S/G is not in the HIGH POWER mode. (High Power mode selection is described below.) While in this mode, feed water and steam flow trouble alarms are inhibited.

To ensure a quick and accurate response to changes in N/R level, the output of the Low Power Controller is modified by an additional error signal generated from a comparison of the Selected W/R Level to a fixed level value. This additional error signal results from the W/R Level transmitter response to changes in power. The W/R level trends "predict" the response of N/R level. Westinghouse has taken advantage of this level change phenomenon to provide an anticipatory (feed forward) action needed when steam and feed flow are not used in the level control process. Thus, any trend in W/R level will always be true in direction and this error signal should correctly affect valve position response.

High Power Mode

Once the plant reaches sufficient power to provide reliable Steam and Feed Flow measurements, the High Power Mode automatically takes over control of the Feed Demand signal development. This controller is in use when Selected CF Flow on the associated S/G is greater than or equal to 20% on a power increase, and remains until CF flow drops to 17%. At this time control is restored to the LOW POWER controller.

The High Power Controller has two main inputs:

- The N/R Level Error
- CF/SM Flow Rate Mismatch.

The mismatch is based on the rate of change of the steam and feed water flows.

The Flow Rate Mismatch Error signal is combined with the N/R Level Error signal and this Total Error signal is sent to the High Power Controller.

Controller Bumpless Transfer

In order to ensure that both the High and Low power controllers are correctly set when they are switched into operation, the SGWLC incorporates a tracking system, which forces the unused controller to track the output of the one in use. When the swap over CF Flow setpoint is reached, the outputs of both controllers are the same and no transients occur. The controller that is no longer in use now tracks the operating controller.

Flow Demand Signal

The output of the High and Low Power Controllers is the Total CF Flow Demand signal. This Total CF Flow Demand signal will be used to position individual feed water regulating valves (FRV) and feed regulating bypass valves (FRBV) for the respective S/G. Valve demand is related to the Total Flow Demand output of the High or Low Power controller, i.e. the greater the controller output, the greater the valve open demand.

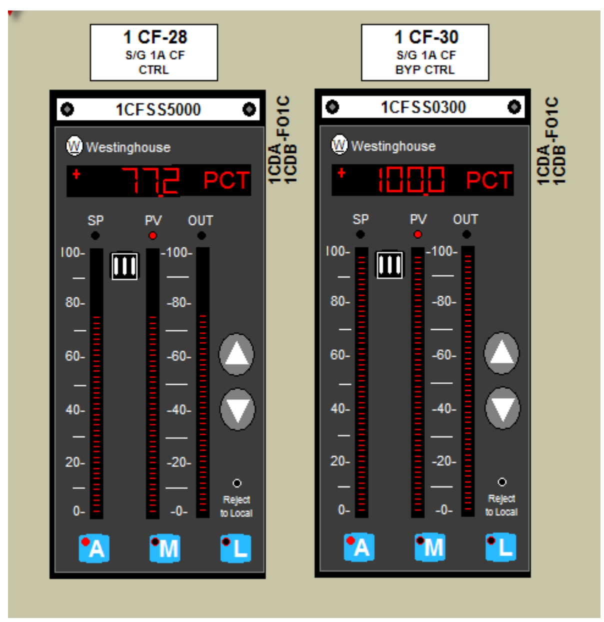
Additionally, each S/G Total flow demand signal is compared and the auctioneered high signal is sent to each CFPT speed controller.

4.6 FRV and FRBV Valve Programmers and Controls

Objective 4D

Each FRV and FRBV controller has a performance curve built in that requires the valve to be at a certain position for any given input signal. When a FRV or FRBV is in AUTO, this input signal is provided by the SGWLC System. Feedback is received from actual valve position to ensure the valve is responding to the demand signal. Significant deviation from demanded position will cause an alarm.

During the initial power increase when the FRBV reaches 55% demanded position, the SGWLC system will insert a small negative signal to the FRBV that will cause the FRV to open slightly. The continued power increase will then be using both the FRV and the FRBV to maintain desired CF flow.



To help prevent a single failure from causing an FRV to fail closed, redundancy is built into the controls. The Automatic control for FRVs consists of duel digital valve positioners and a swapping solenoid. One of these is the Primary and the other is the Backup.

The control signal is sent to the Primary and Backup Valve Positioners. If the Primary Valve Positioner should fail, the Backup Valve Positioner is immediately placed in control with the same signal that was supplied to the Primary Valve Positioner.

9. UNIT DIFFERENCES SUMMARY

Objective 9

S/G Water Level Protection Setpoints

Signal	Unit 1	Unit 2
S/G Hi-Hi Level (P-14)	83%	77%
S/G Lo-Lo Level Rx Trip and CA Auto Start	11%	37%

S/G Water Level Control System Setpoints

Signal	Unit 1	Unit 2
N/R Program Level (0 to 100% power)	39% to 65%	62% to 67%
Low Level Alert (5% above the program Lo-Lo Level	16%	42%

CFPT Trip on Unit 2

For Unit 2 only, the response for a single feed pump trip if Nuclear Power is greater than 65% for SGWLC is different. Following a single feed pump trip with Rx power > 65%, S/G water levels will drop to values less than the program minimum, its return to the program level could come in at a rate that could overshoot the program band and challenge the S/G high-high level P-14 value (77%). Because of this, the control circuit looks at a combination of:

- Rx power > 65%
- Either Feed pump A or B tripped
- Validated S/G level < 52% (7% dead band) and does not exceed 59% within the 30 second time period.

Distractor Part 2

Once these conditions exist, for 30 seconds, a 600 second (10 minutes) timer combines with the condition that if S/G level has now risen greater than 53% to change the S/G level setpoint to 53%. Once the 10 minute timer drops out, the controlling circuit now inserts a slow ramp back from 53% to the current programmed value between 62 and 67%.

Question 47

- The plant was at 100% power when an ATWS event occurred
- The crew entered FRP-S.1, RESPONSE TO NUCLEAR POWER GENERATION ATWS
- All SG Narrow Range levels are OFF-Scale LOW

IAW FRP-S.1, AFW flow must be greater than a MINIMUM value of ____(1)____ GPM to establish a Secondary Heat Sink?

In regards to core age, the most decay heat generated after a reactor trip is present at ____(2)____ of life.

A. (1) 300
(2) beginning

B. (1) 300
(2) end

C. (1) 600
(2) beginning

D. (1) 600
(2) end

Original Question

MODIFIED

Question 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Verify Reactor Trip:** __ • All rod bottom lights - LIT __ • All reactor trip and bypass breakers - OPEN __ • I/R power - TRENDING DOWN.	**Perform the following:** a. Trip reactor. b. **IF** reactor will not trip, **THEN** concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • **GO TO** EP/1/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS).
Correct Flow Path	Distractor Flow Path
3. **Verify Turbine Trip:** __ • All turbine stop valves - CLOSED.	**Perform the following:** __ a. Trip turbine. b. **IF** turbine will not trip, **THEN** perform the following: __ 1) Depress "MANUAL" pushbutton on turbine control panel. __ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons. 3) **IF** control valves will not close, **THEN** CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves.

Question 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. **Control S/G levels as follows:**	
__ a. Verify total CA flow - GREATER THAN 450 GPM. Correct Answer	a. Perform the following: 1) **IF** N/R level in all S/Gs less than 11% (29% ACC), **THEN** perform the following: __ • Start CA pumps __ • Ensure correct valve alignment. 2) **IF** N/R level in all S/Gs less than 11% (29% ACC) **AND** feed flow greater than 450 GPM cannot be established, **THEN** concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • **GO TO** EP/1/A/5000/FR-H.1 (Response to Loss of Secondary Heat Sink).
__ b. **WHEN** each S/G N/R level greater than 11% (29% ACC), **THEN** control feed flow to maintain that S/G N/R level between 11% (29% ACC) and 50%.	
__ 20. **Verify all CA isolation valves on intact S/Gs - OPEN.**	__ **OPEN valve(s).**
__ 21. **Verify S/I equipment status based on monitor light panel(s) - IN PROPER ALIGNMENT.**	__ **Align equipment.**

Question 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Control S/G levels as follows:**	
__ a. Verify N/R level in at least one S/G - GREATER THAN 11% (29% ACC).	a. Perform the following:
Distractor	__ 1) **IF** total CA flow less than 1000 GPM, **THEN** start pumps and align valves as required.
	__ 2) Maintain total CA flow greater than 1000 GPM until N/R level greater than 11% (29% ACC) in at least one S/G.
	__ 3) **WHEN** N/R level greater than 11% (29% ACC) in at least one S/G, **THEN** perform Step 9.b.
	__ 4) **GO TO** Step 9.c.
__ b. THROTTLE feed flow to maintain all S/G N/R levels between 11% (29% ACC) and 50%.	
__ c. **WHEN** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).	
10. **Ensure all dilution paths isolated as follows:**	
__ a. Place NC makeup control switch to "STOP".	
__ b. Place reactor makeup water pumps to "OFF".	

Question 47

STEP 19: Control S/G levels as follows:

PURPOSE:

To ensure CA flow to the steam generators.

APPLICABLE ERG BASIS:

Correct Answer

CA flow is necessary for secondary heat sink. If S/G level is in the narrow range in at least one S/G, a heat sink is available. Therefore, CA flow is needed only to maintain level. If adequate CA flow for decay heat removal cannot be established, the transition to the FR-H.1, Response To Loss Of Secondary Heat Sink, is necessary to establish an alternate source of feed flow or an alternate heat sink.

A range for S/G level control was specified so that the operator would not be so involved in maintaining an exact level in the S/Gs that other operator actions would be delayed (DW-84-010).

PLANT SPECIFIC INFORMATION:

This step allows the operators to reset and throttle CA flow to S/Gs after adequate heat sink is verified. This direction has been added for the following reasons:

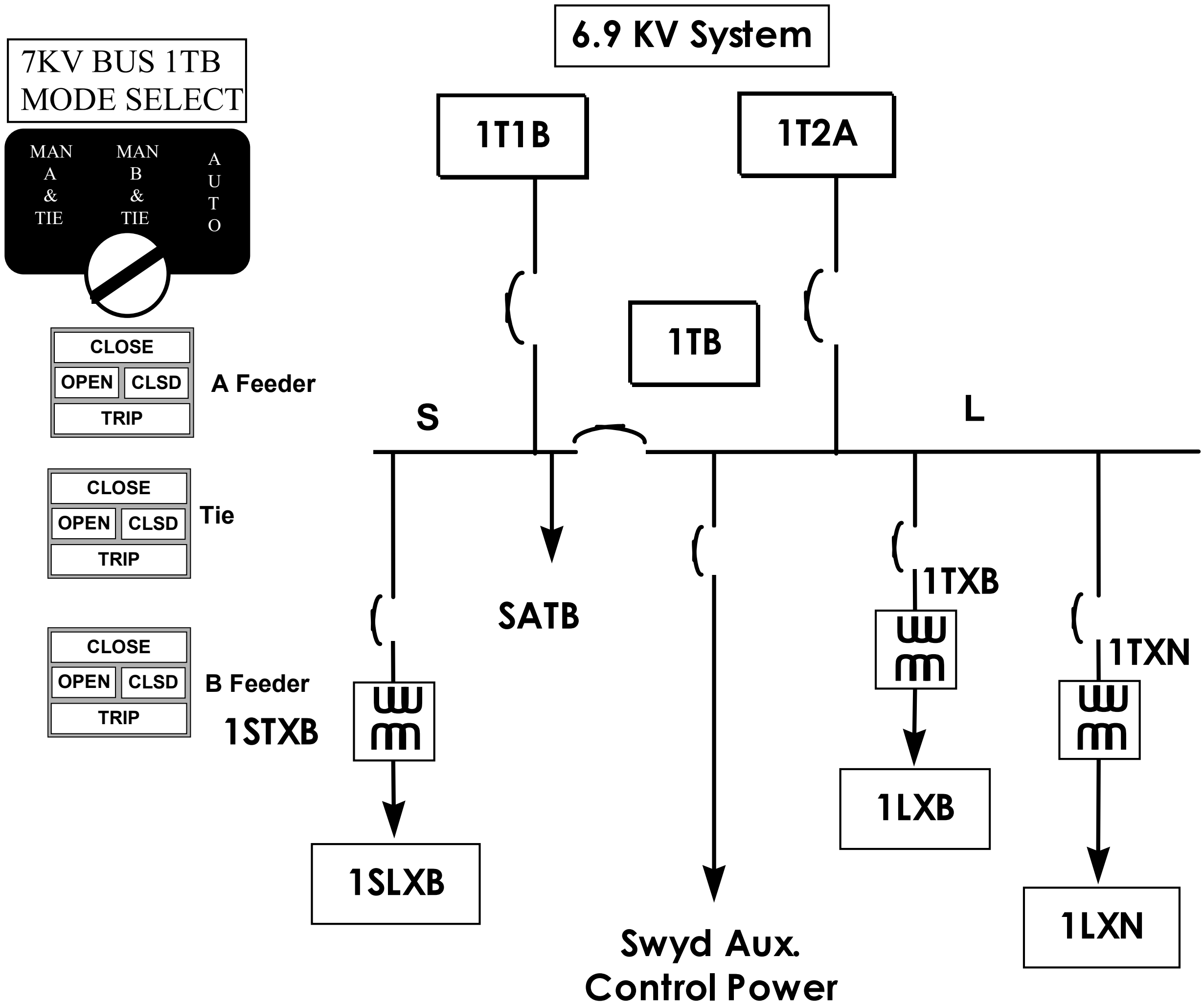
- Several operator feedback items have been received concerning overfilling S/Gs. By providing procedural direction to control feed flow to S/Gs at this point, the potential to limit overfilling the S/Gs is reduced. This is especially a concern if the event started at less than 100% power or low decay heat levels.
- Attempting to control S/G levels at this point also may aid the operator in diagnosing a SGTR in subsequent steps based on the inability to control a particular S/G level.

Unit 1 and Unit 2 difference - The S/G upper control band N/R level setpoint is based on the ERG generic upper limit for Unit 1 and the no-load level for Unit 2 since different model S/Gs are used for each unit.

Continued On Next Page.

Question 48

2.1.2 1TB contains 1T1B (short side- **S**) and 1T2A (long side- **L**). Each 6900V bus is powered from 2 auxiliary transformers using 2 incoming breakers normally closed with the tie breaker open.



Objective # 4, All

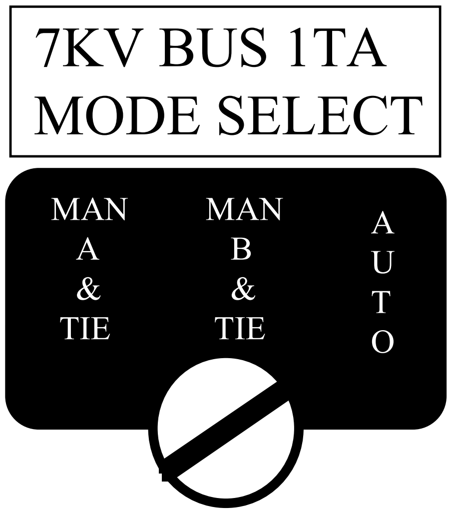
- Short Side (1T1B)
  - SATB
  - 1STXB
  - 1B NC Pump

- Long Side (1T2A)
  - 1 YV-C-2
  - 1B RC Pump
  - 1B Hotwell Pump
  - 1B Condensate Booster Pump
  - B RL Pump
  - 1TXB
  - 1TXN
  - Switchyard Auxiliaries Feeder

Question 48

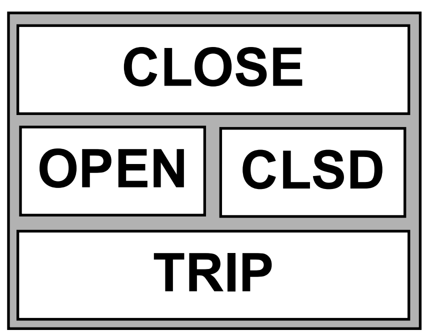
Objective # 5B, All

The incoming and tie breakers are controlled from MC-11 in the control room. A 3-position transfer switch, (See one below, typical of four switches on each Unit) when in AUTO, allows auto swap; when it is in MAN A & TIE it allows operating the "A" incoming breaker and the tie breaker; when it is in MAN B & TIE it allows operating the "B" incoming breaker and the tie breaker.



TRIP – CLOSE pushbuttons operate the breaker

OPEN – CLSD light indication provides the position of the breaker



To supply a 6900V bus completely from one auxiliary unit transformer, the tie breaker can be closed and one incoming breaker opened. This transfer can occur automatically or manually.

For Auto Transfer, ensure Mode select switch is in AUTO. The auto transfer would then be initiated by a Zone Lockout or 75% UV on the incoming line with no fault on the affected bus. If in sync, a fast transfer will occur. The incoming breaker opens before the tie breaker closes, but the transfer occurs within a few cycles with no loss of load.

Question 48

If not in sync, a slow transfer will occur. The transfer is delayed to allow voltage to decay to a point that synchronization is not a concern (25% voltage). This will take approximately one second and a loss of loads on the bus will occur.

PT/1(2)/A/4350/005 (6900V Normal Auxiliary Power Automatic Transfer Test) is performed each outage to ensure the auto swap occurs as designed.

Automatic Fast Transfer Switch

The Automatic Fast Transfer switch will allow defeat of the fast transfer when the unit is off line. This switch is located in the 6900V bus rooms on the control panels with the under voltage transfer relays. The purpose of this switch is to eliminate possible equipment damage from surges on fast transfer. The switch has two positions, ENABLE or DEFEAT.

With the Main Generator off-line, this switch is placed in the DEFEAT position by the Unit Shutdown procedure and then only slow transfers will occur. This switch controls the tie breaker. The defeat removes the sync check relays (25) from the circuit and provides for only a slow transfer.

Distractor Part 2

3.3 600V Unit Bus Abnormal Operation

Objectives # 5C, All

Both incoming breakers on an MCC are normally closed. One feeder breaker from a load center to an MCC will be closed and the alternate feeder to the MCC from the other load center will be open. On loss of power at the “normal” load center the supply from the “alternate” load center will auto close to supply the MCC.

LXC/LXD Unit Buses

If power is lost to 1(2)LXC and 1(2)LXD simultaneously, 1(2)LXC will swap to 1(2)TXS transformer supplied by 1(2)TA. 1(2)LXD is time delayed to allow this to occur. The alternate incoming breakers are interlocked to prevent both from closing at the same time. Each incoming breaker has a MAN-AUTO switch and an OPEN-NORMAL-CLOSE switch. An auto transfer of power will occur on low voltage on the incoming line. If the sources are in sync the alternate breaker closes and then the normal breaker trips. This is a Hot Bus Transfer.

- D/G Control Panels
- D/G CO2 Fire Protection
- Control Rod Drive Control Panel

2.7 Static Inverters (KXIA, KXIB, KUI)

Objective 3D, All

The Static Inverters convert 125 VDC to 120 VAC power and supply the AC portion of Auxiliary Control Power System. A DC input and an AC output breaker is provided.

The normal power supply to Static Inverters KXIA and KXIB is the associated 125 VDC Distribution Panel.

The normal power supply to Static Inverter KUI is the output of 125 VDC Auctioneering Diodes ADA and ADB. Inverter KUI normally supplies 120 VAC regulated panel P-1 (OAC power supply) through an automatic static transfer switch, manual transfer switch, and isolation transformer KUT.

Some of the indications provided on the inverter are:

- AC volts (upstream of inverter output breaker)
- AC amps (upstream of inverter output breaker)
- Frequency (upstream of inverter output breaker)
- INVERTER OUTPUT LOW VOLTAGE light
- IN SYNC - amber light will be lit when the normal and alternate sources are synchronized.

2.8 Automatic Static Transfer Switch (KXAA, KXAB, KUA)

Objective 3E, All

An automatic static transfer switch (KXAA, KXAB, and KUA) associated with each static inverter provides automatic power transfer to an alternate power source (regulated power) on low inverter output voltage. Manual swap capability is also provided via pushbuttons on the Automatic Transfer Switch. **Correct Answer Part 2**

Once the Auto Transfer Switch has automatically swapped to alternate source, a 60-second delay is initiated. After 60 seconds, if the transfer was due to a total loss of inverter output voltage, the switch will transfer back to the inverter whenever voltage and frequency have returned to normal. However, a manual transfer of the static transfer switch will require the operator to manually re-transfer the inverter back to normal.

Distractor Part 2

The Auto Transfer Switch can be bypassed using the Manual Bypass Switch.

Question 49

1. Limits and Precautions

1.1 Smoking, open flames, or evolutions that create sparks are **NOT** allowed in the battery area.

1.2 An inverter may **NOT** be restarted within 60 seconds after taking it out of service in order to allow capacitors and rectifiers to cool down.

1.3 IAE provides corrective action when voltages are **NOT** within the ranges given.

2. Initial Conditions

_____ Verify 1KXIB is in service per Enclosure 4.3 (Inverter Startup).

3. Procedure

_____ 3.1 Verify the "IN SYNC" indicator lights on the following panels are lit:

- ☐ 1KXMB
- ☐ 1KXAB

_____ 3.2 Depress the "ALTERNATE AC SOURCE TO LOAD" pushbutton on 1KXAB.

_____ 3.3 Verify the following indicator light status on 1KXAB:

- ☐ "ALTERNATE AC SOURCE SUPPLYING LOAD" is lit
- ☐ "INVERTER SUPPLYING LOAD" is dark

_____ 3.4 Place 1KXIB B2 (Inverter Output) in the "OFF" position.

_____ 3.5 Place 1KXIB B1 (Battery Input) in the "OFF" position.

NOTE:
1. At this point, 1KXIB is shutdown with 1RDB supplying 1KXPB through 1KXAB.
2. Subsequent steps are to return 1KXIB to service and 1KXAB to normal condition.
3. Step 3.7 must be performed immediately after Step 3.6 is completed, thus place keeping for both steps is performed after both complete. Failure to close the "BATTERY INPUT" breaker immediately after the "PRECHARGE" pushbutton is released may result in blown inverter input fuses.

_____ 3.6 Depress and hold the "PRECHARGE" pushbutton on 1KXIB until the "PRECHARGE" indicator light (above pushbutton) has been lit for a minimum of 5 seconds.

_____ 3.7 Place 1KXIB CB1 (Battery Input) in the "ON" position.

Question 50

_____ 3.52 Open 1VN-1 (1A D/G Eng Exhaust Silencer Drain) (DB-557, CC-38).

Distractor Part 1

3.53 Drain any accumulated oil out of the crankcase vent drip leg as follows:

_____ 3.53.1 Verify 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain) (DB-556, DD-39) is closed.

_____ 3.53.2 Remove pipe cap downstream of 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain).

_____ 3.53.3 Place the waste oil container downstream of 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain).

_____ 3.53.4 Open 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain) to drain any accumulated oil.

_____ 3.53.5 **WHEN** oil has drained, close 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain).

_____ 3.53.6 Replace pipe cap downstream of 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain).

_____ 3.54 **IF** the D/G run was normal **AND** the D/G did **NOT** trip, perform the following at the Alarm Monitor Panel (1ELMC0029):

☐ 3.54.1 Depress and hold the "Alarm Ack" button for 1 second.

☐ 3.54.2 Verify that any "A" or "B" lights that were flashing are now illuminated solid.

_____ 3.55 Verify AC CONTROL POWER ON light LIT on 1DECPA.

3.56 Verify DC Control Power on 1DECPA by performing one of the following:

_____ 3.56.1 Verify DC CONTROL POWER ON light LIT.

OR

_____ 3.56.2 **IF** DC CONTROL POWER ON light is **NOT** LIT,
THEN perform the following:

- _____ Verify 1.47 Bypass Panels "D/G A BYPASS" light is green.
- _____ Have IAE verify voltage at the DC CONTROL POWER light socket.

_____ 3.57 Verify 1A D/G has adequate fuel oil storage tank volume (≥ 77,100 gallons) per one of the following:

Question 50

_____ 3.3 **IF** the D/G is to be barred, perform the following:

_____ 3.3.1 Select the air supply to be used from the following:

For D/G 1A

☐ 1VG-13 (1A D/G Eng Starting Air Tank 1A1 Drain) (DB 558, DD-38)

☐ 1VG-14 (1A D/G Eng Starting Air Tank 1A2 Drain) (DB 558, DD-38)

For D/G 1B

☐ 1VG-57 (1B D/G Eng Starting Air Tank 1B1 Drain) (DB 558, BB-38)

☐ 1VG-58 (1B D/G Eng Starting Air Tank 1B2 Drain) (DB 558, BB-38)

3.3.2 Perform the following to the air supply selected in step 3.3.1:

_____ 3.3.2.1 Open the valve to drain the moisture

_____ 3.3.2.2 **WHEN** moisture has been drained, close the valve.

_____ 3.3.2.3 Attach one end of a 250 psig rated hose.

_____ 3.3.3 Remove Danger Stay Clear Rotating Equipment sign and rope which limits access to the D/G flywheel area for barring the D/G.

_____ 3.3.4 Attach other end of hose to the barring device coupling.

_____ 3.3.5 Open the indicator cock on each cylinder.

Cylinder	☑
1L	☐
2L	☐
3L	☐
4L	☐
5L	☐
6L	☐
7L	☐
8L	☐

Cylinder	☑
1R	☐
2R	☐
3R	☐
4R	☐
5R	☐
6R	☐
7R	☐
8R	☐

_____ 3.3.6 **IF** liquid issues from any of the cocks, Engineering shall be notified to decide if any corrective action shall be taken before proceeding. {PIP 98-0441}

Question 50

_____ 12.12 **<u>IF</u>** all the following are true, bar and roll D/G with air per OP/1/A/6350/002 (Diesel Generator Operation): {PIP 96-1185}

- ☐ D/G 1A has been shutdown for greater than 4 hours.
- ☐ D/G 1A has **<u>NOT</u>** been barred and rolled on air within the last 12 hours.{PIP 00-324}
- ☐ D/G 1B is operable.
- ☐ Tech Spec assessment will allow D/G 1A inoperability.

_____ 12.13 **<u>IF</u>** this is the first start of D/G 1A in the month of March, swap D/G 1A lube oil strainers per OP/1/A/6350/002 (Diesel Generator Operation).

Lube Oil Strainer placed in service ____________

_____ 12.14 **<u>IF</u>** Step 12.12 was **<u>NOT</u>** performed, verify the Fuel Pump Linkage Auto Shutdown Cylinder Shaft is retracted.

12.15 Perform the following:

- _____ Verify 1A1 and 1A2 VG Compressors maintain VG Tank pressure is $\geq$ 235 psig.
- _____ Verify "CONTROL AIR PRESS" on 1DECPA is 58 - 62 psig.
- _____ Close 1VN-1 (1A D/G Exhaust Silencer Drain) (DB-559, DD-39).

12.16 Perform the following at the Diesel Engine Shutdown Logic Monitoring Panel (1ELMC0029):

_____ 12.16.1 Depress and hold the "Alarm Ack" button for 1 second.

_____ 12.16.2 Verify that any "A" or "B" alarm lights that were flashing are now lit solid.

_____ 12.17 **<u>IF AT ANY TIME</u>** the D/G run is aborted prior to completion of:

- ☐ Step 2.38, 2.39 or 2.40 of Enclosure 13.1 (D/G 1A Operation From D/G Room)

<u>OR</u>

- ☐ Step 2.39, 2.40 or 2.41 of Enclosure 13.2 (D/G 1A Operation From Control Room) perform the following:

_____ 12.17.1 Notify Unit/WCC SRO immediately to make a determination of operability.

_____ 12.17.2 **<u>IF</u>** attempting subsequent restart, a re-verification of Step 12.1 through Step 12.16 shall be performed.

5.4 Containment Ventilation Isolation (S_H)

Objective 8F

(S_H) Containment Ventilation Isolation

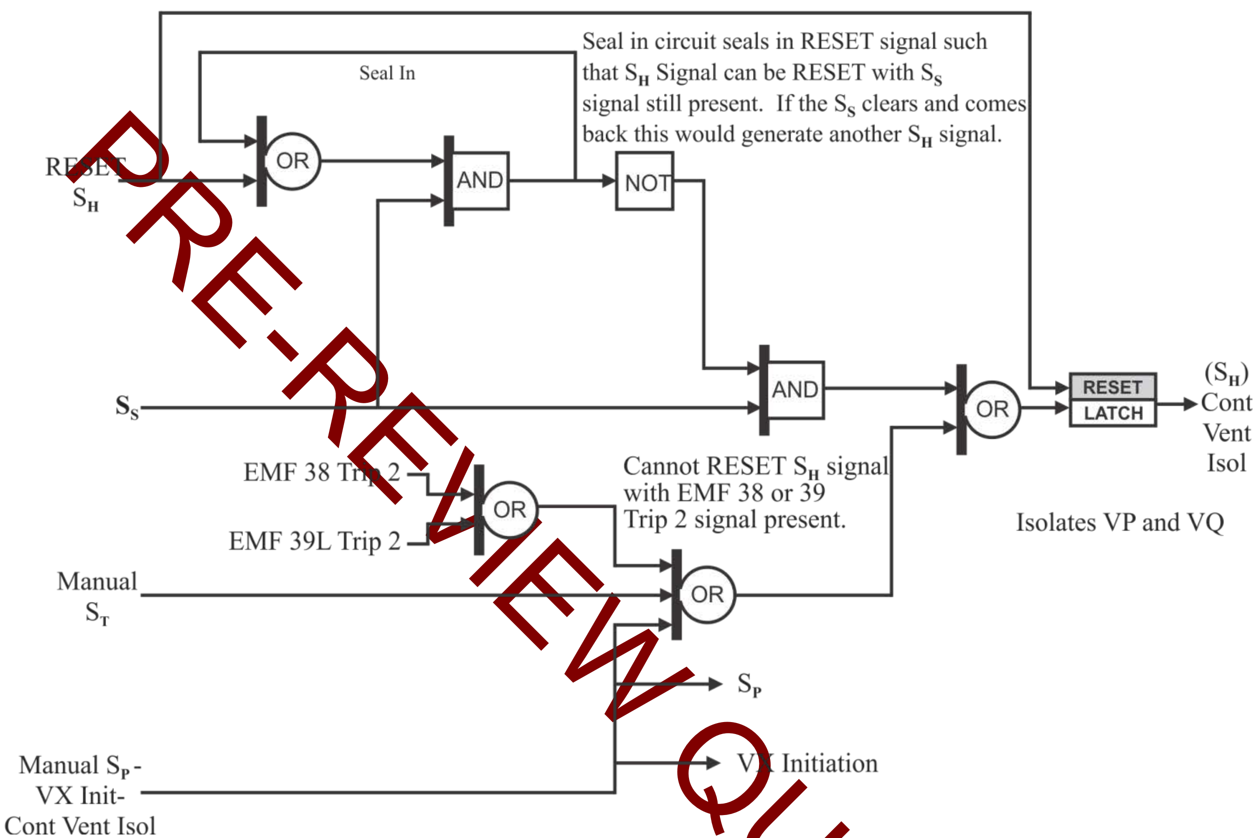


Figure 18

S_H shuts down and isolates VP and isolates VQ Containment Isolation valves.

Four Signals can actuate an S_H:

- Manual "Phase A" (S_T): Train A (B) S_T will actuate train A (B) S_H.
- Manual "Phase B", VX Initiate, Cont Vent Isol": Train A (B) (Phase B, VX Initiate, Cont Vent Isol) will actuate train A (B) SH. This is a single pushbutton that actuates three functions. (Figure 11)
- S_S Signal: Train A (B) S_S will actuate Train A (B) S_H.
- EMF 38, or 39 L, TRIP 2: High Containment Particulate or Gas will actuate BOTH Trains of S_H.

There is one Cont Vent Isol Reset pushbutton per train. The reset is functional with S_S present as long as the EMF Signals are not present. S_H cannot be reset with the EMF Trip 2 signals present.

Question 52

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Verify both NS pumps - OFF.**	**Perform the following:** a. <u>IF</u> all the following conditions met: __ • NS in service __ • NS suction aligned to containment sump __ • RN established to associated NS Hx, __ **<u>THEN RETURN TO</u>** procedure section and step in effect. __ b. Ensure both NS pumps - OFF.
2. **CLOSE the following valves:** __ • 1NS-20A (NS Pump 1A Suct From FWST) __ • 1NS-3B (NS Pump 1B Suct From FWST).	
__ 3. **Verify containment pressure - GREATER THAN 3 PSIG.**	**Perform the following:** __ a. Wait up to 20 seconds for 1NS-20A and 1NS-3B to close. __ b. OPEN 1NS-18A (NS Pmp A Suct From Cont Sump). __ c. OPEN 1NS-1B (NS Pmp B Suct From Cont Sump). __ d. **<u>IF AT ANY TIME</u>** containment pressure goes above 3 PSIG, **<u>THEN</u>** perform Enclosure 2 (Aligning NS for Recirculation). __ e. **<u>RETURN TO</u>** procedure section and step in effect.

Question 52

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. (Continued)

CAUTION **Exceeding 4650 GPM RN flow through an NS Hx will cause damage to the Hx tubes.**

k. Align RN to NS Hx 1A as follows:

1) Verify at least one of the following:

__ • All Unit 1 and Unit 2 RN pumps - ON

OR

__ • RN System - ALIGNED FOR SINGLE SUPPLY HEADER OPERATION.

RESPONSE NOT OBTAINED:

1) Perform the following to support NS Hx cooling flow:

__ a) **IF** only one A train RN pump on, **THEN** CLOSE Unit 2 2RN-48B (RN Supply X-Over Isol).

b) **IF** only A train RN pumps on, **THEN** CLOSE one of the following Unit 2 valves:

__ • 2RN-47A (RN Supply X-Over Isol)

OR

__ • 2RN-48B (RN Supply X-Over Isol).

__ 2) OPEN 1RN-144A (NS Hx 1A Inlet Isol).

__ 3) **WHEN** 1RN-144A begins to open, **THEN** OPEN 1RN-148A (NS Hx 1A Otlt Isol).

Question 53

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. **IF AT ANY TIME VI pressure less than 55 PSIG AND trending down, THEN perform the following:**

   __ a. Trip reactor.

   __ b. **WHEN** reactor power less than 5%, **THEN** depress "CLOSE" pushbutton for all MSIVs.

   __ c. Continue in this procedure as time permits.

   __ d. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Question 53

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 2. **Verify reactor trip breakers - CLOSED.**	**Perform the following:** a. **IF** ND System in RHR mode, **THEN** control ND flow and cooldown rate for operating ND train(s) as follows: • Train A: __ 1) Place "PWR DISCON FOR 1NI-173A" switch in "THROT" position. __ 2) THROTTLE 1NI-173A (ND Hdr 1A To Cold Legs C&D) to maintain desired flow and cooldown rate. __ 3) Raise output for 1ND-26 (ND Hx 1A Outlet Ctrl) to 100%. __ 4) Place 1ND-27 (ND Hx 1A Bypass Ctrl) in manual and closed. • Train B: __ 1) Place "PWR DISCON FOR 1NI-178B" switch in "THROT" position. __ 2) THROTTLE 1NI-178B (ND Hdr 1B To Cold Legs A&B) to maintain desired flow and cooldown rate. __ 3) Raise output for 1ND-60 (ND Hx 1B Outlet Ctrl) to 100%. __ 4) Place 1ND-61 (ND Hx 1B Bypass Ctrl) in manual and closed. __ b. **GO TO** Step 4.

compressed air enters an intercooler which removes the heat generated during the first compression stage. After leaving the second stage, the hot compressed air enters an aftercooler and water separator. These components remove the heat and moisture generated from the second stage of compression. The centrifugal compressors contain an additional moisture separator after the intercooler.

The Recirculated Cooling Water (KR) System normally provides cooling water to the three compressors. The Fire Protection System (RF) is provided as a manually aligned backup cooling water source for compressors E and F during a blackout. When cooling water is aligned, it cools the aftercooler, oil cooler and intercooler.

The air leaving the compressors discharges into their respective **Instrument Air Receivers** which are used to smooth out surges and act as storage volume. An alternative flow path, designated the alternate wet header, may be used from the receivers to the dryer package. The alternate wet header piping utilizes a separate coalescing prefilter prior to entering the desiccant dryer towers. The alternate wet header will permit subsequent VI system piping repair/replacement with no loss of system functionality.

The bulk air supply to the Unit 1 and Unit 2 condensate polishing and steam generator blowdown demineralizers branches off the instrument air compressor discharge header upstream of the instrument air dryers. Two check valves, with a trap between them, are located in this line to prevent the backflow of water into the Instrument Air System.

Moisture is removed from the instrument air by one of the coalescing prefilters. The instrument air is then dried to meet the system required -23°F dew point by one of three desiccant **Air Dryers E, F, or G** piped in parallel. The air then passes through one of the afterfilters. Downstream of the instrument air afterfilters, the instrument air system forms a common header which supplies air to the plant. The air is again filtered through a filter regulator prior to each air operated valve or instrument.

The Instrument Air System is capable of supplying compressed air to both the Instrument Air and the Station Air systems. Self-contained, back pressure control valve **1VI-500** is installed in the crossover header between the instrument air and station air headers. If the instrument air system pressure drops below 80 psig, valve 1VI-500 will close to terminate air supply to the Station Air System while maintaining air supply to the Instrument Air System.

The Station Air System is capable of supplying compressed air to the Instrument Air System upon loss of instrument air header pressure. An instrument air header pressure below 76 psig will open valve **1VS-78** such that the station air compressors will back up the instrument air compressors. The air supplied through the Station Air System passes through two oil removal filters prior to entering the Instrument Air System. The Station Air connection is located in the instrument air discharge header upstream of the instrument air dryers. **Distractor Part 2**

Question 54

Given the following:

- Unit 1 is at 100% power.
- A steam break occurred on the Main Steam Equalization Header.
- Safety Injection 1B failed to actuate.

Which one of the following describes the status of the containment isolation valves associated with Phase A and Phase B before any operator actions?

A. Only Phase A (S_T) Train 1A valves close.
Only Phase B (Sp) Train 1A valves close.

B. Only Phase A (S_T) Train 1A valves close.
No Phase B (Sp) valves close.

C. All Phase A (S_T) Train 1A and 1B valves close.
Only Phase B (Sp) Train 1A valves close.

D. All Phase A (S_T) Train 1A and 1B valves close.
No Phase B (Sp) valves close

Original Question

MODIFIED

Question 54

5.2 Phase A (S_T)

Objective 8D

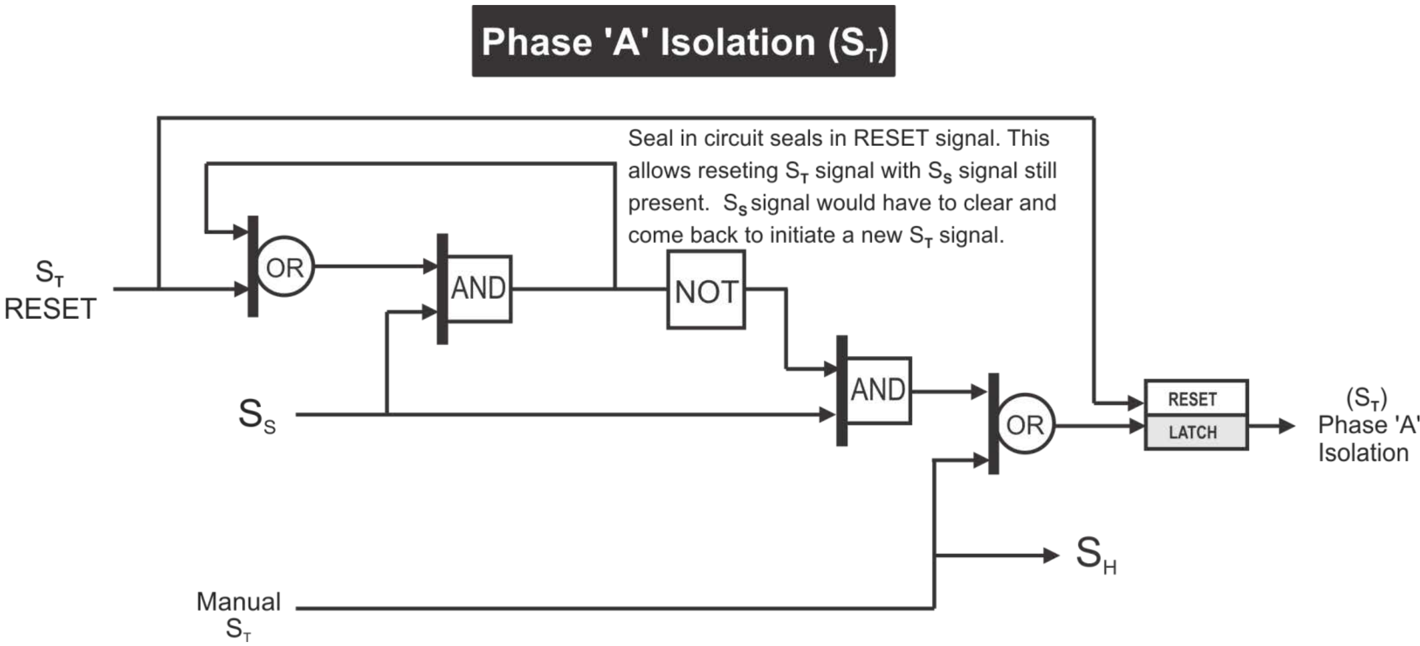


Figure 16

Phase "A" Isolation isolates all containment penetrations, which are non-essential to reactor / containment safety or cooling.

Two signals can generate an S_T signal:

- Manual: One pushbutton per train under clear plastic cover on MC11 (Figure 11).
- Safety Injection (S_S) Train A (B) directly actuates Train A (B) S_T.

Annunciator "PHASE A CONTAINMENT ISOLATION" (Panel AD-13, Window B3) illuminates on an S_T.

Question 54

5.3 Phase B (SP)

Objective 8E

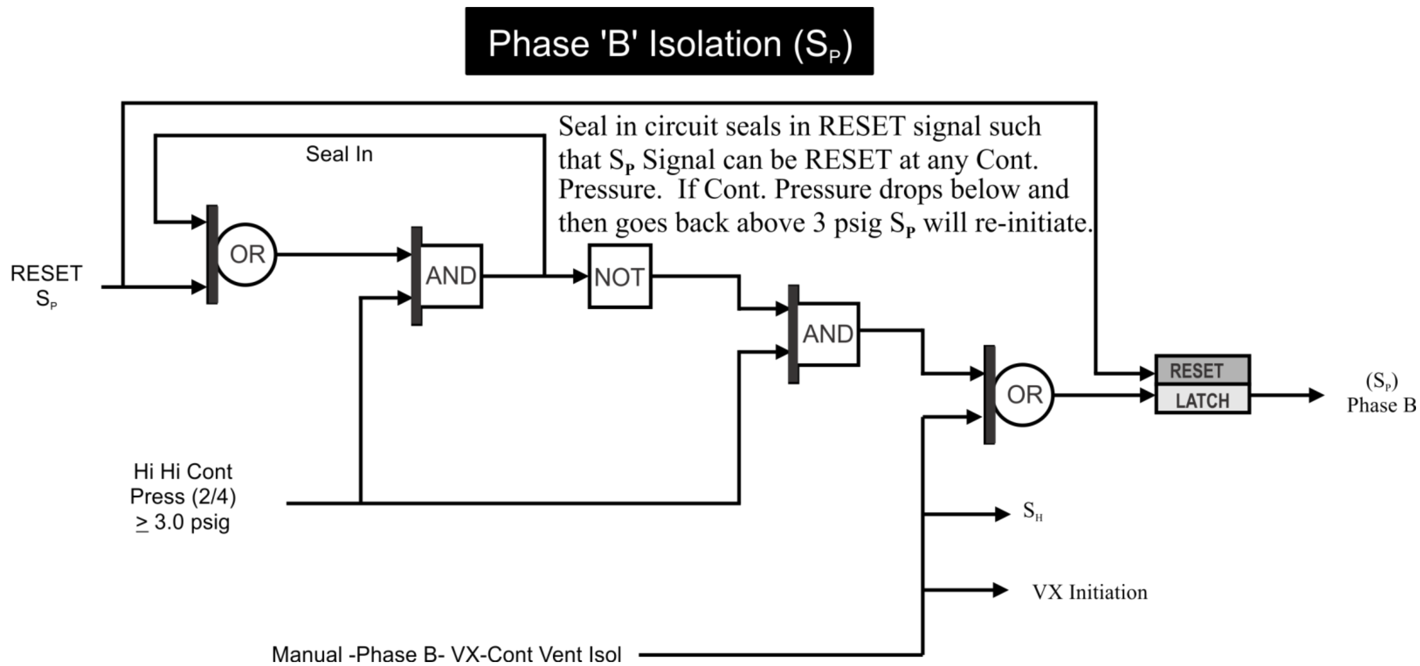


Figure 17

Phase "B" Isolation completes the isolation of non-essential containment penetrations including KC to the NCP's.

Two signals can generate an SP signal:

- Manual: One pushbutton per train under clear glass cover on MC11 (Phase "B", VX Initiate, Cont Vent Isol Button). The "Phase "B", VX Initiate and Cont Vent Isol" Button, when depressed, will initiate both functions. (Figure 11)
- Hi-Hi Containment Pressure: 2/4 Containment Pressure Channels greater than or equal to 3.0 PSIG.

There is one Phase "B" Reset Button for each train to allow manual control of SP valves. The SP reset is functional with any pressure in containment.

Phase "B" Isolation indication:

- Annunciator "CONTAINMENT ISOLATION PHASE B" (Panel AD-13, Window D4) illuminates on an SP.
- Also, Annunciator "CONT HI-HI PRESS" (Panel AD-13, Window B4) signals Control Room on HI HI Containment Pressure on any one channel.

- Prevents tripping the YV chillers on low flow after the CRDMs are de-energized. Failure to operate the LCVUs in "MAX COOL" mode after de-energizing the CRDMs can also result in over pressurization of the YV piping causing relief valves to lift.

Cooling Water Bypass Valve

Objective 4C, All

A "BYPASS" valve or "FULL FLOW" valve is installed in parallel with the normal cooling water flow control valve. This valve will automatically open for each of the following conditions:

1. An LCVU is selected to HIGH speed.
2. Containment pressure rises to greater than or equal to 0.5 psig. Refer to Figure 4.

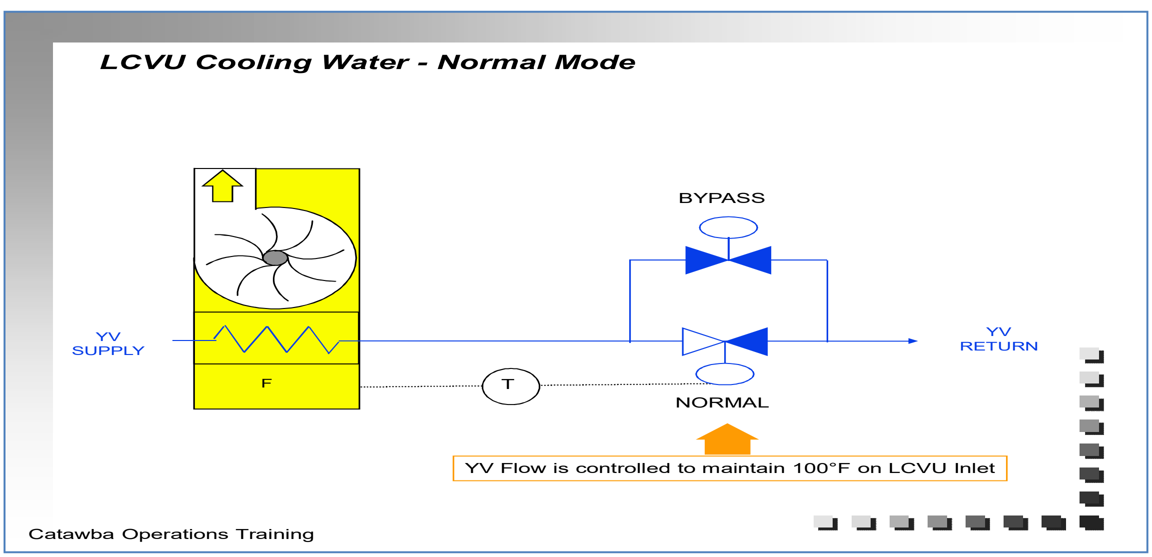
2.1.2 Two (2) pipe tunnel booster fans (PTBFs)

PTBFs are used to circulate lower containment air outside of the crane wall. Like the LCVUs. The PTBFs are two speed fans with "OFF-LOW-HIGH" selector switches on the back of the main control board. The PTBFs do NOT contain cooling water coils. The PTBFs draw cool air from the "D" LCVU discharge duct, prior to the supply header, and discharge to the pipe tunnel and DRPI cabinet area.

Flow / PTBF (Hi/Lo Speed)	9,520 / 4,760 CFM (Unit 1) 9,520 / 5,600 CFM (Unit 2)

Normally only one of the two PTBFs is in service operating at the same speed as the in-service LCVUs. Indication of fan discharge air flow is provided on a local panel in the auxiliary building. PTBF vibration resets are provided on this same panel.

Figure 2: LCVU – Normal Mode



LOWER CONT PRESS 0.5 PSIG INITIATE HI SPEED B/12

SETPOINT: 0.5 psig rising

ORIGIN: NSPS5150, NSPS5230 – ½ Channels

PROBABLE CAUSE: LOCA

Main steam line rupture.

AUTOMATIC ACTIONS: If it's associated ventilation unit is in low speed, the following valves open to double the flow to the lower containment ventilation units:

- RN-473 (LCVU A Full Flow Valve)
- RN455 (LCVU B Full Flow Valve)
- RN447 (LCVU C Full Flow Valve)
- RN481 (LCVU D Full Flow Valve)

IMMEDIATE ACTIONS: Ensure automatic action occurs.

SUPPLEMENTARY ACTIONS: Place additional lower containment ventilation units in operation per OP/1(2)/A/6450/001

Monitor containment pressure and adjust fan speeds as necessary

Refer to TS 3.6.4 (Containment Pressure)

Protection Set III is supplied from 120 VAC Panelboard ERPC. This set powers the following:

- Process Control Channel III (PCS Cabinet 3)
- NIS Channel III
- SSPS Channel III Input Bay (Train A & B)
- NC Pump Monitor Panel Channel III (This channel monitors NCP B.)
- Turbine Trip Relays Channel III
- 48 VDC and 15 VDC Power Supplies (SSPS - Train B)

Protection Set IV is supplied from 120 VAC Panelboard ERPD. This set powers the following:

- Process Control Channel IV (PCS Cabinet 4)
- NIS Channel IV
- SSPS Channel IV Input Bay (Train A & B)
- NC Pump Monitor Panel Channel IV (This channel monitors NCP D.)
- Turbine Trip Relays Channel IV
- 48 VDC and 15 VDC Power Supplies (SSPS - Train B)
- SSPS Train B Output Cabinet

NOTE: In the event of an Extended Loss of ALL AC Power (ELAP), ECA-0.0 (Loss Of ALL AC Power) load sheds vital batteries, resulting in down-powering SSPS Panel, affecting a control logic scheme for the CLAs, (P-11 Permissive). IAE must place a jumper on contacts to allow closing and maintaining CLA Isolation valves closed

3.2 Reactor Trip and Bypass Breaker Power Supply

Train A breaker control power is supplied by 125 VDC Panelboard EPA, and Train B breaker control power is supplied by 125 VDC Panelboard EPD.

3.3 Impact of Loss of Power on Reactor Protection

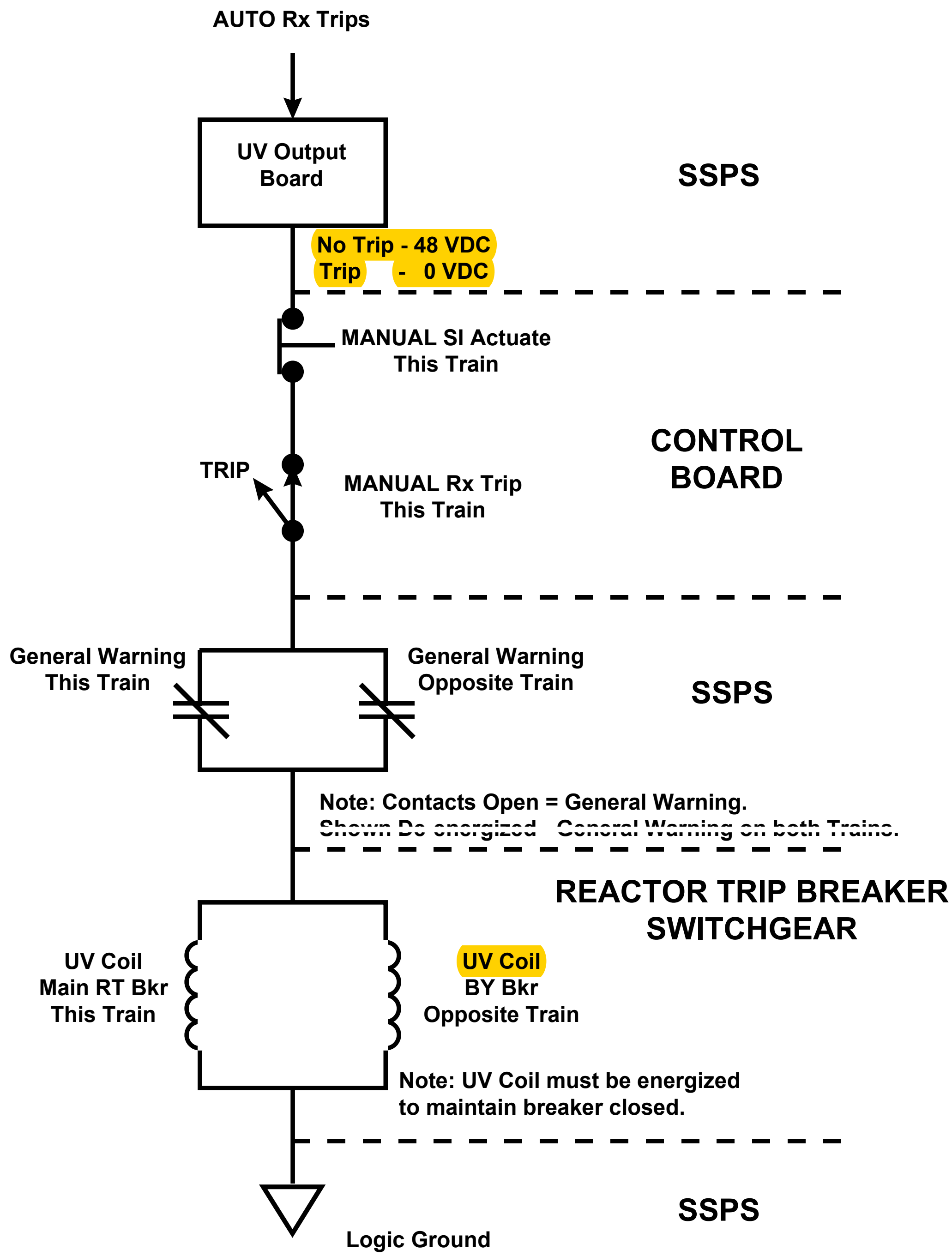
Objective 6, Licensed

Upon loss of a Vital 120 VAC Panelboards, all bistables on the affected channel will de-energize giving trip signal inputs while the control board bistable lights for the affected channel will be lit.

The trip signal for these bistables will change the trip logic. With one trip signal present, the 2/4 logic now becomes 1/3, and the 2/3 logic now becomes 1/2. The power loss should not cause a reactor trip unless another channel's bistable was in TRIP prior to the power loss.

Question 56

10.6 Figure 6: Reactor Trip Breaker Undervoltage Circuit - One Train



2. COMPONENT/SYSTEM DESCRIPTION

2.1 Pzr Level Channels

Objective 2A

Hot Calibrated Channels

Three Pzr Level Channels, 1, 2, and 3, are the “Hot” channels calibrated for the normal Pzr temperature of 653ºF. Their indicated level will be greater than actual level when Pzr temperature is less than 653ºF. See Attachment 4 for the process for determining actual level values. **Figures 1 and 2**

These Pzr level channels are used during normal operation for protection, control, and indication. Selected Pzr Levels are developed by median selector circuits in the Distributed Control System (DCS).

Two Selected Pzr level signals are developed to provide redundant control signals, designated Selected Pzr Level -1 and Selected Pzr Level -2. Looking at Attachment 4 Graphs you will see how the indicated level changes with Pzr water temperature changes. **Figures 3 and 4**

Cold Calibrated Channel

The “Cold Cal” channel is calibrated at 100ºF and 14.7 psia (~atmospheric pressure). It’s indicated level will be less than actual level when Pzr temperature is greater than 100ºF. See Attachment 4 for the process for determining values.

The cold cal channel is used during plant startup, shutdown, and refueling for indication and is considered a non-safety related indication.

Indication

The meter indication for all Pzr levels is 0 – 100% and is located in the Control Room (CR). The three Pzr Level Channels (Hot Cal) are on MC10 with the Pzr Reference Level indication next to them. The Pzr Cold Level indication is on MC5.

The DCS graphics page NC – Pressurizer and PRT has both selected Pzr levels and the setpoint (SP) or reference level. The DCS graphics page NV – Charging Flow Control has the Pzr Level Master with SP, PV (process variable) and O (output).

Separate transmitters feed both Auxiliary Shutdown Panels (ASP) and the Standby Shutdown Facility (SSF) indication. (3 transmitters)

On MC1 there is a 2-pen recorder providing indication of Selected Pzr Level-1 and Pzr Reference Level. These are normally used to trend Pzr level.

120 VAC Vital I&C Panelboards **ERPA** (N41), **ERPB** (N42), **ERPC** (N43), **ERPD** (N44), supply power to the Power Range detectors and circuitry.

For the PR detectors, instrument power supplies the meters, circuit processing components, high voltage and detector power. Instrument Power Fuses provide overcurrent protection for power supply circuits.

Control power supplies the lights on the drawer and 120 VAC to the bistable relay drivers to the plant relays. Control power fuses provide overcurrent protection for the control signal circuit transformers.

A trip signal will be generated when either instrument or control power fuses are removed. The instrument power fuse trip can be prevented by blocking or bypassing; the control fuse trip is prevented by blocking only.

Loss of ERPA, B, C or D

At 100% power, if only one power supply fails (ERPA, B, C or D), all of the bistables for the affected channels will go to the tripped condition. Since most of the actions that occur at 100% power require 2/4 channels, no automatic action should occur with the exception of OVERPOWER ROD STOP which requires only 1/4 channels.

If more than one power supply is lost to the Power Range detectors, the 2/4 Power Range logic for reactor trips will be met and the reactor will trip.

PR Instrument Power Failure

For a loss of instrument power only (Instrument Power Fuse failure), all light indications on the PR 'A' drawer will be present indicating that all of the bistables are in the tripped condition.

PR Control Power Failure

The bistables for the affected nuclear instruments will be in the tripped condition. The PR drawer bistable lights will NOT be lit.

Distractor Part 2

At 100% power, if two or more channels of power are affected, a reactor trip will occur.

Below P-10, if an Intermediate Range is affected by a loss of control power, the reactor will trip regardless of the affected channels bypass switch position.

Below P-6, if a Source Range is affected by a loss of control power, the reactor will trip regardless of the affected channels bypass switch position.

If the reactor trip from Source Range has been blocked on MC1 with the unit above P-6, then a Source Range control power failure will NOT result in a reactor trip. If unit goes below P-6, or the affected Source Range is manually unblocked below P-10, then the reactor will trip if the control power failure has not been corrected first.

If the reactor trip from Intermediate Range has been blocked on MC1 with the unit above P-10, then an Intermediate Range control power failure will NOT result in a reactor trip. If unit

Question 59

16.6 ENGINEERED SAFETY FEATURES

16.6-3 Inlet Door Position Monitoring System

COMMITMENT The Inlet Door Position Monitoring System shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3, and 4.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Inlet Door Position Monitoring System non-functional.	A.1.1 Verify the Ice Bed Temperature Monitoring System is FUNCTIONAL.	Immediately
	AND	
	A.1.2 Verify ice bed temperature is $\leq$ 27°F.	Once per 4 hours
	AND	
	A.1.3 Restore the Inlet Door Position Monitoring System to FUNCTIONAL status.	14 days
	OR	
	A.2 Restore the Inlet Door Position Monitoring System to FUNCTIONAL status.	48 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 4.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

The sump solution will have considerable depth, which inhibits the ready diffusion of hydrogen from solution. This retention of hydrogen in solution will have a significant effect in reducing the hydrogen yields to the containment atmosphere. The build-up of hydrogen concentration in solution will enhance the back reaction to formation of water and lower the net hydrogen yield.

Zirc - Water Reaction - The complete reaction of zirconium with steam can be characterized by the following:

$Zr + 2\ H_20 \rightarrow\ Zr0_2 + 2H_2 + Heat$

For every mole of zirconium reacted, one mole of zirconium dioxide and two moles of hydrogen are produced. Moreover, for each mole of zirconium reacted, two moles of steam are required. The reaction is exothermic, i.e., net energy (heat) is released.

Metal Water - Steel Steam Reaction

The presence of large amounts of stainless steel in a reactor vessel makes further consideration of the oxidation of steel by high temperature steam advisable. The mechanism of oxidation of steel is highly complex and several oxide forms are possible. As in the case of zirconium the reaction results in the generation of hydrogen and heat. The rate of oxidation of stainless steel is low at temperatures below approximately 1000°C but it becomes larger than that of zirconium at temperatures approaching the melting point of steel ~2550°F. During core heat-up and Zr clad reaction, high temperature (≈ 3000°F) steam and H_2 exit the core and heat-up thinner walled stainless steel components. Also heat radiation from the core contributes. During core heatup and zirc clad reaction, 150 pounds of hydrogen will be produced from metal-water reaction.

The DBA LOCA hydrogen analysis calculation (CNC-1552.08-00-0194, Revision 3) provides the technical details that support the Catawba UFSAR Section 6.2.5 (Combustible Gas Control in Containment). In this calculation it is documented that there are four potential sources of hydrogen during an accident -- (1) Zirc-Water reaction, (2) Dissolved primary system hydrogen, (3) Radiolysis - Core and Sump, and (4) Corrosion of metals - Zinc and Aluminum. Of these potential sources, dissolved primary system hydrogen is by far the least amount contributing to the overall hydrogen sources. In addition, from this calculation the **zirc-water reaction is assumed to contribute a maximum of 21,000 scf of hydrogen** throughout the accident based on the 10CFR50.46 criteria for evaluation of the ECCS, which requires zirc-water reaction be limited to 1% by weight of the total quantity of zirconium in the core.

HYDROGEN SOURCES

- Radiolysis of water
- Core
- Sump
- Zirc-Water reaction (highest production rate)
- Metal-Water (Stainless Steel/Aluminum)
- Paint
- Galvanized metal
- Concrete (decomposition if large core damage)
- Dissolved in coolant **Distractor Part 1**

HYDROGEN CONTROL

- Sampling - Hydrogen Analyzers
  - 2 trains
  - Read out in Control Room and Post Accident Sample Panel
- Hydrogen glow plugs
  - 72 plugs
  - 2 trains
- Hydrogen Recombiners
  - Maintain hydrogen concentrations <4% **Distractor Part 2**
  - Used in conjunction with skimmer fans
- Hydrogen purge
  - Used in conjunction with the VE system
  - Used only after consulting with the TSC

Question: 70
(1 point)

Given the following Unit 1 timeline:

1000
- The Unit has experienced a runback, from 100% power, following a trip of 1A CFPT

1003
- Main Turbine target load has been reached
- 1AD-2 A/9 (Control Rod Bank Lo Limit) illuminates

1005
- 1AD-2 B/9 (Control Rod Bank Lo-Lo Limit) illuminates
- Steam Dumps have closed
- Temperature Error meter indicates (+) 1.8° F

Entry into the Action Statement of Tech Spec 3.1.6 (Control Bank Insertion Limits) is FIRST required at _____(1)_____ .

Per the conditions provided at **1005,** OMP 1-7 _____(2)_____ state that control rods should be placed in MANUAL.

Which ONE of the following completes the statements above?

A. 1. 1003
   2. does

B. 1. 1003
   2. does NOT

C. 1. 1005
   2. does

D. 1. 1005
   2. does NOT

Question 61

7.7.1.3.3 Control Bank Rod Insertion Monitoring

When the reactor is critical, the normal indication of reactivity status in the core is the position of the control bank in relation to reactor power (as indicated by the Reactor Coolant System loop ΔT) and coolant average temperature. These parameters are used to calculate insertion limits for the control banks. Two alarms are provided for each control bank.

Distractor Part 1

1. The "low" alarm alerts the operator of an approach to the rod insertion limits requiring boron addition by following normal procedures with the Chemical and Volume Control System.
2. The "low-low" alarm alerts the operator to take immediate action to stop any dilution in progress. Shutdown margin is subsequently verified above the required minimum or boron is added, and the control bank(s) is restored above its insertion limit setpoint.

The purpose of the control bank rod insertion monitor is to give warning to the operator of excessive rod insertion. The insertion limit maintains sufficient core reactivity shutdown margin following reactor trip and provides a limit on the maximum inserted rod worth in the unlikely event of the hypothetical rod ejection, and limits rod insertion such that acceptable nuclear peaking factors are maintained. Since the amount of shutdown reactivity required for the design shutdown margin following a reactor trip increases with increasing power, the allowable rod insertion limits must be decreased (the rods must be withdrawn further) with increasing power. Two parameters which are proportional to power are used as inputs to the insertion monitor. These are the ΔT between the hot leg and the cold leg, which is a direct function of reactor power, and T_{avg}, which is programmed as a function of power. The rod insertion monitor uses parameters for each control rod bank as follows:

Correct Answer Part 1

$$Z_{LL} = A(\Delta T)_{auct} + B(T_{avg})_{auct} + C$$

where:

Z_{LL}	=	Maximum permissible insertion limit for affected control bank
$(\Delta T)_{auct}$	2nd Highest ΔT of all loops	
$(T_{avg})_{auct}$	2nd Highest T_{avg} of all loops	
A,B,C	= Constants chosen to maintain $Z_{LL} \geq$ actual limit based on physics calculations	

The control rod bank demand position (Z) is compared to Z_{LL} as follows:

$$\text{If } Z - Z_{LL} \leq D \text{ a low alarm is actuated}$$

$$\text{If } Z - Z_{LL} \leq E \text{ a low} - \text{low alarm is actuated.}$$

Since the 2nd highest value of T_{avg} and ΔT are chosen by auctioneering, a conservatively high representation of power is used in the insertion limit calculation.

Actuation of the low alarm alerts the operator of an approach to a reduced shutdown reactivity situation. Administrative procedures require the operator to add boron through the Chemical and Volume Control System. Actuation of the low-low alarm requires the operator to stop any dilution in progress. Shutdown margin is subsequently verified above the required minimum or boron is added, and the control bank(s) is restored above its insertion limit setpoint. The value for "E" is chosen such that the low-low alarm would normally be actuated before the insertion limit is exceeded. The value for "D" is chosen to allow the operator to follow normal boration procedures. Figure 7-19 shows a block diagram representation of the control rod bank insertion monitor. The monitor is shown in more detail on the functional diagrams shown in Figure 7-2,

Question 61

Attachment 11.1
General Statements of Philosophy

13. Manual Initiation of Safeguards Actions

ROs and SROs are expected to manually initiate safeguards actions if an automatic action setpoint is being approached, to avoid challenging the automatic safeguards function. An example of this is to manually initiate safety injection if pressure is decreasing in an uncontrolled manner to 1845 psig.

Exceptions to this philosophy are listed below:

- Do not initiate Phase B earlier than required. Procedural guidance is provided to initiate VX if containment pressure has not exceeded 3 psig and greater than 1 psig.

- During an ATWS, do not initiate S/I in "anticipation" of an S/I signal if the reactor will not trip, since this will cause a loss of CF flow to the S/Gs. This exception is stated in the APs that manually initiate S/I in "anticipation" of an S/I signal. (e.g. AP/1/A/5500/010 Enclosure 1)

The operator is expected to manually initiate any action which should have automatically occurred if the automatic function fails, such as the Safety Injection fails to initiate during an uncontrolled Reactor Coolant depressurization at 1845 psig (even during an ATWS) or an ECCS pump fails to start on a Safety Injection signal.

14. Placing Control Rods in manual following a load rejection

Following a load rejection/turbine runback, the control room crew should place control rods in manual once the steam dumps have closed. This will stop auto rod insertion with Tavg ~ 3°F higher than Tref. With the negative reactivity from the Xenon transient still causing Tavg to decrease, this should allow the crew the time to have a focus brief on borating the NC system and restoring rods above insertion limits (if necessary) without Tavg decreasing excessively below Tref.

Question 61

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. <u>Operator Actions</u>	
__ 1. **Verify turbine load - TRENDING DOWN IN AUTOMATIC.**	**Perform the following:** __ a. Select "MANUAL" on turbine control panel. __ b. Depress "CONTROL VALVES LOWER" pushbutton and reduce turbine load as required.
2. **Verify proper reactor response:** __ • Control rods - IN "AUTO" AND STEPPING IN __ • P/R neutron flux - TRENDING DOWN. Distractor Part 2	**<u>IF</u> <u>AT</u> <u>ANY</u> <u>TIME</u> T-Avg greater than 1.5°F higher than T-Ref, <u>THEN</u> perform the following:** __ a. Insert control rods as required to maintain T-Avg within 1°F of T-Ref. b. **<u>IF</u>** control rods will **<u>NOT</u>** insert, **<u>THEN</u>** perform the following: __ 1) Trip Reactor. __ 2) **<u>GO</u> <u>TO</u>** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Question 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- Reducing turbine generator load to stabilize vacuum is only effective when low vacuum is due to reduced RC cooling.

Correct Answer Part 1

- CA System is inoperable when available Condensate Storage System water temperature is greater than 136°F.

4. **Lower turbine load as required to stabilize vacuum as follows:**

Distractor Part 1

__ a. Verify vacuum loss due to reduced RC cooling.

a. Perform the following:

__ 1) **IF AT ANY TIME** vacuum loss due to reduced RC cooling, **THEN RETURN TO** Step 4.

__ 2) **GO TO** Step 5.

b. **IF** rapid power reduction required, **THEN** perform the following:

NOTE
- Any load reduction rate of greater than 25 MW/Min must be performed in manual mode.
- Unloading rates greater than 60 MW/Min (5%/minute) will meet C-7A interlock and may result in steam dump actuation.
- In manual mode, the control valves are capable of full travel within 3 minutes.

__ 1) Select "MANUAL" and "CONTROL VALVE LOWER" to reduce turbine load as required.

__ 2) **REFER TO** AP/1/A/5500/009 (Rapid Downpower).

__ 3) **GO TO** Step 5.

Question 62

NOTE If trend down in vacuum is rapid, using the reactor trip guidance for loss of CF pumps may be more appropriate.

1. **Reactor Trip Criteria:**

- **IF** reactor power greater than or equal to 69% **AND** main condenser vacuum trending down to 22 in. Hg in any condenser section imminent, **THEN** perform the following:

    a. Trip reactor. **Correct Answer Part 2**

    b. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

- **IF** in Mode 1 or 2 **AND** vacuum trending down to 16.9 in. Hg in both CF pump condensers imminent, **THEN** perform the following:

    a. Trip reactor.

    b. **WHEN** reactor trip verified, **THEN** trip CF pumps.

    c. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

2. **Turbine Trip Criteria:**

- **IF** reactor power less than 69% **AND** main condenser vacuum trending down to 22 in. Hg in any condenser section imminent, **THEN** perform the following:

    a. Trip turbine.

    b. **REFER TO** AP/1/A/5500/002 (Turbine Generator Trip).

- **IF** turbine impulse pressure less than or equal to 370 psig **AND** exhaust hood temperature trending up to 225°F imminent, **THEN** perform the following:

    a. Trip turbine.

    b. **REFER TO** AP/1/A/5500/002 (Turbine Generator Trip).

- **IF** turbine load less than or equal to 360 MWs **AND** main condenser vacuum trending down to 24.3 in. Hg in any condenser section imminent, **THEN** perform the following:

    a. Trip turbine. **Distractor Part 2**

    b. **REFER TO** AP/1/A/5500/002 (Turbine Generator Trip).

Question 63

A. Purpose

- **To verify proper response in the event of high activity in the NC System.**

B. Symptoms

- **Primary sample results indicate increasing activity level**
- **NC filter area monitors alarm - LIT:**
  - 1RAD-3, D/3 "1EMF 18 568 KK - 56 NC FILTER 1A"
  - 1RAD-3, D/4 "1EMF 19 568 KK,LL - 56 NC FILTER 1B".

Part 2 Correct Answer

- **1RAD-3, E/4 "1EMF 48 NC SAMPLE LINE REACTOR COOLANT" - LIT**
- **Dose Equivalent Iodine (DEI) has reached Action Level 3 limit as defined in AD-NF-ALL-0806, Nuclear Fuel Integrity.**

Distractor B.2 & D.2

5.2. 1EMF12 - Control Room

1EMF12 - Control Room	
Area Monitor - Measures the dose rates within the control room.	
Tech Spec or SLC	None
Power Supply	1RPA
Auto Actions	None
Notes	• Per the annunciator response immediate actions, all non-essential personnel should be evacuated because only 6 respirators are available for the control room personnel. • A valid reading of 15 mR/hr places the site in an Alert per AD-EP-ALL-0101.

5.3. 1EMF15 (2EMF4) - Spent Fuel Building Refueling Bridge **Distractor A.1 & B.1**

1EMF15 (2EMF4) - Spent Fuel Building Refueling Bridge	
Area Monitor - Measures activity over the fuel pool (for inadvertent criticality or fuel damage).	
Tech Spec or SLC	**SLC 16.7-10 (Radiation Monitoring for Plant Operations)** **Monitor 3: 1EMF-15, 2EMF-4) Fuel Storage Pool Area** **EMF Requirement:** 1 per unit with fuel in the fuel storage pool areas
Power Supply	1(2) RPA
Auto Actions	Automatically stops new fuel elevator from being raised
Notes	• Symptom for entry into AP/1(2)/A/5500/25 (Damaged Spent Fuel) • An unplanned lowering water level with corresponding rise in radiation levels places the site in an Unusual Event. • An unplanned valid Trip 2 alarm with damge to irradiated fuel resulting in a release of radioactivity places the site in an Alert

5.5. 1EMF18 and 19 (2EMF5 and 6) - Reactor Coolant Filter A and B

1EMF18 (2EMF5) - Reactor Coolant Filter A **1EMF19 (2EMF6) - Reactor Coolant Filter B** Correct Answer	
Area monitor - Measures dose rate next to the 1(2) A(B) reactor coolant filter housing which would result from activated particulate in the NCS or from particulate released from the NV demineralizers.	
Tech Spec or SLC	None
Power Supply	1(2) RPA
Auto Actions	None
Notes	• Symptom for entry into AP/1(2)/A/5500/18 (High Activity in Reactor Coolant) • Alarm setpoint set to ensure filter is removed from service and replaced before radiation level exceeds the shielding capacity of the transfer cask.

6.17. 1(2) EMF48- Reactor Coolant **Distractor A.1 & B.1**

1(2) EMF48- Reactor Coolant	
Process Monitor -Measures activity in the NC sample flow in the NM system. This EMF is NOT a process monitor depicted in Illustration 2 but is a GM detector mounted adjacent to the sample line. High activity would indicate a fuel cladding failure or crud burst**.**	
Tech Spec or SLC	**TS 3.3.3 (Post Accident Monitoring Instrumentation)** **EMF-48, NC System Monitor** **EMF Requirement:** 1 required channel in Modes 1, 2, and 3
Power Supply	1RPA
Auto Actions	None
Notes	• Dose Equivalent Iodine > 300µCi/cc is a Loss of the Fuel Clad Barrier per AD-EP-ALL-0101 Fission Product Barrier Matrix. During an event where fuel damage is possible, if 1(2) EMF-48 alarms, then the SRO should ensure Chemistry analysis is performed so the fission product barrier may be evaluated in a timely manner. • Symptom for entry into AP/1(2)/A/5500/018 (High Activity in Reactor Coolant)

Question 64

Given the following:

- Units 1 & 2 are at 100% RTP
- 1A RN pump is in service
- 1A and 2A KC trains are in service

Subsequently:

- The following Unit 1 annunciators are lit
  - 1AD-12 B/2 “RN PIT A Screen Hi D/P”
  - 1AD-12 B/1 “RN Pump Intake Pit A Level – LO”
  - 1AD-12 E/2 “RN Pit A Swap to SNSWP”
- The crew has entered AP/0/A/5500/020 (Loss of Nuclear Service Water), Case II (Loss of RN Pit Level)

Enclosure 2 (RN Valve Alignment for RN Swap to SNSWP) will direct the BOP to ensure _____(1)_____ is closed .

Following system stabilization, the BOP is directed to “Ensure KC Hx Outlet Mode Switches – Properly Aligned”. In response, the BOP _____(2)_____ required to reposition 1RN-291 (KC Hx 1A Outlet Throttle Valve) from its original alignment.

A. 1. 1RN-47A (RN Supply X-Over Isol)
   2. is

B. 1. 1RN-47A (RN Supply X-Over Isol)
   2. is NOT

C. 1. 1RN-48B (RN Supply X-Over Isol)
   2. is

D. 1. 1RN-48B (RN Supply X-Over Isol)
   2. is NOT

Original Question

MODIFIED

Question 64

2. **Ensure the following valves - CLOSED:**

__ • 1RN-1A (RN P/H Pit A Isol From Lake)

__ • 1RN-2B (RN P/H Pit A Isol From Lake)

__ • 1RN-5A (RN P/H Pit B Isol From Lake)

__ • 1RN-6B (RN P/H Pit B Isol From Lake)

__ • 1RN-53B (Station RN Disch Hdr X-Over)

__ • 1RN-54A (Station RN Disch Hdr X-Over)

__ • 1RN-57A (Station RN Disch To RL Sys)

__ • 1RN-843B (Station RN Disch To RL Sys)

__ • 1RN-49A (Non-Ess Supply Hdr Isol)

__ • 1RN-50B (Non-Ess Supply Hdr Isol)

__ • 2RN-49A (Non-Ess Supply Hdr Isol)

__ • 2RN-50B (Non-Ess Supply Hdr Isol)

__ • 1RN-847A (D/G 1A Hx Ret To Lake)

__ • 1RN-849B (D/G 1B Hx Ret To Lake)

__ • 2RN-847A (D/G 2A Hx Ret To Lake)

__ • 2RN-849B (D/G 2B Hx Ret To Lake).

__ 3. **IF RN supply header in single supply header operation, THEN return to procedure and step in effect.**

4. **IF RN Pit A initiated auto or manual swap to SNSWP, THEN ensure the following valves closed:**

__ • 1RN-48B (RN Supply X-Over Isol)

__ • 2RN-48B (RN Supply X-Over Isol).

15.2 Emergency Low Pit Level

Objective 16B, All

The Emergency Low Pit Level provides automatic transfer of the suction source for RN from Lake Wylie to the SNSWP. Upon receipt of an emergency low level actuation, the isolation valves from the SNSWP and Lake Wylie open and close, respectively.

The emergency low level setpoint only applies to the safety related instruments (three channels in each pit):

- 1/3 channels in either pit actuates a computer alarm.
- 2/3 channels in either pit actuates an annunciator alarm, actuates a computer alarm, and separates the RN loops (except through1RN-36A and 1RN-37B), isolates RN from Lake Wylie, and aligns RN to the SNSWP.

Actions interlocked to either pit:

- Start all 4 RN pumps (1A, 1B, 2A, and 2B). **Correct Answer Part 2**
- Opens
  - RN Pumphouse suction valves from SNSWP.
  - RN Return Header valves to the SNSWP
  - DG HX Returns to the SNSWP (Both units)
- Closes
  - RN Pumphouse Pit Isolations from the Lake
  - RN Non-essential Supply Isolations (both units)
  - Station RN Discharge Header Cross-Overs
  - Station RN Discharge to RL System
  - D/G HX Returns to RL System (Both units).

Actions interlocked to Pit A instruments only:

- Closes the opposite train RN Supply Cross-Over Isolations (Both units)

Actions interlocked to pit B instruments only: **Correct Answer Part 1**

- Closes the opposite train RN Supply Cross-Over Isolations (Both units)

L. Manual Preaction Systems (Turbine Bearing Deluge Systems)

1. The Main Turbine Bearings are protected by a Manual Preaction sprinkler system consisting of the following:
   a) Isolation (control) valve
   b) Mulsifyre clapper (deluge) valve
   c) Supervisory air supply (VS)
   d) Dry piping distribution system
   e) Closed sprinkler heads

Correct Answer Part 2

2. The purpose of this type of system is to deliver large quantities of Fire Protection (RF) water to totally engulf equipment while reducing the potential of equipment damage due to inadvertent discharge of water. (**Obj. #20**)
3. The Isolation and Mulsifyre Deluge valves are the same type used in the Mulsifyre (automatic water spray) systems discussed earlier; however, the actuation of the deluge valve is accomplished manually and not by the heat detectors. Also, the sprinkler heads are not open, they are the same as those used in the Automatic Wet Pipe Sprinkler Systems.
4. The thermal detectors associated with the Main Turbine Bearings fire protection system only provide an alarm to the Control Room. Upon receipt of the alarm it can be decided if the deluge valve should be opened. Opening the deluge valve can be accomplished from the remote pull stations (Control Room or next to MTOT), or from the local pull station at the valve. Opening the deluge valve allows water to enter the distribution piping, however water will not be discharged unless sufficient heat is present at the sprinkler heads to cause the heat sensitive elements to have been destroyed. (**Obj. #21, 22**)

Correct Answer Part 1

5. The distribution piping is normally pressurized by the Station Air (VS) System. Pressure is monitored, and an alarm is provided when pressure is low indicative of a fused sprinkler head or leak in the piping. Again, upon receipt of the alarm, it can be decided if the deluge valve should be opened (as described above). (**Obj. #23**)

M. Automatic Pre-action System (Waste Solidification Building)

1. The Waste Solidification Building is protected by an Automatic Preaction sprinkler system consisting of the following:
   a) Isolation (control) valve
   b) Mulsifyre clapper (deluge) valve
   c) Supervisory air supply (VS)
   d) Dry piping distribution system
   e) Closed sprinkler heads
2. In order to reduce the potential of freezing the water in the Waste Solidification Building fire protection system piping, an Automatic Preaction system is utilized.

Question 66

5.6 Standing Instructions (continued)

2. In a timely manner, all Operations personnel review and initial, acknowledging they understand the issued Standing Instruction, unless exempted (see below).

   a. Operations personnel **NOT** on shift will review new Standing Instructions within seven days from issuance or within seven following the return to work (e.g., vacation, extended time off).

   b. All shift Operations personnel reviews issued Standing Instructions before taking the watch. This may occur shortly after when allowed by the CRS or Shift Manager.

   c. If the Standing Instruction only impacts certain individuals in Operations, the SM can allow an exemption for that Standing Instruction review (e.g., a Standing Instruction that discusses EAL classification may not be applicable to AO's and the SM can waive their requirement to review).

   d. The required population or population exempted to review a Standing Instruction shall be documented in Attachment 1, Standing Instructions.

   e. A template for Standing Instruction review is provided in Attachment 16, Standing Instruction / OSIP Review Sheet

3. If any of the following are issued to provide additional monitoring criteria for plant operation, then communicate the guidance to the shifts through Standing Instructions:

   - External department guidance documents that will be in place for greater than 72 hours
   - Operational Decision Making (ODM) Evaluations that do not have an associated Adverse Condition Monitoring Plan (ACMP)
   - NCR immediate or interim corrective actions that warrant communication to the shift

4. Standing Instructions shall **NOT** be used to alter, change, or compensate for lack of appropriate procedures.

5. Standing Instructions shall **NOT** conflict with or be used as a replacement for Operating Procedures or other existing procedures.

Question 66

5.6 Standing Instructions (continued)

6. Standing Instructions shall **NOT** conflict with or be used in place of an incorrect or inaccurate procedure.

   a. If an Operating Procedure is technically inaccurate, then place the procedure on hold in accordance with AD-DC-ALL-0201, Development and Maintenance of Controlled Procedure Manual Procedures.

   b. If an EOP or AOP has an identified Deviation per AD-OP-ALL-1001, Conduct of Abnormal Operations, then a Standing Instruction can be used to inform the crew until the EOP or AOP is changed.

7. Standing Instructions are written and prepared by any knowledgeable person and are approved and signed by Operations management (SM or higher).

8. Use Attachment 1, Standing Instructions to develop a Standing Instruction.

   a. Attach supporting documentation as needed (e.g., NCRs, External Department Guidance) for clarification.

   b. Enter actions required by the shifts (e.g., additional monitoring, controls in manual) in the "Required Actions" section of Attachment 1.

9. Use Attachment 2, Standing Instruction Index to maintain an index of Standing Instructions.

10. Standing Instructions shall be numbered in consecutive order based on the current year (e.g., 17-001, 17-002, 17-003).

11. Attachment 1, Standing Instructions, shall indicate each unit to which the Standing Instruction is applicable.

    a. For a shared item (e.g., common system), a copy of the Standing Instruction shall be placed at each unit that is checked.

12. If a change to the information or required actions is necessary, then revise the Standing Instruction.

    a. Revisions to Standing Instructions can be made by a knowledgeable person.

    b. Indicate updates by adding the next revision number to the Standing Instruction Number (e.g., 07-01 Rev 1).

    c. If a revision to a Standing Instruction is issued, then Operations Management (SM or higher) reviews and signs the revision as if it were a new Standing Instruction.

Question 67

4.4.4 Senior Reactor Operators (continued)

7. Delegates implementation of reactivity plans to the Reactivity Manager when the Reactivity Manager position is staffed, but must maintain broad oversight of the plant.

4.4.5 Reactivity Manager

1. Provides dedicated oversight during Reactivity Manipulations to ensure that the expected reactor response is obtained, applicable error reduction tools are used and Control Room distractions are minimized. {7.1.2} {7.1.6} {7.1.7}

2. Attends pre-job brief and ensures it is conducted prior to manipulation of reactivity controls.

3. Ensures core parameters are maintained within prescribed limits.

4. [BWR] Ensures the reactivity effect of any reactor pressure and feedwater temperature adjustments are understood during control rod and recirculation pump manipulations. {7.1.2} {7.1.6} {7.1.7}

5. Ensures reactivity manipulations are performed by dedicated reactor operators with no concurrent duties in a careful, deliberate manner and expected results are verified by redundant indications. {7.1.2} {7.1.6} {7.1.7}

6. Assumes responsibility for receipt of alarms directly associated with the reactivity manipulation.

7. Oversees manipulation of control rods, Reactor Coolant System pressure, feedwater parameters and [BWR] recirculation flow.

4.4.6 Reactor Operators

1. [BWR] Maintain cognizance of plant conditions that are near the Scram Avoidance Region (OPRM operable) or Region II (OPRM inoperable).

2. Assume no other duties while supervising trainees in the operation of controls that affect reactivity. The trainee and reactor operator are considered one operator. A second licensed operator is required to perform peer checks for reactivity manipulations performed by a trainee/RO combination.

3. Are not involved in any potentially distracting activities during reactivity manipulations involving movement of control rods, [PWR] RCS Makeup System or [BWR] recirculation flow. Complete attention shall be given to proper setup and operation of the reactivity control system and monitoring reactor response. {7.1.1} {7.1.2} {7.1.6} {7.1.7} [7.3.4]

4. Inform the CRS of any unexpected core parameter or reactivity change.

Question 67

5.4.7 Abnormal Operations (continued)

5. Any time reactor power unexpectedly changes from steady state conditions, the OATC is expected to:

   a. Communicate to the CRS the initial alarm followed by reactor power level.

   b. Continuously monitor reactor power throughout the transient.

6. The OATC shall place appropriate control stations in manual if the plant response is not valid as indicated by any of the following:

   - NI power increasing above the pre-transient power level
   - Failed instrumentation is diagnosed
   - Invalid input exists and SRO directs control systems be placed in manual

7. Peer-checks during Abnormal Operations:

   a. Peer-checks are not required for reactivity manipulations during performance of Abnormal and Emergency Procedures unless additional manpower is available.

   b. The CRS is prohibited from performing peer-check of reactivity related manipulations.

   c. Peer-checks are not required when responding to a failure requiring manipulation of Reactivity Management related components.

8. [PWR] During abnormal conditions requiring manual control of the reactor, ROs shall inform the CRS of the following (notification to the CRS is not required prior to taking the action):

   - The first manual control rod insertion
   - All manual control rod withdrawals
   - The first manual adjustment made to the Turbine
   - The first manual adjustment made to Feedwater
   - Any water addition made to the VCT/LDST
   - No further communications of manual adjustments are necessary. When normal operation is resumed, communications return to normal operational requirements for reactivity changes.

Question 68

_____ 3.1.6 Slowly open 1AS-4 (Main Steam to Aux Steam Header Control Bypass) (TB-610, 1M-32).

_____ 3.1.7 Verify Unit 1 SM Header pressure is approximately equal to AS Header Pressure.

NOTE: If turbine prewarming is **NOT** to begin at this time, continue with the procedure and sign off Steps 3.1.8 and 3.1.9 when ready to begin prewarming the high pressure turbine shell.

_____ 3.1.8 Verify turbine initial conditions contained in the turbine generator startup enclosure of OP/1/B/6300/001 (Turbine-Generator) have been signed off.

CAUTION: Do **NOT** exceed 50 psig in the high pressure shell while doing Step 3.64 until PZR pressure is greater than 1945 psig (PZR Low Pressure Rx Trip Setpoint, P-11). This is to prevent P-13 (57 psig impulse pressure) enabling P-7 to unblock the Lo Power Reactor Trips (PZR High Level, PZR Low Pressure, Low NC Flow on 2/4 Loops, NC Pump Undervoltage, NC Pump Underfrequency).

NOTE: Step 3.64 may be done at anytime during plant heatup to 2235 psig/557°F. The purpose of this step is to use any extra steam available to start warming the high pressure shell of the turbine when the NC System heatup is being performed with the MSIV's open. The warming does **NOT** have to be at 60-100 psig as required by OP/1/B/6300/001 (Turbine-Generator). This warming is to reduce the amount of time required before placing the Turbine-Generator on the line.

_____ 3.64 **IF** MSIVs are open **AND** S/G pressure > 100 psig, begin warming of the turbine per OP/1/B/6300/001 (Turbine-Generator). (R.M.)

NOTE: Step 3.65 is designed to supplement the AS header being supplied by AEB(s) with Unit 1 SM to aid in maintaining both Unit 1 and 2 steam seals. This additional steam source will also aid in supplying AS to the Unit 1 CFPT that is in service. (R.M.)

_____ 3.65 **IF** AEB(s) are supplying Unit 1 AS **AND** it is desired to supplement this steam supply with Unit 1 SM during heatup perform following:

_____ 3.65.1 Ensure SM pressure is greater than or equal to 500 psig.

_____ 3.65.2 Verify 1AS-2 (Main Stm To Aux Stm) is closed. (1MC13)

_____ 3.65.3 Open 1AS-1 (SM To AS Inlet). (1MC13) **Distractor Part 1**

_____ 3.65.4 Notify Operator(s) at the AEB(s) that AS on Unit 1 will be supplemented by SM.

_____ 3.65.5 Adjust 1AS-2 (Main Stm To Aux Stm), in manual, as required to match steam pressure being supplied by the AEB(s) (140 to 165 psig on OAC Pt C1A0962 (As Crosstie Pressure)). (R.M.)

NOTE: Step 3.65.6 provides assurance that SM from Unit 1 is providing expected results.

_____ 3.65.6 Adjust 1AS-2 (Main Stm To Aux Stm) to obtain a slight increase in AS Header pressure as seen on OAC Pt C1A0962 (AS Crosstie Pressure)). (R.M.)

Question 68

5.1.2 Unplanned Reactivity Manipulations

1. A reactivity manipulation that does not allow time to establish the required controls for reactivity evolutions as outlined in this procedure (R1, R2, R3).

2. The required controls for planned reactivity evolutions are not applicable during emergency conditions.

3. Operators will place the plant in a stable, known safe condition and reestablish formal reactivity control as soon as practical following the event.

5.2 Reactivity Manipulation Significance

5.2.1 Major Reactivity Manipulation (R1 Activities)

1. A planned activity that significantly affects core power or reactivity requiring the highest level of operator attention and dedication.

2. Refer to Attachment 2, Planned Reactivity Evolution Category Examples, to assist with determination of the evolution category.

3. Approved Reactivity Manipulation Plan per AD-NF-ALL-0201, Reactivity Management Plan Development {7.1.2}

   a. A Reactivity Manipulation Plan is not required for Reactivity Manipulations made for Beginning-Of-Cycle (BOC) Startup and Power Escalation in accordance with site-specific procedures and AD-NF-ALL-0201, Reactivity Manipulation Plan Development.

4. Pre-job Briefing that includes pertinent OE

5. Operations Manager or designee present in the Control Room to provide management oversight (Shift Manager can not be the designee).

6. Shift Manager present during critical steps to provide oversight.

7. Shift Technical Advisor (STA) present in the Control Room, until all R1 activities are complete, the plant is stable, and SM concurrence is obtained.

8. Reactor Engineering support as follows:

   a. Development of Reactivity Manipulation Plan

   b. Control Room presence for power changes greater than 15%

9. Reactivity Manager oversight (Dedicated SRO, other than the CRS or STA, with no concurrent duties) {7.1.2} {7.1.6} {7.1.7}

Question 68

5.2.1 Major Reactivity Manipulation (R1 Activities) (continued)

10. Dedicated RO with no concurrent duties (can be one of the unit reactor operators) {7.1.2} {7.1.6} {7.1.7}

11. All unrelated parallel Control Room activities shall be evaluated for distractions to the Control Room Crew and approved by the Shift Manager and the Reactivity Manager prior to commencement (minimal impact to control room team, no excessive alarms). {7.1.2} {7.1.6} {7.1.7}

12. Specific monitoring requirements identified

13. Signage or barriers use to limit Control Room access {7.1.2}

14. Consider JITT {7.1.6}

5.2.2 Reactivity Manipulation (R2 Activities)

Distractor Part 2

1. A planned activity that involves routine changes to core reactivity requiring dedicated operator attention.

2. Refer to Attachment 2, Planned Reactivity Evolution Category Examples, to assist with determination of the evolution category.

3. Pre-job briefing that includes pertinent OE

4. Dedicated RO with no concurrent duties (can be one of the unit Reactor Operators) {7.1.2} {7.1.6} {7.1.7}

5. Reactivity Manager oversight (Can be CRS with no concurrent tasks) {7.1.2} {7.1.6} {7.1.7}

6. Reactor Engineering supports as follows:

   a. Development of a Reactivity Manipulation Plan (if requested by Operations)

   b. [BWR] Predicts of power changes

   c. Development of Control Rod Movement Sheets (if utilized)

7. Unrelated parallel Control Room activities shall be suspended during actual component manipulations or a dedicated SRO shall be established for the activity (activities may resume during R2 activity monitoring period) {7.1.2} {7.1.6} {7.1.7}

   a. Deviation to the unrelated parallel activity can be made on a case by case basis with the concurrence of the Reactivity Manager/CRS

Question 68

<< Planned Reactivity Evolution Category Examples >>

1. **R1 Evolutions:**

   a. Activity significantly affecting core power/reactivity which requires significant operator attention. Examples include but are not limited to:

      (1) Zero Power Physics Testing

      (2) Reactor startups

      (3) Power changes greater than or equal to [PWR] 10%, [BWR] 15%

      (4) Reactor Shutdown

      (5) [BWR] Control Rod Sequence Exchange

2. **R2 Evolutions:**

   a. Activity that affects Core Reactivity or the normal control of it, but is of minor effect. Examples include but are not limited to:

      (1) Removing/Restoring Steam Generator Blowdown

      (2) [PWR] Saturating an unsaturated demineralizer

      (3) Power changes less than [PWR] 10% [BWR] 15%

      (4) [PWR] Control Rod Drive Movement Tests

      (5) [PWR] Routine adjustments in boron concentration

      (6) [BWR] Adjustments to Reactor Recirculating pump including 'Preconditioning Ramp'

      (7) Evolution requiring reactivity adjustments while automatic controllers of reactivity components or systems are in manual

Question 69

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR for four loop operation; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.14 for the WRB-2M CHF correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080 degrees F, decreasing 58 degrees F for every 10,000 MWd/mtU of fuel burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

Question 69

BASES

BACKGROUND (continued)

The high and intermediate head subsystems of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a safety injection actuation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a small break LOCA and there is a high level of probability that the criteria are met following a large break LOCA:

Distractor Part 1

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

Question 70

5.5.7 Critical Parameters (continued)

4. Critical Parameters shall be monitored using multiple indications (if available).

   a. Backup or redundant instrumentation is available to monitor the Critical Parameter. {7.1.10}

5. As time permits, Critical Parameters should be displayed in the Control Room and contain the following information:

   - Parameter (Critical Parameter) being monitored
   - Control band or single parameter value
   - Reporting frequency

6. As time permits, announce Critical Parameter assignment in a Crew Update or brief to ensure that the entire crew understands the Critical Parameter and owner.

7. Critical Parameters under manual control shall receive heightened crew awareness and prioritization.

   a. The CRS shall consider reassigning any additional transient actions to other Operators not assigned to Critical Parameter manual control.

8. If a Critical Parameter has been or is expected to be exceeded, then inform the CRS (a Crew Update is preferred) and take actions as prescribed when the Critical Parameter was established.

5.5.8 Deleting Computer Alarms

1. Ensure deleted OAC computer alarms are logged in the eSOMS narrative log, unless being deleted per an approved procedure that will restore them at the conclusion of the procedure.

   a. Until the alarm is restored, the eSOMS log entry should be an open log entry.

2. All computer alarms are to be initially believed and a thorough review/response shall be performed to verify that indications are valid and proper actions are implemented.

Question 70

5.5.8 Deleting Computer Alarms (continued)

4. A weekly audit of deleted computer alarms will be performed to ensure that the issue is captured in an active Work Order/Work Request with appropriate priority assigned.

5.5.9 Guidelines for Bypassing of Safety Systems

1. Safety systems must be allowed to perform their automatic function when required for transient mitigation.

2. Safety systems must not be bypassed before automatic actuation, except as follows:

   a. Safety systems may be bypassed when directed by operating procedures for normal plant cooldown or when directed by procedures for testing.

   b. Safety systems may be bypassed when directed by Event Procedures for specific transients.

3. Equipment automatically actuated by a safety system must not be repositioned except as follows:

   a. Equipment may be overridden and repositioned when directed by Event Procedures for specific transients.

   b. If both of the following conditions are met and SRO approval is obtained, then equipment may be overridden and repositioned outside of the procedures:

      (1) The Safety System is not required to perform its intended safety function.

      (2) Continued operation of the Safety System could elevate the severity of the transient, damage equipment, or cause unnecessary operator burden.

   c. If a safety system has been bypassed or overridden, the operator assumes the responsibility to reactuate the system if necessary for transient mitigation.

Question 70

Saturday Day Shift Routine Activities for Unit 1

	Perform the following PT(s)/OP(s):
	• **Semi-Daily Surveillance (PT/1/A/4600/002) PIP C-94-0738, CSD 3.0.24 (Scope 2.1)**
	• **Drain System Analyzer Rack Traps (OP/0/A/6500/115)**
	• **Any AMT has been on HFS for >/= 3days Sample and conduct LWR (OP/0/B/6500/060)**
	• **Any AMT full and HFS is available then place AMT on HFS (OP/0/B/6500/061)**
	Perform the following weekly:
	• **Perform applicable schedule review (T-3 if on days after training, T-8 if on nights after training)**
	• **Review and Walk Down Online Clearances for the two weeks of weekday nights after training. Clearances are located in the OPS Online Office Bins.**
	• **Review the Active ODM's with the shift including trigger points and current trends.**
	Perform the Following as required:
	• **On first Saturday of the quarter, perform inspection of OPS Storage Areas, ensure all equipment properly stored.**
	• **Lower Personnel Airlock Leak Rate Test (PT/1/A/4200/001 F), if required to clear an active LCO tracking entry.**
	• **On the third Saturday of each month, perform a control board W/O, W/R Sticker audit. Print ST1613 OStick Audit Report from "Nuclear Reports – MS Reporting Services 2008" on the DAE. Ensure OOS stickers for Unit 0 & 1 match with this report (ie – remove sticker for completed WR/WO, ensure items on report have sticker). OAC points on the report do NOT need to be reviewed as part of this audit. Notify OWPG OPS Onlines of any discrepancies that cannot be resolved.**

Distractor Part 2

Individual items are to be initialed when complete.
Ensure any PT not performed an NCR is generated, LCO Tracking Entry Made, and notify SM (or designee).
Generate NCR (if needed) for item(s) not completed.
Reference SOMP 1-13

Question 71

5.1 Emergency Exposure Determination (continued)

NOTES

- Emergency exposure limits are exclusive of current occupational exposure.
- Only one emergency exposure is allowed per lifetime.
- Dose to the lens of the eye is limited to three times the listed value.
- Dose close to other organs, including skin and body extremities, is limited to ten times the listed value.

Table 1, Emergency Exposure Limits

TEDE Limit (Rem)	Activity
5	All activities during the emergency. Distractor Part 1
10	Protecting valuable property when lower dose is not practicable.
25	Lifesaving or protection of large populations when lower dose is not practical per EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.
Greater Than 25	Lifesaving or protection of large populations, only if individuals receiving exposure is a volunteer, and fully aware of risks involved.

6. If emergency exposure is needed and is estimated to be greater than 5 Rem but less than 25 Rem, then identify individual(s) who will perform the activity.

   a. Complete sections A and B of Attachment 1, Emergency Exposure Authorization.

   b. Brief the individual(s) on the activities to be performed and inform personnel of their assigned Emergency Exposure Limit.

Correct Answer Part 1

Question 72

5.3.5 Dose Monitoring Warning Flags and SRD Alarms (continued)

2. SRD Dose and Dose Rate Alarms:

   a. SRD alarms are provided to help limit dose and keep dose ALARA.

   b. SRDs are programmed to alarm at a predetermined dose and dose rate during the log-on process.

   c. The alarm set-points are specified by the RWP.

   d. The dose alarm consists of an audible alarm and a visual alarm.

      (1) When the dose set-point is exceeded, then an audible alarm and a red light will flash on the SRD.

      Correct Answer Part 2

      (2) When the worker logs out at the ACS, then the audible alarm and the flashing red light will **NOT** stop until the SRD is reset.

   e. The dose rate alarm consists of an audible alarm and a visual alarm.

      (1) Expect an audible alarm and a red light to flash on the SRD when the dose rate set-point is exceeded.

      (2) The dose rate alarm automatically resets when the dose rate drops below 80% of the dose rate alarm set-point.

   f. Other Alarms:

      Distractor Part 2

      (1) The SRD is programmed to alarm when SRD is activated for 16 hours or when RWP specific stay time is exceeded.

      (2) The SRD is programmed to alarm on low battery.

   g. SRD Dose Alarm Response:

      (1) Workers are expected to immediately inform co-workers, exit the RCA, log out at the ACS, and call RP upon receiving an SRD dose alarm. Re-entry is **NOT** permitted until the alarm is cleared by RP.

Question 72

5.3.5 Dose Monitoring Warning Flags and SRD Alarms (continued)

h. SRD Dose Rate Alarm Response:

(1) Work can continue following a travel path dose rate alarm providing the alarm clears prior to arriving at the work location.

Distractor Part 1

(a) RP will discuss this expected alarm during RWP briefings.

(2) Workers are expected to stop work, exit the area and notify RP upon receiving a third anticipated dose rate alarm unless otherwise directed by RP.

(3) Workers are expected to stop work, exit the area, and notify RP immediately for unexpected dose rate alarms (any dose rate alarm that is **NOT** briefed by RP prior to beginning work).

Correct Answer Part 1

i. SRD malfunctions:

(1) Workers are expected to immediately exit the RCA and notify RP about a malfunctioning SRD.

3. Alternate alarm indicators will be provided to those workers who self identify as having trouble hearing audible SRD alarms.

4. SRD dose-rate alarms may be expected by RP due to higher dose rates in the travel path to the work location or a worker being in close proximity to a radiation source.

5. Anticipated dose rate alarms are discussed during the RP brief prior to beginning work.

Question 73

During an emergency event:

The on-site emergency facility that assumes responsibility for communications with offsite agencies including the NRC once it is activated is the _____(1)_____.

The MINIMUM level of emergency classification that always REQUIRES an evacuation of all non-essential personnel from the site is a _____(2)_____.

Which ONE (1) of the following completes the statements above?

A. 1. Technical Support Center (TSC)
   2. Site Area Emergency

B. 1. Technical Support Center (TSC)
   2. General Emergency

C. 1. Operations Support Center (OSC)
   2. Site Area Emergency

D. 1. Operations Support Center (OSC)
   2. General Emergency

Original Question

MODIFIED

- Provide Protective Action Recommendations (PARs)
- Provide Offsite notifications
- Continued assessment of actual or potential consequences both on-site and off-site throughout the evolution of the emergency condition.
- Effective implementation of emergency measures in the environs including protective actions for affected areas, implementation of emergency monitoring teams and facilities to evaluate the environmental consequences of the emergency condition, prompt notification and communications with of-site authorities
- Continued maintenance of an adequate state of emergency preparedness until the emergency situation has been effectively managed and the station is returned to a normal or safe operating condition.

- The following EC responsibilities cannot be delegated:
  - Classification of the Event
  - Emergency Dose Extensions

  The following are the responsibility of the EC, until EOF activation:
  - Offsite Notifications
  - Protective Action Recommendations (PARs)

4. Emergency Classification

Guidance provided in AD-EP-ALL-0101 (Emergency Classification) and CSD-EP-CNS-0101-02 (EAL Wallcharts)

Objective #1 & 4, ALL

4.1 **Emergency Classification Levels (ECL):**

- Unusual Event (UE): (Least Severe) Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.
- Alert: Events are in progress, or have occurred, which involve an actual or potential substantial degradation of the level of safety of the plant OR a security event that involves probable life-threatening risk to site personnel or damage to site equipment because of hostile action. Any releases are expected to be small fractions of the EPA Protective Action Guideline exposure levels.
  - Requires activation of the OSC, TSC, EOF
- Site Area Emergency (SAE): Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the SITE BOUNDARY.

Question 73

  - Requires activation of the OSC, TSC, and EOF
  - Requires a Site Assembly for personnel accountability
  - Requires a Site Evacuation **Correct Answer Part 2**
- General Emergency (GE): (Most Severe) Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile actions that result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.
  - Requires activation of the OSC, TSC, and EOF
  - Requires a Site Assembly for personnel accountability
  - Requires a Site Evacuation **Distractor Part 2**

4.2 Event Classification

Objective #2, LPSO/LOR

The time between when indications exist that an EAL threshold has been exceeded and declaration of the event will not exceed 15 minutes, unless extraordinary conditions prevail. AD-EP-ALL-0101 (Emergency Classification) will direct classification using the EAL Wallchart.

The EAL classification information is formatted as follows:

The first letter indicates the EAL category:

- **R** – Abnormal Radiation Levels or Radiological Effluents
- **H** – Hazards
- **E** – ISFSI (Dry Cask Fuel Storage)
- **S** – System Malfunctions (Hot Conditions)
- **F** – Fission Product Barriers (Hot Conditions)
- **C** – Cold Shutdown / Refueling System Malfunctions (Cold Conditions)

The second letter indicates the emergency level:

- U – Unusual Event
- A – Alert
- S – Site Area Emergency
- G – General Emergency

Last is a number in #.# format, with the first being the subcategory and the second being the selected EAL in that subcategory.

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

1	2	3	4	5	6	DEF

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

For example, the EAL HA1.2 indicates a classification in the Hazard Category at an Alert emergency level. Subcategory 1 represents a Security event and EAL 2 indicates an NRC-validated aircraft attack threat within 30 minutes of the site as indicated on the Wallchart.

- Complete accountability of site personnel within 30 minutes of a Site Area Emergency or General Emergency declaration or the decision to conduct accountability.
- Notification of event declaration to the NRC is required 'as soon as possible' following notification to State and Counties, but no later than 60 minutes after an event declaration.
- Activation of the NRC Emergency Response Data System (ERDS) data link is required within 60 minutes of an Alert or higher event declaration.
- Follow-up Notifications to the state(s) and local agencies are to occur within approximately 60 minutes from the first contact of the previous notification until a new time period is agreed upon by all offsite agencies

5.2 AD-EP-ALL-0111 INITIAL ACTIONS

Objective #5, LPSO/LOR

5.2.1 Immediate Actions

- If due to a Security threat or Hostile Action:
  - If an airborne threat, maintain continuous communications with NRC
  - If an EAL has been exceeded, then classify the event
  - If security-based event, notify the NRC and if <30 minutes away, perform a rapid evacuation of plant personnel
- Offsite Communicator initiates actions as directed

5.2.2 Action if EAL Exceeded

- Within 15 minutes: Classify and Declare the event
- If General Emergency, evaluate PARs
- If Site Area Emergency or General Emergency, take action to ensure onsite protective actions are complete within 30 minutes
  - Site Assembly required for Site Area Emergency or higher

5.2.3 Actions within 5 Minutes of Event Declaration

- If an Alert or Higher is declared, activate the ERO. **Correct Answer Part 1**
  - OSC, TSC, EOF activation
  - ERO can be activated for unusual event at EC discretion
- Make Plant Announcement
- Evaluate on-site protective actions

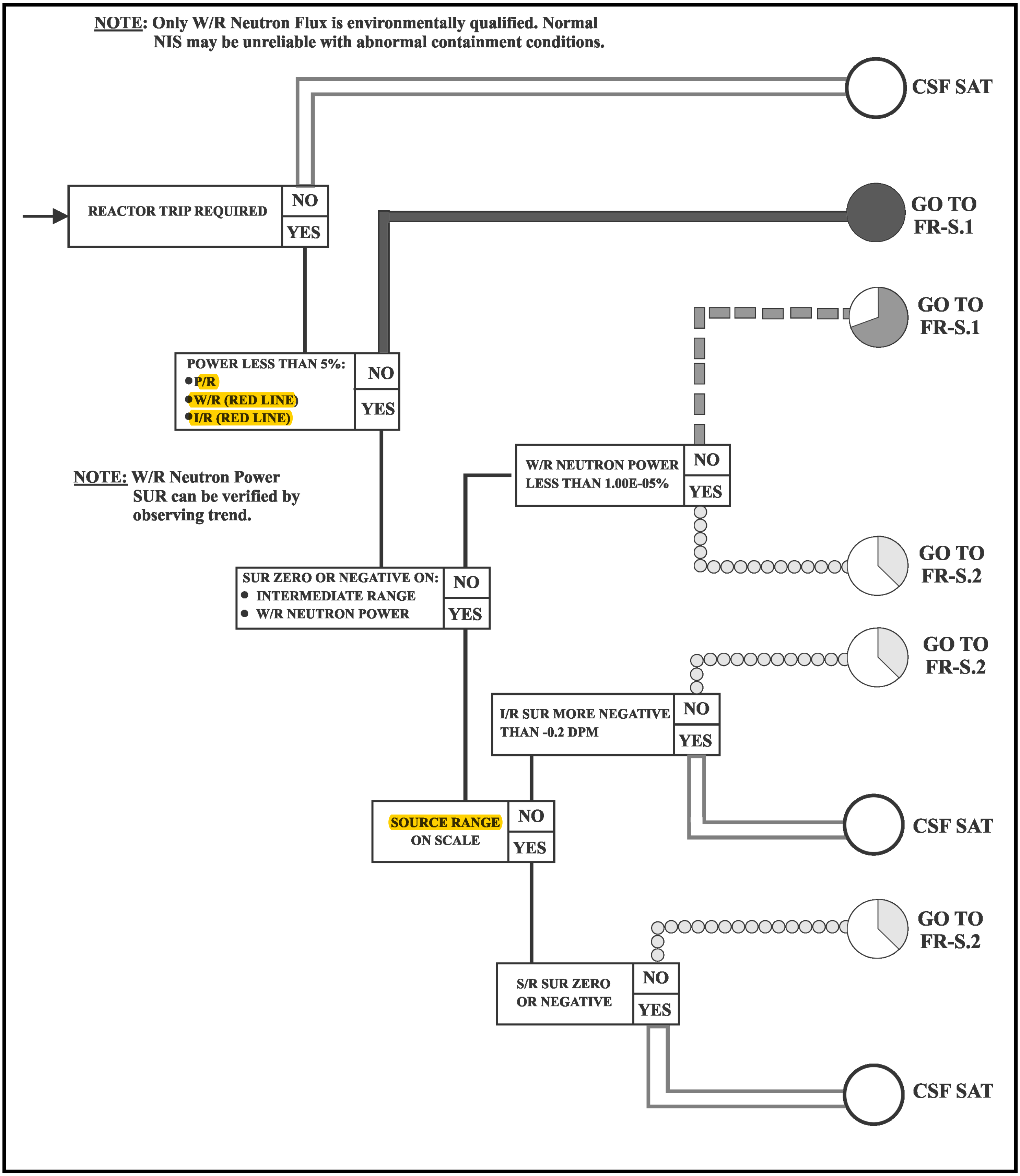
Distractor Part 1

5.2.4 Actions within 15 Minutes of Event Declaration

- Fill out Emergency Notification Form and provide to Offsite Communicator for notification
- If release is in progress, direct performance of a dose assessment
  - If General Emergency, make follow-up notification within 30 minutes to include dose assessment and meteorological data

Question 74

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES SUBCRITICALITY - Page 1 of 1	PAGE NO. 3 of 11 Revision 9



Question 74

Table 3.3.3-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS
1.	Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	2	B,D,F,H
2.	RCS Cold Leg Temperature (Wide Range)	2	B,D,F,H
3.	RCS Pressure (Wide Range)	2	B,D,F,H
4.	Reactor Vessel Water Level	2	B,D,F,H
5.	Containment Sump Water Level (Wide Range)	2	B,D,F,H
6.	Containment Pressure (Wide Range)	2	B,D,F,H
7.	Containment Area Radiation (High Range)	1	B,D
8.	Not Used		
9.	Pressurizer Level	2	B,D,F,H
10.	Steam Generator Water Level (Narrow Range)	2 per steam generator	B,D,F,H
11.	Core Exit Temperature - Quadrant 1	2[(a)]	B,D,F,H
12.	Core Exit Temperature - Quadrant 2	2[(a)]	B,D,F,H
13.	Core Exit Temperature - Quadrant 3	2[(a)]	B,D,F,H
14.	Core Exit Temperature - Quadrant 4	2[(a)]	B,D,F,H
15.	Auxiliary Feedwater Flow	1 per steam generator	C,D,E,H
16.	RCS Radiation Level	1	B,D
17.	RCS Subcooling Margin Monitor	2	B,D,F,H
18.	Steam Line Pressure	2 per steam generator	B,D,F,H
19.	Refueling Water Storage Tank Level	2	B,D,F,H
20.	Neutron Flux (Wide Range) **Correct Answer**	2	B,D,F,H
21.	Steam Generator Water Level (Wide Range)	1 per steam generator	C,D,E,H

(a) A channel consists of two core exit thermocouples (CETs).

Question 75

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
32. (Continued)	
	5) **WHEN** all intact S/Gs pressure within 50 psig of steam header pressure, **THEN** perform the following: __ a) OPEN all MSIVs on intact S/Gs. __ b) CLOSE all MSIV bypass valves. __ c) Restore any alignments made using Enclosure 15 (Equalizing Across MSIVs). __ d) Dump steam to condenser while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour. __ e) **WHEN** condenser dumps established, **THEN** S/G PORVs may be closed.
	__ 6) **GO TO** Step 32.g to dump steam using S/G PORVs while pressure equalizing across MSIVs.
__ d. Verify steam dumps - IN PRESSURE MODE.	d. Place steam dumps in pressure mode as follows: __ 1) Place "STM DUMP CTRL" in manual. __ 2) Place steam dumps in pressure mode.
__ e. Dump steam to condenser while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour.	__ e. **IF** steam cannot be dumped to condenser, **THEN GO TO** Step 32.g.

Correct Answer Part 1

Question 75

Enclosure 4.10

Placing An ND Train in Service for Cooldown During Emergency Conditions

1. Limits and Precautions

1.1 When manually operating any motor operated valve, minimize the torque applied to the handwheel.

1.2 After manual operation, maintenance or packing adjustment of any motor operated safety related valve, it shall be cycled electrically to ensure reliable automatic operation.

Distractor Part 1

1.3 NC System cooldown rate of less than 80°F in any hour period is administratively recommended. The Tech Spec cooldown limit specified in TS 3.4.3 Figure 3.4.3-2 shall **NOT** be exceeded. (TS 3.4.3)

1.4 Avoid operation of an ND Pump with total pump flow rates (flow to Cold Legs, Letdown and PZR Spray as appropriate) between 2000 and 3000 gpm. If operation in this range is unavoidable, increased wear will occur on the pump motor lower bearings. The pump shall be evaluated for proper operation (no unusual flow noises or vibration) if operating in this range.

1.5 The ND pumps shall **NOT** be operated at flows less than 500 gpm for more than 3 continuous hours.

1.6 Due to the lack of positive seal seating surfaces, some minimal valve leak by may be experienced on valves 1ND-26, 1ND-27, 1ND-60, and 1ND-61 when they are in the full closed position.

2. Initial Conditions

____ 2.1 Verify the KC System is in operation per OP/1/A/6400/005 (Component Cooling System).

____ 2.2 Verify NC System pressure is less than 385 psig.

____ 2.3 Verify NC System temperature is less than 350°F. Correct Answer Part 2

____ 2.4 Verify the unit is presently under the control of an Emergency Procedure.

____ 2.5 Verify the ND System has been injecting into the NC System.

2.6 Verify the following:

- ____ ND Pump suctions are aligned to the containment sump.
- ____ Both trains of ND are in operation.
- ____ The ND Train to remain in cold or hot leg recirculation is capable of supplying suction flow to the NV and NI pumps.

Question 76

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3. **Correct Answer Part 1**

ACTIONS

---NOTES---
Separate Condition entry is allowed for each PORV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One or two PORVs inoperable and not capable of being manually cycled.	B.1 Close associated block valves.	1 hour
	AND	
	B.2 Remove power from associated block valves.	1 hour
	AND	
	B.3 Restore PORV(s) to OPERABLE status.	72 hours

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of two pumps (charging pumps, safety injection pumps, or charging and safety injection pumps) capable of injecting into the RCS, the accumulators isolated, reactor coolant pump operation limited as specified in Table 3.4.12-1 and either a, b, or c below:

a. Two power operated relief valves (PORVs) with nominal lift setting = 400 psig (as left calibrated), allowable value $\leq$ 425 psig (as found), with RCS cold leg temperature $\geq$ 70°F; or

b. Two residual heat removal (RHR) suction relief valves with lift settings $\geq$ 417 psig and $\leq$ 509 psig with an indicated RCS cold leg temperature $\geq$ 70°F; or

c. A combination of any one PORV and one RHR suction relief valve, each with lift settings as described above.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is $\leq$ 210°F, **Distractor Part 2**
MODE 5,
MODE 6 when the reactor vessel head is on.

---NOTE---
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in Specification 3.4.3.

Question 76

BASES

LCO (continued)

a. Two OPERABLE PORVs (NC-32B and NC-34A); or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the specified limit and testing proves its automatic ability to open at this setpoint, and motive power is available to the valve and its control circuit. The following restrictions are placed on PORV OPERABILITY for LTOP due to commonalities between the PORV power supplies and letdown isolation:

Distractor Part 2

- NC-32B is not OPERABLE for LTOP if excess letdown is in service.
- NC-32B is not OPERABLE for LTOP if normal letdown is in service and centrifugal charging pump B is in operation.
- NC-34A is not OPERABLE for LTOP if normal letdown is in service.

b. Two OPERABLE RHR suction relief valves (ND-3 and ND-38); or

An RHR suction relief valve is OPERABLE for LTOP when both of its RHR suction isolation valves are open, its setpoint is at or between 417 psig and 509 psig, and testing has proven its ability to open in this pressure range.

c. One OPERABLE PORV and one OPERABLE RHR suction relief valve.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 210°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 210°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 210°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows

Question 76

BASES

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for manual RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are assumed to operate in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV automatic actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. Events that assume this condition include uncontrolled bank withdrawal at power, uncontrolled bank withdrawal from subcritical, and single rod withdrawal at power (Ref. 2). (This statement clarifies that worst case DNBR calculations are analyzed by assuming automatic PORV operation. This statement is to bound the worst case DNBR calculations based on all possible plant conditions and is not a requirement for automatic PORV operation for system OPERABILITY.)

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

Correct Answer Part 2

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. Three PORVs are required to be OPERABLE to meet RCS pressure boundary requirements. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage. Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2.

Continued From Previous Page.

Question 77

- Thermal stresses on the steam generator tubes should also be considered before feeding a faulted steam generator. If the steam generator is dry, cold feed flow may stress the hot tubes causing tube failures. Further discussion on feeding a hot, dry steam generator is provided in the background document for FR-H.5 (Response To Steam Generator Low Level).

Distractor Part 1

- As an alternative cooldown method, one could steam a ruptured steam generator. In addition to increasing radiological releases, this will result in continued primary-to-secondary leakage. If the tube failure is large, the reactor coolant makeup supply could be depleted before ND system cooling can be established. This may also result in a steam generator overfill condition. Hence, before steaming a ruptured steam generator, one must consider potential radiological consequences, including availability of the condenser, reactor coolant activity, and meteorological conditions, and also the rate of accumulation of water in the ruptured steam generator and reactor coolant makeup supply.

STEP 42: Verify NC pump status as follows

PURPOSE:

Question 77

To alert the operator that NC pump seal damage may have occurred if NC pump cooling had previously been lost. In that case, starting the affected NC pump may further damage the seal and NC pump.

To establish forced coolant flow, if possible, or to verify natural circulation flow if NC pumps cannot be started.

APPLICABLE ERG BASIS:

Correct Answer Part 1

NC pump 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. This step provides guidance on establishing conditions for starting an NC pump to prevent NC pump damage and minimize any perturbations in NC system conditions.

Depressurization of the NC system may generate a steam bubble in the upper head region of the reactor vessel if no NC pump is running. This bubble could rapidly condense during pump startup, drawing liquid from the pressurizer and reducing reactor coolant subcooling. If pressurizer inventory is not sufficient, level may decrease off span. In addition, local flashing of reactor coolant could occur if NC system subcooling is not adequate. These conditions would require S/I reinitiation, thereby increasing leakage into the ruptured steam generator, and may confuse the operator if such behavior was unexpected.

Continued On Next Page.

Continued From Previous Page. **Question 77**

If an NC pump cannot be manually rotated due to containment access limitations or restrictions, the plant should be taken to Cold Shutdown conditions under natural circulation to permit pump disassembly and visual inspection as part of the status evaluation. An NC pump should not be started without this status evaluation since any seal misalignment or crud blockage could aggravate NC pump seal damage, potentially propagating into NC pump seal damage and increased seal leakage flow. However, there is an exception to this requirement for a status evaluation prior to NC pump start. The NC pump should be started even without a status evaluation if an extreme (red level) or severe (orange level) challenge to a Critical Safety Function is diagnosed via Status Tree monitoring and the operator is instructed to start an NC pump in the associated Function Restoration Guideline. Under these conditions, the NC pump support systems should be restored to as near normal conditions as possible and the NC pump started.

Correct Answer Part 2

A step has been added to evaluate RVLIS level prior to NC pump restart. After Safety Injection is terminated, it is possible voided conditions could occur in the upper head region at the time these steps are performed. If an NC pump restart is attempted under these conditions, pressurizer level could decrease off-scale low and/or NC system subcooling could be lost due to the collapse of the upper head void and filling of the upper head region with primary coolant. This would result in the need to reinitiate the S/I signal, consequently hindering plant recovery. In E-3, the RVLIS full range indication has always been used to determine if any voids exist in the upper head. If so, pressurizer level and NC system subcooling requirements are verified or established. These minimum values were intended to ensure plant conditions would be maintained following NC pump restart under voided conditions. If the RVLIS indication determines that a upper head void is not present, the restrictions on pressurizer level and NC subcooling do not apply. A second concern addressed for these steps is the need to provide guidance to use the pressurizer heaters to saturate the pressurizer prior to starting an NC pump with a void present in the upper head. Saturated conditions are desired to limit the pressure decrease upon NC pump restart under voided conditions but specific guidance was not originally provided. The original procedure relied on general operator training/knowledge to ensure saturated conditions existed prior to NC pump restart.

NC pumps should not be started prior to a status evaluation unless an extreme (red) or severe (orange) CSF challenge is diagnosed. Under such a CSF challenge, the "rules of usage" apply and an NC pump should be started if so instructed in the associated FRG. Under a CSF challenge, potential NC pump damage is an acceptable consequence if NC pump start is required to address a CSF challenge (e.g., to mitigate an inadequate core cooling condition). This is consistent with the intent of these FRGs which attempt to first establish support conditions to start an NC pump, but then start an NC pump whether or not the support conditions are established.

If NC pump seal cooling is lost for only a few minutes, the inventory of cold water in the seal area should prevent excessive seal heat up. For longer periods of time, seal and bearing temperatures may increase greater than 300°F. If excessive temperatures develop, the affected NC pump should not be restarted prior to a complete NC pump evaluation.

Continued On Next Page.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 7. **Verify total CA flow - GREATER THAN 450 GPM.**	**Perform the following:** __ a. Ensure CA Pump #1 - RUNNING. __ b. **IF** flow less than 450 GPM due to operator action to control CA flow, **THEN GO TO** Step 8. __ c. Ensure all CA Pump #1 flow control valves - AT 100% DEMAND POSITION. __ d. **IF** flow less than 450 GPM, **THEN** dispatch operator to verify proper CA Pump #1 valve alignment. **REFER TO** Enclosure 2 (Local Valve Alignment for CA Pump #1).
8. **Attempt to restore power to 1ETA or 1ETB as follows:** __ a. Start both D/Gs from Control Room. __ b. Verify both D/Gs running.	 b. Perform the following: __ 1) Initiate both trains S/I. __ 2) **IF** at least one D/G starts, **THEN GO TO** Step 8.c. __ 3) **GO TO** Step 9.

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. (Continued)	
__ c. Verify both D/G load sequencers - AUTOMATICALLY LOADING BUS.	c. Perform the following for affected train(s): • 1ETA: 1) **IF** 1ETA de-energized, **THEN** perform the following: a) Ensure the following breakers - OPEN: __ • "ETA NORM FDR FRM ATC" __ • "ETA ALT FDR FRM SATA". b) **IF** 1ETA still de-energized, **THEN** perform the following: __ (1) **IF** D/G 1A running, **THEN** depress and hold D/G "OFF" pushbutton. __ (2) Dispatch operator to open 1EDE-F01F (Diesel Generator Load Sequencer Panel 1DGLSA) (AB-577, BB-46, Rm 496). __ (3) **WHEN** 1EDE-F01F open, **THEN** ensure D/G "OFF" pushbutton released. __ 2) **IF** 1ETA energized, **THEN** ensure at least one A Train RN pump on. (RNO continued on next page)

PRE-REVIEW QUESTION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

8. (Continued)

- 1ETB:

N/A

1) **IF** 1ETB de-energized, **THEN** perform the following:

a) Ensure the following breakers - OPEN:

__ • "ETB NORM FDR FRM ATD"

__ • "ETB ALT FDR FRM SATB".

b) **IF** 1ETB still de-energized, **THEN** perform the following:

__ (1) **IF** D/G 1B running, **THEN** depress and hold D/G "OFF" pushbutton.

__ (2) Dispatch operator to open 1EDF-F01F (Diesel Generator Load Sequencer Panel 1DGLSB) (AB-560, BB-46, Rm 372).

__ (3) **WHEN** 1EDF-F01F open, **THEN** ensure D/G "OFF" pushbutton released.

__ 2) **IF** 1ETB energized, **THEN** ensure at least one B Train RN pump on.

__ d. Verify 1ETA or 1ETB - ENERGIZED.

__ d. **GO TO** Step 9.

__ e. Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees).

__ f. **RETURN TO** procedure and step in effect.

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 4. **Verify 1ETA and 1ETB - ENERGIZED.**	**Perform the following:** __ a. **IF** 1ETA **AND** 1ETB de-energized, **THEN GO TO** EP/1/A/5000/ECA-0.0 (Loss of All AC Power). __ b. **WHEN** time allows, **THEN** attempt to restore power to de-energized switchgear while continuing with this procedure. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power).

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE The ESPS D/G can restore one essential bus within one hour. Energizing 1ETA or 1ETB through a shared transformer will require an operator in the 6.9KV switchgear room and require a second operator to rack in the standby breaker in the 4.16KV switchgear room.

19. **Restore power to 1ETA or 1ETB using any of the following while continuing with this procedure:**

- 1ETA:

Distractors C.2 & D.2

__ • **IF** desired, **THEN** energize 1ETA from D/G. **REFER TO** Enclosure 5 (ETA Power Restoration From D/G)

OR

__ • **IF** desired, **THEN** align Unit 1 normal power to 1ETA through 1ATC. **REFER TO** Enclosure 7 (Aligning Normal Power to 1ETA (1ATC))

OR

__ • **IF** desired, **THEN** align Unit 2 alternate power to 1ETA through SATA. **REFER TO** Enclosure 8 (Aligning Unit 2 Alternate Power to 1ETA (SATA))

OR

__ • **IF** desired, **THEN** align Unit 1 alternate power to 1ETA through SATA. **REFER TO** Enclosure 9 (Aligning Unit 1 Alternate Power to 1ETA (SATA))

OR

__ • **IF** desired **AND** ESPS D/G available, **THEN** align ESPS power to 1ETA through 1ATC. **REFER TO** Enclosure 49 (Aligning ESPS to 1ETA (1ATC)).

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE If NC pump seal cooling was previously isolated, further cooling of the NC pump seals will be established by natural circulation cooldown as directed in subsequent procedures.	
45. **Select recovery procedure as follows:**	
__ a. Verify NC subcooling based on core exit T/Cs - GREATER THAN 0°F.	__ a. **GO TO** EP/1/A/5000/ECA-0.2 (Loss of All AC Power Recovery With S/I Required).
__ b. Verify Pzr level - GREATER THAN 11% (30% ACC).	__ b. **GO TO** EP/1/A/5000/ECA-0.2 (Loss of All AC Power Recovery With S/I Required).
c. Verify the following valves - CLOSED: __ • 1NI-9A (NV Pmp C/L Inj Isol) __ • 1NI-10B (NV Pmp C/L Inj Isol).	__ c. **IF** any NV pump on, **THEN GO TO** EP/1/A/5000/ECA-0.2 (Loss of All AC Power Recovery With S/I Required).
__ d. **GO TO** EP/1/A/5000/ECA-0.1 (Loss of All AC Power Recovery Without S/I Required).	Distractors B.1 & D.1

END

Question 79

Given the following conditions on Unit 1:

- Unit is at 100% RTP
- 1ERPA has de-energized due to an inverter failure
- The crew is preparing to re-energize 1ERPA via 1VRD per AP/1/A/5500/029 (Loss of Vital or Aux Control Power)

Upon the loss of 1ERPA, _____(1)_____ VCT makeup capability was lost.

Once 1ERPA is aligned to 1VRD, the crew _____(2)_____ exit the action statement of Tech Spec 3.8.7 (Inverters – Operating).

Which ONE (1) of the following completes the statements above?

A. 1. auto ONLY
   2. will

B. 1. auto ONLY
   2. will NOT

C. 1. auto AND manual
   2. will

D. 1. auto AND manual
   2. will NOT

Original Question

MODIFIED

Question 79

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

7. (Continued)

__ g. Ensure "MANUAL BYPASS SWITCH" on 1EME (Manual Bypass Switch For Swing Inverter 1EIE) - SELECTED TO "ALTERNATE AC SOURCE TO LOAD".

NOTE Padlock key interlock will provide a control to prevent closing of more than one output breaker on 1EME.

__ h. Unlock breaker 1EME B4 (1EME Output To 1EMA).

__ i. Ensure breaker 1EME B4 (1EME Output To 1EMA) - ON.

__ j. Ensure breaker 1EMA B4 (1EMA Output To 1ERPA) - ON.

__ k. Ensure "MANUAL BYPASS SWITCH" on 1EMA - SELECTED TO "ALTERNATE AC SOURCE TO LOAD".

Correct Answer Part 1

Question 79

Manual Bypass Switches

Distractor Part 1

Objective # 4F, ISS, LICENSED

Six Manual Bypass Switches - EMA, EMB, EMC, EMD, EME, and EMF - are utilized in this system to manually transfer the inverter to the alternate power source without power interruption (make-before-break contacts) to the 120 VAC panelboards (ERPA, ERPB, ERPC and ERPD). Reference **Figure 16** (Manual Bypass Switch)

The normal and alternate sources should be synchronized (in sync light lit) prior to transfer. However, there is no interlock to prevent transfer if the two sources are not synchronized. Each Manual Bypass Switch has an “In Sync” indication to let the operator know that the normal and alternate power supplies to the 120 VAC panelboards are synchronized.

The Manual Bypass Switch has two positions: “INVERTER TO LOAD” and “ALTERNATE SOURCE TO LOAD.” The power source for each of these positions for each inverter is listed below:

MANUAL BYPASS SWITCH	“INVERTER TO LOAD”	“ALTERNATE SOURCE TO LOAD”	MANUAL BYPASS SWITCH OUTPUT SUPPLIES:
EMA	EIA Output	EME Output	ERPA
EMB	EIB Output	EMF Output	ERPB
EMC	EIC Output	EME Output	ERPC
EMD	EID Output	EMF Output	ERPD

Question 79

ACTIONS (continued)

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its voltage regulated transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

Correct Answer Part 2

If the channel-related inoperable inverter is replaced by its train's swing inverter, the 24 hour limit does not apply (unless the swing inverter is also inoperable).

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital bus energized from the inverter. The verification of proper indicated voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

Question 79

LCO (continued)

Distractor Part 2

by their associated battery powered inverters provide uninterruptible supply of AC electrical power to associated loads even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the AC vital bus be powered by its channel-related inverter, or swing inverter. When the redundant train of Class 1E AC vital bus electrical power distribution subsystem is required by LCO 3.8.10, the power source for these AC vital buses may consist of 1) the associated channel-related inverter powered by its associated battery; 2) the constant voltage source transformer; or 3) a swing inverter powered by one of the train-related batteries. This ensures the availability of sufficient power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, "Distribution Systems—Shutdown," the remaining OPERABLE Inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances,

Question 80

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1 **Loss of Essential AC Power**	Prolonged loss of **all** offsite and **all** onsite AC power to essential buses 1 2 3 4 **SG1.1** Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB **AND** SSF fails to supply NC pump seal injection **OR** CA supply to SGs **AND EITHER**: - Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1) - Core Cooling RED PATH conditions met	Loss of **all** offsite and **all** onsite AC power to essential buses for 15 minutes or longer 1 2 3 4 **SS1.1** Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1) Unit 2 Condition at 1015 Correct Answer Distractor A.2 & C.2	Loss of **all but one** AC power source to essential buses for 15 minutes or longer 1 2 3 4 **SA1.1** AC power capability, Table S-1, to essential 4160V buses 1(2)ETA and 1(2)ETB reduced to a single power source for ≥ 15 min. (Note 1) **AND** **Any** additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS Unit 1 Condition at 1015 Distractor A.1 & B.1	Loss of **all** offsite AC power capability to essential buses for 15 minutes or longer 1 2 3 4 **SU1.1** Loss of **all** offsite AC power capability, Table S-1, to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1) Unit 2 has Not los ATC or ATD but has lost SATA and SATB since aligned via Unit 1 Power Supply
2 **Loss of Vital DC Power**	Loss of **all** essential AC and vital DC power sources for 15 minutes or longer 1 2 3 4 **SG1.2** Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. **AND** Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses EDA, EDC, EDB and EDD for ≥ 15 min. (Note 1)	Loss of all vital DC power for 15 minutes or longer 1 2 3 4 **SS2.1** Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses EDA, EDC, EDB and EDD for ≥ 15 min. (Note 1)	None	None
3 **Loss of CR Indications**	Unit 2 Condition at 1015 Correct Answer None	None	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress 1 2 3 4 **SA3.1** An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1) **AND** Any significant transient is in progress, Table S-3	UNPLANNED loss of Control Room indications for 15 minutes or longer 1 2 3 4 **SU3.1** An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

Table S-1 AC Power Sources
Offsite
- ATC (Train A)
- SATA (Train A) (if already aligned)
- ATD (Train B)
- SATB (Train B) (if already aligned)
Onsite
- D/G A (Train A)
- D/G B (Train B)

Table S-3 Significant Transients
- Reactor trip

Question 81

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE • After low steamline pressure main steam isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation. • OAC graphic SMRATES to monitor S/G pressure rates can be accessed via a hot button in the center of the SM graphic.	
11. **Initiate NC System cooldown to Cold Shutdown as follows:**	
__ a. Verify "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT.	__ a. **<u>GO</u> <u>TO</u>** Step 11.g.
__ b. Verify MSIVs on all intact S/Gs - OPEN.	b. Perform the following: __ 1) **<u>IF</u>** any S/G faulted, **<u>THEN</u> <u>GO</u> <u>TO</u>** Step 11.g. __ 2) **<u>IF</u>** intact MSIVs required closed to isolate leak, **<u>THEN</u> <u>GO</u> <u>TO</u>** Step 11.g. 3) Reset Main Steam Isolation signal as follows: __ a) Ensure manual loaders for all MSIV bypass valves - ADJUSTED TO 0%. __ b) Reset SM Isolation. __ c) Reset S/G PORVs. (RNO continued on next page)

Question 81

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

11. (Continued)

4) Place steam dumps in pressure mode as follows:

__ a) Place "STM DUMP CTRL" in manual.

__ b) Adjust "STM DUMP CTRL to 0% demand.

__ c) Place steam dumps in pressure mode.

5) Perform the following to equalize pressure across MSIVs on intact S/Gs:

__ a) OPEN MSIV bypass valve on intact S/Gs.

__ b) **IF AT ANY TIME** pressure does not equalize as required, **THEN** isolate steam loads off main steam header. **REFER TO** Enclosure 7 (Equalizing Across MSIVs).

(RNO continued on next page)

Question 81

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
	6) **WHEN** all intact S/Gs pressure within 50 psig of steam header pressure, **THEN** perform the following:
	__ a) OPEN all MSIVs on intact S/Gs.
	__ b) CLOSE all MSIV bypass valves.
	__ c) **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK".
	__ d) Dump steam to condenser while maintaining cooldown rate based on NC T-Colds as close as possible without exceeding 100°F in an hour.
	__ e) **WHEN** condenser dumps established, **THEN** S/G PORVs may be closed.
	__ 7) **GO TO** Step 11.g to dump steam using S/G PORVs while pressure equalizing across MSIVs.
__ c. **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK".	
__ d. Verify steam dumps - IN PRESSURE MODE.	d. Place steam dumps in pressure mode as follows:
	__ 1) Place "STM DUMP CTRL" in manual.
	__ 2) Place steam dumps in pressure mode.

Question 81

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
Distractor Part 2	
NOTE CSF should not be implemented until directed by this procedure.	
2. **Verify at least one of the following annunciators - LIT:**	**IF both alarms dark, THEN perform the following:**
__ • 1AD-20, B/2 "CONT. SUMP LEVEL >2.5 ft"	a. Ensure S/I - RESET:
OR	__ 1) ECCS.
__ • 1AD-21, B/2 "CONT. SUMP LEVEL >2.5 ft".	__ 2) D/G load sequencers.
	__ b. Ensure ND pumps - OFF.
	c. **IF** either ND pump continues to run, **THEN** isolate affected trains discharge path as follows:
	1) **IF** train "A" affected, **THEN** CLOSE:
	__ • 1NI-173A (ND Hdr 1A To Cold Legs C&D)
	__ • 1ND-32A (ND Train 1A Hot Leg Inj Isol).
	2) **IF** train "B" affected, **THEN** CLOSE:
	__ • 1NI-178B (ND Hdr 1B To Cold Legs A&B)
	__ • 1ND-65B (ND Train 1B Hot Leg Inj Isol).
	(RNO continued on next page)

APPLICABLE SAFETY ANALYSES (continued)

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Analyses are performed in regard to static rod misalignment, single rod withdrawal, dropped rod, and dropped group of rods (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps. Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(X,Y,Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}(X,Y)$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in the safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36 (Ref. 6).

Question 88

In accordance with the following Technical Specification BASES:

(1) T.S. 2.1.2 (RCS Pressure SL)
Pressurizer Safeties, Reactor Trip Setpoints, and ______(1)______ are required in order to ensure Reactor Coolant System Pressure Safety Limits are not exceeded.

(2) T.S. 3.3.1 (Reactor Trip System Instrumentation)
Only three channels of pressurizer level (vs. four) are required because ___________(2)___________.

A. (1) Pressurizer PORVs
(2) pressurizer level does NOT provide a backup signal to any other reactor trips

B. (1) Pressurizer PORVs
(2) of the slow rate of charging that is available

C. (1) Steam Generator Safeties
(2) pressurizer level does NOT provide a backup signal to any other reactor trips

D. (1) Steam Generator Safeties
(2) of the slow rate of charging that is available

Original Question

MODIFIED

Question 83

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level-High

Distractor Part 2

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The setpoints are based on percent of instrument span. The LCO requires three channels of Pressurizer Water Level-High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

Correct Answer Part 2

10. Reactor Coolant Flow-Low

a. Reactor Coolant Flow-Low (Single Loop)

The Reactor Coolant Flow-Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48% RTP, a loss of flow in any RCS loop will actuate a reactor trip. The

APE033 2.4.50 - Loss of Intermediate Range Nuclear Instrumentation
APE033 GENERIC
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

Given the following initial conditions on Unit 1:

- The Unit is conducting a startup at the POAH
- Intermediate Range Channel N-35 begins to operate erratically
- AP/1/A/5500/016 (Malfunction of Nuclear Instrumentation System) Case III (Intermediate Range Malfunction) is entered

Subsequently:

- Power is stable at the POAH
- AP/16 actions are complete

The “1/N-35A I/R CHANNEL 1 TRIP BYPASS” status light on 1SI-19 _____(1)_____ LIT.

<u>Per Tech Spec 3.3.1</u> (RTS Instrumentation), the startup to MODE 1 _____(2)_____ continue.

Which ONE (1) of the following completes the statements above?

A. 1. is
2. may

B. 1. is
2. may NOT

C. 1. is NOT
2. may

D. 1. is NOT
2. may NOT

Question 84

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

__ 1. **Verify reactor power - GREATER THAN 10%.**

__ **Stop any power increase.**

__ 2. **Verify 1AD-2, C/3 "I/R HI FLUX LEVEL ROD STOP" - DARK.**

__ **Adjust turbine load to maintain T-Avg at T-Ref.**

3. **Identify affected I/R channel:**

__ • N-35

OR

__ • N-36.

NOTE 1AD-2, C/4 "N/I SYS S/R & I/R TRIP BYPASS" will actuate in the following step if malfunction is due to instrument power failure.

__ 4. **At affected I/R drawer, place "LEVEL TRIP" switch for affected channel in "BYPASS".**

Part 1 Correct Answer

__ 5. **Verify affected I/R channel trip bypass status light (1SI-19) - LIT.**

Perform the following:

__ a. Notify Shift Manager that due to loss of control power, I/R "HIGH LEVEL TRIP" cannot be bypassed. Reactor power reduction below P-10 I/R BLOCK PERMISSIVE will result in Reactor Trip due to "N/I HI FLUX IR RX TRIP".

__ b. **GO TO** Step 7.

__ 6. **Verify 1AD-2, C/4 "N/I SYS S/R & I/R TRIP BYPASS" - LIT.**

ACTIONS (continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux-Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-High; and
- SG Water Level-Low Low.

A known inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-three logic for actuation of the two-out-of-four trips. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 11.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 11.

F.1 and F.2

Distractor A.2 & C.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below

ACTIONS (continued)

Distractor A.2 & C.2

the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

<u>G.1 and G.2</u>

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint.

Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip. Required Action G.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this

ACTIONS (continued)

Part 2 Correct Answer

Action.

H.1

Condition H applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is below the P-6 setpoint and one or two channels are inoperable. Below the P-6 setpoint, the NIS source range performs the monitoring and protection functions. The inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. The NIS intermediate range channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10.

I.1

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately. Required Action I.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action.

J.1

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition and the unit exits this Condition.

Question 85

16.9 AUXILIARY SYSTEMS

16.9-5 Fire Rated Assemblies

COMMITMENT All required Fire Rated Assemblies (walls, floors/ceilings, cable enclosures and other fire barriers) and all sealing devices in fire rated assembly penetrations (fire doors, fire dampers, and penetration seals) as shown on the CN-1105 drawing series shall be FUNCTIONAL.

APPLICABILITY: At all times.

--NOTE--
Non-functional or breached fire barrier features (walls, floors, ceilings, doors, dampers, and penetration seals) in the diesel generator rooms and the auxiliary feedwater pump rooms may affect CO_2 System FUNCTIONALITY. See SLC 16.9-3, "CO_2 Systems".

REMEDIAL ACTIONS

N/A IF the required Fire Rated Assembly sealing device is a Fire Door, see Table 16.9-5-1

N/A IF the required Fire Rated Assembly sealing device is a Fire Damper see Table 16.9-5-2

IF required Fire Rated Assembly is a Fire Barrier or Penetration Seal:

✓ 1. Identify the location of the impaired fire protection feature by elevation, column, and building

✓ 2. Verify the wall, floor/ceiling is a committed boundary on the CN-1105 drawing series (if not a committed boundary, SLC 16.9-5 does not apply)

✓ 3. Refer to CN-1209-10 series drawings to identify the Fire Area on both sides of the impaired feature

N/A 4. IF either of the Fire Areas is identified as High Safety Significant (HSS) (see Table 16.9-5-3) then implement the REQUIRED ACTION **CONDITION A**

✓ 5. IF the Fire Areas are not HSS, then identify the associated shutdown trains/methods of the Fire Areas on each side of the barrier using Table 16.9-5-4 and implement the REQUIRED ACTION as identified in the following Chart:

Shutdown Train (Side 1 & Side 2)	*A*	*B*	*SSS*	*A or B*	*A and B*
A	**CONDITION C**	**CONDITION B**	**CONDITION B**	**CONDITION C**	**CONDITION B**
B	**CONDITION B**	**CONDITION C**	**CONDITION B**	**CONDITION C**	**CONDITION B**
SSS	**CONDITION B**	**CONDITION B**	**CONDITION C**	**CONDITION B**	**CONDITION B**
A or B	**CONDITION C**	**CONDITION C**	**CONDITION B**	**CONDITION C**	**CONDITION B**
A and B	**CONDITION B**	**CONDITION B**	**CONDITION B**	**CONDITION B**	**CONDITION C**

Question 85

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1TB-FD038	TB1	16-17/V	594+0	TB1/SRV	DID	C
1TB-FD039	TB1	16-17/V	594+0	TB1/SRV	DID	C
1TB-FD040	TB1	16/V	594+0	TB1/SRV	DID	C
1TB-FD043	TB1	30-31/1J-1K	568+0	TB1/OTT	DID	C
1TB-FD044	TB1	32/1J-1K	594+0	TB1/MTOT	DID	C
1TB-FD045	TB1	30/1J-1K	594+0	TB1/MTOT	DID	C
1TB-FD046	TB1	32/1K-1L	568+0	TB1/OTT	DID	C
2TB-FD013	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD014	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD015	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD016	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD017	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD018	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD019	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD020	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD021	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD022	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD023	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD024	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD031	TB2	32/2K-2L	568+0	TB2/OTT	DID	C
2TB-FD032	TB2	18/P	594+0	TB2/SRV	DID	C
2TB-FD036	TB2	16-17/P	594+0	TB2/SRV	DID	C
2TB-FD038	TB2	17-18/P	594+0	TB2/SRV	DID	C
2TB-FD039	TB2	32/2J-2K	594+0	TB2/MTOT	DID	C
2TB-FD040	TB2	30/2J-2K	594+0	TB2/MTOT	DID	C
2TB-FD041	TB2	30-31/2J/2K	568+0	TB2/OTT	DID	C

*2VA-FD070 is exempt from inspection requirements (SLC TR 16.9-5-5) for ALARA reasons

Table 16.9-5-3

HIGH SAFETY SIGNIFICANT (HSS) FIRE AREAS*

Distractor Part 1

FIRE AREA	BLDG	ELEVATION	DESCRIPTION
6	AUX	560+0	Unit 1 Electrical Pen Room El 560
12	AUX	577+0	Unit 2 Electrical Pen Room El 577
13	AUX	577+0	Unit 1 Electrical Pen Room El 577
14	AUX	577+0	Unit 2 4160V Essential Swgr Room (2ETA)
15	AUX	577+0	Unit 1 4160V Essential Swgr Room (1ETA)
16	AUX	574+0	Unit 2 Cable Room El 574
17	AUX	574+0	Unit 1 Cable Room El 574
21	AUX	594+0	Main Control Room El 594

*High Safety Significant (HSS) Fire Areas are defined as the areas with HSS fire barrier features in accordance with the Catawba NFPA 805 Monitoring Program.

Question 85

Table 16.9-5-4

FIRE AREAS AND SHUTDOWN TRAIN / METHOD

FIRE AREA	FIRE AREA DESCRIPTIONS	ASSURED SHUTDOWN TRAIN / METHOD
1	ND & NS Pump Room El 522 (Common)	SSS
2	Unit 2 CA Pump Room El 543	SSS
3	Unit 1 CA Pump Room El 543	SSS
4	Aux Bldg. Gen Area & NV Pump Room El 543 (Common)	SSS
5	Unit 2 Electrical Pen Room El 560	A
6	Unit 1 Electrical Pen Room El 560	A
7	Unit 2 4160V Essential SWGR Room El 560	A
8	Unit 1 4160V Essential SWGR Room El 560	A
9	Unit 2 Battery Room El 554	SSS
10	Unit 1 Battery Room El 554	SSS
11	Aux Bldg. Gen Area & U1 KC Pump Room El 560 (Common)	SSS
12	Unit 2 Electrical Pen Room El 577	B
13	Unit 1 Electrical Pen Room El 577	B
14	Unit 2 4160V Essential SWGR Room El 577	B
15	Unit 1 4160V Essential SWGR Room El 577	B
16	Unit 2 Cable Room El 574	SSS
17	Unit 1 Cable Room El 574	SSS
18	Aux Bldg. Gen Area & U2 KC Pump Room El 577 (Common)	SSS
19	Unit 2 Electrical Pen Room El 594	A
20	Unit 1 Electrical Pen Room El 594	A
21	Control Room El 594 (Common)	SSS
22	Aux Bldg. Gen Area El 594 (Common)	SSS
23	Unit 2 Fuel Storage Area El 605	A
24	Unit 1 Fuel Storage Area El 605	A
25	Diesel Generator Bldg. 1A El 556	B
25A	Diesel Generator Bldg. 1A Stairwell	B
26	Diesel Generator Bldg. 1B El 556	A
26B	Diesel Generator Bldg. 1B Stairwell	A
27	Diesel Generator Bldg. 2A El 556	B
27A	Diesel Generator Bldg. 2A Stairwell	B
28	Diesel Generator Bldg. 2B El 556	A
28B	Diesel Generator Bldg. 2B Stairwell	A
29	Train A RN Pump Structure El 600 (Common)	B
30	Train B RN Pump Structure El 600 (Common)	A
31	Unit 2 Train A Aux Shutdown Panel El 543	B
32	Unit 1 Train A Aux Shutdown Panel El 543	B
33	Unit 2 Train B Aux Shutdown Panel El 543	A
34	Unit 1 Train B Aux Shutdown Panel El 543	A
35	Control Room Tagout Area El 594	A or B
36	Unit 2 Turbine Driven CA Pump Control Panel Room El 543	B
37	Unit 1 Turbine Driven CA Pump Control Panel Room El 543	B
38	Unit 1 Fuel Storage Area HVAC Room El 631	A or B

(continued)

Catawba Nuclear Station

ILT-17 NRC Written Exam CNS SRO NRC Examination

Question: 86
(1 point)

Given the following conditions on Unit 1:

- The Unit is stable at 100% RTP
- Operators are performing PT/1/A/4150/001 A (NC Pump Seal Injection Flow Verification)
- 1NV-294 is fully open
- Seal Injection flow is 42 gpm and stable

Based on the conditions above, the Action Statement of Tech Spec 3.5.5 (Seal Injection Flow) ______(1)______ required to be entered.

In accordance with the Bases of Tech Spec 3.5.5, the Seal Injection Flow Limit ______(2)______ based on the safety analysis assumptions for minimum ECCS Injection flow.

Which ONE of the following completes the statements above?

A. 1. is
   2. is

B. 1. is NOT
   2. is

C. 1. is
   2. is NOT

D. 1. is NOT
   2. is NOT

Original Question

Question 86

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be ≤ 40 gpm with centrifugal charging pump operating and the charging flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow not within limit.	A.1 Adjust manual seal injection throttle valves to give a flow within limit with centrifugal charging pump operating and the charging flow control valve full open.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

Question 86

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

This LCO is applicable only to those units that utilize the centrifugal charging pumps for safety injection (SI). The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but do not set the limits on their flow requirements. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

This LCO ensures that seal injection flow of ≤ 40 gpm, with centrifugal charging pump operating and charging flow control valve full open, will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical for a large LOCA. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

Catawba Nuclear Station

ILT-17 NRC Written Exam CNS SRO NRC Examination

Question: 87
(1 point)

Given the following Unit 2 conditions:

- The Unit is at 100% RTP
- A loss of 2ERPD has occurred
- The crew has entered AP/2/A/5500/029 (Loss of Vital or Aux Control Power)
- NO Tech Spec actions have been addressed

The current Containment Pressure channel logic for the remaining Containment Pressure channels which will cause a **Phase B** actuation is _____(1)_____ .

In accordance with Tech Spec 3.3.2 (ESFAS Instrumentation) LCO Actions, when the failed channel is removed from service, I&E will place the Containment Pressure **Hi-Hi** Bistable in _____(2)_____ .

Which ONE of the following completes the statements above?

A. 1. 2/3
   2. Trip

B. 1. 2/3
   2. Bypass

C. 1. 1/3
   2. Trip

D. 1. 1/3
   2. Bypass

Original Question

Question 87

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

containment isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase B containment isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase B Isolation-Containment Pressure - High-High

Containment Pressure - High-High uses four channels in a two-out-of-four logic configuration. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip.

Containment Pressure - High-High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure - High-High setpoints.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

Question 87

ACTIONS (continued)

D.1, D.2.1, and D.2.2

Condition D applies to:

- Containment Pressure-High;
- Pressurizer Pressure-Low;
- Steam Line Pressure-Low;
- Steam Line Pressure-Negative Rate-High;
- Loss of offsite power (refer to Condition D footnote);
- SG Water level—Low Low; and
- SG Water level—High High (P-14) for the Feedwater Isolation Function.

Distractor Part 2

If one channel is inoperable, 72 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements. The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition is justified in Reference 13.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 12 hours for surveillance testing of other channels. The 12 hours allowed for testing is justified in Reference 13.

Question 87

ACTIONS (continued)

E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Phase B Isolation Containment Pressure-High High; and

Correct Answer Part 2

- Steam Line Isolation Containment Pressure - High High.

Neither of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious isolation initiation. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate isolation.

To avoid the inadvertent actuation of Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 72 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 72 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 12 hours for surveillance testing. Placing a second channel in the bypass condition for up to 12 hours for testing purposes is acceptable based on the results of Reference 13.

Question 88

A. **Purpose**

- **To verify proper response to a loss of feedwater supply to the S/Gs.**
- **To verify proper response to a loss of normal supply of auxiliary feedwater.**
- **To verify proper response to an abnormal S/G feedwater control condition.**

B. **Symptoms**

Case I. Loss Of CF Supply To S/Gs:

- CFPT A and B - TRIPPED
- 1AD-3, C/6 "CF ISOL TRN A" - LIT
- 1AD-3, D/6 "CF ISOL TRN B" - LIT
- Any S/G lo level alert alarm on 1AD-4 - LIT
- Any S/G flow mismatch lo CF flow alarm on 1AD-4 - LIT.

Case II. Loss Of Normal CA Supply:

- Any CA start resulting from abnormal plant conditions (auto or manual)
- 1AD-8, B/1 "UST LO LEVEL" lit **AND** 1CA-4 (CA Pmps Suct From UST) open
- CACST less than 50% **AND** 1CA-6 (CA Pmps Suct From CACST) open.

Case III. CF Control Not in Auto:

- Any unanticipated CF controller in manual
- Any S/G lo level alert alarm on 1AD-4 - LIT
- Any S/G flow mismatch lo CF flow alarm on 1AD-4 - LIT
- 1AD-2, F/9 "DCS ALTERNATE ACTION" - LIT.

Question 88

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Verify reactor power - LESS THAN 5%.** Correct Answer	**IF AT ANY TIME all CF supply to S/G(s) lost, THEN perform the following:** __ a. Trip reactor. __ b. GO TO EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ 2. **Verify all S/G hi-hi level alert alarms (1AD-4) - DARK.**	**IF 2/4 S/G N/R levels on any one S/G greater than 83%, THEN perform the following:** __ a. Verify all Feedwater Isolation status lights (1SI-5) - LIT. b. IF any Feedwater Isolation status light not lit, THEN perform the following: __ 1) Initiate Feedwater Isolation. __ 2) IF proper status light indication not obtained, THEN CLOSE affected valve(s).
__ 3. **Verify 1AD-2, F/9 "DCS ALTERNATE ACTION" - DARK.**	**IF all the following conditions exist:** __ • 1AD-2, F/9 in alarm for CF control function in alternate action __ • At least one CF pump - IN SERVICE __ • 1AD-3, C/6 "CF ISOL TRN A" - DARK __ • 1AD-3, D/6 "CF ISOL TRN B" - DARK, __ THEN GO TO Case III (CF Control Not In Auto).

Question 88

A. Purpose

- **To verify the proper response in the event of a turbine generator trip.**
- **To place the plant in a stable condition.**

B. Symptoms

- **Any Turbine Trip alarm on 1AD-1 - LIT**
- **Turbine stop valves closed status lights (1SI-2) - LIT**
- **Zone G Lockout Trip alarm on 1AD-11 (F/4) - LIT.**

Question 88

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Verify reactor power - LESS THAN 69%.** Distractor	**Perform the following:** __ a. Ensure reactor - TRIPPED. __ b. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
2. **Verify Turbine Trip:** __ • All turbine stop valves - CLOSED.	**Perform the following:** __ a. Trip turbine. b. **IF** turbine will not trip, **THEN** perform the following: __ 1) Depress "MANUAL" pushbutton on turbine control panel. __ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons. 3) **IF** turbine will not runback, **THEN** perform the following: __ a) Trip reactor. b) CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves. __ c) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Question 88

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Steam Line Pressure-Negative Rate-High must be OPERABLE in MODE 3 when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure-Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, stop the excessive flow of feedwater into the SGs, and to limit the energy released into containment. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows. Feedwater Isolation serves to limit the energy released into containment upon a feedwater line or steam line break inside containment.

The Functions are actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine;

Distractor

- Trips the MFW pumps;
- Initiates feedwater isolation; and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves.

Turbine Trip and Feedwater Isolation signals are both actuated by SG Water Level-High High, or by an SI signal. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. A Feedwater Isolation signal is also generated by a

Question 88

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(1) Feedwater Isolation-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Distractor

(2) Feedwater Isolation-Steam Generator Water Level-High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic. The setpoints are based on percent of narrow range instrument span.

(3) Feedwater Isolation-Safety Injection

Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements. Item 5.b.(1) is referenced for the applicable MODES.

(4) Feedwater Isolation - RCS T_{avg}- Low coincident with Reactor Trip (P-4)

This signal provides protection against excessive cooldown, which could subsequently introduce a positive reactivity excursion after a plant trip. There are four channels of RCS T_{avg} - Low (one per loop), with a two-out-of-four logic required coincident with a reactor trip signal (P-4) to initiate a feedwater isolation. The P-4 interlock is discussed in Function 8.a.

Question 88

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Engineered Safety Feature Actuation System Interlocks—Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker is open. Operators are able to reset SI 60 seconds after initiation. If a P-4 is present when SI is reset, subsequent automatic SI initiations will be blocked until the RTBs have been manually closed. This Function allows operators to take manual control of SI systems after the initial phase of injection is complete while avoiding multiple SI initiations. The functions of the P-4 interlock are:

Correct Answer

- Trip the main turbine;
- Isolate MFW with coincident low T_{avg};
- Prevent reactuation of SI after a manual reset of SI;
- Transfer the steam dump from the load rejection controller to the unit trip controller; and
- Prevent opening of the MFW isolation valves if they were closed on SI or SG Water Level—High High.

Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated power. To avoid such a situation, the noted Functions have been interlocked with P-4 as part of the design of the unit control and protection system.

None of the noted Functions serves a mitigation function in the unit licensing basis safety analyses. Only the turbine trip Function is explicitly assumed since it is an immediate consequence of the reactor trip Function. Neither turbine trip, nor any of the other four Functions associated with the reactor trip signal, is required to show that the unit licensing basis safety analysis acceptance criteria are not exceeded.

The RTB position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable

Question 89

F Fission Product Barriers	FG1.1 1 2 3 4	FS1.1 1 2 3 4	FA1.1 1 2 3 4
	Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1)	Loss or potential loss of **any** two barriers (Table F-1)	**Any** loss or **any** potential loss of either Fuel Clad or NCS (Table F-1)

Time After S/D	FC Loss	NCS Loss	Loss
0 - 1	550	8.8	5500
1 - 2	400	8.4	4000
2 - 8	160	7.0	1600
>8	100	6.2	1000

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (NCS) Barrier		Containment (CMT) Barrier	
	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
A. NCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER**: • UNISOLABLE NCS leakage • SG tube RUPTURE	1. CSFST Integrity-RED PATH conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B. Inadequate Heat Removal	1. CSFST Core Cooling-RED PATH conditions met	1. CSFST Core Cooling-ORANGE PATH conditions met 2. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED PATH conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C. CMT Radiation / NCS Activity	1. EMF53A/B > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. EMF53A/B > Table F-2 column "NCS Loss"	None	None	1. EMF53A/B > Table F-2 column "CMT Potential Loss"
D. CMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of NCS leakage outside of containment	1. CSFST Containment-RED PATH conditions met 2. Containment hydrogen concentration > 6% 3. Containment pressure > 3 psig with < one full train of containment cooling operating per design for > 15 min. (Notes 1, 10)
E. EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Question 89

<< Emergency Coordinator Checklist >>

1.5 Actions to Complete Within 30 Minutes of Site Area Emergency (or Higher) Declaration

1. **IF** Site Area Emergency or General Emergency, **THEN evaluate** concurrently Attachment 1 Section 3.1, Site Assembly and Accountability and Attachment 1 Section 3.2, Site Evacuation, to ensure onsite protective actions are taken. ...._____

1.6 Actions to Perform As Soon As Possible, But No Later Than 60 Minutes After Event Declaration

NOTE

Initial notification of the NRC is to occur as soon as possible following State and Counties notifications, but not to exceed one hour, following the declaration of an emergency or a change in the classification level. .... ☐

1. Initial notification of the NRC is to occur as soon as possible following State and Counties notifications, but not to exceed 1 hour, following the declaration of an emergency or a change in the classification level. ...._________

   a. **Obtain** copy of the completed and approved ENF. .... ☐

   b. Using the ENF, **direct** Offsite Communicator to notify NRC in accordance with Attachment 3, NRC Communications Checklist. .... ☐

1.7 Actions to Perform Within 60 Minutes of Alert or Higher Event Declaration

1. **Ensure** the NRC Emergency Response Data System (ERDS) data link is activated or transmitting data. ...._________

   a. **IF** ERDS is **NOT** currently activated, **THEN activate** ERDS. ...._________

Question 89

<< NRC Communications Checklist >>

1.1 Mobilization (continued)

2. **Notify** Shift Manager/Emergency Coordinator (SM/EC) of arrival on station. .... ☐

   a. **IF** directed to initiate ERONS, **THEN initiate** ERONS to activate the ERO in accordance with AD-EP-ALL-0301, Activation of the Emergency Response Organization Notification System (ERONS), using the appropriate Template ID and Message Title. .... ____/**TAB 2**

   b. **IF** directed to initiate ERONS **AND** ERONS is **NOT** available, **THEN activate** the ERO using the back-up system per CSD-EP-ALL-0111-01, ERONS Backup Job Aid. .... ____/**TAB 5**

1.2 Establish Control Room / NRC interface

NOTE
Emergency Response Data System (ERDS) is required to be activated within 1 hour of declaration of an Alert or higher emergency classification level. .... ☐

1. **Ensure** ERDS activation. .... ☐

   a. **IF** ERDS has **NOT** been activated, **THEN activate** ERDS using station specific procedure. .... ____/**TAB 3**

2. **Obtain** the following: .... ☐

   - NRC authentication code .... ☐

NOTE
Emergency Notification System (ENS) phone numbers are provided on ENS phones and in CSD-EP-ALL-0104-01, Emergency Telephone Directory. .... ☐

3. **IF** initial notification from the Control Room due to declaration of any emergency classification, **THEN** perform the following: .... ________

   a. **Obtain** copy of completed and approved Emergency Notification Form (ENF). .... ☐

Question 89

<< NRC Communications Checklist >>

1.2 Establish Control Room / NRC interface (continued)

b. Using the ENF, **notify** the NRC Operations Center by one of the following means: .... ☐

Correct Answer Part 2

- Primary: Emergency Notification System (ENS) Phone .... ☐
  - ◊ 1-301-816-5100

OR

- Alternate: Commercial Telephones .... ☐
  - ◊ 1-800-532-3469
  - ◊ 1-301-951-0550
  - ◊ 1-301-415-0550
  - ◊ 1-301-415-0553
  - ◊ 1-800-449-3694

AND

- Facsimile .... ☐
  - ◊ 1-301-816-5151

4. **IF** requested by the NRC Operations Center, **THEN maintain** continuous communications with the NRC Operations Center. .... _________

5. **Notify** the NRC of an upgrade in classification as soon as possible and within one hour. .... ☐

6. **WHEN** TSC is activated, **THEN provide** information necessary for turnover and **notify** Shift Manager/Emergency Coordinator (SM/EC) that TSC is activated and assuming NRC communication responsibilities. .... _________

7. **WHEN** TSC is activated, **THEN notify** NRC that the TSC is activated, assuming NRC communication responsibilities and provide contact information. .... _________

Question 89

<< NRC Communications Checklist >>

2.0 ONGOING ACTIONS

2.1 NRC Communications

1. **Provide** plant condition and event information to NRC as requested. .... ☐

2. **Notify** SM/EC of planned NRC activities (e.g., dispatch of a site team, specific requests). .... ☐

3. **Consult** with SME/EC for information about emergency and planned or in-progress mitigating actions to communicate to NRC. .... ☐

2.2 NRC Notifications

1. **Notify** the NRC of an upgrade in classification as soon as possible and within one hour. .... ☐

2. **Complete** NRC Event Notifications using site-specific procedure. .... ____/**TAB 4**

Distractor Part 2

   a. **Obtain** SM/EC review and approval of completed form. .... ☐

   b. **Transmit** form to the NRC. .... ☐

NOTE
The NRC Headquarters Operation Officer (HOO) receives information about an upgrade in classification. .... ☐

3. **IF** the HOO has left ENS Bridge Line, **THEN request** for HOO to rejoin Bridge Line. .... ________

1.15 When paralleling a D/G to the Essential Switchgear, the in-service KC and/or NV Pumps are aligned to the opposite train to prevent the simultaneous loss of NC Pump Seal Injection and Thermal Barrier Cooling if a D/G failure results in the loss of the Essential Switchgear. If both KC and NV in-service Pumps are powered from the same bus as the D/G being tested, KC is swapped to the opposite train. (PIP 99-3510)

1.16 Do **NOT** operate the D/G at no load or light loads for long periods of time to prevent buildup of carbon and sludge in the engine.

Part 1 Correct Answer

1.17 LD heat trace is designed to only maintain the LD stagnant loop temperature after the stagnant loop is heated to the Tech Spec range. The LD heat trace **CANNOT** be used to heatup the LD system.

2. Initial Conditions

_____ 2.1 Verify D/G 1A is aligned per Enclosure 4.1 (Diesel Alignment for ES Actuation).

NOTE:
- Verifying lube oil temperature and jacket water temperatures at the engine inlets and outlets are 140-190°F helps ensure long term reliability of the engine. Under normal starting conditions lube oil inlet and outlet temperatures and jacket water inlet and outlet temperatures of 140-190°F is the preferred starting temperature. Operability temperature is 120-190°F. (PIP C-14-02352)
- The following readings can be accessed from the OAC (e.g., using OAC graphic DGAKDLD).

2.2 Verify lube oil temperature and jacket water temperatures at the engine inlets and outlets per one of the following conditions:

_____ • **IF** starting under normal plant conditions verify lube oil temperature and jacket water temperatures at the engine inlets and outlets are 140-190°F.

_____ • **IF** starting under abnormal or emergency plant conditions verify lube oil temperature and jacket water temperatures at the engine inlets and outlets are 120-190°F.

_____ 2.3 Verify the Control Room has been notified that a LOCAL START of D/G 1A is being performed.

Question 90

_____ 3.9 **IF** starting D/G 1A for Aux Safeguards Testing, start D/G 1A per PT/1/A/4200/009A (Auxiliary Safeguards Test Cabinet Periodic Test) to simulate an emergency signal.

_____ 3.10 **IF** performing a manual slow start, perform the following:

_____ 3.10.1 Establish a dedicated operator for the "SLOW START ENABLED" switch on 1DECPA.

Dedicated Operator ______________________________

_____ 3.10.2 **IF AT ANYTIME** while the "SLOW START ENABLED" switch is in the "ON" position, an emergency start signal is received, place " SLOW START ENABLED" switch in "OFF" position.

_____ 3.10.3 Place the "SLOW START ENABLED" switch on 1DECPA in the "ON" position.

_____ 3.10.4 Verify the "SLOW START ENABLED" light on 1DECPA is lit.

NOTE:
1. The "Turbo Oil Sol Vlv" will automatically close at 95% D/G speed.
2. Step 3.11 shall be performed during routine tests and starts where time permits. It shall be omitted during emergency situations due to the nature of the start. This step is to allow oil to be supplied to the turbochargers to prolong bearing life.
3. The following step will result in D/G 1A Annunciator Panel, B/6 "TURBO OIL SOL VLV OPEN" alarming.

CAUTION: The "Turbo Oil Sol Vlv" shall **NOT** be open for more than 5 minutes to prevent oil overflowing from the turbocharger bearing to the exhaust manifold potentially causing a turbocharger fire.

_____ 3.11 Unless directed otherwise, 3-5 minutes before engine start, place the "Turbo Oil Sol Vlv" switch on 1DECPA in the "OPEN" position.

_____ 3.12 Direct the Control Room Operator to announce the start of 1A Diesel Generator per AD-OP-ALL-1000 (Conduct of Operations).

NOTE: The following step will result in D/G 1A Annunciator Panel, F/2 "AUX EQUIP NOT IN AUTO".

_____ 3.13 Place "L. O. Pump & Heater" switch on 1DECPA in "RUN" position.

Distractor A.1 & B.1

Question 90

BACKGROUND (continued)

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Approximately 1 minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Part 2 Correct Answer

Ratings for Train A and Train B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is 7000 kW with 10% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

7000 KW + 10% = 7700 KW

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the UFSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the Accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during Accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

The AC sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 6).

LCO

Two qualified circuits between the offsite transmission network and the onsite Essential Auxiliary Power System and separate and independent DGs for each train ensure availability of the required power to shut down

- Low Suction Pressure Protection for Manual Start of CA

Should the CA system be operating in manual or "CA SYS VLV" be RESET, the auto open feature is blocked and the CA pump that is running will trip. This trip feature operates on a time delay of five (5) seconds. The low pressure condition must exist continuously for 5 seconds to initiate a trip.

The Low Suction Pressure trips can be defeated by use of switches on the main control board. This action is done when aligning the CA suctions to the Hotwell per AP/1/A/5500/006 (Loss of S/G Feedwater).

The CA pumps' trip on low suction pressure is blocked in Case II of AP/1/A/5500/006 (Loss of Normal CA Supply) so that the hotwell can be used as a CA suction source. The following switches on MC10 are used to perform that function:

CA TD PMP Lo Suct Press Trip Block

CA PMP 1A Lo Suct Press Trip Block

CA PMP 1B Lo Suct Press Trip Block

7.2. CAPT OVERSPEED PROTECTION

Objective 9B, Licensed

- Electronic Overspeed

At 115% of Rated Speed, the CAPT Trip Throttle valve trips closed to stop the pump. The operator must re-open the valve by depressing the OPEN pushbutton on MC10 or at the Local Panel to reset the pump. The electronic overspeed indicating light will stay lit until the condition clears and the CAPT Trip Throttle valve is re-opened. Annunciator AD-5, E/3 "CAPT STOP VLV CLSD" will alarm when the CAPT Trip Throttle valve is closed.

- Mechanical Overspeed **Distractor B.2 & C.2**

At 125% of Rated Speed, the CAPT Trip Throttle valve trips closed and the linkage "opens" to stop the pump. For the mechanical overspeed trip the operator must at the CAPT manually move the trip linkage towards the CAPT Trip Throttle valve (away from the turbine); reference OP/1/A/6250/002 (Auxiliary Feedwater System).

The CAPT Trip Throttle valve may then be reopened manually by operating the valve's handwheel or locally at the CAPT control panel (located behind the turbine driven pump auxiliary shutdown panel) by depressing the OPEN pushbutton, or from the Control Room using the OPEN pushbutton on MC10.

7.3. LOSS OF VI

Objective 9C, Licensed

The discharge flow control valves are pneumatically operated valves provided with air accumulators for each valve to allow the valves to be remotely operated on a loss of VI. The accumulators are sized so the flow control valves can be operated for one hour to allow Operators enough time to manually secure flow to a S/G following a S/G Tube Rupture (SGTR) Design Basis Event in order to prevent overfill of the S/G.

RECOVERY/RESTORATION TECHNIQUE:

The objective of the recovery/restoration technique incorporated into guideline FR-H.1 is to restore and/or maintain adequate secondary heat removal capability and to establish NC system feed and bleed heat removal if secondary heat removal capability cannot be maintained.

The following subsections provide a summary of the major categories of operator actions and key utility decision points for guideline FR-H.1, Response To Loss Of Secondary Heat Sink.

MAJOR ACTION CATEGORIES IN E-0:

- Attempt Restoration of Feed Flow to Steam Generators

The operator attempts to restore or establish auxiliary feedwater flow, main feedwater flow, and condensate flow (in that order) while checking symptoms for a loss of secondary heat sink. Auxiliary feedwater flow restoration is attempted first and, if unsuccessful, NC pumps are tripped to extend the available time to establish feed flow from the main feedwater and condensate systems.

- Initiation of NC System Feed and Bleed Heat Removal

If symptoms for loss of secondary heat sink are reached, NC system feed and bleed heat removal is initiated through S/I actuation (feed path) and opening the pressurizer PORVs (bleed path). Feed and bleed heat removal is maintained until the secondary heat sink is reestablished and verified.

- Restore and Verify Secondary Heat Sink

After NC system feed and bleed heat removal is established, the operator continues attempts to restore narrow range level in at least one S/G. After level is established, the effectiveness of the secondary heat sink is verified by decreasing NC system temperatures.

- Termination of NC System Feed and Bleed Heat Removal

With a verified secondary heat sink, the operator performs a coordinated sequence for S/I flow reduction and closing of pressurizer PORVs. After the completion of the sequence, the operator is transferred to ES-1.1, SI Termination, for plant recovery.

Continued From Previous Page.

Question 91

KNOWLEDGE/ABILITY:

Step was simplified to quickly verify NV S/I flow, prior to opening Pzr PORVs. ECCS pumps and valve positions are not checked at this time. If NV S/I flow is indicated, pumps and valves are expected to be aligned to establish S/I flow. If NV S/I flow is not indicated then the RNO provides actions necessary to restore ECCS flow.

- Initiating Feed and Bleed is time critical. CNC-1552.08-00-0285 (McGuire and Catawba Nuclear Stations- All Units, Loss of Feedwater Feed and Bleed Analysis) assumes that feed and bleed will be initiated within 4 minutes of meeting initiation criteria, or within 8 minutes of Rx trip on Lo-Lo S/G Level. The intent of this step is to ensure that ECCS flow is established before Pzr PORVs are opened. This intent will be met by either NV S/I flow indication or, if initiation criteria occurs after 90 minutes, one NI pump running. NI pumps are normally aligned for S/I mode, so just checking the pump on is adequate.

- Note that this step is also intended to obtain maximum available ECCS flow. This intent is met by the previous step to initiate S/I (initiating S/I will start all available ECCS pumps and align valves), and by STEP 29, which will check all ECCS pumps and alignments.

- Note that it is not justified to delay opening Pzr PORVs (required to establish alternate decay heat removal) to first check all ECCS alignments. As long as ECCS flow from NV (or NI if 90 minute requirement met) has been established, it is acceptable to open Pzr PORVs (per ERG background document). After the time critical action to establish feed and bleed is completed, it is then appropriate to check for maximum ECCS capability. Delaying time critical actions to address the possibility of multiple failures on both ECCS trains during an event involving loss of both trains of aux feedwater is not appropriate. This is based on operator response time assumed in plant specific analysis.

Time Critical Operator Action sequence - See STEP 26.

Distractor Part 1

Correct Answer Part 1

Question 91

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 36. **Verify KC flow to ND heat exchangers - INDICATING FLOW.**	__ **IF AT ANY TIME ND pump operating with flow less than 1000 GPM to NC loops AND KC to associated ND HX isolated, THEN stop affected ND pump within 3 hours.**
37. **Align CA to establish control of S/G feed as follows:**	
__ a. Ensure CA System valve control - RESET.	
__ b. CLOSE CA flow control valves on S/Gs not presently being fed.	
38. **Continue attempts to establish secondary heat sink in at least one S/G as follows:**	
__ • CA flow. **REFER TO** Steps 6 through 7	
__ • CF or CM flow. **REFER TO** Steps 10 through 18	

NOTE RY System should not be used unless all other feedwater sources are unavailable.

__ • RY flow. **REFER TO** Step 19.	
__ 39. **Verify N/R level in at least one S/G - GREATER THAN 11% (29% ACC).**	__ **RETURN TO Step 38.**
40. **Verify NC System temperatures as follows:**	__ **RETURN TO Step 38.**
__ • Core exit T/Cs - TRENDING DOWN	
__ • NC T-Hots associated with S/Gs being fed - TRENDING DOWN.	

Question: 76
(1 point)

Given the following conditions on Unit 2:

- Unit initially operating at 100% RTP
- Containment pressure rapidly increased to 3.4 PSIG
- 2B S/G completely depressurized
- Automatic reactor trip did not occur and manual trip from the control room was unsuccessful
- OATC is manually inserting control rods per EP/2/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS)

Current conditions:

- While inserting rods, the OATC notes the following:
  - DRPI indication for Control Bank 'D' rod M4 is YELLOW with a DATA 'B' indication above the rod
  - DRPI indication for rod M4 indicates 198 steps
  - Demand position counters for Groups 1 & 2 are blinking with position indication of 210 steps

In accordance with T.S. 3.1.7 (Rod Position Indication) bases, LCO 3.1.7 _____(1)_____ met.

When reactor shutdown has been verified, the NEXT required procedure transition from FR-S.1 is to GO TO _____(2)_____.

Which ONE (1) of the following completes the statements above?

A. 1. is
2. E-0 (Reactor Trip or Safety Injection)

B. 1. is NOT
2. E-0 (Reactor Trip or Safety Injection)

Original Question

C. 1. is
2. E-2 (Faulted Steam Generator Isolation)

MODIFIED

D. 1. is NOT
2. E-2 (Faulted Steam Generator Isolation)

Question 92

APPLICABLE SAFETY ANALYSES (continued)

worth, and with at least minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that one DRPI System and one Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

a. The DRPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits";

b. For the DRPI System either Data A or Data B is OPERABLE for each rod; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore,

Question 92

6.4 **IF** a control rod fails to withdraw or a single rod is dropped during approach to criticality perform one of the following:

Correct Answer Part 2

- **IF** malfunction is in Control Bank, insert all Control Banks.
- **IF** malfunction is in Shutdown Bank, insert all Control and Shutdown Banks.

This guidance is more conservative than that given in AP/1(2)/5500/015, Rod Control Malfunction, and therefore shall take precedence. {PIP C-06-4287} Distractor Part 2

6.5 **IF** more than one control rod drops during the approach to criticality, **MANUALLY TRIP** the reactor per AP/1(2)/A/5500/14, Control Rod Misalignment.

6.6 **IF** an alarm is received on the Rod Control System or DRPI requiring rod withdrawal to be halted **AND** IAE cannot determine cause for alarm and repair problem or determine if further rod withdrawal is permissible, reinsert all Control Banks.

- **IF** Startup is **NOT** xenon-free (xenon worth ≥ 100 pcm), take action within 30 minutes of malfunction.
- **IF** startup is xenon-free (xenon worth < 100 pcm), take action within 60 minutes of malfunction.
- Obtain ICRR data at 10-minute intervals for the duration of the delay.

6.7 To reduce uncertainties in achieving criticality, T-AVG shall be maintained between 555 and 559 °F during approach to criticality.

6.8 **IF** diluting with BDMS enabled, periodically monitor and reset the BDMS actuation setpoint.

6.9 **IF** abnormal changes in count rate (i.e., irregular count rates, instrument drift, etc.) are observed on either Source Range or BDMS detector, rod withdrawal shall be suspended. Rod withdrawal may be resumed only after the source of the abnormality has been identified and it has been determined that it will not jeopardize plant safety.

6.10 **IF** it is expected that criticality will be achieved above the Upper Allowable Limit (UAL)/Rod Withdrawal Limits **OR** below the Lower Allowable Limit (LAL)/Rod Insertion Limits, insert all control banks and contact Reactor Systems Engineering Supervisor or designee.

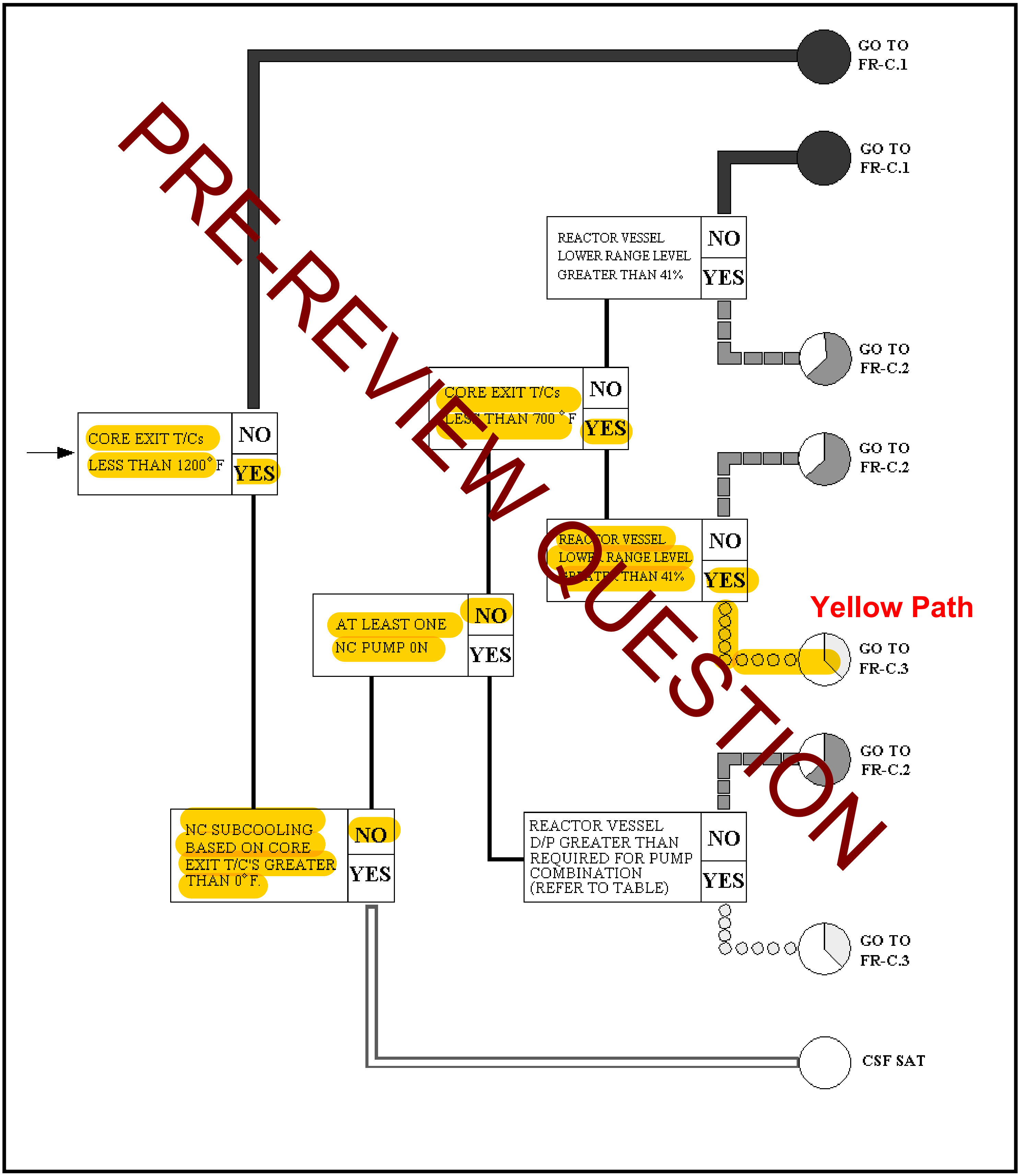
6.11 Ensure that the NC boron sample used for reactivity balance calculations is representative of current NC system boron (i.e. taken with all four NC pumps operating, sufficient time allowed for mixing after last boron change, etc.).

6.12 Prerequisite steps in Sections 4, 7, and 8 may be signed off in any order.

Question 93

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES CORE COOLING - Page 1 of 1	PAGE NO. 4 of 11 Revision 9

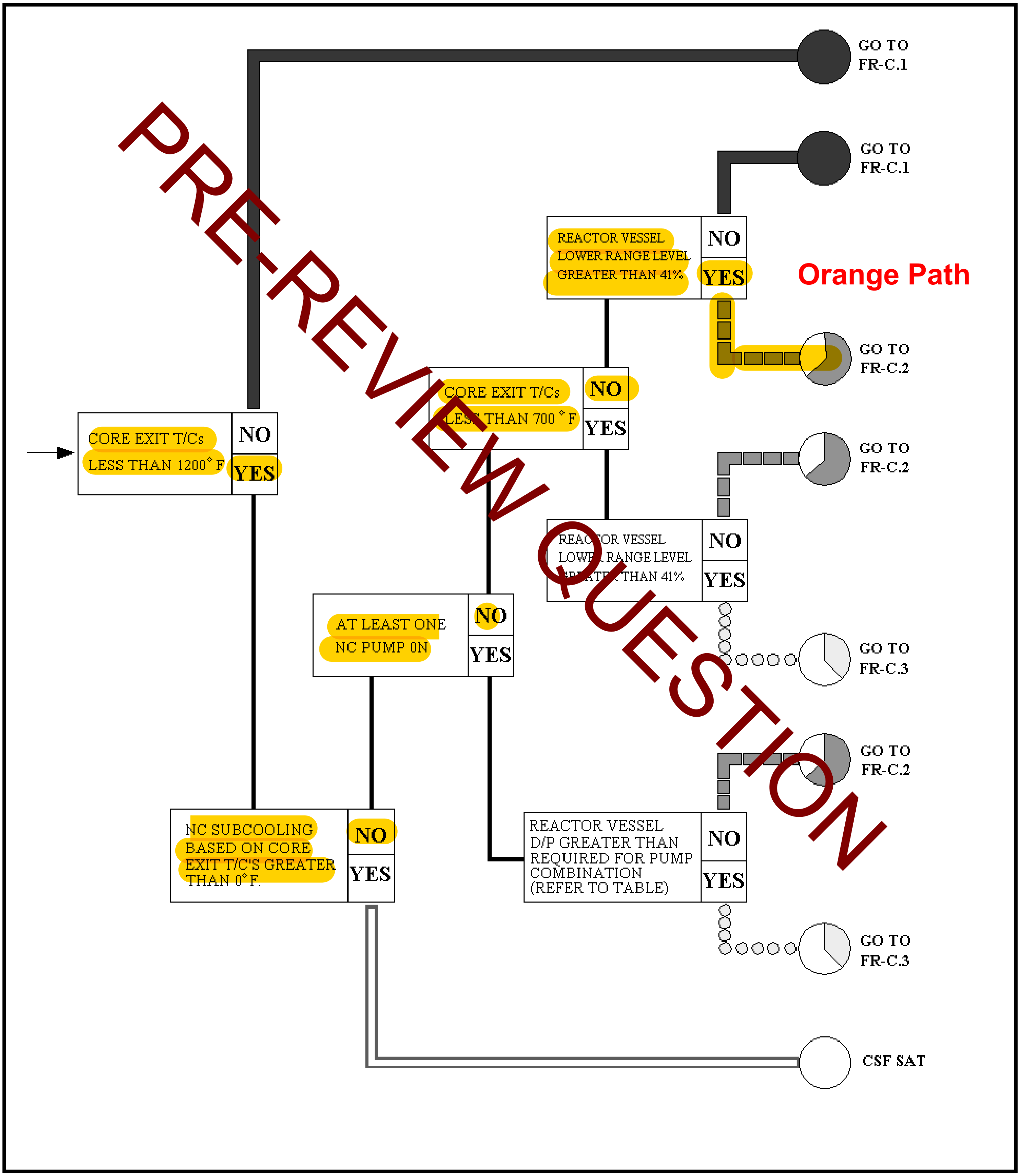
Given Indications at 0930



Question 93

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES CORE COOLING - Page 1 of 1	PAGE NO. 4 of 11 Revision 9

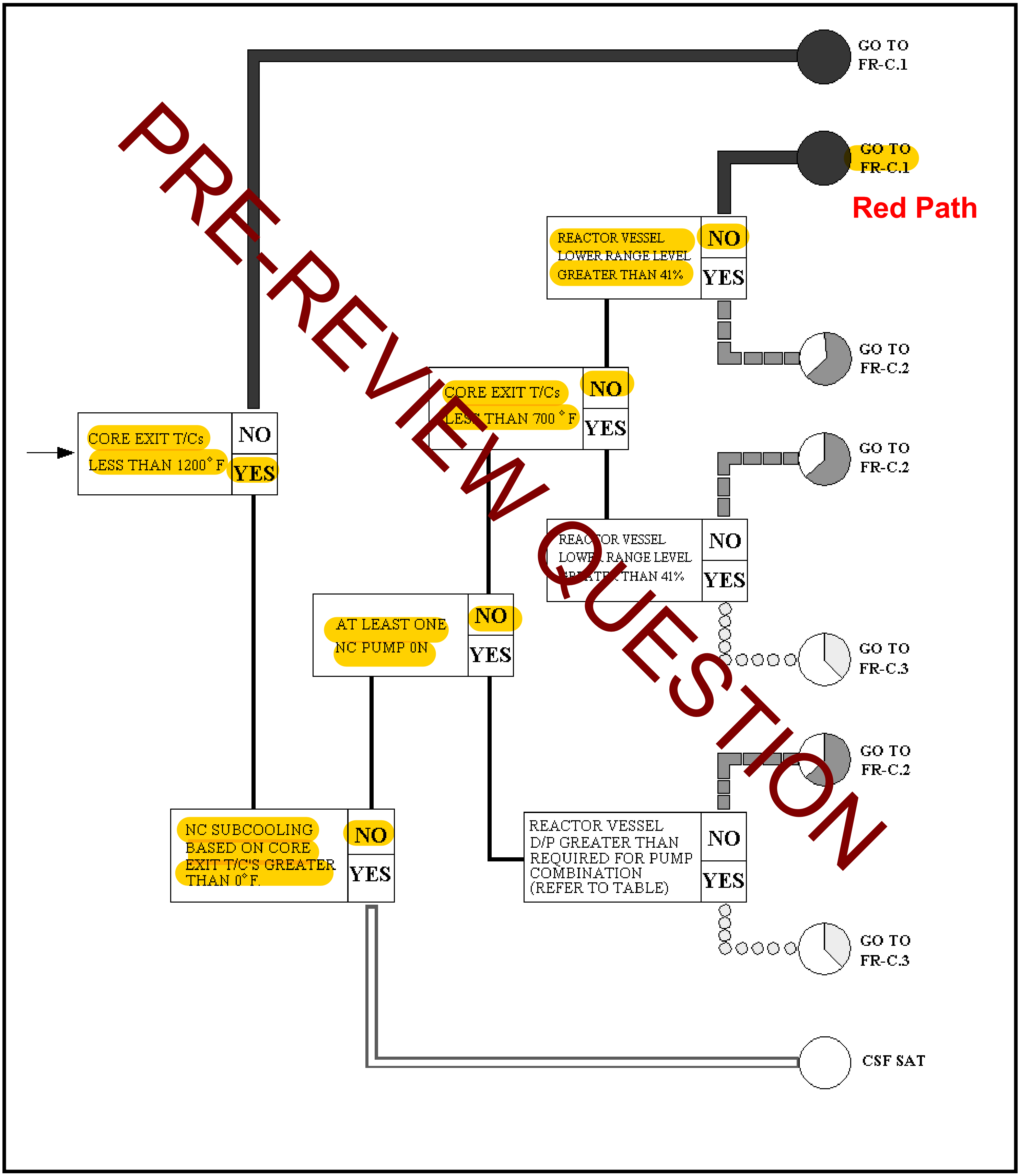
Given Indications at 0945



Question 93

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES CORE COOLING - Page 1 of 1	PAGE NO. 4 of 11 Revision 9

Given Indications at 1015



Question 93

F Fission Product Barriers	FG1.1 1 2 3 4	FS1.1 1 2 3 4	(Note 11, 12) FA1.1 1 2 3 4
	Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1) Correct Classification at 1015	Loss or potential loss of **any** two barriers (Table F-1) Correct Classification at 0945	**Any** loss or **any** potential loss of either Fuel Clad or NCS (Table F-1) Initial Alert Classification 0910

Time After S/D	FC Loss	NCS Loss	CMT Potential Loss
0 - 1	550	8.8	5500
1 - 2	400	8.4	4000
2 - 8	160	7.0	1600
>8	100	6.2	1000

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (NCS) Barrier		Containment (CMT) Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A. NCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER**: • UNISOLABLE NCS leakage • SG tube RUPTURE	1. CSFST Integrity-RED PATH conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B. Inadequate Heat Removal	1. CSFST Core Cooling-RED PATH conditions met	1. CSFST Core Cooling-ORANGE PATH conditions met 2. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED PATH conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C. CMT Radiation / NCS Activity	1. EMF53A/B > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. EMF53A/B > Table F-2 column "NCS Loss"	None	None	1. EMF53A/B > Table F-2 column "CMT Potential Loss"
D. CMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of NCS leakage outside of containment	1. CSFST Containment-RED PATH conditions met 2. Containment hydrogen concentration > 6% 3. Containment pressure > 3 psig with < one full train of containment cooling operating per design for > 15 min. (Notes 1, 10)
E. EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Question 94

5.2 Procedure Compliance Standards During Abnormal and Emergency Operations (continued)

e. If a Variance is utilized, then at a minimum, enter the following relevant information into the Operations Narrative Logs:

- Variance description
- Any SRO concurrence obtained
- Basis for the Variance (e.g., public health and safety, personnel injury, or plant equipment damage)

f. Ensure necessary measures to maintain Plant Status Control are performed.

g. Initiate a NCR to evaluate the effectiveness of the transient response procedures, including Variances implemented during simulator scenarios.

(1) Ensure the word "Variance" is in the NCR subject to support NCR trending.

7. Deviations - 10 CFR 50.54x Departure from Licensed Condition for Protection of Public Health and Safety

a. Deviations from Technical Specifications or License Condition during normal plant operation are **NOT** allowed.

(1) In the event of an emergency, the SM has the authority and responsibility to take action necessary to protect the health and safety of the public as allowed by 10 CFR 50.54(x), 10 CFR 50.54(y), and 10 CFR 72.

b. In order to invoke a Deviation, no action consistent with License Conditions and Technical Specifications that can provide adequate or equivalent protection is immediately apparent.

c. If the TSC is activated, then the TSC Emergency Coordinator determines to depart from license conditions.

(1) If emergency responsibilities are assumed, then the TSC Emergency Coordinator has higher authority for command and control.

(2) Prior to exiting the license conditions, the TSC Emergency Coordinator should consult with a SRO, but the TSC Coordinator is **NOT** required to be a SRO and does **NOT** require SRO concurrence for departing license conditions

5.2 Procedure Compliance Standards During Abnormal and Emergency Operations (continued)

d. If the TSC is **NOT** activated, then a SRO must approve a Deviation prior to taking action. If time allows, then concurrence from a second SRO should be obtained prior to taking action.

e. Enter the relevant information concerning the Deviation into the Unit Log as soon as conditions allow.

f. Notify the NRC per 10 CFR 50.72 requirements due to invoking 50.54(x).

(1) Notify the NRC Operations Center using the Emergency Notification System (ENS).

(2) When time permits, then notify the NRC Operations Center prior to taking the action.

(3) Otherwise, notify the NRC Operations Center as soon as possible and in all cases, within 1 hour after taking the action.

g. Notify the Resident NRC Inspector as soon as practical.

h. Submit an NCR to review the circumstances and determine whether a change to the procedure is required to prevent a Deviation in the future.

(1) Include the word 'Deviation' in the NCR title to support trending.

8. Deviation - 10 CFR 50.54(dd) Departure from Licensed Condition for National Security

a. In the event of a national security emergency, the SM has the authority to take action necessary to implement national security objectives as stated in 10 CFR 50.54(dd) which reads as follows:

(1) A licensee may take reasonable action that departs from license condition or a technical specification in a national security emergency when this action is immediately needed to implement national security objectives as designated by the national command authority through the NRC.

AND

(2) No action consistent with license conditions and technical specifications that can meet national security objectives is immediately apparent.

Question 95

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	$\geq$ 0.99	> 5	NA
2	Startup	$\geq$ 0.99	$\leq$ 5	NA
3	Hot Standby	< 0.99	NA	$\geq$ 350
4	Hot Shutdown[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	$\leq$ 200
6	Refueling[c]	NA	NA	NA

Distractor Part 1

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

Question 95

Catawba 1 Cycle 26 Core Operating Limits Report

2.16 Standby Shutdown System - (SLC-16.7-9)

2.16.1 Minimum boron concentration limit for the spent fuel pool required for Standby Makeup Pump Water Supply. Applicable for MODES 1, 2, and 3.

Parameter	Limit
Spent fuel pool minimum boron concentration for TR 16.7-9-3.	2,700 ppm

2.17 Boration Systems Borated Water Source – Shutdown (SLC 16.9-11)

2.17.1 Volume and boron concentrations for the Boric Acid Tank (BAT) and the Refueling Water Storage Tank (RWST) during MODE 4 with any RCS cold leg temperature ≤ 210 °F, and MODES 5 and 6.

Distractor Part 2

NOTE: When cycle burnup is ≥ 445 EFPD, Figure 6 may be used to determine the required BAT Minimum Level.

Parameter	Limit
BAT minimum boron concentration	7,000 ppm
Volume of 7,000 ppm boric acid solution required to maintain SDM at 68°F	2,000 gallons
BAT Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-11)	13,086 gallons (14.9%)
RWST minimum boron concentration	2,700 ppm
Volume of 2,700 ppm boric acid solution required to maintain SDM at 68 °F	7,000 gallons
RWST Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-11)	48,500 gallons (8.7%)

Correct Answer Part 2

Question 95

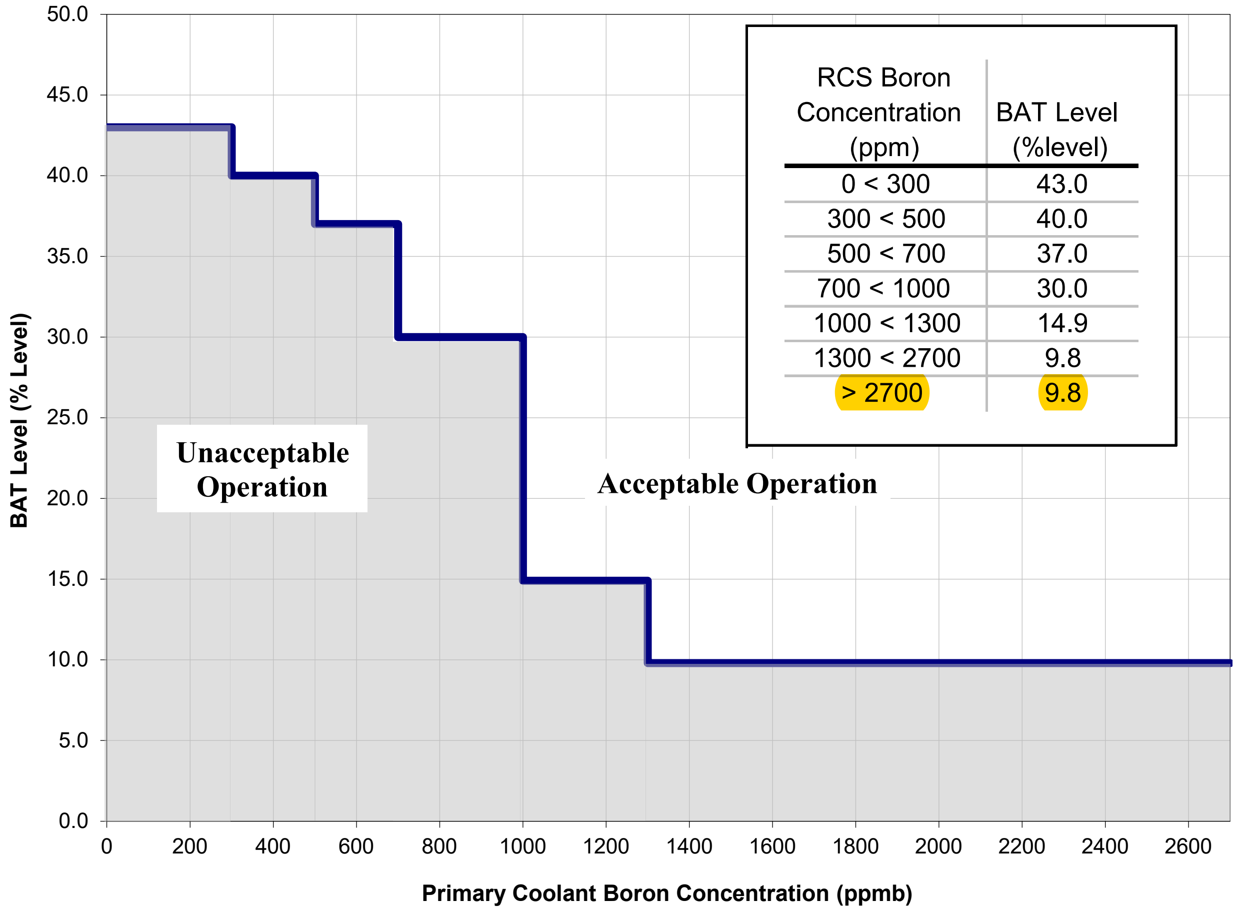
Catawba 1 Cycle 26 Core Operating Limits Report

Figure 6

**Boric Acid Storage Tank Indicated Level Versus
Primary Coolant Boron Concentration**

(Valid When Cycle Burnup is ≥ 445 EFPD)

This figure includes additional volumes listed in SLC 16.9-11 and 16.9-12



RCS Boron Concentration (ppm)	BAT Level (%level)
0 < 300	43.0
300 < 500	40.0
500 < 700	37.0
700 < 1000	30.0
1000 < 1300	14.9
1300 < 2700	9.8
> 2700	9.8

Distractor Part 2

Question 95

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding to approximately 75 percent of its required level and then using the refueling water pump to complete the filling process or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

Question 96

4.6 **Shift Manager**

1. Determines appropriateness of the use of clearances in times of emergency.
2. Ensures plant status control is maintained in accordance with AD-OP-ALL-0204, Plant Status Control, and this procedure.

Correct Answer Part 2

3. Approves Exceptional Clearances along with a Work Group Supervisor.

4.7 **Clearance Approver**

1. Maintains an active or current SRO license.

Distractor Part 2

2. Approves EEI's, and the hang, temp lifts, removal of clearances.
3. Maintains Control Room personnel informed of all plant status changes prior to establishing or removing a clearance.
4. Ensures closeout of clearances.
5. Ensures the impact of clearances on the station is understood.

4.8 **Clearance Requestor**

1. Requests a clearance in accordance with the guidance of this procedure and AD-MN-ALL-0005, Nuclear Planning.
2. Typically this role is performed by the Maintenance Planner, with support from the craft supervisor as described in the Work Order Planning Process. However, other personnel may request a clearance.

4.9 **Clearance Preparer**

1. Designs the protection boundary for the work.
2. Verifies the clearance request contains a scope statement of sufficient detail to communicate the exact work scope to be included in the clearance boundary.
3. Applies systems and component knowledge, as well as knowledge of procedures, operating experience, and regulations to develop the clearance boundary.
4. Contacts the Clearance Requestor or Work Group Supervisor as necessary to ensure understanding of the work scope and clearance boundary needs.

Question 96

5.6 Clearance Reviewer

1. For Active Clearances being reviewed from a Model Clearance, use the guidance in Section 1.0 of Attachment 5, Clearance Hang Preparation Checklist. For all other Clearances or new Model Clearance generation continue in this section.

    a. If the Active Clearance is being reviewed from a Model and does not meet any of the reviewer criteria in Section 1.0 of Attachment 5, Clearance Hang Preparation Checklist, then the Active Clearance shall be deleted or prepared and reviewed using section 2.0 if Attachment 5. Additionally, action shall be taken to revise the Model.

2. The purpose of the clearance review is to ensure that the design of the clearance is technically correct, that good isolation practices have been used, and that appropriate procedural requirements have been met.

    a. The clearance reviewer is charged with making an informed decision about the technical accuracy and safety adequacy, in all respects, of the clearance that is presented for review.

    b. The clearance reviewer shall maintain an independent perspective on the clearance.

    Distractor Part 1

    c. For all clearances (except Keowee), the review shall be performed by a licensed operator or a previously licensed operator at that station.

        (1) For Keowee Operations clearances, the review shall be performed by a Keowee Operations Tech 3 or exempt Keowee personnel.

3. A detailed understanding of the system is needed by the clearance reviewer to identify parallel flowpaths, remote controlling locations, electrical backfeed sources, proximity to energized sources, multiple power sources, and so on. When in doubt, the appropriate technical help should be sought. In making an informed decision about the clearance, the following techniques are ways to ensure an adequate technical review by improving the independence of the clearance reviewer's perspective:

    a. Do not discuss the clearance with the clearance preparer until the clearance reviewer has developed his or her own idea of the clearance boundary.

    b. References used by the clearance preparer may be used, but no marked up items are addressed until after a review of the task and the creation of an independent boundary has been completed. Previously marked up prints may be used for review of outage block/master clearances.

    c. Independently assess and understand the hazards.

Question 97

<< Defense-In-Depth (DID) Status Sheet Guidelines >>

NOTE

The Defense In Depth Sheets combine the generic needs and risk analysis methods common to all sites. The site specific DEFENSE-IN-DEPTH Basis documents give site specific details to guide Operators and Work Control personnel to correctly assess and manage risk for that site. Compliance with the defense in depth plan is verified at least once per shift and before major safety systems or components are removed from service that affects the Defense-in-Depth sheet. {7.1.5}

1.0 GUIDANCE FOR DEFENSE-IN-DEPTH (DID) STATUS SHEETS

1. During outage development, Site Outage Management will develop DID sheets for unique configurations for review during the Independent Review Team outage risk assessment using Attachment 2, [BWR] Defense-in-Depth (DID) Status Sheet.

2. When entering Mode 4 (PWR) or Mode 3 (BWR), the risk will be analyzed using the DID sheet.

3. During an outage an Outage Risk Analyst will generate a DID sheet for each of the following:

   a. Prior to entering a new configuration to ensure risk will be acceptable.

   b. When an un-scheduled change occurs such as equipment failure or new external event.

   c. Once every 12 hours (per shift).

4. During Outage planning and before entering Mode 4 (PWR) or Mode 3 (BWR), the Shift Manager and Work Control Outage Manager shall review the configuration requirements and notes. {7.1.7}

   a. The Shift Manager and Work Control Outage Manager then concur that the status of plant equipment agrees with the planned shutdown Risk Management configuration and the Shift Manager will direct mode entry. {7.1.10}

5. Before manipulating the plant into the next configuration, the Shift Manager will review and approve the DID status. {7.1.7} {7.1.10} {7.1.11}

Question 97

5.4 Plant Condition Mode Changes (PCMC) Coding

1. The purpose of PCMC coding is to aid tracking and review of the status of equipment required by Technical Specifications (TSs), Selected Licensee Commitments (SLCs), or other mode dependent commitments for specific plant conditions or modes to assist Operations in verifying conditions to make a Mode Change. [7.3.2]

   Distractor Part 2

2. During schedule development and revision all WOTs shall be PCMC coded.

3. The following are the required minimum conditions to apply PCMC codes. Additional PCMC codes can be used in conjunction with these if needed.

   - **MODE / Plant Condition Not Applicable:** The value entered in CAS is 0 to show the WO had been reviewed as N/A to Mode or Plant Condition requirements.
   - **MODE 1:** The work must be completed before Mode 1 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 1.
   - **MODE 2:** The work must be completed before Mode 2 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 2.
   - **MODE 3:** The work must be completed before Mode 3 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 3.
   - **MODE 4:** The work must be completed before Mode 4 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 4.
   - **MODE 5:** The work must be completed before Mode 5 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 5.
   - **MODE 6 Fuel Movement:** Work that must be completed before initiating fuel movement.
   - **MODE 6 Core Alterations:** The work must be completed prior to beginning core alterations in MODE 6.
   - **Less than 23 Feet in the Refueling Cavity:** The work must be completed before draining below 23 feet in the refueling cavity.
   - **Low Temperature Overpressure Protection (LTOP):** The work must be completed before entering LTOP.

Question 97

5.4 Plant Condition Mode Changes (PCMC) Coding (continued)

- **Mode 3T:** Turbine Driven Auxiliary Feedwater Pump-Testing that must be completed in MODE 3 at the appropriate steam pressure before declaring the Turbine Driven Auxiliary Feedwater Pump fully operable. All other work and testing on the Turbine Driven Auxiliary Feedwater pump that can be completed independent of the pump IWP must be completed prior to MODE 3.

- **Mode 5T - Rod Control:** The work must be completed prior to making the Control Rods capable of being withdrawn. This will include work on rod position indication, Source Range Instrumentation, Automatic Trip and Interlock Logic, Manual Reactor Trip, and Reactor Trip Breakers.

4. The Operations SOCs, a Senior Reactor Operator (SRO), or individuals designated by Operations Management shall review all required WOTs to be worked during an outage and apply proper PCMC codes. [7.3.1]

   a. PCMC is not intended to track operability of equipment required for the current plant condition or mode. [7.3.2] **Correct Answer Part 2**

   b. The AOM-Work Control, a SRO, or designated persons by Operations Management shall review all WOTs to be work worked during an outage to ensure proper PCMC coding has been applied.

   c. This applies to WOTs coded prior to the outage start as well as scope identified during the outage time period.

5. Operations SOCs or an SRO shall perform a second review of PCMC codes of WOTs that are amended after the initial review to ensure proper PCMC coding. [7.3.1]

5.5 <u>Outage Window/Segment Templates</u>

1. Establish site refueling outage window/segment templates at each site with a goal of standardizing window/segment durations and work practices across the fleet to the greatest extent feasible to support fleet outage coordination.

2. Schedule templates will be the minimum Critical Path schedule to shutdown, offload fuel, shuffle fuel, reload fuel, heat up, and perform required Technical Specification testing. No other work is assumed.

3. Base schedule templates and activity durations with plants of similar design on best times achieved across the fleet verses best times at a particular site.

Question 98

TESTING REQUIREMENTS

---NOTE---
The Liquid Radwaste Treatment System shall be demonstrated FUNCTIONAL by meeting SLC 16.11-1 and SLC 16.11-3.

TEST		FREQUENCY
TR 16.11-4-1	Project liquid release doses from each unit to UNRESTRICTED AREAS, in accordance with the methodology and parameters in the ODCM, when the Liquid Radwaste Treatment System is not being fully utilized.	31 days

BASES

The FUNCTIONALITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept “as low as is reasonably achievable”. This COMMITMENT implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

This SLC applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared radwaste treatment systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the radwaste treatment system. For determining conformance to COMMITMENTS, these allocations from shared radwaste treatment systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

REFERENCES

1. Catawba Offsite Dose Calculation Manual.

2. 10 CFR Part 50, Appendix A.

Question 99

5.1.5 Reactor Building Entry (Modes 3 & 4) Distractor Part 1

- Access is normally controlled by the WCCSRO.
- Control may be transferred to another responsible group if that group posts a person at the personnel hatch.
  - This person shall be responsible for maintaining the Containment Access Log Book and to account for persons inside Containment.
- Each time responsibility for Containment access is transferred to another group from or to Operations, the date, time and name of the other group shall be entered in the Unit Unified Logbook.

5.1.5.1 Upper Containment

A. Entry into Upper Containment requires pre-job authorization for each job from the Radiation Protection Supervisor or designee.

B. Dependent on location of the Containment Access Log Book, personnel entering Upper Containment shall notify the WCCSRO or Personnel Hatch Watch.

C. A Task Preview shall be given by WCCSRO or Personnel Hatch Watch.

5.1.5.2 Lower Containment

A. Entry into Lower Containment requires pre-job authorization for each job from Radiation Protection Management dependent upon Radiation Protection postings/survey results.

B. Dependent on location of Containment Access Log Book, personnel entering Lower Containment shall notify the WCCSRO or Personnel Hatch Watch.

C. A Task Preview shall be given by WCCSRO or Personnel Hatch Watch.

Question 99

BASES

APPLICABLE SAFETY ANALYSES (continued)

containment was designed with an allowable leakage rate of 0.30% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.30% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 14.68 psig following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is

ACTIONS (continued)

preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

Note that for the purpose of Required Action A.1, A.2 and A.3, the bulkhead associated with an air lock door is considered to be part of the door. For example, an air lock door may be declared inoperable if the equalizing valve becomes inoperable or if it is replaced. It is appropriate to treat the associated bulkhead as part of the door because a leak path through the bulkhead is no different than a leak path past the door seals.

- If a valid orange path is encountered, the operator is expected to scan all of the remaining trees, and then, if no red path is encountered, to promptly implement the corresponding EP. If during the performance of an orange path procedure, any red condition or higher priority orange condition arises, then the red or higher priority orange condition shall be addressed first, and the original orange path procedure suspended.

- Once a procedure is entered due to a valid red or orange condition, that procedure shall be performed to completion unless preempted by some higher priority condition. It is expected that the actions in the procedure will clear the red or orange condition before all the operator actions are complete. However, these procedures shall be performed to the point of the defined transition to a specific procedure. At this point, any lower priority red or orange paths currently indicating or previously started but not completed shall be addressed.

- If a CSF procedure directs the operator to return to the procedure and step in effect and the corresponding status tree continues to display the off normal condition, then the corresponding CSF procedure does not have to be implemented again since all recovery actions have already been completed. However, if the same status tree subsequently changes to a valid higher priority condition, then the corresponding CSF procedure shall be implemented as required by its priority.

- Certain CSF procedures are used to address both orange and red path conditions for the same parameters. If the procedure is already in progress due to the orange path condition, it is not required to return to the first step if the condition becomes red. Also, at the completion of the procedure, the procedure does not have to be implemented again, since all recovery actions have already been implemented.

Question 22

Operations Liquid Waste Release

1. Purpose

To aid the operator in the correct methods of performing steps in Radwaste procedure OP/0/B/6500/015 (Discharging a Monitor Tank to the Environment) and Radiation Protection procedure HP/0/B/1004/004 (Radioactive Liquid Waste Release). Also to aid the operator as to limits and results expected while these procedures are being performed.

2. Limits and Precautions

2.1 Ensure that RN is discharging through at least one RL header.

2.2 Ensure that RN is **NOT** discharging to SNSWP.

2.3 If the pre-set radiation levels are exceeded on EMF-49 or the dilution flow rate drops below the setpoint for 0RLP5080 (RL Discharge Total Flow), 1WL-124 (Waste Monit Tnk Pmps Disch) will trip closed.

2.4 Releases that are interrupted by EMF-49 "HI-RAD" trips may be initiated up to a maximum of three times, including original initiation, without re-sampling per HP/0/B/1004/004 (Radioactive Liquid Waste Release).

2.5 Turbine Building Sump releases are secured if the pre-set levels are exceeded on 1/2EMF-31.

3. Procedure

Refer to Section 4 (Enclosures)

4. Enclosures

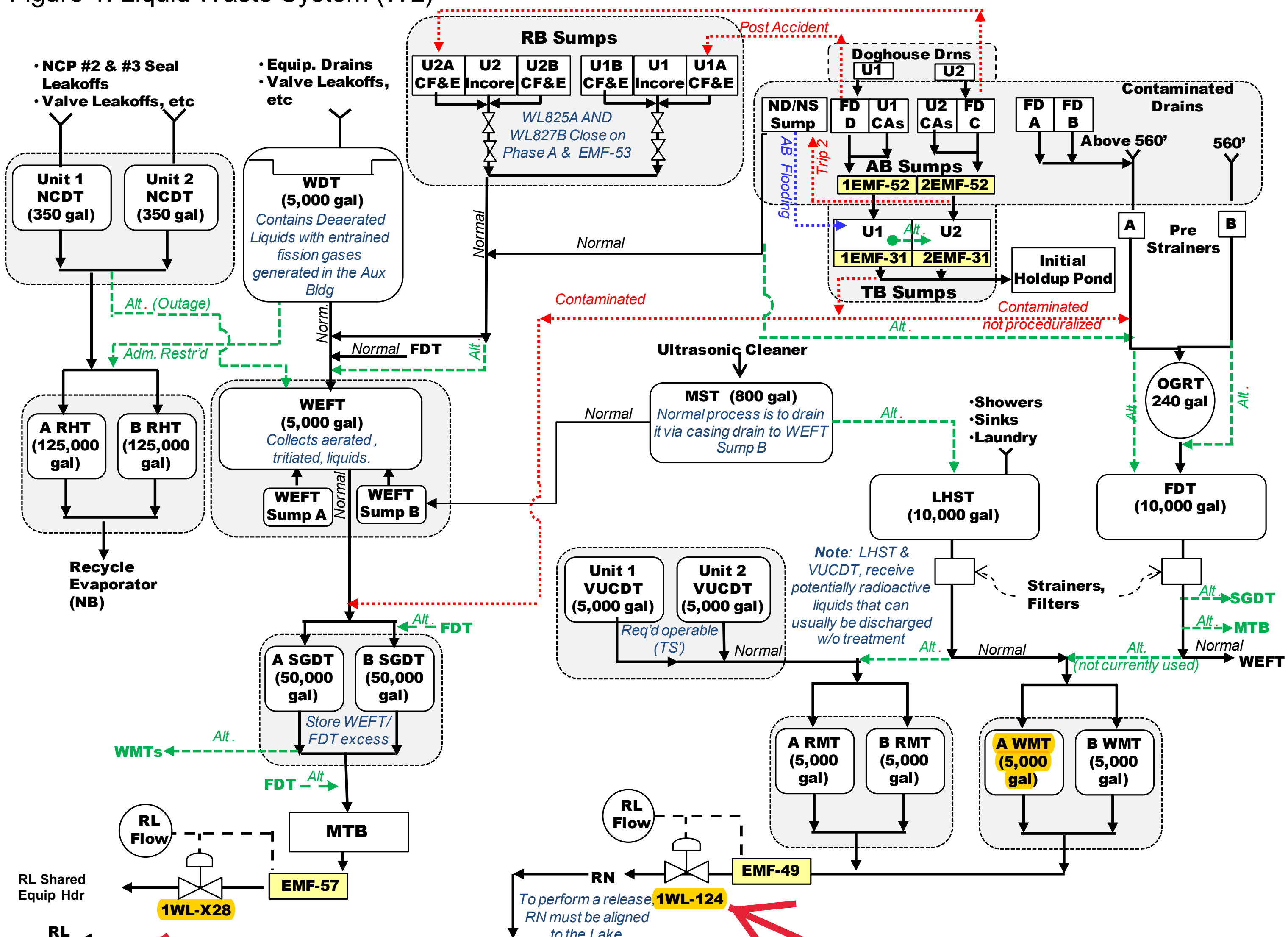
4.1 Liquid Waste Release from a Monitor Tank

4.2 Discharging a Contaminated Turbine Building Sump to Holdup Pond

Question 22

28. FIGURES

28.1 Figure 1: Liquid Waste System (WL)



Distractor Part 2

Correct Answer Part 2

Question 23

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Attempt to identify and isolate leak as follows:**	
a. Verify the following conditions - NORMAL: __ • Containment temperature __ • Containment pressure __ • Containment humidity __ • Containment floor & equipment sump level.	a. Perform the following: __ 1) Evacuate containment. 2) Perform the following: __ a) Start all lower containment ventilation units in low speed. __ b) Start all upper containment ventilation units. __ c) Place all upper and lower containment ventilation units in "MAX" cooling. 3) **IF AT ANY TIME** containment pressure reaches 1.2 PSIG, **THEN** perform the following: __ a) Ensure Unit 1 reactor tripped. __ b) Ensure S/I initiated. c) CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves. __ d) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). __ 4) **GO TO** Step 10.
__ b. Dispatch operators to locate and identify source of steam leak.	

Question 24

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16. **WHEN "P-11 PZR S/I BLOCK PERMISSIVE" status light (1SI-18) lit, THEN perform the following:**	
__ a. Depress ECCS steam pressure "BLOCK" pushbuttons.	
__ b. Verify main steam isolation blocked status lights (1SI-13) - LIT.	

NOTE
- Partial uncovering of the S/G tubes is acceptable in the following steps.
- After low steamline pressure main steam isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation.
- OAC graphic SMRATES to monitor S/G pressure rates can be accessed via a hot button in the center of the SM graphic.

C-9 NOT lit due to loss of offsite power

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. **Depressurize all intact S/Gs as follows:**	
a. Verify condenser available as follows: __ • "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT __ • MSIV on intact S/G(s) - OPEN.	__ a. **GO TO** Step 17.d RNO.
__ b. Verify steam dumps - IN PRESSURE MODE.	b. Place steam dumps in pressure mode as follows: __ 1) Place "STM DUMP CTRL" slim station in manual. __ 2) Place steam dumps in pressure mode.
__ c. **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK".	

Question 24

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ d. Dump steam to condenser at maximum rate while attempting to avoid Main Steam Isolation. **Distractor Part 2**	__ d. Dump steam from all intact S/G(s) with S/G PORV(s) at maximum rate. **Correct Answer**
__ e. Verify all intact S/G pressures - LESS THAN 140 PSIG.	e. Perform the following: __ 1) **IF** S/G pressure trending down, **THEN RETURN TO** Step 13. __ 2) **GO TO** Step 24.
__ f. Verify at least two NC T-Hots - LESS THAN 370°F.	f. Perform the following: __ 1) **IF** NC T-Hots trending down, **THEN RETURN TO** Step 13. __ 2) **GO TO** Step 24.
__ g. Stop S/G depressurization and maintain S/G pressures stable.	

Question 24

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15. (Continued) __ d. Dump steam to condenser while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour. **Distractor Part 1**	d. Perform the following: __ 1) Maintain cooldown rate based on NC T-Colds less than 100°F in an hour while dumping steam in the following steps. __ 2) Dump steam using intact S/G PORV(s). 3) **IF** any intact S/G PORV cannot be operated from Control Room, **THEN** perform the following: __ a) Dispatch operator(s) to operate affected S/G(s) PORV. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 10 (Local Operation of S/G PORVs). __ b) Obtain sound powered phone from storage box on rear wall of Control Room. __ c) Connect sound powered phone to jack on 1MC-11. __ d) Monitor sound powered phone for communication from Doghouse(s). 4) **IF** any intact S/G PORV unavailable, **THEN** evaluate using the following to dump steam: __ • OPEN MSIVs and dump steam to condenser. **REFER TO** Enclosure 4 (Condenser Dump Operation). __ • Start CA Pump #1.

Question 25

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Verify S/I termination criteria:**	
__ a. NC subcooling based on core exit T/Cs - GREATER THAN 0°F.	__ a. **<u>GO</u> <u>TO</u>** Step 9.f.
ACC Conditions apply due to Containment pressure > 3.0 psig	
b. Secondary heat sink:	__ b. **<u>GO</u> <u>TO</u>** Step 9.f.
__ • Any intact S/G N/R level - GREATER THAN 11% (29% ACC) OR __ • Total feed flow to intact S/Gs - GREATER THAN 450 GPM.	
__ c. NC pressure - STABLE OR TRENDING UP.	__ c. **<u>GO</u> <u>TO</u>** Step 9.f.
__ d. Pzr level - GREATER THAN 11% (30% ACC).	d. Perform the following: __ 1) **<u>IF</u>** NC pressure trending up **<u>AND</u>** normal Pzr spray available, **<u>THEN</u>** attempt to stabilize NC pressure using normal Pzr spray. __ 2) **<u>GO</u> <u>TO</u>** Step 9.f.
__ e. **<u>GO</u> <u>TO</u>** EP/1/A/5000/ES-1.1 (Safety Injection Termination).	
__ f. Monitor S/I termination criteria. **<u>REFER</u> <u>TO</u>** Enclosure 2 (S/I Termination Criteria).	
__ g. **<u>IF</u> <u>AT</u> <u>ANY</u> <u>TIME</u>** S/I termination criteria met while in this procedure, **<u>THEN</u> <u>RETURN</u> <u>TO</u>** Step 9.	

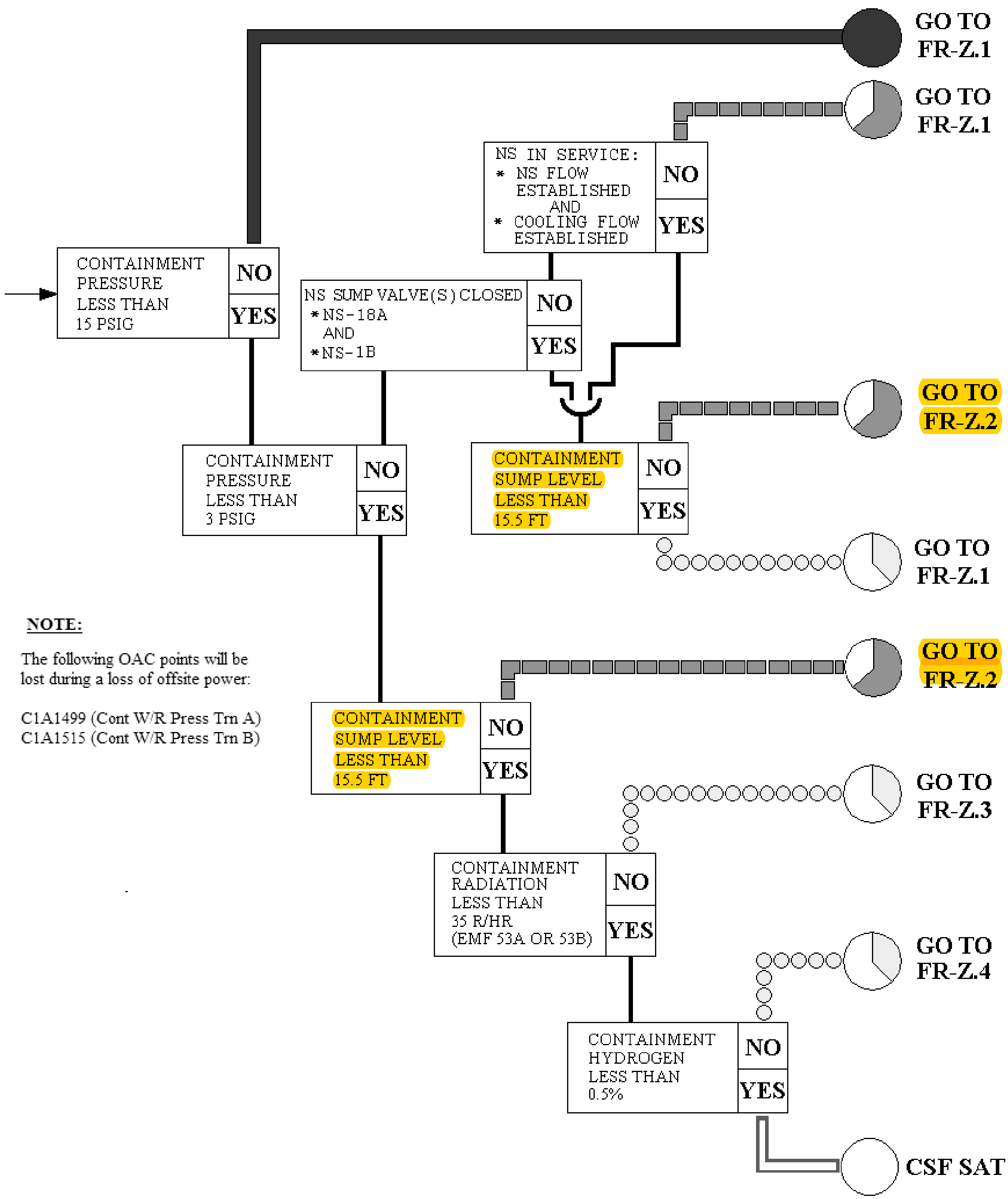
Correct Answer Part 1

Correct Answer Part 2

Distractor Part 2

Question 26

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES CONTAINMENT - Page 1 of 1	PAGE NO. 9 of 11 Revision 9



NOTE:

The following OAC points will be lost during a loss of offsite power:

C1A1499 (Cont W/R Press Trn A)
C1A1515 (Cont W/R Press Trn B)

Question 26

Containment boundary penetrations for isolation valves, personnel air locks and equipment hatches are considered in the maximum leak rate calculation (L_a).

The isolation valves or devices are passive or active (automatic). Automatic devices provide two isolation barriers that are closed on a containment isolation signal. Examples of manual devices include de-activated automatic valves, closed manual valves, blind flanges, check valves, or other automatic valves designed to close without operator action.

Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses.

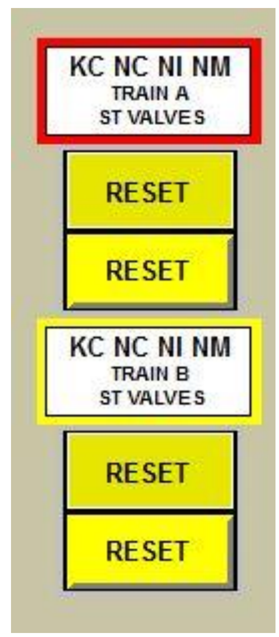
Containment Isolation Valves

Containment isolation valves are identified in UFSAR table 6-77 (Containment Isolation Valve Data). UFSAR Table 6-77 lists all containment isolation valves and indicates the appropriate Tech Spec condition for inoperable valves. Most are automatically operated valves. Some are manual valves that are normally locked closed. A few are manually operated and normally open.

Post-Accident Valve Submergence

A list of active valves in containment that are below maximum flood elevation is presented in UFSAR Table 6-96.

Some valve operators were not qualified for submergence. These valves close on Containment Isolation Phase A (S_T) signals. There is sufficient time for them to close before being flooded.



To prevent possible repositioning after flooding, the valves motor controls circuits have been modified. One relay per train will be energized by a Phase A (S_T) signal and mechanically latched in. Normally closed contacts from this relay will be wired between the limit switches and the open motor starter coils of valves of the corresponding train. These contacts will open on S_t and prevent any spurious limit switch operation from repositioning the valves.

These relays have manual reset capability in the control room. (KC, NC, NI, NM S_T)

Correct Answer Part 2

Catawba Nuclear Station

2019 CNS RO NRC Examination

Question: 43
(1 point)

Given the following conditions on Unit 1:

- A LOCA has occurred
- Containment pressure peaked at 3.3 PSIG, and is now 2.5 PSIG and slowly lowering
- Crew has entered EP/1/A/5000/ES-1.2 (Post LOCA Cooldown and Depressurization) and is performing the initial cooldown


LOOP DATA

		LOOP A	LOOP B	LOOP C	LOOP D
CURRENT T-COLD, BEST	(DEG F)	546.0	546.4	546.1	546.1
ADMINSTRATIVE LIMIT	(DEG F)	479.8	480.2	479.8	479.8
TECH SPEC LIMIT	(DEG F)	459.8	460.2	459.8	459.8
T-COLD CURRENT MINUS 1 HR T-COLD MAXIMUM	(DEG F)	-12.8	-13.6	-12.3	-12.1
15-MIN RATE	(DEG F/HR)	-38	-38	-36	-36
5-MIN RATE	(DEG F/HR)	-131	-134	-128	-127
1-MIN RATE	(DEG F/HR)	-113	-107	-111	-102

In accordance with ES-1.2:

The **INITIAL** cooldown will be started using the _____(1)_____.

With rates established, per the graphic above, the cooldown _____(2)_____ continue.

Which ONE (1) of the following completes the statement above?

A. 1. S/G PORVs
   2. can NOT

B. 1. S/G PORVs
   2. can

Original Question

C. 1. Condenser Steam Dumps
   2. can NOT

MODIFIED

D. 1. Condenser Steam Dumps
   2. can

Question 27

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ g. Dump steam to condenser while maintaining cooldown rate based on NC T-Colds as close as possible without exceeding 100°F in an hour.	__ g. **IF** steam cannot be dumped to condenser, **THEN GO TO** Step 10.i.
Correct Answer Part 1	**Distractor Part 1**
__ h. **GO TO** Step 11.	
__ i. Dump steam from intact S/G PORVs while maintaining cooldown rate based on NC T-Colds as close as possible without exceeding 100°F in an hour.	i. **IF** any intact S/G PORV cannot be operated from Control Room, **THEN** perform the following: __ 1) Dispatch operator(s) to dump steam from S/G(s) PORV. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 10 (Local Operation of S/G PORVs). __ 2) Obtain sound powered phone from storage box on rear wall of Control Room. __ 3) Connect sound powered phone to jack on 1MC-11. __ 4) Monitor sound powered phone for communication from the Doghouse(s).
If Containment pressure exceeded 3 psig condenser dumps would not be available	
__ 11. **Verify NC subcooling based on core exit T/Cs - GREATER THAN 0°F.**	__ **GO TO Step 29.**

Catawba Nuclear Station

ILT-17 NRC Written Exam CNS RO NRC Examination

Question: 28
(1 point)

Given the following Unit 1 conditions:

- The Unit is at 100% RTP
- Total charging flow is currently 90 gpm
- 1NV-294 (NV Pmps A&B Disch Flow Ctrl) is in MANUAL
- 1NV-309 (Seal Water Injection Flow) is in AUTO

Assuming stable plant conditions, as 1NV-294 is throttled OPEN, 1NV-309 will throttle in the ______(1)______ direction.

In order to restore automatic control of the Pressurizer Level Control system ______(2)______ must be placed in AUTO.

Which ONE of the following completes the statements above?

A. 1. OPEN
2. 1NV-294 ONLY

B. 1. CLOSED
2. 1NV-294 ONLY

C. 1. OPEN
2. 1NV-294 AND PZR Level Master

D. 1. CLOSED
2. 1NV-294 AND PZR Level Master

Original Question

MODIFIED

Question 28

2.3.8 Charging Header Flow Control Valve, NV-294

Charging Header Flow Control Valve NV-294 is an air operated globe valve which controls charging flow on the discharge of the NV pumps based on pressurizer level.

NV-294 is controlled from MC10 by a SLIM station or via DCS Soft Controls. The SLIM station is normally in the “AUTO” position which allows the Pressurizer Level Control System to control valve position. NV-294 will maintain a minimum flow while in AUTO. This minimum flow is set by the operator. Normally it is set for 47 gpm (32 gpm for seal injection flow and 15 gpm for cooling flow through the regenerative heat exchanger). Anytime NV-294 is placed in Manual, the Pressurizer Level Master Controller will also transfer to Manual to prevent windup in the Pressurizer Level Master Controller (discussed in more detail in the ILE Lesson Plan).

On transfer of control to the ASP, control of NV-294 from the control room is disabled by operator action to swap a plug connection.

Upon loss of air or power, NV-294 fails to the open position to ensure charging, so there is no net loss of inventory from the NV System. If control of NV-294 is lost from the control room, a manual valve, NV-295, located just downstream of NV-294, is available to re-establish throttling.

2.3.9 NCP Seal Injection Flow Control Valve, NV-309

NCP Seal Injection Flow Control Valve, NV-309 is an air operated globe valve which maintains backpressure on the charging header to ensure adequate seal water is provided to the NC Pumps. NV-309 is normally maintained in automatic to provide 32 gpm seal injection. NV-309 is located in the charging flow path, so closing this valve diverts more flow to the NCP seals.

Correct Answer

In addition to MC10, this valve may be controlled from ASP A or ASP B. On transfer of control to the ASP, control of NV-309 from the control room is disabled by operator action to swap a plug connection.

Upon a loss of air or power, NV-309 fails to the open position to ensure sufficient charging so there is no net loss of inventory from the NC system. A manual valve (NV-308) can be used to isolate NV-309, and another manual valve (NV-311) in parallel with NV-309 is available to re-establish throttling.

Question 28

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be ≤ 40 gpm with centrifugal charging pump operating and the charging flow control valve full open.

Distractor Part 2

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow not within limit.	A.1 Adjust manual seal injection throttle valves to give a flow within limit with centrifugal charging pump operating and the charging flow control valve full open.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

Two VCT Level Control signals are used to perform the control functions described above. The following table lists the specific functions of each level control signal.

Control Signal	Associated VCT Level Channel*	Control Function
Selected VCT Level-1	LT-5761	• Auto Makeup start and stop signals • Modulate signal to NV-172 • Swap to FWST on low level** • High Level Alarm • Low Level Alarm
Selected VCT Level - 2	LT-5760	• Provides backup signal to fail open NV-172 on high VCT level • Swap to FWST on low level** • High Level Alarm • Low Level Alarm
*In case of a failure, the associated VCT Level Channel could be swapped to the non-failed channel via DCS controls. **Both Selected VCT Level-1 and -2 are required to swap to the FWST (2/2 logic)		

2.2.2 VCT Level Instrument Failures

The four VCT level indication channels have a common reference and variable leg tap coming off on the VCT (**Figure 3**). Thus, the response of the VCT level instrument failure depends on where the failure occurs. The VCT Level instruments used for control, LT-5760 and LT-5761, have bellows installed on the reference and common legs.

If the difference between LT-5760 and LT-5761 exceeds **3%,** a DCS Trouble Alarm will be generated. When the discrepancy between LT-5760 and LT-5761 reaches **6%,** a DCS Alternate Action will occur. Because only two VCT level channels are used for control, DCS is unable to determine which signal is BAD if either channel fails. Thus, DCS puts both Selected VCT Level -1 and Selected VCT-Level 2 in Alternate Action if either channel fails. (Alternate Actions are discussed in more detail later in the lesson plan.)

Correct Answer Part 1

The impacts of the following VCT level failures were previously analyzed by Engineering:

Failure of VCT Level Common Variable Leg

- Should this common tap become ruptured or isolated, VCT level will be erroneous. If ruptured or leaking, the **level indication for all channels will fail low**. Because the control level channels will agree, an Alternate Action from DCS will not occur.

Loss of ERPA **Correct Answer Part 2**

Auto and Manual Makeup Capability to the VCT is unavailable. If VCT level decreases below 23%, then, by procedure, the NV Pump suction will be manually aligned to the FWST and a downpower will be initiated.

The following valves fail closed if selected to auto:

- NV-181A (B/A Blender Otlt to VCT)
- NV-186A (B/A Blender Otlt to VCT Otlt)
- NV-242A (RMWST to B/A Blender Ctrl)
- NV-238A (B/A to Blendr Ctrl Vlv).

BAT Pumps A and B will not operate if in auto

Reactor Makeup Pumps A and B will not operate

Train A SMM Boron Dilution Interlock will be disabled with switch in "Enable"

Loss of ERPD

Reactor Makeup Pumps A and B will be disabled

Train B SMM Boron Dilution Interlocks will not function as designed.

Loss of CDB

- NV-172A (3-way Divert Vlv to VCT-RHT) will fail to the VCT position.

Loss of KXPA

Automatic VCT makeup capability is lost. (Manual makeup is still available)

Loss of start capability for the following pumps:

- Reactor Makeup Water Pump A
- Reactor Makeup Water Pump B.

Loss of KXPB

NV-849, NV-148, and NV-309 will fail open on a loss of KXPB. In order to control seal water flow, an operator will be dispatched to isolate NV-309 via upstream manual isolation valve NV-308 and manually throttle NV-311 (located in parallel with NV-309) to establish 32 gpm total seal water flow.

Question: 31
(1 point)

Given the following conditions on Unit 2:

- Unit is in Mode 4
- 2A ND train in service in RHR Mode
- 2B ND train remains in Injection Mode

Subsequently:

- Instrument Air is lost to 2ND-26 (ND Hx 2A Outlet Ctrl)

ND system flow _____(1)_____ automatically adjust to compensate for the change caused by this malfunction.

In accordance with OP/2/A/6200/004 (Residual Heat Removal System), 2B ND train _____(2)_____ be realigned to RHR mode.

Which ONE (1) of the following completes the statements above?

A. 1. will
2. can

B. 1. will NOT
2. can

C. 1. will
2. can NOT

D. 1. will NOT
2. can NOT

Original Question

MODIFIED

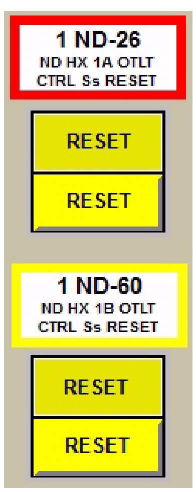
ND Heat Exchanger Outlet Control Valves, ND-26, -60

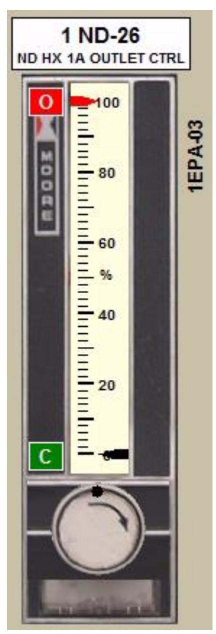
Valves ND-26 and ND-60 are air operated butterfly control valves located downstream of their respective ND Heat Exchanger. These valves may be manually controlled from the Control Room and are used to control the reactor coolant flow through the respective ND Heat Exchanger (ND-26 to A Hx, ND-60 to B Hx). These valves are used to establish the heat removal rate, thereby establishing the NC System cooldown rate. Generally, only one train is in service for cooldown at a time.

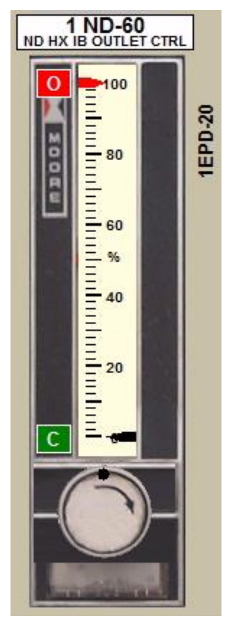
ND-26 and ND-60 automatically open on a Safety Injection signal (S_s) to ensure an ND flow path through the respective ND Heat Exchanger during ECCS operation. Each valve can be controlled manually from the Control Room via manual loaders on control board MC-11. Each valve can also be controlled locally via manual loader.

After receipt of a Safety Injection signal (S_s), the control room operator must depress the ECCS Reset Pushbutton for the appropriate train and push the associated valve Reset Pushbutton to regain control of the valve.

During normal plant operation, ND-26 and ND-60 are in an open position, so that the ND System is aligned for ECCS operation. During normal cooldown, the valves are in a throttled position. Upon loss of instrument air during normal cooldown, the valves again fail open to assure an ND flow path through the ND Heat Exchangers during residual heat removal operation. Should this occur, flow control capability may be re-established by throttling valves NI-173A and NI-178B.







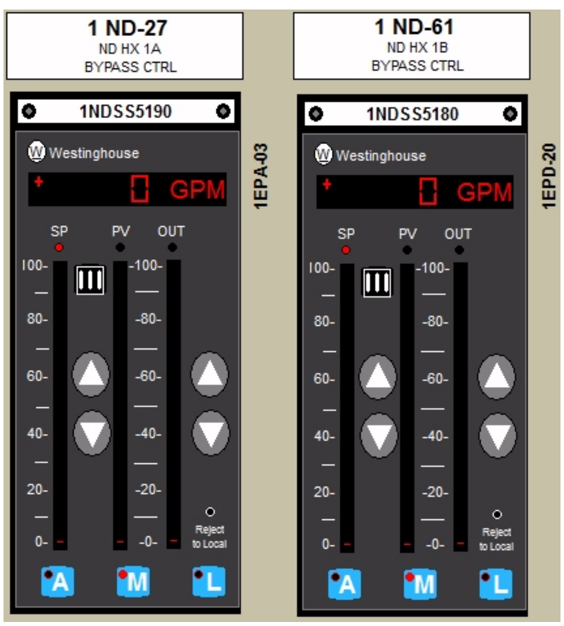
Question 30

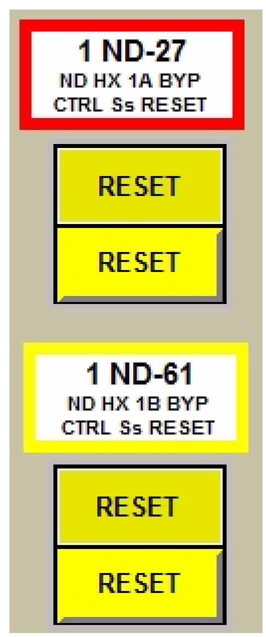
ND Heat Exchanger Bypass Control Valves, ND-27, -61

Valves ND-27 and ND-61 are air operated butterfly control valves located in the ND Heat Exchanger A (B) bypass line. During NC System heatup and cooldown, ND-27 (-61) automatically modulate to maintain the desired total ND Train A (B) flow rate (approximately 3300 gpm), based on the setpoint selected in the Control Room. ND-27 (-61) operates in conjunction with ND Heat Exchanger A (B) flow control valve ND-26 (-60) to establish the NC System cooldown rate by controlling the temperature of the reactor coolant returned to the NC System.

ND-27 (-61) fails closed upon receipt of an S_S signal to prevent bypassing of ND Heat Exchanger A (B) during a Design Basis Event. This instrumentation can be controlled via Manual/Auto (M/A) Stations on control board MC-11 and a soft M/A station on DCS. It can also be controlled following transfer to Local control via a local M/A Station. The Local controller, located outside the ND and NS Heat Exchanger Rooms, is automatically placed in service when the associated ASP is placed in "LOCAL".

If an alternate action occurs on the ND Heat Exchanger outlet flow, the ND-27 (ND-61), M/A Station will transfer to Manual and hold the “last good value.” An alternate action would occur if DCS senses that the ND Heat Exchanger outlet flow signal quality input is BAD.



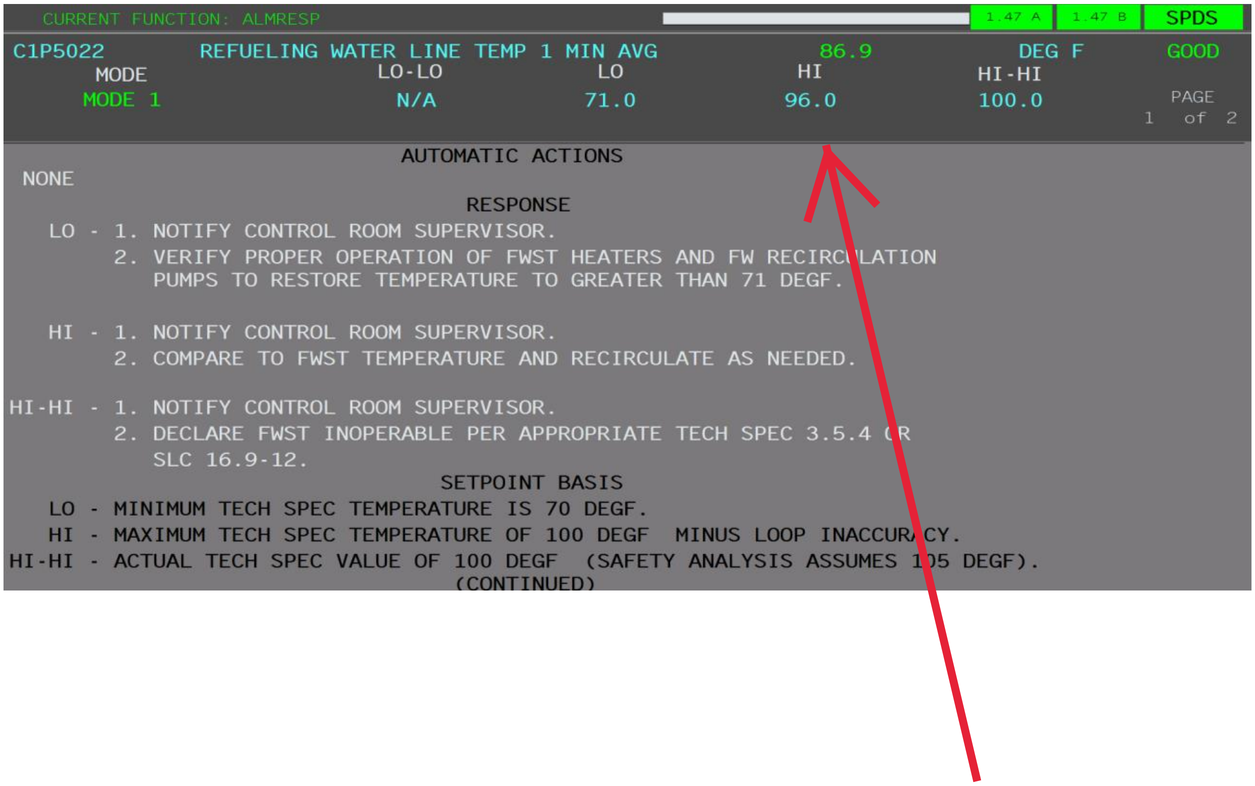


Question 31

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature is ≥ 70°F and ≤ 100°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is ≥ 377,537 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program

Question 31


CURRENT FUNCTION: ALMRESP | 1.47 A | 1.47 B | SPDS

C1P5022 REFUELING WATER LINE TEMP 1 MIN AVG 86.9 DEG F GOOD

MODE	LO-LO	LO	HI	HI-HI	PAGE
MODE 1	N/A	71.0	96.0	100.0	1 of 2

AUTOMATIC ACTIONS

NONE

RESPONSE

LO - 1. NOTIFY CONTROL ROOM SUPERVISOR.
2. VERIFY PROPER OPERATION OF FWST HEATERS AND FW RECIRCULATION PUMPS TO RESTORE TEMPERATURE TO GREATER THAN 71 DEGF.

HI - 1. NOTIFY CONTROL ROOM SUPERVISOR.
2. COMPARE TO FWST TEMPERATURE AND RECIRCULATE AS NEEDED.

HI-HI - 1. NOTIFY CONTROL ROOM SUPERVISOR.
2. DECLARE FWST INOPERABLE PER APPROPRIATE TECH SPEC 3.5.4 OR SLC 16.9-12.

SETPOINT BASIS

LO - MINIMUM TECH SPEC TEMPERATURE IS 70 DEGF.
HI - MAXIMUM TECH SPEC TEMPERATURE OF 100 DEGF MINUS LOOP INACCURACY.
HI-HI - ACTUAL TECH SPEC VALUE OF 100 DEGF (SAFETY ANALYSIS ASSUMES 105 DEGF).
(CONTINUED)

Distractor Part 1

Question 31

16.9 AUXILIARY SYSTEMS

16.9-12 Boration Systems Borated Water Sources - Operating

COMMITMENT The following borated water source(s) shall be FUNCTIONAL as required by SLC 16.9-8:

a. A Boric Acid Tank (BAT) with:

1) A minimum contained borated water volume as specified in the CORE OPERATING LIMITS REPORT (COLR),

2) A minimum boron concentration as specified in the COLR, and

Distractor Part 2

3) A minimum solution temperature of 65°F.

b. The Refueling Water Storage Tank (RWST) with:

1) A minimum contained borated water volume as specified in the COLR or Technical Specification Surveillance Requirement 3.5.4.2, whichever is larger,

2) A minimum boron concentration as specified in the COLR,

3) A minimum solution temperature of 70°F, and

4) A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all Reactor Coolant System (RCS) cold leg temperatures > 210°F.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required BAT non-functional.	A.1 Restore the required BAT to FUNCTIONAL status.	72 hours

(continued)

Question 32

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be:

a. $\geq$ 75°F and $\leq$ 100°F for the containment upper compartment, and

Correct Answer b. $\geq$ 100°F and $\leq$ 120°F for the containment lower compartment.

--NOTE--
The minimum containment average air temperature in MODES 2, 3, and 4 may be reduced to 60°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limits.	A.1 Restore containment average air temperature to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Question 32

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

Distractor

LCO 3.6.4 Containment pressure shall be $\geq$ -0.1 psig and $\leq$ +0.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	In accordance with the Surveillance Frequency Control Program

Question 32

PANEL: 1AD-19

C/9

VQ CONTAINMENT PRESSURE ALERT

SETPOINT:
1. ≥ 0.25 psig   Distractor
2. ≤ -.06 psig

ORIGIN:
1. 1VQEM5040 (high pressure signal)
2. 1VQEM5041 (low pressure signal)

PROBABLE CAUSE:

During Modes 1-4:

High Pressure:
1. Plant heatup
2. Steam/air leaks

Low Pressure:
1. Plant cooldown
2. Excess containment cooling capacity in service.

During Modes 5 and 6:

High Pressure:
1. Improper balance of VP supply and exhaust flows.
2. VP has tripped or VP Exhaust Fans have tripped.

Low Pressure:
1. Improper balance of VP supply and exhaust flows.
2. VP Supply Fans have tripped.
3. VF Supply Fans have tripped.

AUTOMATIC ACTIONS: None

IMMEDIATE ACTIONS:

IF in Modes 1-4, perform the following:
1. Verify containment pressure high or low on 1VQP5040 on 1MC5.

IF in Mode 5 **OR** Mode 6 when movement of non-recently irradiated fuel or CORE ALTERATIONS **NOT** in progress, perform the following:

1. Verify containment pressure high or low on 1VQP5040 on 1MC5.
2. **IF** VP is in service, rebalance VP supply and exhaust flows to clear the annunciator per OP/1/A/6450/015 (Containment Purge System).
3. **IF** VP is **NOT** in service, initiate a VQ release or addition per OP/1/A/6450/017 (Containment Air Release and Addition System) as necessary.

CONTINUED ON THE NEXT PAGE

5.6 PRT Response to a Phase A Isolation Signal

Objective 6B

The following PRT- related valves will close on a Phase A (S_T) Isolation Signal:

- 1(2)NC-56B (RMW Pump Disch Cont Isol)
- 1(2)NC-53B (N2 To PRT Cont Isol)
- 1(2)NC-54A (N2 To PRT Cont Isol)

KC water to the NCDT heat exchanger will also be isolated on a Phase A. With both Reactor Makeup Water and KC to the NCDT Heat Exchanger, there is no available method to cool the PRT with the Phase A signal present.

6. LIMITS AND PRECAUTIONS

Objective 7

Reference the limit and precautions for the following procedure:

- OP/1(2)/A/6150/004, Pressurizer Relief Tank

7. OPERATING EXPERIENCE

7.1 Committed OE

None.

7.2 Non-Committed OE

7.2.1 Unexpected PRT Level Response During NCS Venting (AR 01497768)

While performing Unit 2 Reactor Coolant System Venting per OP/2/A/6150/001 Enclosure 4.2, PRT level indication did not increase as expected after 2NC-251B and 2NC-253A were opened. All 4 of the Reactor Head Vent and Reactor Head Vent Block valves were previously opened per the procedure to vent the reactor vessel head. When 2NC-251B and 2NC-253A were opened, PRT Level was monitored by OAC point C2A0879 and level increased as expected. When PRT level increased by at least 1%, only 2NC-251B and 2NC-253A were closed to suspend venting. After ~ 2-3 minutes, 2NC-251B and 2NC-253A were reopened to resume venting the reactor head, PRT level did not increase as expected. Both OAC and control board indications showed the valves to be open. The CRS, OSM, and Primary Group were contacted and a decision was made to cycle Kerotest valves 2NC-251B and 2NC-253A to ensure the disc was separated from the body. Once the affected valves were cycled, PRT level began to increase as expected.

The component cooling surge tanks have sufficient capacity to accommodate thermal transients and leakage into or out of the KC System. The vent lines on the surge tanks are maintained open during all operational modes and are sized large enough to prevent excessive vacuum in the tanks should cold water from the KC drain sump be added to the surge tank at its maximum temperature. In addition, this vent line is also sufficient to prevent over pressurization of the tank in the event of a Thermal Barrier Tube rupture, evaluated to be approximately 260 gpm. Surge tank overflow is directed to the component cooling drain sumps via loop seal.

Each units two surge tanks are connected by a separate 8 inch overflow line at approximately the 97% level that allows one surge tank to overflow to the tank on the opposite train. This 8 inch line will equalize variations in surge tank level.

Distractor Part 2

An assured makeup source to KC is provided by the RN system, which can be aligned to the KC pumps suction header via manual valves. Normal makeup is provided by YM demineralized water and is not assured.

Correct Answer Part 2

The Control Room Operators will be alerted of leakage from the system via a KC Surge Tank Lo Level annunciator at 37.3%.

Each surge tank has safety related level instrumentation that automatically isolates the KC non-essential Auxiliary and Reactor Building header isolation valves (train related) upon Lo-Lo level at 34%; achieving train separation. This assures that at least one train of KC will have sufficient NPSH if an out leakage develops when both essential and non-essential trains are aligned together. The Lo-Lo level at 34% uses a 1/1 instrument per tank to cause the train isolations. A separate instrument is used for indication and alarms.

KC Pumps

Objective 4a and 5, All

Refer to figure 4 for KC Pump Layout

Four Component Cooling pumps are provided per unit with each train pair powered from a separate assured power source. Each train pair has 100% flow capacity (50% per pump) and can provide the minimum requirements of both the Essential and Non-essential Header loads.

The KC Pumps are horizontal shaft, centrifugal pumps equipped with mechanical seals to minimize leakage. They can be operated from the Control Room or ASP by two position START/STOP pushbuttons. Normal KC pump discharge pressure is 100 psig. The KC pumps are located in the Auxiliary Building on the 560" level (Unit 1 pumps) and the 577' level (Unit 2 pumps)

Flow through the KC system will be dependent on the components in service. The KC pumps minimum flow rate is 1100 gpm and the maximum is 5700 gpm per pump (manufacturer's recommended minimum flow and pump run-out conditions).

will also open when Unit 1(2) ASP A and B are transferred to “LOCAL” and the train related pump is running.

Reactor Building Non-Essential Header cools the following:

- NC Pumps
- NCDT
- Excess Letdown Hx's

Component Cooling cools the NC pump Thermal Barrier Hx, and the upper and lower bearing oil coolers of the NC Pumps. The NC Pump thermal barrier and oil cooler flows for both units can be read on gauges on the 543 elevation of Aux building. **Correct Answer Part 1**

The Thermal Barrier Hx has an inlet check valve and the outlet valve that can auto close in the event there are indications of a thermal barrier rupture. All the piping between the check valve inlet and outlet valve and the valves themselves are rated for NCS pressure and temp with a relief set at 2485 psig. Output from the relief valve is directed to the Containment Floor and Equipment Sump. The outlet valve will auto close @ 60 gpm after 30 seconds. (The 30 sec. time delay prevents the valve from closing on surge, during a Pump Start).

This arrangement of an inlet check valve and auto closed outlet valve should isolate any Thermal Barrier HX leak. Flow to the Thermal Barrier is manually throttled to 40 gpm/pump, with a high flow alarm at 60 gpm and a low flow alarm at 35 gpm. Westinghouse performed specific analysis in 2009 to determine the bounding leak rate for the thermal barrier heat exchangers used at Catawba and McGuire. The analysis determined that the bounding leak rate is 10.55 gpm (AR 01540034 Corrective Action #1).

The NC pump upper and lower bearing oil coolers are also cooled by Component cooling. The upper Bearing flow is controlled at 165 gpm/pump by a flow controller located in Aux. building on 543' elevation. There is a high flow alarm at 200 gpm and a low flow alarm at 140 gpm on the pump upper bearing Component Cooling flow.

The NC pump lower bearing oil cooler KC flow is manually adjusted to 6 gpm/pump. There is a low flow alarm at 5 gpm.

The KC Supply Header Flow to NCP's has a Low alarm @ 425 GPM. The Annunciator alarm on AD20/21 requires verification of flowpath and monitoring the motor bearing temperatures. Refer to AP/021 (Loss of KC) if alarm is due to a loss of flow.

On a Unit 1(2) Sp signal the NC Pump Containment Supply and Return isolation valves (1(2)KC-338B, 1(2)KC-424B, and 1(2)KC-425A) Isolations will close.

Backup Heaters

Objective 2F

The backup heaters are made up of three groups labeled A, B, and D. The A and B Groups each contain 6 banks of heaters worth 416 kW. The D Group contains 7 banks of heaters worth 484 KW. One of the D Group heater banks contains 3 heaters and is powered from SMXG. This Shared Motor Control Center SMXG is normally powered from SLXG, but can also be powered from the SSF D/G. Correct Answer

These Group D heaters #28, 55, and 56, can be controlled at the SSF. They are 70kW and must be energized within 15 hours of a reactor trip. The heaters are required for SSF functionality which is to maintain NC System pressure control and to ensure that any steam bubble in the pressurizer does not migrate to the reactor vessel.

Manual Heater Control

The backup heaters are either on or off - they cannot have their output varied like the C heaters. They can be manually energized but, after a Blackout condition, that is delayed 12 minutes. Groups A, B, D can be controlled from MCB and Groups A & B can be controlled from the Auxiliary Shutdown Panels (ASP). The one bank in Group D that is powered from the SSF can be controlled there too.

Distractor Part 1

Auto Heater Control

Objective 3A

A low pressure deviation of 25 psi below setpoint (normally 2210 psig) will automatically turn on the heaters.

Upon low-low pressurizer level of 17% the heaters will turn off. This condition blocks control of the backup heaters from the MCB but Groups A & B can still be controlled from the ASP. When the low-low PZR level condition clears the heaters will auto re-energize if a low pressure demand signal is still present.

Pressurizer Operability

Operability of the PZR is predicated on two conditions.

1. The water level must be less than or equal to 92%. This ensures a bubble exists which preserves the steam space needed for adequate pressure control.

Objective 2F, 5

2. Two Groups of Backup heaters, A and B; each with a capacity greater than or equal to 150 KW and capable of being powered from an Emergency Power Supply. Only these two Groups can be powered from an Emergency Power Supply, as they are supplied by 600 VAC Blackout load centers LXI and LXH that are powered from the Blackout 4KV buses FTA and FTB. The 4KV Blackout buses can be powered from ETA and ETB during a Station Blackout.

The failed channel pressure signal will be removed by the median select (MSS) circuitry in the DCS

The IPE DCS circuit will actuate a DCS Trouble Alarm on 1AD-2, F/10

- For a single channel failing low the following annunciators will be received:

  1AD-6, D/8, PZR LO PRESS ALERT

  1AD-6, E/8, PZR LO PRESS SI ALERT

- For a single channel failing high, the annunciator 1AD-6, A/8, PZR HI PRESS ALERT will be received.

- Reactor Trip

  When a reactor trip occurs Tavg is reduced very quickly as the rods fall into the core.

  This decrease in T_{AVG} causes PZR level to decrease due to NCS density increase. As a result of the PZR level decrease, the steam-space to water-space ratio has greatly increased in the PZR which causes PZR pressure to decrease.

  The PZR pressure control system should respond as required to bring PZR pressure back to normal operating pressure over a period of time by energizing PZR heaters.

- Reactor Coolant System Heatup and Pressurization

  During plant startup from Mode 5 to Mode 3 the pressurizer pressure control system is used to raise the NC system pressure to normal operating pressure. First, the PZR heaters are energized. After placing the heaters in service a constant spray flow is established via normal spray from the NC system or alternate spray from either the ND system or the NV system. By establishing spray flow a constant outflow of water is maintained through the PZR surge line.

  The purpose of the constant outflow is to prevent an insurge of water affecting the surge line temperature thus affecting the indicated PZR heatup rate. When NC system pressure reaches 1700 psig the PZR Pressure Master Controller can now be utilized to control system pressure. The PZR Pressure Master may be operated manually with the raise and lower pushbuttons or automatically by adjusting the setpoint. This control of the heaters and sprays allows control of the rate of change of pressurization of the NCS. When normal system pressure of 2235 psig is reached the PZR Pressure Master Controller setpoint is verified correct and the PZR Pressure Master controller is placed in automatic.

6.4 Alternate Actions

Objective 9

Selected Pressurizer Pressure-1 (SPP-1)

- Pressurizer Pressure Master Controller goes to MANUAL with Last Good Value ('C' heater operation and spray valve position do not change)

Question 37

Enclosure 13.4

Control Bank Withdrawal

NOTE:	This enclosure is performed by the SRO and RO dedicated to reactor startup.

Initials	Printed Name

NOTE:	Step 13.4.1 may be performed out of sequence.

13.4.1 **WHEN** "P-6 S/R BLOCK PERMISSIVE" lamp on 1(2)SI-18 is lit, perform the following:

_____ 13.4.1.1 Block Source Range high flux level trip as follows:

- ☐ "TRN A S/R SELECT" switch to "BLOCK". Correct Answer Part 1
- ☐ "TRN B S/R SELECT" switch to "BLOCK".

_____ 13.4.1.2 Verify following permissive lamps on 1(2)SI-18 are lit:

- ☐ "S/R TRAIN A TRIP BLOCKED"
- ☐ "S/R TRAIN B TRIP BLOCKED"

_____ 13.4.2 Review estimated critical position with Reactor Engineer.

_____ 13.4.3 Verify lowest NC Loop T-Avg is ≥ 551 °F. {S.R. 3.4.2.1}

_____ 13.4.4 Withdraw Control Banks until: (R.M.)

☐ Source Range count rate doubles ____________ N31 ____________ N32

OR

☐ BDMS count rate doubles ____________ BDMS A ____________ BDMS B

OR

☐ Control Bank B reaches fully withdrawn position

OR

☐ Stable startup rate reaches 0.5 DPM.

_____ 13.4.5 Review projected critical position with Reactor Engineer.

Reactor Trip Interlocks

Several reactor trips are only active above or below specific power levels. The following table lists the interlocks that are associated with the Reactor Trips.

INTERLOCK	SETPOINT	FUNCTIONS
P-6	1/2 IR $\geq$ ~10^{-5}% power	• Allows manual block of the Source Range Neutron Flux Rx Trip above P-6 • Automatically enables Source Range Neutron Flux trip below P-6
P-7	P-10 or P-13	Automatically enables the 5 "At Power" Trips above P-7: • NCP Undervoltage • NCP Underfrequency • NCP 2 Loop Loss of Flow • Pressurizer Pressure - Low • Pressurizer Level - High Automatically blocks these trips below P-7.
P-8	2/4 Power Range detectors $\geq$ 48% power	• Automatically enables the Single Loop Reactor Coolant Flow-Low reactor trip above P-8. • Automatically blocks this trip below P-8.
P-9	2/4 Power Range detectors $\geq$ 69% power	• Automatically enables the reactor trip on turbine trip above P-9. • Automatically blocks this trip below P-9.
P-10	2/4 Power Range detectors $\geq$ 10% power	• Input to P-7 (see above) • Automatically provides a backup signal to block the SR Neutron Flux trip above P-10 • Allows manual block of the following above P-10: ○ C-1 rod stop ○ Intermediate Range Neutron Flux Rx Trip ○ Power Range Neutron Flux Low Rx Trip • Automatically reinstates the C-1 rod stop, Intermediate Range Rx Trip, and Power Range Low Rx Trip below P-10.
P-13	1/2 Turbine Impulse Pressure $\geq$10% power	• Input to P-7

9. NORMAL SYSTEM OPERATION

9.1 Normal Operation

Objective 11, ALL

The normal operation mode of System VA shall have the supply air system in operation with the Containment Chilled Water (YV) System and Plant Heating Water (YH) available as required to temper supply air to achieve the desired area temperature. Both units will normally have both Train A and B Filtered Exhaust, Unfiltered Exhaust and Supply in operation.

- All 4 Filtered Exhaust Fans running in bypass mode.
- All 4 Unfiltered Exhaust Fans running
- All 4 Supply Fans running
- Counting Room Supply and Filter Unit running controlled independently
- 4 Radwaste Area AHUs running
- Filter and Demineralizer Room Exhaust Fans running - receive permissive from Auxiliary Bldg Filtered Exhaust Fans.
- Auxiliary Shutdown Panel Air Conditioning Unit's compressor starts if room temp is greater than 68°F.
- Evaporator Room Supply Units placed in "COOL".
- Restricted Instrument Shop Supply Unit runs as necessary.
- PD Pump Room Supply Unit placed in "STOP".
- Doghouse Vent Fans controlled by thermostat in "AUTO".
- UHI Building Vent Fan controlled by room thermostat in "START".
- Fume Hood Exhaust Fans in the Sample Room (3) - permissive from unfiltered exhaust fans (Unit 1 or 2).

10. ABNORMAL OPERATIONS

10.1 Safety Injection Operation

Objective 12A, ALL

Upon receipt of a Safety Injection signal to initiate LOCA operation the following train-related actions take place in the VA System (i.e. a Unit 1 Train A Safety Injection will perform the below actions for Unit 1 A Train and Unit 2 A Train Components.):

- Filtered Exhaust Fans continue operating or start if they are shut down
- Filter Units operate in the filtered mode of operation

- Inlet vanes (vortex dampers) to the Filtered Exhaust Fans will go to a throttled position reducing the filtered exhaust design flow from approximately 30,000 to approximately 6,540 cfm per train.
- Non- safety portion of the Filtered Exhaust are isolated by electro hydraulic isolation dampers that fail to a closed position
- Unfiltered Exhaust Fans trips which cause the associated Supply Fans to trip

The following actions also occur on a Safety Injection signal, but these signals are unit-related:

- Unit and train related ASP supply units are tripped off and then restarted.
- Trips Unit Related Filter Room Exhaust Fans.

The VA System will continue to operate in this mode until the safety injection signal is reset.

The VA System will not go to the LOCA mode of operation if tornado isolation is activated.

The following two options can be used to establish Auxiliary Building ventilation following an S_S signal: Return to Normal Alignment or Post LOCA Operation.

Return to Normal Alignment

The return to normal alignment is directed by EP/ES-1.1 (Safety Injection Termination) and is accomplished by using an enclosure from OP/0/A/6450/003 (Auxiliary Building Ventilation). This process involves a damper reset key switch and two resets for the ASP supply units per train. After this is done, VA may be returned to normal alignment.

Post LOCA Operation

Objective 12B, All

Post LOCA operation is not directed by any EP or AP. This mode should only be used with TSC concurrence (may be used if dose levels in the Auxiliary Building preclude the use of return to normal alignment). Procedure guidance is provided using an enclosure from OP/0/A/6450/003 (Auxiliary Building Ventilation). This process involves moving a connector from one receptacle to another within LOCA Control Panels A and B and placing a selector switch to “ON”.

Activation of the Post LOCA mode of operation will cause the following responses in the VA System:

- Allow the Auxiliary Building unfiltered and supply fans to operate
- Isolation dampers for the non-safety filtered exhaust ducting to the Auxiliary Building will open allowing air to be exhausted from these areas.

All Post LOCA Controls will be disabled upon receipt of a LOCA signal. Additionally, the Post LOCA Controls do not bypass any VA System interlocks (i.e. fan interlocks, smoke detectors, EMF interlocks)

4. SYSTEM INTERRELATIONSHIPS

4.1 Containment Chilled Water System (YV)

Objective 7A, All

YV provides normal cooling to VV components per OP/1/A/6450/020, Containment Chilled Water System. The YV System containment chillers are then in turn cooled by the Low Pressure Service Water System (RL). The YV system is covered in the RN Lesson Plan.

4.1.1 YV design flow to each VV system AHU cooler follows:

- LCVUs – 825 gpm
- UCVUs – 18 gpm
- IIRVU – 10 gpm

4.2 Nuclear Service Water (RN)

Objective 7B, All

RN may be automatically or manually aligned to VV when YV is unavailable. YV will not be available to provide cooling to the UCVUs during a loss of offsite power.

5. NORMAL SYSTEM OPERATION

5.1 Normal VV Cooling Alignment Modes 1 – 4

Objective 8A, All

The normal VV component configuration alignment in Modes 1-4 per OP/1(2)/A/6450/001(Containment Ventilation Systems) is as follows:

- Three (3) LCVUs in “LOW” speed and in MAX COOL
- One (1) PTBF in LOW speed
- Three (3) CRDM vent fans ON
- One (1) IIRVU in NORM
- One (1) or two (2) UCVUs in NORM and associated RAFs in AUTO

15.3 Safety Injection Signal

Objective 16C, All

Upon receipt of a safety injection signal on either unit , all four RN pumps will start and their associated discharge isolation valves and pump motor cooler isolation valves will open. The Unit related KC HXs Control Valves fail open to assure sufficient heat transfer will be available through the KC HX when it is needed during a design basis event. The RN Chemical Addition Dilution Water Crossover Isolation Valves close for train separation. The Auxiliary Building Vent Unit Supply and Return Header Isolation valves close to conserve flow for containment cooling or essential heat loads..

The D/Gs start on a Safety Injection, and the D/G Engine Jacket Water Cooler RN Supply Isolation Valves are interlocked to open whenever a D/G starts and close if the D/G stops (or fails to start)

15.4 Phase B Signal

Objective 16D, All

The following automatic actions occur upon receipt of a Phase B signal. Item 1 occurs upon a Phase B signal from either unit. Items 2-6 occur on the unit which generated the Phase B signal.

1. SNSWP Return Isolation Valves 1RN-58B and 1RN-63A open. These valves open on an SP signal to provide an assured RN discharge flow path if the normal RN discharge flow path to RL is unavailable due to a failed isolation valve.
2. RN Supply Crossover Isolation Valves 1(2)RN-47A and 1(2)RN-48B close. The automatic closure for the unit specific valve isolates RN flow to nonessential components to conserve flow for essential components on that unit. However, if the RN system is aligned in Single Supply Header Operation, 1(2)RN-47A and 1(2)RN-48B remain open and the RN trains remain cross-connected. Therefore, RN trains A and B do not isolate and remain cross-connected. This ensures that RN cooling water flow is available to all four essential headers while the RN system is aligned in Single Supply Header Operation.
3. RN Nonessential Supply Header Isolation Valves 1(2)RN-49A and 1(2)RN-50B close. The automatic closure on the SP signal separates that unit's RN nonessential header from the RN essential headers, and conserves flow for safety related components on the RN essential headers.
4. RN Nonessential Return Header Isolation Valves 1(2)RN-51A and 1(2)RN-52B close to separate that unit's nonessential header from the essential headers, and to conserve flow for safety related components on the essential headers.
5. Upper Containment Isolation Valve 1(2)RN-404B close. This valve is normally open to provide cooling water to containment HVAC units during normal operation. If containment HVAC is lost during normal operation, the resulting increase in containment pressure could generate an "artificial" ESF actuation.

Consequently, valve 1(2)RN-404B is required to remain open until a phase B isolation signal is initiated.

6. Lower Containment Isolation Valves 1(2)RN-437B, 1(2)RN-484A, 1(2)RN-487B close. These valves are normally open to provide cooling water to containment HVAC units during normal operation. If containment HVAC is lost during normal operation, the resulting increase in containment pressure could generate an "artificial" ESF actuation. Additionally, these valves are open to allow cooling water flow to the NC Pump Motor Air Coolers. Without the cooling water flow, motor stator temperatures would increase, possibly resulting in the loss of the NC pumps. Consequently, these valves are required to remain open until a phase B isolation signal is initiated.

15.5 Transfer to ASP

Objective 16E, All

RN is required to operate upon transfer of control to the Auxiliary Shutdown Panels. In order to provide the most assured means of shutdown, automatic actions that occur upon the transfer of controls for RN includes:

- RN Suction(s) and discharge(s) align to SNSWP (Unit 1 ASP's ONLY)
- A/B train discharges split (Unit 1 ASP's ONLY)
- Diesel discharge(s) swap to SNSWP (unit/train related, panel specific valve)
- Essential header supply valves(s) open (unit/train related, panel specific valve)
- Full RN flow is aligned through KC Hx's (unit/train related, panel specific valve)

15.6 SNSWP Ice Melt

Objective 16F, All

Ice formation in the source and intake section is felt to be impossible because the intake structures are well below the surface of Lake Wylie and the SNSWP. In cold weather the RN system suction and discharge can be aligned to the SNSWP to prevent severe ice accumulation on the surface of the pond. Any WL discharge in progress from the Waste Monitoring Tank Building (1WL-124) must be secured prior to making this alignment.

This evolution is accomplished by performing the following:

- Swapping RN returns to the SNSWP
  - Open the RN Header A and B return valves to the SNSWP (1RN-63A &58B)
  - Open the DG 1A, 1B, 2A, 2B HX returns to the SNSWP valves (1(2)RN-846A & 848B
  - Close the DG 1A.1B, 2A, 2B HX returns to Lake Wylie valves. (1(2)RN-847A & 849B)

Glycol Containment Isolation Valves

NF supply containment isolation **NF-228A** is located outside containment. NF-228A is an air operated valve and is controlled from the control room.

NF-228A automatically closes on the following:

- Loss of Instrument Air
- Loss of Power
- Low-Low Expansion Tank Level (A key switch is provided on the local NF control panel to allow the Low-Low Expansion Tank Level interlock to be bypassed.)
- Phase A signal, S_T

NF Return Containment Isolation Valve **NF-233B** is located inside containment. NF-233B is a motor operated valve that automatically closes on a Phase A Containment Isolation (St) signal. A bypass allows glycol that is trapped in the penetration to bleed back into containment in the event of containment isolation.

The outside NF Return Containment Isolation Valve, **NF-234A**, is an air operated valve that automatically closes on the following (same as NF-228A):

- Loss of Instrument Air
- Loss of Power
- Low-Low Expansion Tank Level (A key switch is provided on the local NF control panel to allow the Low-Low Expansion Tank Level interlock to be bypassed.)
- Phase A signal, S_T

Both NF-234A and NF-233B are controlled from the control room as described above.

Some evolutions may require securing glycol flow to containment for a short period. If the NF containment isolation valves are closed and/or the penetration is isolated, action must be taken to:

- Minimize ice bed heatup and/or
- Provide over pressure protection for the isolated penetration

5. AUTOMATIC ACTIONS SUMMARY

Objective 8, Licensed

Containment Isolation Automatic Closures

The following table summarizes the three containment isolation valve actions on the plant conditions shown.

Valves CLOSE on:	Loss of VI	Loss of Power	Low-Low Expansion Tank Level	St Signal
NF-228A Containment Supply Isol	X	X	X	X
NF-233B Containment Return Isol.				X
NF-234A Containment Return Isol.	X	X	X	X

NF Ventilation Chiller Trips

- Low Glycol Flow ≤ 2.5 psid
- Low KR Flow
- Low Compressor Oil level ≤15 psid
- Low Suction Press
- High Compressor Discharge Pressure

Catawba Nuclear Station

2019 CNS RO NRC Examination

Question: 41
(1 point)

Given the following conditions on Unit 1:

- Unit is at 100% RTP
- It has been determined that eight Ice Condenser Intermediate Deck doors will not open due to excessive ice buildup

Based on the conditions listed above, peak pressure following a Design Basis Accident will be reached _____(1)_____ than normal.

Containment design pressure is _____(2)_____.

Which ONE (1) of the following completes the statements above?

A. 1. sooner
2. 20 psig

B. 1. later
2. 20 psig

C. 1. sooner
2. 15 psig

D. 1. later
2. 15 psig

Original Question

MODIFIED

Ice fabrication and conveyor sub-systems

The initial ice load and make-up to the ice condenser baskets is provided by the ice fabrication and ice conveyor sub-systems. Periodic makeup is performed during plant outages due to loss of ice as a result of sublimation. This makeup is performed using ice conveyor lines which are temporally installed for this evolution.

Refrigeration Sub-system

During normal operation, the ice bed temperature is maintained at approximately 15ºF by the refrigeration system which passes cold air through the ice condenser and glycol through the ice condenser floor cooling coils.

Design Basis Requirements

Objective 1, Licensed Only

Limiting initial peak containment pressure to less than design pressure

During normal operation the ice condenser must be operable. This ensures sufficient heat removal capability to condense the NC system volume released during a LOCA. The peak containment pressure transient is <15 psig during a design basis event (DBE) LOCA. The ice bed is sized for an energy absorption capability of more than twice the energy initially released in any incident.

Continuing with lowering containment pressure after leak initiation

A continuation of cooling is provided through the NF ice condenser by the two 40,000 cfm containment air return system fans which circulate upper containment air into the lower compartment.

Controlling Iodine by lowering pH in the containment sump

The function of the post LOCA iodine removal is accomplished by chemically controlling the alkaline ice to a pH range of 9.0 to 9.5. During the accident, the melting ice provides a medium for removal of iodine from the containment atmosphere. This solution is then collected in the containment sump.

Maintaining boron concentration in the containment sump

The condenser ice baskets are filled with ice containing approximately 2000 ppm boron. This assures that the baskets contain sufficient boron (Sodium Tetraborate) to preclude dilution of the containment sump. The boron concentration of the ice is maintained between 1800 and 2330 ppm.

The solution drains into the containment sump, where it is available for residual core heat removal.

Question 41

3.6 CONTAINMENT SYSTEMS

3.6.13 Ice Condenser Doors

LCO 3.6.13 The ice condenser lower inlet doors, intermediate deck doors, and top deck doors shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

---NOTES---

1. Separate Condition entry is allowed for each ice condenser door.

2. Entry into Condition B is not required due to personnel standing on or opening an intermediate deck or top deck door for short durations to perform required surveillances, minor maintenance such as ice removal, or routine tasks such as system walkdowns.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ice condenser lower inlet doors inoperable due to being physically restrained from opening.	A.1 Restore lower inlet door to OPERABLE status.	1 hour
B. One or more ice condenser doors inoperable for reasons other than Condition A or not closed.	B.1 Verify maximum ice bed temperature is $\leq$ 27°F.	Once per 4 hours
	<u>AND</u>	
	B.2 Restore ice condenser door to OPERABLE status and closed position.	14 days

(continued)

Question 42

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. **Verify at least one of the following annunciators - LIT:**	**Perform the following:**
__ • 1AD-20, B/3 "CONT. SUMP LEVEL >3.3 ft" OR __ • 1AD-21, B/3 "CONT. SUMP LEVEL >3.3 ft".	__ a. **WHEN** at least one "CONT. SUMP LEVEL >3.3 ft" annunciator - LIT, **THEN GO TO** Step 5. __ b. Do not continue in this enclosure until at least one annunciator - LIT.
5. **Align NS train 1A to containment sump as follows:**	
__ a. Verify NS pump 1A - AVAILABLE TO RUN.	__ a. **GO TO** Step 6.
__ b. Verify 1NI-185A (ND Pump 1A Cont Sump Suct) - OPEN.	__ b. **GO TO** Step 6.
__ c. Verify NS pump 1B - OFF.	c. **IF** NS pump 1B running **AND** RN established to NS Hx 1B, **THEN** perform the following: __ 1) Ensure 1NS-20A (NS Pump 1A Suct From FWST) - CLOSED. __ 2) Ensure 1NS-18A (NS Pmp A Suct From Cont Sump) - OPEN. __ 3) **GO TO** Step 7.
Enclosure 2 will provide direction to start only one NS pump. If the other pump is in service, guidance will be provided to bypass pump start steps.	
__ d. OPEN 1NS-29A (NS Spray Hdr 1A Cont Isol).	__ d. **GO TO** Step 6.
__ e. OPEN 1NS-32A (NS Spray Hdr 1A Cont Isol).	__ e. **GO TO** Step 6.

Question 42

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
k. Isolate FWST from NV and NI pumps as follows:	
__ 1) Place "PWR DISCON FOR 1NI-100B" switch in "ENABLE".	
__ 2) CLOSE 1NI-100B (NI Pmps Suct From FWST).	
3) CLOSE the following valves:	
__ • 1NV-252A (NV Pumps Suct From FWST)	
__ • 1NV-253B (NV Pumps Suct From FWST).	
	<u>**NOTE**</u> An invalid SPDS orange path may briefly exist between opening NS suction valve from sump and starting NS pump. FR-Z.1 should not be entered unless NS pump fails to start.
__ 7. **Verify Enclosure 2 (Aligning NS for Recirculation) - PREVIOUSLY COMPLETED.**	__ **Align NS for recirc. <u>REFER TO</u> Enclosure 2 (Aligning NS for Recirculation).**

Question 43

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
__ 3) CLOSE 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol).	3) Perform the following: __ a) CLOSE 1CA-36 (CA Pump #1 Flow To S/G 1D). __ b) Dispatch operator to close 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol) (DH-584, DD-EE, 43-44, Rm 591).
__ 10. **<u>WHEN</u> NC T-Hots start to increase, <u>THEN</u> dump steam from intact S/G PORVs to stabilize NC T-Hots.**	
11. **Verify conditions allowing alignment of one ND train for Aux Containment Spray as follows:**	
__ a. At least one ND train - ALIGNED AND OPERATING IN COLD LEG RECIRC MODE.	a. Perform the following: __ 1) **<u>WHEN</u>** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation) completed, **<u>THEN</u>** perform Steps 11 and 12. __ 2) **<u>GO TO</u>** Step 13.
__ b. Containment pressure - GREATER THAN 15 PSIG.	__ b. **<u>GO TO</u>** Step 13.

Question 43

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
__ c. Elapsed time since Reactor Trip - GREATER THAN 50 MIN.	c. Perform the following: __ 1) Designate someone to notify Control Room Supervisor when 50 min from Reactor Trip has elapsed. __ 2) **WHEN** time since Reactor Trip greater than 50 min, **THEN** perform Steps 11 and 12. __ 3) **GO TO** Step 13.
d. Verify the following valves - CLOSED: __ • 1NS-43A (ND Pmp 1A To Cont Spray Hdr) __ • 1NS-38B (ND Pmp 1B To Cont Spray Hdr).	__ d. **IF** ND Aux Containment Spray aligned to operating ND train with RN and KC cooling, **THEN GO TO** Step 12.

Question 43

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

24. (Continued)

d. **<u>IF</u>** NI Pump 1B in service, **<u>THEN</u>** ensure the following valves - OPEN:

__ • 1NI-135B (NI Pump 1B Suct)

__ • 1NI-150B (NI Pump 1B C-Leg Inj Isol)

__ • 1NI-162A (NI To C-Legs Inj Hdr Isol)

__ • 1NI-100B (NI Pmps Suct From FWST).

e. **<u>IF</u>** both of the following conditions exist, **<u>THEN</u>** **<u>GO</u>** **<u>TO</u>** Step 25:

__ • Any NI pump in service **<u>AND</u>** S/I flowpath established

<u>AND</u>

Distractor D

__ • Time between reactor trip and implementation of this procedure - GREATER THAN 90 MINUTES.

(RNO continued on next page)

Question: 45
(1 point)

Given the following Unit 1 initial conditions:

- The Unit is at 25% RTP following a refueling outage.
- AP/1/A/5500/028 (Secondary Steam Leak) has been entered following a discovery of a leak on the Unit 1 Main Turbine Crossover line

Subsequently:

- The Unit 1 Main Turbine is tripped to isolate the leak
- Reactor power is currently 11%

At this time, ______(1)______ steam dumps are operating to control NC temperature at ______(2)______ degrees F.

Which ONE of the following completes the statement above?

A. 1. ONLY condenser
 2. 557

B. 1. ONLY condenser
 2. 560

C. 1. condenser AND atmospheric
 2. 557

D. 1. condenser AND atmospheric
 2. 560

Original Question

MODIFIED

3.1.2 Atmospheric Dumps

The Atmospheric Dump valves are made of two banks numbered 4 and 5, with four valves in Bank 4 and five valves in Bank 5. The total capacity of these two banks is ~35% RTP. The atmospheric valves do not start opening until the condenser dump valves are full open, this requires a minimum opening signal of 49%. The atmospheric dump valves require signal strength of 100% to be full open. Actuation of the atmospheric dump valves requires a combination of several arming signals. **Figure 3**

Arming Signals

Objective 4A

All three of the following are needed in order to open the Atmospheric Steam Dump valves:

- T-AVG Mode selected
- No Train 'A' P-4 contact
- C-7B Loss of Load Interlock Atmospheric Dump

3.2 Load Rejection Controller

Objective 3B

The Load Rejection Controller is used during a load rejection event to prevent a large T-AVG increase on a loss of load. The controller is enabled by the Steam Dump Select switch being in the "T-AVG" position and no reactor trip has occurred as sensed by the P-4 Train 'B' contact. **Figure 4**

The controller compares Selected T-AVG-2 to Tref and sends a control signal to modulate all banks as necessary. A lead/lag circuit conditions the Selected T-AVG signal. This is to make the steam dumps respond in an anticipatory manner based on the rate of change in the signal. This controller is capable of developing a 100% valve demand signal and therefore all steam dump valves, condenser and atmospheric, could be opened fully. A 3°F deadband exists on the controller to allow rod control insertion to aid in decreasing T-AVG. The controller sends a signal to modulate condenser dump valves open sequentially with some overlap between Banks 1 and 2. When Bank 2 is full open then all valves in Bank 3 start opening (in parallel), then Bank 4, then Bank 5. **Figure 5 & 6**

Loss of Load Interlocks (C-7A & C-7B)

Objective 4B

The Loss of Load interlocks are derived from the three Turbine Impulse Pressure channels using three separate circuits. The current value of impulse pressure is compared to a lagged value (120 second time constant) to determine if there is a difference. This difference value is compared to a fixed negative setpoint of -10% and -30%.

C-7A

The C-7A Loss of Load Interlock Condenser Dump requires greater than a -10% step or -5%/minute ramp difference in 2 out of 3 Turbine Impulse Pressure circuits to allow the condenser dump valves to open. The C-7A signal will open the Arming Solenoid valve to supply air to the condenser dump valves air actuators with C-9 (Condenser available for steam dump) activated. C-7A will arm Banks 1, 2, and 3 by energizing the arming solenoid on the air supply valves.

The C-7A actuation also illuminates Status light, C-7A LOSS OF LOAD INTLK COND DMP VLVS, on MC1. This Status light will stay lit until the solenoid relay is reset. To reset the C-7A interlock the STEAM DUMP SELECT switch is taken to the RESET position. **Figure 4**

The DCS Graphics page for STEAM DUMP has a “C-7A” status light that will go dark as soon as the signal clears.

C-7B

The C-7B Loss of Load Interlock Atmospheric Dump requires greater than a -30% step difference or -15%/minute rate in 2 out of 3 Turbine Impulse Pressure circuits to allow the atmospheric dump valves to open. The C-7B signal will open the Arming Solenoid valve to supply air to the atmospheric dump valves air actuators. C-7B will arm Banks 4 and 5 by energizing the arming solenoid on the air supply valves.

The C-7B actuation also illuminates Status light, C-7B LOSS OF LOAD INTLK ATMOS DMP VLVS, on MC1. This Status light will stay lit until the solenoid relay is reset. To reset the C-7B interlock the STEAM DUMP SELECT switch is taken to the RESET position.

The DCS Graphics page for STEAM DUMP has a “C-7B” status light that will go dark as soon as the signal clears.

3.3 Plant Trip Controller

Objective 3C

The Plant Trip Controller is used to reduce T-AVG to T no-load following a Reactor trip. Tno-load is the equivalent of T-AVG for 0% power or 557°F. The controller is enabled by the STEAM DUMP SELECT switch being in the "T-AVG" position and a Reactor Trip condition has been sensed by the P-4 Train ‘B’ contact. **Figure 4**

The controller compares Selected T-AVG-2 to T no-load and modulates the three banks of the Condenser Dump valves (See **Figure 7**). A lead/lag circuit conditions the Selected T-AVG signal. This circuit initially boosts the magnitude of any change in Selected T-AVG. This is to make the steam dumps respond in an anticipatory manner based on the rate of change in the signal. The controller output is limited to 49% steam dump demand, which is insufficient to open the atmospheric steam dump valves.

STEAM PRESSURE CONTROLLER

- Enabled by 'PRESS' selected on Steam Dump Select switch
- Selected Steam Pressure-1 (2nd highest) (SSP-1) used for controller.
- Steam Equalization Header Pressure used if SSP-1 is in Alternate Action for controller.
- Can only actuate the condenser dumps (49% output max)
- Must be in Steam Press Mode in order to use SLIM Station or soft control
- For startup, dumps will control steam pressure at 1092 psig and T-AVG will increase as power increases.

LOAD REJECTION CONTROLLER

- Enable by Steam Dump Select Switch in 'T-AVG' mode AND NO P-4 Train 'B'
- Compares Selected T-AVG-2 (ST-AVG-2) to Tref (Selected Turbine Impulse Pressure- 2 (STIP-2) and sends signal to modulate all banks as necessary (100% output max).
- 3°F deadband exists to allow rod control to actuate first and last

NOTE: 3˚F dead band prevents any Steam dump actuation. I.E. on a full load rejection steam dumps would close with T-AVG at 560˚F if Tref was 557˚F.

- Three channels of Turbine Impulse Pressure are used for developing C-7A and C-7B (2/3 logic for both interlocks - not median select)
- C-7A → 10% Step or 5%/minute ramp
- C-7B → 30% Step or 15%/minute ramp
- To Reset C-7A/B place STEAM DUMP SELECT switch to 'RESET'

560 degrees corresponds to ~ 1130 psia or 1115 psig

PLANT TRIP CONTROLLER

- Used to reduce T-AVG to T no-load following a Rx Trip
- Enabled by 'T-AVG' mode AND P-4 Train 'B'
- Compares T-AVG (Sel T-AVG-2) to T no-load and modulates Condenser dumps (49% max output)

P-12 LO-LO T-AVG INTERLOCK

- Provides a block to keep from inadvertently cooling down to <553˚F. Keeps NC above 551˚F → Minimum Temperature for Criticality.
- Set @ 553˚F 2/4 NC loops
- Steam Dump INTLK BYP switches, Train 'A'&'B' →
  - Either switch to 'OFF' blocks steam dump actuation
  - Both switches to 'BYP INTLK"
    - Bypasses P-12 block signal for Bank 1 Dump valves
    - If BOTH switches taken to 'OFF/RESET' – Resets bypass signal

Never isolate more than 3 steam dump valves: Atmospheric Dump or Condenser Dump - any combination.

NOTE: If P-12 clears (3/4 NC loops >553˚F) bypass is automatically RESET

NOTE: At 340˚F switches in SSPS cabinets can be used to allow opening all condenser dump valves for increased C/D capability.

Question 45

- Unit shutdowns
- Major power transients

2. Due to resulting limitations in the various measurement methods, specific methods have been designated for use under various operating conditions as follows:

   a. Modes 2, 3, 4 - Leak rate monitoring based on tritium concentration from secondary system grab samples.

   b. Mode 1 at greater than or equal to 5% and less than or equal to 40% Reactor Power - Leak rate monitoring based on condenser off-gas radiation monitor readings. Suspend power escalation until new data can be input into the OAC calculation (reactor coolant Xe-133 Equivalent activity, condenser off-gas flow rate) if the radiation monitor alarms.

   c. Mode 1 at greater than 40% and less than 95% Reactor Power or stable operation (defined as less than 10% power change in one hour) at any power level greater than 40% Reactor Power - Leak rate monitoring using main steam line N-16 monitors, or condenser off-gas radiation monitor subject to the knowledge that radiation monitor counts will increase as the primary source term activity increases.

      1) If a monitor alarms, then suspend power escalation to allow the system to stabilize and to validate the leak rate with a second method.

   d. Mode 1 Normal Operation (greater than or equal to 95% to 100% Reactor Power, or stable operation (defined as less than 10% power change in one hour) at any power level greater than 40% Reactor Power) - Leak rate monitoring using condenser off-gas radiation monitor, main steam line N-16 monitors, or condenser off-gas grab sampling (listed in order of preference).

3. During startups, reactor coolant Xe-133 Equivalent activity and condenser off-gas flow rate should be obtained and entered into the OAC calculation to minimize alarms due to changing conditions during startup.

   a. Obtain and enter values prior to reaching 5% power and at approximately 90% power.

   b. Updated values can also be entered into the OAC at other times as needed to avoid nuisance alarms as conditions change.

4. During startups and until 48 hours after steady-state power, the alarm set points may be set at a level higher than normal as needed to offset nuisance alarms.

5. The alarm set points shall not exceed the lowest shutdown limit at any time during startups.

4.5 High and Low Power Controllers

Objective 4C

SGWLC operates in two modes of control: Low Power and High Power (dependent on Feedwater Flow). There are separate controllers used for Low and High Power Modes of operation.

Low Power Mode

This portion of SGWLC is used when the steam and feed flow transmitters are unable to provide SGWLC with reliable input for level control. This controller is in use when the Median-Selected CF Flow on the associated S/G is not in the HIGH POWER mode. (High Power mode selection is described below.) While in this mode, feed water and steam flow trouble alarms are inhibited.

To ensure a quick and accurate response to changes in N/R level, the output of the Low Power Controller is modified by an additional error signal generated from a comparison of the Selected W/R Level to a fixed level value. This additional error signal results from the W/R Level transmitter response to changes in power. The W/R level trends "predict" the response of N/R level. Westinghouse has taken advantage of this level change phenomenon to provide an anticipatory (feed forward) action needed when steam and feed flow are not used in the level control process. Thus, any trend in W/R level will always be true in direction and this error signal should correctly affect valve position response.

High Power Mode

Once the plant reaches sufficient power to provide reliable Steam and Feed Flow measurements, the High Power Mode automatically takes over control of the Feed Demand signal development. This controller is in use when Selected CF Flow on the associated S/G is greater than or equal to 20% on a power increase, and remains until CF flow drops to 17%. At this time control is restored to the LOW POWER controller.

The High Power Controller has two main inputs:

- The N/R Level Error
- CF/SM Flow Rate Mismatch.

The mismatch is based on the rate of change of the steam and feed water flows.

The Flow Rate Mismatch Error signal is combined with the N/R Level Error signal and this Total Error signal is sent to the High Power Controller.

Controller Bumpless Transfer

In order to ensure that both the High and Low power controllers are correctly set when they are switched into operation, the SGWLC incorporates a tracking system, which forces the unused controller to track the output of the one in use. When the swap over CF Flow setpoint is reached, the outputs of both controllers are the same and no transients occur. The controller that is no longer in use now tracks the operating controller.

Flow Demand Signal

The output of the High and Low Power Controllers is the Total CF Flow Demand signal. This Total CF Flow Demand signal will be used to position individual feed water regulating valves (FRV) and feed regulating bypass valves (FRBV) for the respective S/G. Valve demand is related to the Total Flow Demand output of the High or Low Power controller, i.e. the greater the controller output, the greater the valve open demand.

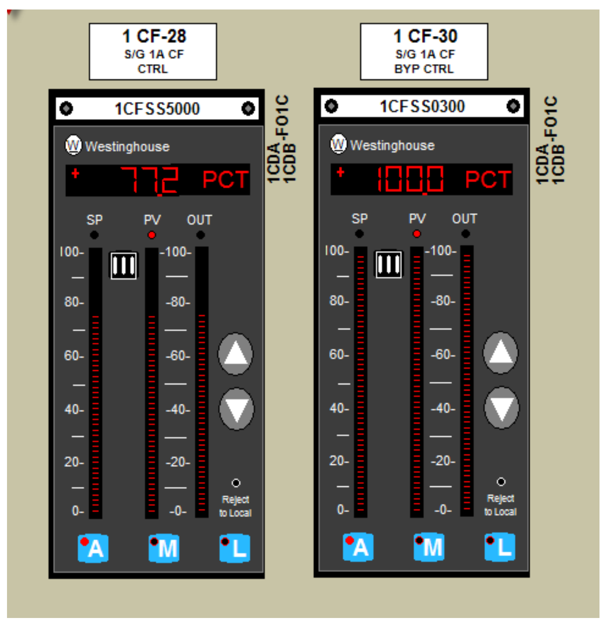
Additionally, each S/G Total flow demand signal is compared and the auctioneered high signal is sent to each CFPT speed controller.

4.6 FRV and FRBV Valve Programmers and Controls

Objective 4D

Each FRV and FRBV controller has a performance curve built in that requires the valve to be at a certain position for any given input signal. When a FRV or FRBV is in AUTO, this input signal is provided by the SGWLC System. Feedback is received from actual valve position to ensure the valve is responding to the demand signal. Significant deviation from demanded position will cause an alarm.

During the initial power increase when the FRBV reaches 55% demanded position, the SGWLC system will insert a small negative signal to the FRBV that will cause the FRV to open slightly. The continued power increase will then be using both the FRV and the FRBV to maintain desired CF flow.



To help prevent a single failure from causing an FRV to fail closed, redundancy is built into the controls. The Automatic control for FRVs consists of duel digital valve positioners and a swapping solenoid. One of these is the Primary and the other is the Backup.

The control signal is sent to the Primary and Backup Valve Positioners. If the Primary Valve Positioner should fail, the Backup Valve Positioner is immediately placed in control with the same signal that was supplied to the Primary Valve Positioner.

9. UNIT DIFFERENCES SUMMARY

Objective 9

S/G Water Level Protection Setpoints

Signal	Unit 1	Unit 2
S/G Hi-Hi Level (P-14)	83%	77%
S/G Lo-Lo Level Rx Trip and CA Auto Start	11%	37%

S/G Water Level Control System Setpoints

Signal	Unit 1	Unit 2
N/R Program Level (0 to 100% power)	39% to 65%	62% to 67%
Low Level Alert (5% above the program Lo-Lo Level	16%	42%

CFPT Trip on Unit 2

For Unit 2 only, the response for a single feed pump trip if Nuclear Power is greater than 65% for SGWLC is different. Following a single feed pump trip with Rx power > 65%, S/G water levels will drop to values less than the program minimum, its return to the program level could come in at a rate that could overshoot the program band and challenge the S/G high-high level P-14 value (77%). Because of this, the control circuit looks at a combination of:

- Rx power > 65%
- Either Feed pump A or B tripped
- Validated S/G level < 52% (7% dead band) and does not exceed 59% within the 30 second time period.

Distractor Part 2

Once these conditions exist, for 30 seconds, a 600 second (10 minutes) timer combines with the condition that if S/G level has now risen greater than 53% to change the S/G level setpoint to 53%. Once the 10 minute timer drops out, the controlling circuit now inserts a slow ramp back from 53% to the current programmed value between 62 and 67%.

Question 47

- The plant was at 100% power when an ATWS event occurred
- The crew entered FRP-S.1, RESPONSE TO NUCLEAR POWER GENERATION ATWS
- All SG Narrow Range levels are OFF-Scale LOW

IAW FRP-S.1, AFW flow must be greater than a MINIMUM value of ____(1)____ GPM to establish a Secondary Heat Sink?

In regards to core age, the most decay heat generated after a reactor trip is present at ____(2)____ of life.

A. (1) 300
(2) beginning

B. (1) 300
(2) end

C. (1) 600
(2) beginning

D. (1) 600
(2) end

Original Question

MODIFIED

Question 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Verify Reactor Trip:**	**Perform the following:**
__ • All rod bottom lights - LIT	a. Trip reactor.
__ • All reactor trip and bypass breakers - OPEN	b. **IF** reactor will not trip, **THEN** concurrently perform the following:
__ • I/R power - TRENDING DOWN.	__ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees)
	__ • **GO TO** EP/1/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS).
Correct Flow Path	Distractor Flow Path
3. **Verify Turbine Trip:**	**Perform the following:**
__ • All turbine stop valves - CLOSED.	__ a. Trip turbine.
	b. **IF** turbine will not trip, **THEN** perform the following:
	__ 1) Depress "MANUAL" pushbutton on turbine control panel.
	__ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons.
	3) **IF** control valves will not close, **THEN** CLOSE the following valves:
	__ • All MSIVs __ • All MSIV bypass valves.

Question 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. **Control S/G levels as follows:**	
__ a. Verify total CA flow - GREATER THAN 450 GPM. Correct Answer	a. Perform the following: 1) <u>**IF**</u> N/R level in all S/Gs less than 11% (29% ACC), <u>**THEN**</u> perform the following: __ • Start CA pumps __ • Ensure correct valve alignment. 2) <u>**IF**</u> N/R level in all S/Gs less than 11% (29% ACC) <u>**AND**</u> feed flow greater than 450 GPM cannot be established, <u>**THEN**</u> concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • <u>**GO TO**</u> EP/1/A/5000/FR-H.1 (Response to Loss of Secondary Heat Sink).
__ b. <u>**WHEN**</u> each S/G N/R level greater than 11% (29% ACC), <u>**THEN**</u> control feed flow to maintain that S/G N/R level between 11% (29% ACC) and 50%.	
__ 20. **Verify all CA isolation valves on intact S/Gs - OPEN.**	__ **OPEN valve(s).**
__ 21. **Verify S/I equipment status based on monitor light panel(s) - IN PROPER ALIGNMENT.**	__ **Align equipment.**

Question 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Control S/G levels as follows:**	
__ a. Verify N/R level in at least one S/G - GREATER THAN 11% (29% ACC).	a. Perform the following:
	__ 1) **IF** total CA flow less than 1000 GPM, **THEN** start pumps and align valves as required.
Distractor	__ 2) Maintain total CA flow greater than 1000 GPM until N/R level greater than 11% (29% ACC) in at least one S/G.
	__ 3) **WHEN** N/R level greater than 11% (29% ACC) in at least one S/G, **THEN** perform Step 9.b.
	__ 4) **GO TO** Step 9.c.
__ b. THROTTLE feed flow to maintain all S/G N/R levels between 11% (29% ACC) and 50%.	
__ c. **WHEN** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).	
10. **Ensure all dilution paths isolated as follows:**	
__ a. Place NC makeup control switch to "STOP".	
__ b. Place reactor makeup water pumps to "OFF".	

Question 47

STEP 19: Control S/G levels as follows:

PURPOSE:

To ensure CA flow to the steam generators.

APPLICABLE ERG BASIS:

Correct Answer

CA flow is necessary for secondary heat sink. If S/G level is in the narrow range in at least one S/G, a heat sink is available. Therefore, CA flow is needed only to maintain level. If adequate CA flow for decay heat removal cannot be established, the transition to the FR-H.1, Response To Loss Of Secondary Heat Sink, is necessary to establish an alternate source of feed flow or an alternate heat sink.

A range for S/G level control was specified so that the operator would not be so involved in maintaining an exact level in the S/Gs that other operator actions would be delayed (DW-84-010).

PLANT SPECIFIC INFORMATION:

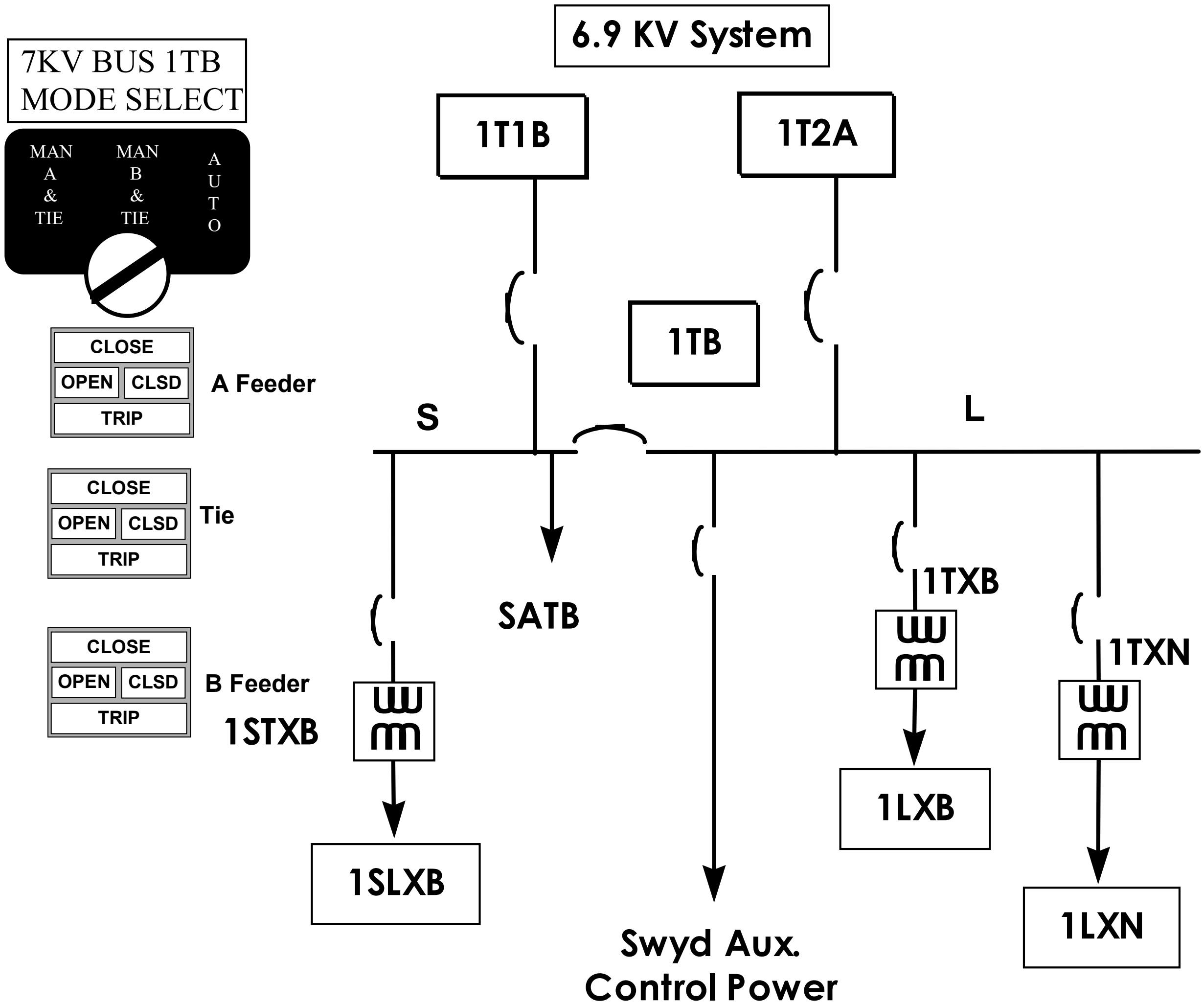
This step allows the operators to reset and throttle CA flow to S/Gs after adequate heat sink is verified. This direction has been added for the following reasons:

- Several operator feedback items have been received concerning overfilling S/Gs. By providing procedural direction to control feed flow to S/Gs at this point, the potential to limit overfilling the S/Gs is reduced. This is especially a concern if the event started at less than 100% power or low decay heat levels.
- Attempting to control S/G levels at this point also may aid the operator in diagnosing a SGTR in subsequent steps based on the inability to control a particular S/G level.

Unit 1 and Unit 2 difference - The S/G upper control band N/R level setpoint is based on the ERG generic upper limit for Unit 1 and the no-load level for Unit 2 since different model S/Gs are used for each unit.

Continued On Next Page.

2.1.2 1TB contains 1T1B (short side- **S**) and 1T2A (long side- **L**). Each 6900V bus is powered from 2 auxiliary transformers using 2 incoming breakers normally closed with the tie breaker open.



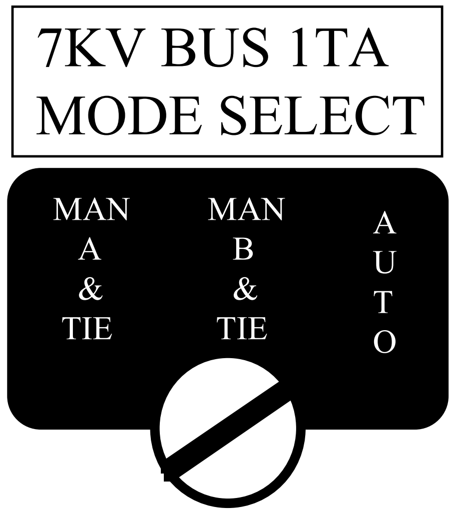
Objective # 4, All

- Short Side (1T1B)
  - SATB
  - 1STXB
  - 1B NC Pump
- Long Side (1T2A)
  - 1 YV-C-2
  - 1B RC Pump
  - 1B Hotwell Pump
  - 1B Condensate Booster Pump
  - B RL Pump
  - 1TXB
  - 1TXN
  - Switchyard Auxiliaries Feeder

Question 48

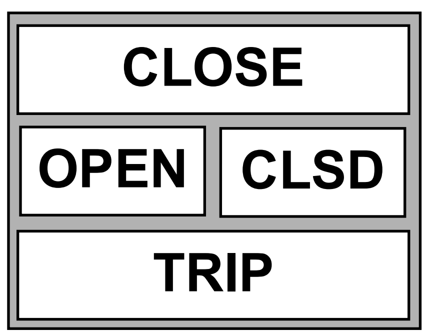
Objective # 5B, All

The incoming and tie breakers are controlled from MC-11 in the control room. A 3-position transfer switch, (See one below, typical of four switches on each Unit) when in AUTO, allows auto swap; when it is in MAN A & TIE it allows operating the "A" incoming breaker and the tie breaker; when it is in MAN B & TIE it allows operating the "B" incoming breaker and the tie breaker.



TRIP – CLOSE pushbuttons operate the breaker

OPEN – CLSD light indication provides the position of the breaker



To supply a 6900V bus completely from one auxiliary unit transformer, the tie breaker can be closed and one incoming breaker opened. This transfer can occur automatically or manually.

For Auto Transfer, ensure Mode select switch is in AUTO. The auto transfer would then be initiated by a Zone Lockout or 75% UV on the incoming line with no fault on the affected bus. If in sync, a fast transfer will occur. The incoming breaker opens before the tie breaker closes, but the transfer occurs within a few cycles with no loss of load.

If not in sync, a slow transfer will occur. The transfer is delayed to allow voltage to decay to a point that synchronization is not a concern (25% voltage). This will take approximately one second and a loss of loads on the bus will occur.

PT/1(2)/A/4350/005 (6900V Normal Auxiliary Power Automatic Transfer Test) is performed each outage to ensure the auto swap occurs as designed.

Automatic Fast Transfer Switch

The Automatic Fast Transfer switch will allow defeat of the fast transfer when the unit is off line. This switch is located in the 6900V bus rooms on the control panels with the under voltage transfer relays. The purpose of this switch is to eliminate possible equipment damage from surges on fast transfer. The switch has two positions, ENABLE or DEFEAT.

With the Main Generator off-line, this switch is placed in the DEFEAT position by the Unit Shutdown procedure and then only slow transfers will occur. This switch controls the tie breaker. The defeat removes the sync check relays (25) from the circuit and provides for only a slow transfer.

Distractor Part 2

3.3 600V Unit Bus Abnormal Operation

Objectives # 5C, All

Both incoming breakers on an MCC are normally closed. One feeder breaker from a load center to an MCC will be closed and the alternate feeder to the MCC from the other load center will be open. On loss of power at the “normal” load center the supply from the “alternate” load center will auto close to supply the MCC.

LXC/LXD Unit Buses

If power is lost to 1(2)LXC and 1(2)LXD simultaneously, 1(2)LXC will swap to 1(2)TXS transformer supplied by 1(2)TA. 1(2)LXD is time delayed to allow this to occur. The alternate incoming breakers are interlocked to prevent both from closing at the same time. Each incoming breaker has a MAN-AUTO switch and an OPEN-NORMAL-CLOSE switch. An auto transfer of power will occur on low voltage on the incoming line. If the sources are in sync the alternate breaker closes and then the normal breaker trips. This is a Hot Bus Transfer.

- D/G Control Panels
- D/G CO2 Fire Protection
- Control Rod Drive Control Panel

2.7 Static Inverters (KXIA, KXIB, KUI)

Objective 3D, All

The Static Inverters convert 125 VDC to 120 VAC power and supply the AC portion of Auxiliary Control Power System. A DC input and an AC output breaker is provided.

The normal power supply to Static Inverters KXIA and KXIB is the associated 125 VDC Distribution Panel.

The normal power supply to Static Inverter KUI is the output of 125 VDC Auctioneering Diodes ADA and ADB. Inverter KUI normally supplies 120 VAC regulated panel P-1 (OAC power supply) through an automatic static transfer switch, manual transfer switch, and isolation transformer KUT.

Some of the indications provided on the inverter are:

- AC volts (upstream of inverter output breaker)
- AC amps (upstream of inverter output breaker)
- Frequency (upstream of inverter output breaker)
- INVERTER OUTPUT LOW VOLTAGE light
- IN SYNC - amber light will be lit when the normal and alternate sources are synchronized.

2.8 Automatic Static Transfer Switch (KXAA, KXAB, KUA)

Objective 3E, All

An automatic static transfer switch (KXAA, KXAB, and KUA) associated with each static inverter provides automatic power transfer to an alternate power source (regulated power) on low inverter output voltage. Manual swap capability is also provided via pushbuttons on the Automatic Transfer Switch. **Correct Answer Part 2**

Once the Auto Transfer Switch has automatically swapped to alternate source, a 60-second delay is initiated. After 60 seconds, if the transfer was due to a total loss of inverter output voltage, the switch will transfer back to the inverter whenever voltage and frequency have returned to normal. However, a manual transfer of the static transfer switch will require the operator to manually re-transfer the inverter back to normal.

Distractor Part 2

The Auto Transfer Switch can be bypassed using the Manual Bypass Switch.

1. Limits and Precautions

1.1 Smoking, open flames, or evolutions that create sparks are **NOT** allowed in the battery area.

1.2 An inverter may **NOT** be restarted within 60 seconds after taking it out of service in order to allow capacitors and rectifiers to cool down.

1.3 IAE provides corrective action when voltages are **NOT** within the ranges given.

2. Initial Conditions

_____ Verify 1KXIB is in service per Enclosure 4.3 (Inverter Startup).

3. Procedure

_____ 3.1 Verify the "IN SYNC" indicator lights on the following panels are lit:

- ☐ 1KXMB
- ☐ 1KXAB

_____ 3.2 Depress the "ALTERNATE AC SOURCE TO LOAD" pushbutton on 1KXAB.

_____ 3.3 Verify the following indicator light status on 1KXAB:

- ☐ "ALTERNATE AC SOURCE SUPPLYING LOAD" is lit
- ☐ "INVERTER SUPPLYING LOAD" is dark

_____ 3.4 Place 1KXIB B2 (Inverter Output) in the "OFF" position.

_____ 3.5 Place 1KXIB B1 (Battery Input) in the "OFF" position.

NOTE:
1. At this point, 1KXIB is shutdown with 1RDB supplying 1KXPB through 1KXAB.
2. Subsequent steps are to return 1KXIB to service and 1KXAB to normal condition.
3. Step 3.7 must be performed immediately after Step 3.6 is completed, thus place keeping for both steps is performed after both complete. Failure to close the "BATTERY INPUT" breaker immediately after the "PRECHARGE" pushbutton is released may result in blown inverter input fuses.

_____ 3.6 Depress and hold the "PRECHARGE" pushbutton on 1KXIB until the "PRECHARGE" indicator light (above pushbutton) has been lit for a minimum of 5 seconds.

_____ 3.7 Place 1KXIB CB1 (Battery Input) in the "ON" position.

Question 50

_____ 12.12 **IF** all the following are true, bar and roll D/G with air per OP/1/A/6350/002 (Diesel Generator Operation): {PIP 96-1185}

- ☐ D/G 1A has been shutdown for greater than 4 hours.
- ☐ D/G 1A has **NOT** been barred and rolled on air within the last 12 hours.{PIP 00-324}
- ☐ D/G 1B is operable.
- ☐ Tech Spec assessment will allow D/G 1A inoperability.

_____ 12.13 **IF** this is the first start of D/G 1A in the month of March, swap D/G 1A lube oil strainers per OP/1/A/6350/002 (Diesel Generator Operation).

Lube Oil Strainer placed in service _____________

_____ 12.14 **IF** Step 12.12 was **NOT** performed, verify the Fuel Pump Linkage Auto Shutdown Cylinder Shaft is retracted.

12.15 Perform the following:

_____ • Verify 1A1 and 1A2 VG Compressors maintain VG Tank pressure is $\geq$ 235 psig.

_____ • Verify "CONTROL AIR PRESS" on 1DECPA is 58 - 62 psig.

_____ • Close 1VN-1 (1A D/G Exhaust Silencer Drain) (DB-559, DD-39).

12.16 Perform the following at the Diesel Engine Shutdown Logic Monitoring Panel (1ELMC0029):

_____ 12.16.1 Depress and hold the "Alarm Ack" button for 1 second.

_____ 12.16.2 Verify that any "A" or "B" alarm lights that were flashing are now lit solid.

_____ 12.17 **IF AT ANY TIME** the D/G run is aborted prior to completion of:

- ☐ Step 2.38, 2.39 or 2.40 of Enclosure 13.1 (D/G 1A Operation From D/G Room)

OR

- ☐ Step 2.39, 2.40 or 2.41 of Enclosure 13.2 (D/G 1A Operation From Control Room) perform the following:

_____ 12.17.1 Notify Unit/WCC SRO immediately to make a determination of operability.

_____ 12.17.2 **IF** attempting subsequent restart, a re-verification of Step 12.1 through Step 12.16 shall be performed.

_____ 3.52 Open 1VN-1 (1A D/G Eng Exhaust Silencer Drain) (DB-557, CC-38).

Distractor Part 1

3.53 Drain any accumulated oil out of the crankcase vent drip leg as follows:

_____ 3.53.1 Verify 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain) (DB-556, DD-39) is closed.

_____ 3.53.2 Remove pipe cap downstream of 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain).

_____ 3.53.3 Place the waste oil container downstream of 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain).

_____ 3.53.4 Open 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain) to drain any accumulated oil.

_____ 3.53.5 **WHEN** oil has drained, close 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain).

_____ 3.53.6 Replace pipe cap downstream of 1ZD-1 (1A D/G Eng Crankcase Vent Drip Leg Drain).

_____ 3.54 **IF** the D/G run was normal **AND** the D/G did **NOT** trip, perform the following at the Alarm Monitor Panel (1ELMC0029):

☐ 3.54.1 Depress and hold the "Alarm Ack" button for 1 second.

☐ 3.54.2 Verify that any "A" or "B" lights that were flashing are now illuminated solid.

_____ 3.55 Verify AC CONTROL POWER ON light LIT on 1DECPA.

3.56 Verify DC Control Power on 1DECPA by performing one of the following:

_____ 3.56.1 Verify DC CONTROL POWER ON light LIT.

OR

_____ 3.56.2 **IF** DC CONTROL POWER ON light is **NOT** LIT,
THEN perform the following:

- _____ Verify 1.47 Bypass Panels "D/G A BYPASS" light is green.
- _____ Have IAE verify voltage at the DC CONTROL POWER light socket.

_____ 3.57 Verify 1A D/G has adequate fuel oil storage tank volume (≥ 77,100 gallons) per one of the following:

Question 50

_____ 3.3 **IF** the D/G is to be barred, perform the following:

_____ 3.3.1 Select the air supply to be used from the following:

For D/G 1A

☐ 1VG-13 (1A D/G Eng Starting Air Tank 1A1 Drain) (DB 558, DD-38)

☐ 1VG-14 (1A D/G Eng Starting Air Tank 1A2 Drain) (DB 558, DD-38)

For D/G 1B

☐ 1VG-57 (1B D/G Eng Starting Air Tank 1B1 Drain) (DB 558, BB-38)

☐ 1VG-58 (1B D/G Eng Starting Air Tank 1B2 Drain) (DB 558, BB-38)

3.3.2 Perform the following to the air supply selected in step 3.3.1:

_____ 3.3.2.1 Open the valve to drain the moisture

_____ 3.3.2.2 **WHEN** moisture has been drained, close the valve.

_____ 3.3.2.3 Attach one end of a 250 psig rated hose.

_____ 3.3.3 Remove Danger Stay Clear Rotating Equipment sign and rope which limits access to the D/G flywheel area for barring the D/G.

_____ 3.3.4 Attach other end of hose to the barring device coupling.

_____ 3.3.5 Open the indicator cock on each cylinder.

Cylinder	☑
1L	☐
2L	☐
3L	☐
4L	☐
5L	☐
6L	☐
7L	☐
8L	☐

Cylinder	☑
1R	☐
2R	☐
3R	☐
4R	☐
5R	☐
6R	☐
7R	☐
8R	☐

_____ 3.3.6 **IF** liquid issues from any of the cocks, Engineering shall be notified to decide if any corrective action shall be taken before proceeding. {PIP 98-0441}

5.4 Containment Ventilation Isolation (S_H)

Objective 8F

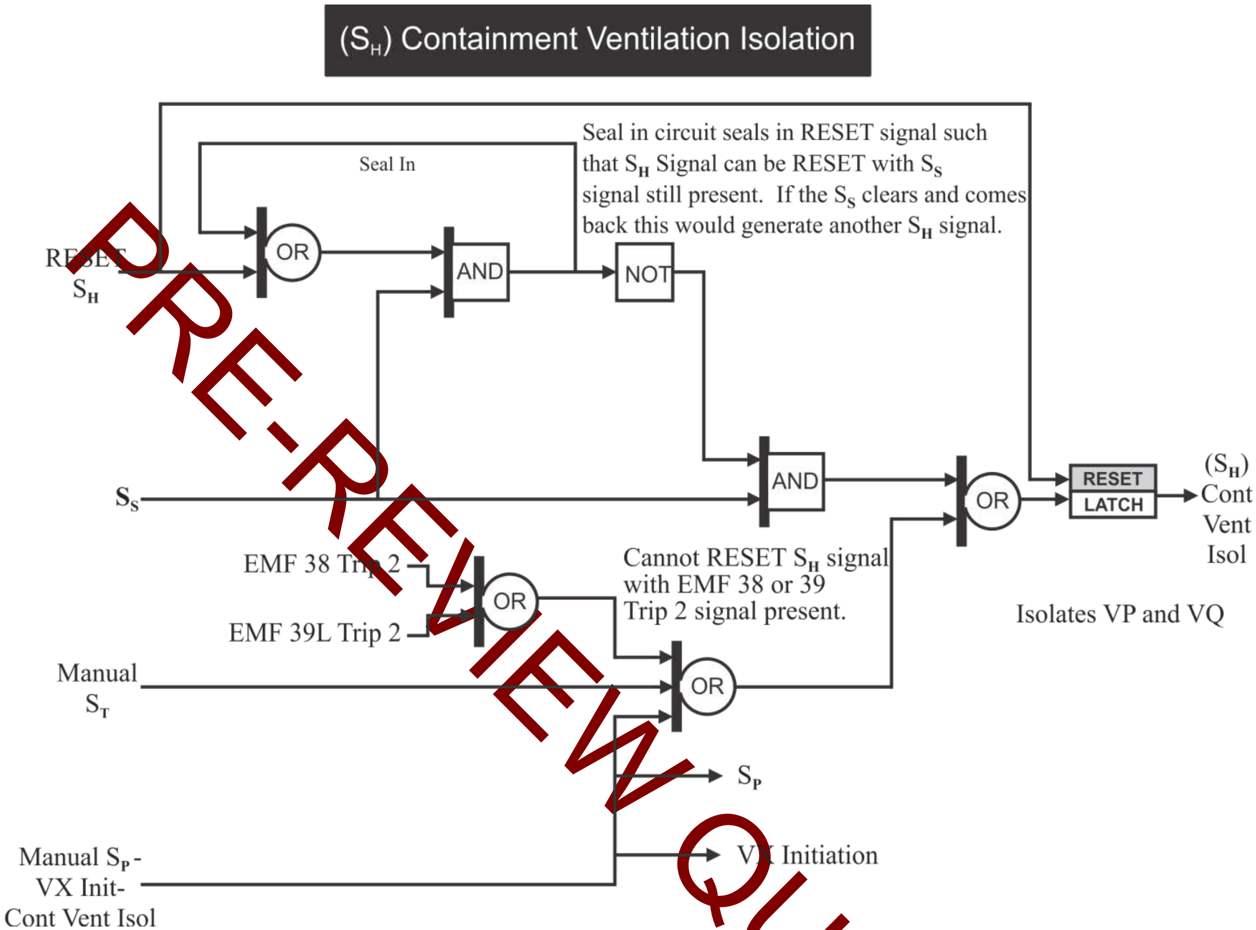


Figure 18

S_H shuts down and isolates VP and isolates VQ Containment Isolation valves.

Four Signals can actuate an S_H:

- Manual "Phase A" (S_T): Train A (B) S_T will actuate train A (B) S_H.
- Manual "Phase B", VX Initiate, Cont Vent Isol": Train A (B) (Phase B, VX Initiate, Cont Vent Isol) will actuate train A (B) SH. This is a single pushbutton that actuates three functions. (Figure 11)
- S_S Signal: Train A (B) S_S will actuate Train A (B) S_H.
- EMF 38, or 39 L, TRIP 2: High Containment Particulate or Gas will actuate BOTH Trains of S_H.

There is one Cont Vent Isol Reset pushbutton per train. The reset is functional with S_S present as long as the EMF Signals are not present. S_H cannot be reset with the EMF Trip 2 signals present.

Question 52

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Verify both NS pumps - OFF.**	**Perform the following:** a. <u>**IF**</u> all the following conditions met: __ • NS in service __ • NS suction aligned to containment sump __ • RN established to associated NS Hx, __ <u>**THEN**</u> <u>**RETURN**</u> <u>**TO**</u> procedure section and step in effect. __ b. Ensure both NS pumps - OFF.
2. **CLOSE the following valves:** __ • 1NS-20A (NS Pump 1A Suct From FWST) __ • 1NS-3B (NS Pump 1B Suct From FWST).	
__ 3. **Verify containment pressure - GREATER THAN 3 PSIG.**	**Perform the following:** __ a. Wait up to 20 seconds for 1NS-20A and 1NS-3B to close. __ b. OPEN 1NS-18A (NS Pmp A Suct From Cont Sump). __ c. OPEN 1NS-1B (NS Pmp B Suct From Cont Sump). __ d. <u>**IF AT ANY TIME**</u> containment pressure goes above 3 PSIG, <u>**THEN**</u> perform Enclosure 2 (Aligning NS for Recirculation). __ e. <u>**RETURN TO**</u> procedure section and step in effect.

Question 52

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. (Continued)

CAUTION **Exceeding 4650 GPM RN flow through an NS Hx will cause damage to the Hx tubes.**

k. Align RN to NS Hx 1A as follows:

1) Verify at least one of the following:

__ • All Unit 1 and Unit 2 RN pumps - ON

OR

__ • RN System - ALIGNED FOR SINGLE SUPPLY HEADER OPERATION.

RESPONSE NOT OBTAINED:

1) Perform the following to support NS Hx cooling flow:

__ a) **IF** only one A train RN pump on, **THEN** CLOSE Unit 2 2RN-48B (RN Supply X-Over Isol).

b) **IF** only A train RN pumps on, **THEN** CLOSE one of the following Unit 2 valves:

__ • 2RN-47A (RN Supply X-Over Isol)

OR

__ • 2RN-48B (RN Supply X-Over Isol).

__ 2) OPEN 1RN-144A (NS Hx 1A Inlet Isol).

__ 3) **WHEN** 1RN-144A begins to open, **THEN** OPEN 1RN-148A (NS Hx 1A Otlt Isol).

Question 53

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **IF AT ANY TIME VI pressure less than 55 PSIG AND trending down, THEN perform the following:**	
__ a. Trip reactor.	
__ b. **WHEN** reactor power less than 5%, **THEN** depress "CLOSE" pushbutton for all MSIVs.	
__ c. Continue in this procedure as time permits.	
__ d. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).	

Question 53

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 2. **Verify reactor trip breakers - CLOSED.**	**Perform the following:**

a. **IF** ND System in RHR mode, **THEN** control ND flow and cooldown rate for operating ND train(s) as follows:

- Train A:

__ 1) Place "PWR DISCON FOR 1NI-173A" switch in "THROT" position.

__ 2) THROTTLE 1NI-173A (ND Hdr 1A To Cold Legs C&D) to maintain desired flow and cooldown rate.

__ 3) Raise output for 1ND-26 (ND Hx 1A Outlet Ctrl) to 100%.

__ 4) Place 1ND-27 (ND Hx 1A Bypass Ctrl) in manual and closed.

- Train B:

__ 1) Place "PWR DISCON FOR 1NI-178B" switch in "THROT" position.

__ 2) THROTTLE 1NI-178B (ND Hdr 1B To Cold Legs A&B) to maintain desired flow and cooldown rate.

__ 3) Raise output for 1ND-60 (ND Hx 1B Outlet Ctrl) to 100%.

__ 4) Place 1ND-61 (ND Hx 1B Bypass Ctrl) in manual and closed.

__ b. **GO TO** Step 4.

compressed air enters an intercooler which removes the heat generated during the first compression stage. After leaving the second stage, the hot compressed air enters an aftercooler and water separator. These components remove the heat and moisture generated from the second stage of compression. The centrifugal compressors contain an additional moisture separator after the intercooler.

The Recirculated Cooling Water (KR) System normally provides cooling water to the three compressors. The Fire Protection System (RF) is provided as a manually aligned backup cooling water source for compressors E and F during a blackout. When cooling water is aligned, it cools the aftercooler, oil cooler and intercooler.

The air leaving the compressors discharges into their respective **Instrument Air Receivers** which are used to smooth out surges and act as storage volume. An alternative flow path, designated the alternate wet header, may be used from the receivers to the dryer package. The alternate wet header piping utilizes a separate coalescing prefilter prior to entering the desiccant dryer towers. The alternate wet header will permit subsequent VI system piping repair/replacement with no loss of system functionality.

The bulk air supply to the Unit 1 and Unit 2 condensate polishing and steam generator blowdown demineralizers branches off the instrument air compressor discharge header upstream of the instrument air dryers. Two check valves, with a trap between them, are located in this line to prevent the backflow of water into the Instrument Air System.

Moisture is removed from the instrument air by one of the coalescing prefilters. The instrument air is then dried to meet the system required -23°F dew point by one of three desiccant **Air Dryers E, F, or G** piped in parallel. The air then passes through one of the afterfilters. Downstream of the instrument air afterfilters, the instrument air system forms a common header which supplies air to the plant. The air is again filtered through a filter regulator prior to each air operated valve or instrument.

The Instrument Air System is capable of supplying compressed air to both the Instrument Air and the Station Air systems. Self-contained, back pressure control valve **1VI-500** is installed in the crossover header between the instrument air and station air headers. If the instrument air system pressure drops below 80 psig, valve 1VI-500 will close to terminate air supply to the Station Air System while maintaining air supply to the Instrument Air System.

The Station Air System is capable of supplying compressed air to the Instrument Air System upon loss of instrument air header pressure. An instrument air header pressure below 76 psig will open valve **1VS-78** such that the station air compressors will back up the instrument air compressors. The air supplied through the Station Air System passes through two oil removal filters prior to entering the Instrument Air System. The Station Air connection is located in the instrument air discharge header upstream of the instrument air dryers. **Distractor Part 2**

Question 54

Given the following:

- Unit 1 is at 100% power.
- A steam break occurred on the Main Steam Equalization Header.
- Safety Injection 1B failed to actuate.

Which one of the following describes the status of the containment isolation valves associated with Phase A and Phase B before any operator actions?

A. Only Phase A (S_T) Train 1A valves close.
Only Phase B (Sp) Train 1A valves close.

B. Only Phase A (S_T) Train 1A valves close.
No Phase B (Sp) valves close.

C. All Phase A (S_T) Train 1A and 1B valves close.
Only Phase B (Sp) Train 1A valves close.

D. All Phase A (S_T) Train 1A and 1B valves close.
No Phase B (Sp) valves close

Original Question

MODIFIED

5.2 Phase A (S_T)

Objective 8D

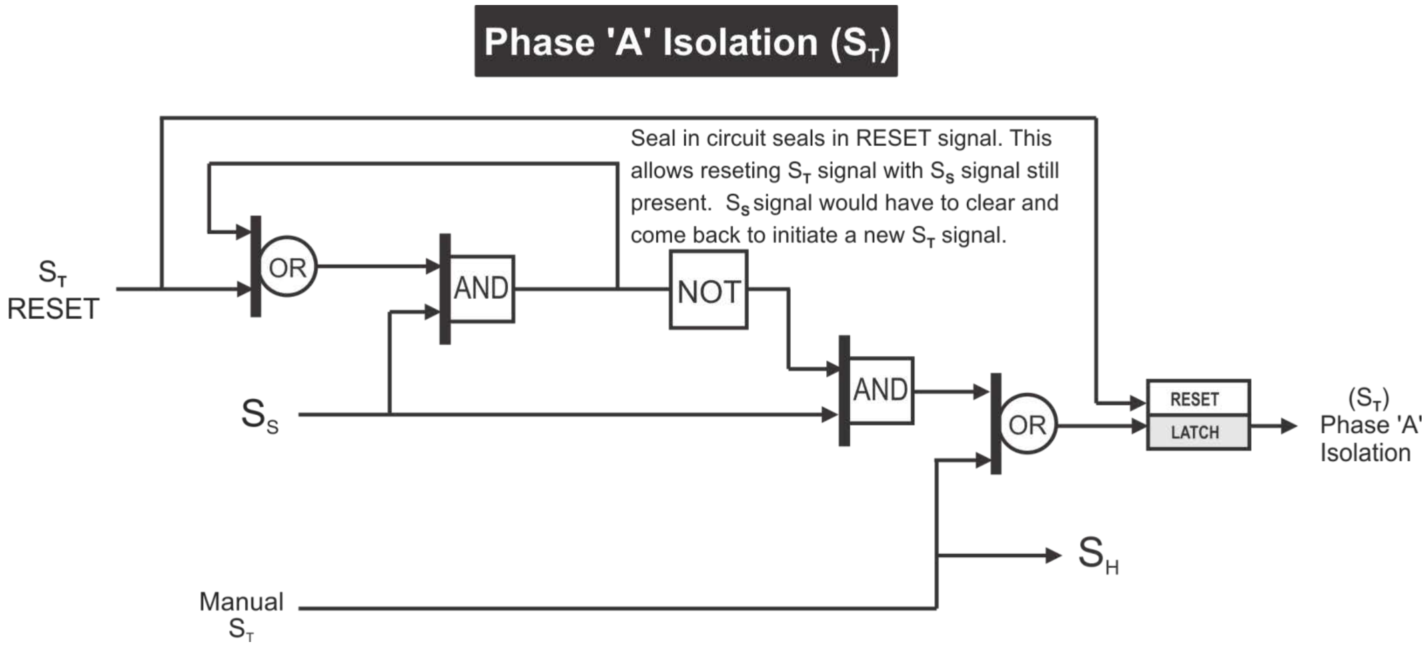


Figure 16

Phase "A" Isolation isolates all containment penetrations, which are non-essential to reactor / containment safety or cooling.

Two signals can generate an S_T signal:

- Manual: One pushbutton per train under clear plastic cover on MC11 (Figure 11).
- Safety Injection (S_S) Train A (B) directly actuates Train A (B) S_T.

Annunciator "PHASE A CONTAINMENT ISOLATION" (Panel AD-13, Window B3) illuminates on an S_T.

Question 54

5.3 Phase B (S_P)

Objective 8E

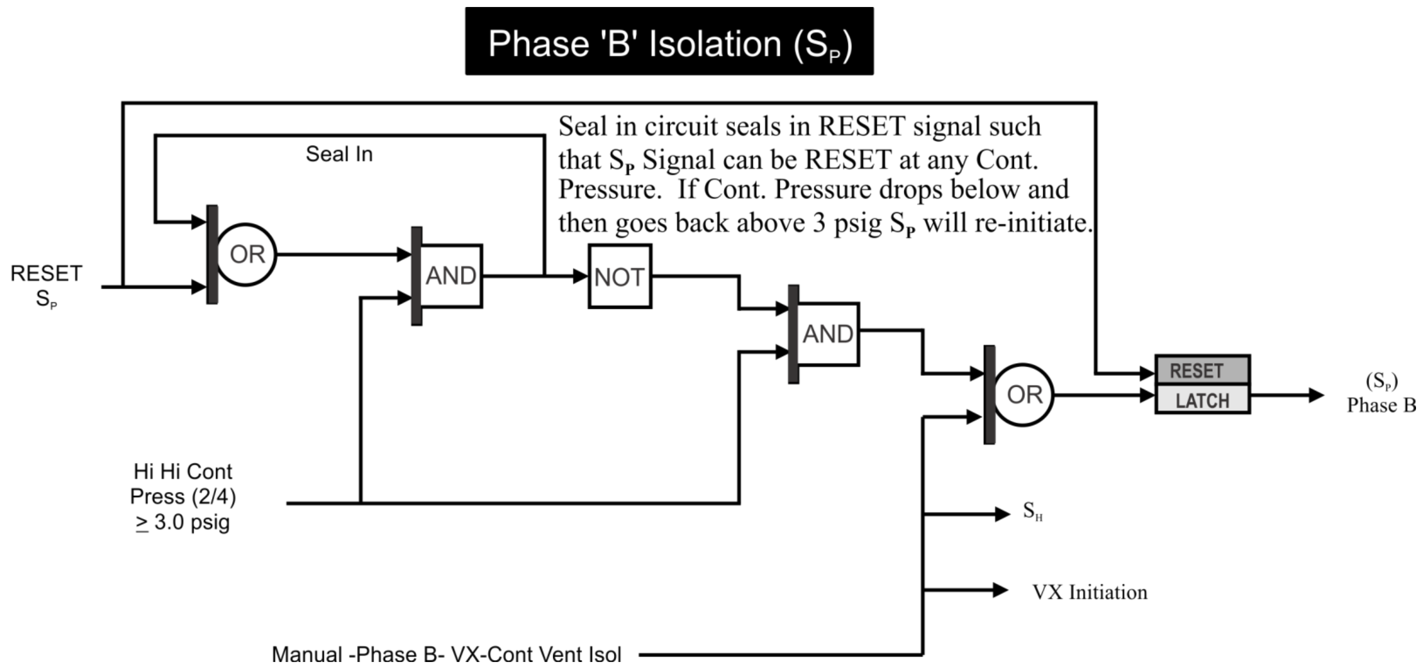


Figure 17

Phase "B" Isolation completes the isolation of non-essential containment penetrations including KC to the NCP's.

Two signals can generate an S_P signal:

- Manual: One pushbutton per train under clear glass cover on MC11 (Phase "B", VX Initiate, Cont Vent Isol Button). The "Phase "B", VX Initiate and Cont Vent Isol" Button, when depressed, will initiate both functions. (Figure 11)
- Hi-Hi Containment Pressure: 2/4 Containment Pressure Channels greater than or equal to 3.0 PSIG.

There is one Phase "B" Reset Button for each train to allow manual control of S_P valves. The S_P reset is functional with any pressure in containment.

Phase "B" Isolation indication:

- Annunciator "CONTAINMENT ISOLATION PHASE B" (Panel AD-13, Window D4) illuminates on an S_P.
- Also, Annunciator "CONT HI-HI PRESS" (Panel AD-13, Window B4) signals Control Room on HI HI Containment Pressure on any one channel.

- Prevents tripping the YV chillers on low flow after the CRDMs are de-energized. Failure to operate the LCVUs in "MAX COOL" mode after de-energizing the CRDMs can also result in over pressurization of the YV piping causing relief valves to lift.

Cooling Water Bypass Valve

Objective 4C, All

A "BYPASS" valve or "FULL FLOW" valve is installed in parallel with the normal cooling water flow control valve. This valve will automatically open for each of the following conditions:

1. An LCVU is selected to HIGH speed.
2. Containment pressure rises to greater than or equal to 0.5 psig. Refer to Figure 4.

2.1.2 Two (2) pipe tunnel booster fans (PTBFs)

PTBFs are used to circulate lower containment air outside of the crane wall. Like the LCVUs. The PTBFs are two speed fans with "OFF-LOW-HIGH" selector switches on the back of the main control board. The PTBFs do NOT contain cooling water coils. The PTBFs draw cool air from the "D" LCVU discharge duct, prior to the supply header, and discharge to the pipe tunnel and DRPI cabinet area.

Flow / PTBF (Hi/Lo Speed)	9,520 / 4,760 CFM (Unit 1) 9,520 / 5,600 CFM (Unit 2)

Normally only one of the two PTBFs is in service operating at the same speed as the in-service LCVUs. Indication of fan discharge air flow is provided on a local panel in the auxiliary building. PTBF vibration resets are provided on this same panel.

Figure 2: LCVU – Normal Mode

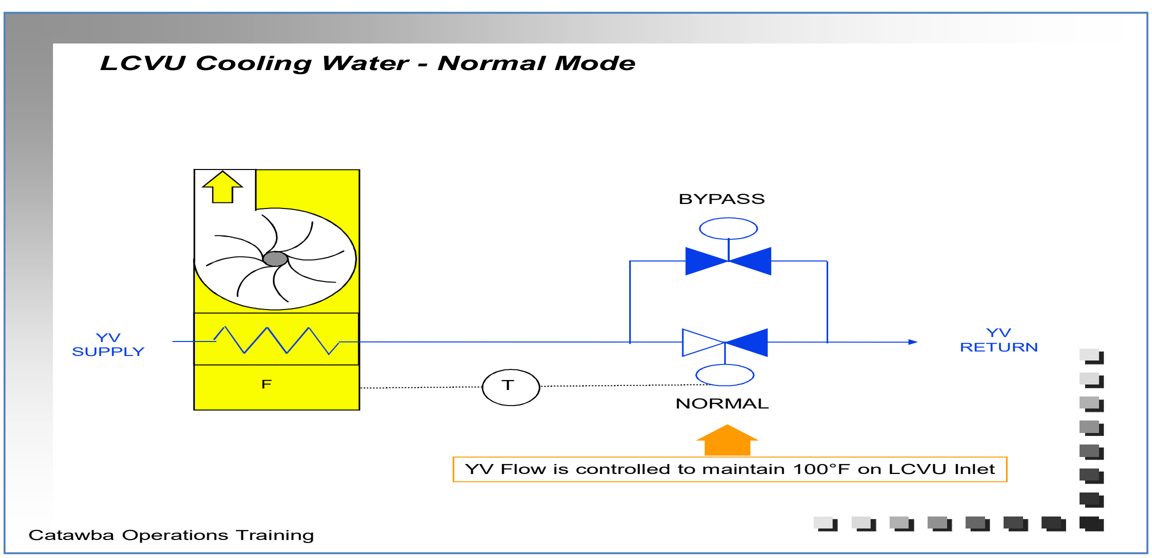


Figure 3: LCVU – MAX COOL Mode

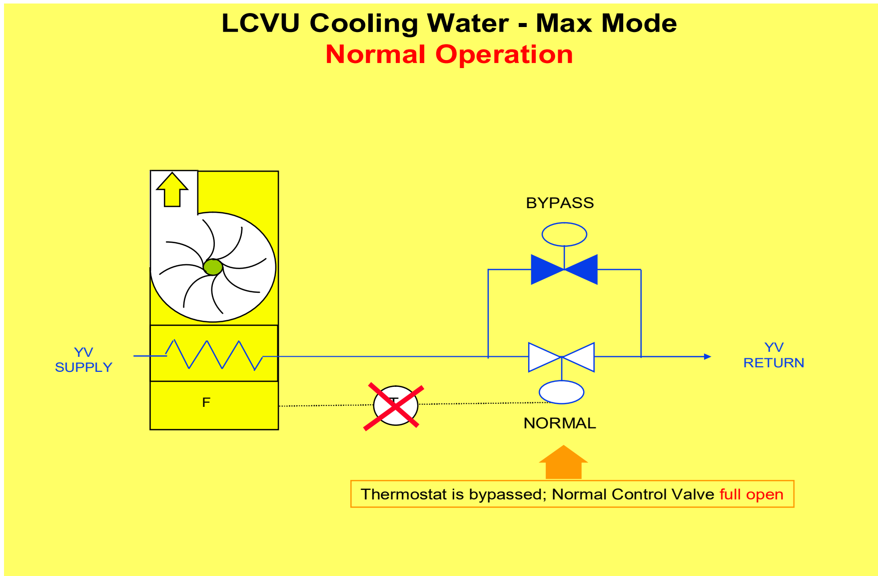
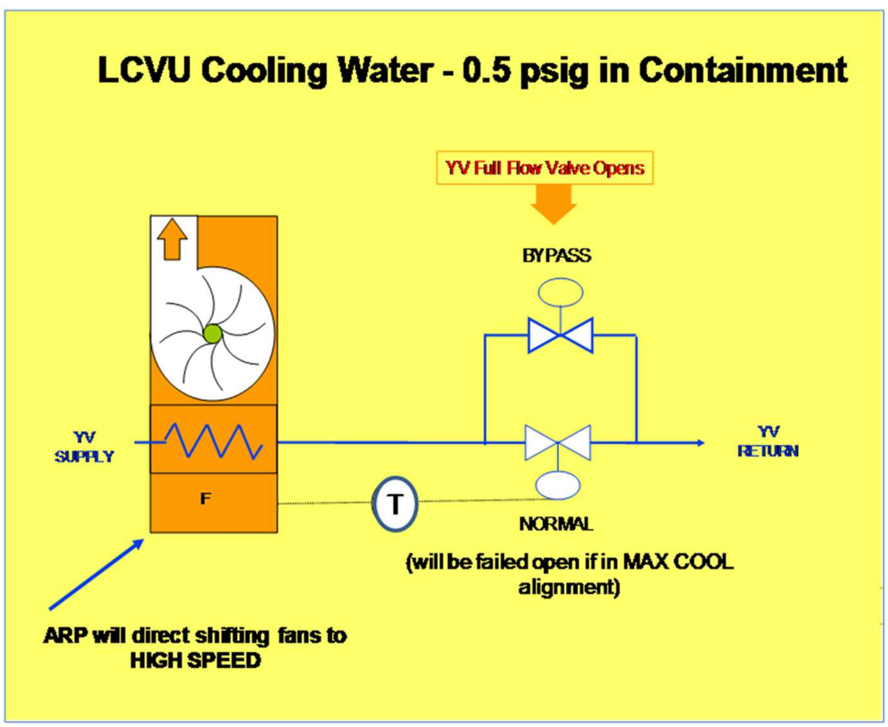


Figure 4: LCVU – 0.5 psig in Containment



LOWER CONT PRESS 0.5 PSIG INITIATE HI SPEED B/12

SETPOINT: 0.5 psig rising

ORIGIN: NSPS5150, NSPS5230 – ½ Channels

PROBABLE CAUSE: LOCA

Main steam line rupture.

AUTOMATIC ACTIONS: If it's associated ventilation unit is in low speed, the following valves open to double the flow to the lower containment ventilation units:

- RN-473 (LCVU A Full Flow Valve)
- RN455 (LCVU B Full Flow Valve)
- RN447 (LCVU C Full Flow Valve)
- RN481 (LCVU D Full Flow Valve)

IMMEDIATE ACTIONS: Ensure automatic action occurs.

SUPPLEMENTARY ACTIONS: Place additional lower containment ventilation units in operation per OP/1(2)/A/6450/001

Monitor containment pressure and adjust fan speeds as necessary

Refer to TS 3.6.4 (Containment Pressure)

Protection Set III is supplied from 120 VAC Panelboard ERPC. This set powers the following:

- Process Control Channel III (PCS Cabinet 3)
- NIS Channel III
- SSPS Channel III Input Bay (Train A & B)
- NC Pump Monitor Panel Channel III (This channel monitors NCP B.)
- Turbine Trip Relays Channel III
- 48 VDC and 15 VDC Power Supplies (SSPS - Train B)

Protection Set IV is supplied from 120 VAC Panelboard ERPD. This set powers the following:

- Process Control Channel IV (PCS Cabinet 4)
- NIS Channel IV
- SSPS Channel IV Input Bay (Train A & B)
- NC Pump Monitor Panel Channel IV (This channel monitors NCP D.)
- Turbine Trip Relays Channel IV
- 48 VDC and 15 VDC Power Supplies (SSPS - Train B)
- SSPS Train B Output Cabinet

NOTE: In the event of an Extended Loss of ALL AC Power (ELAP), ECA-0.0 (Loss Of ALL AC Power) load sheds vital batteries, resulting in down-powering SSPS Panel, affecting a control logic scheme for the CLAs, (P-11 Permissive). IAE must place a jumper on contacts to allow closing and maintaining CLA Isolation valves closed

3.2 Reactor Trip and Bypass Breaker Power Supply

Train A breaker control power is supplied by 125 VDC Panelboard EPA, and Train B breaker control power is supplied by 125 VDC Panelboard EPD.

3.3 Impact of Loss of Power on Reactor Protection

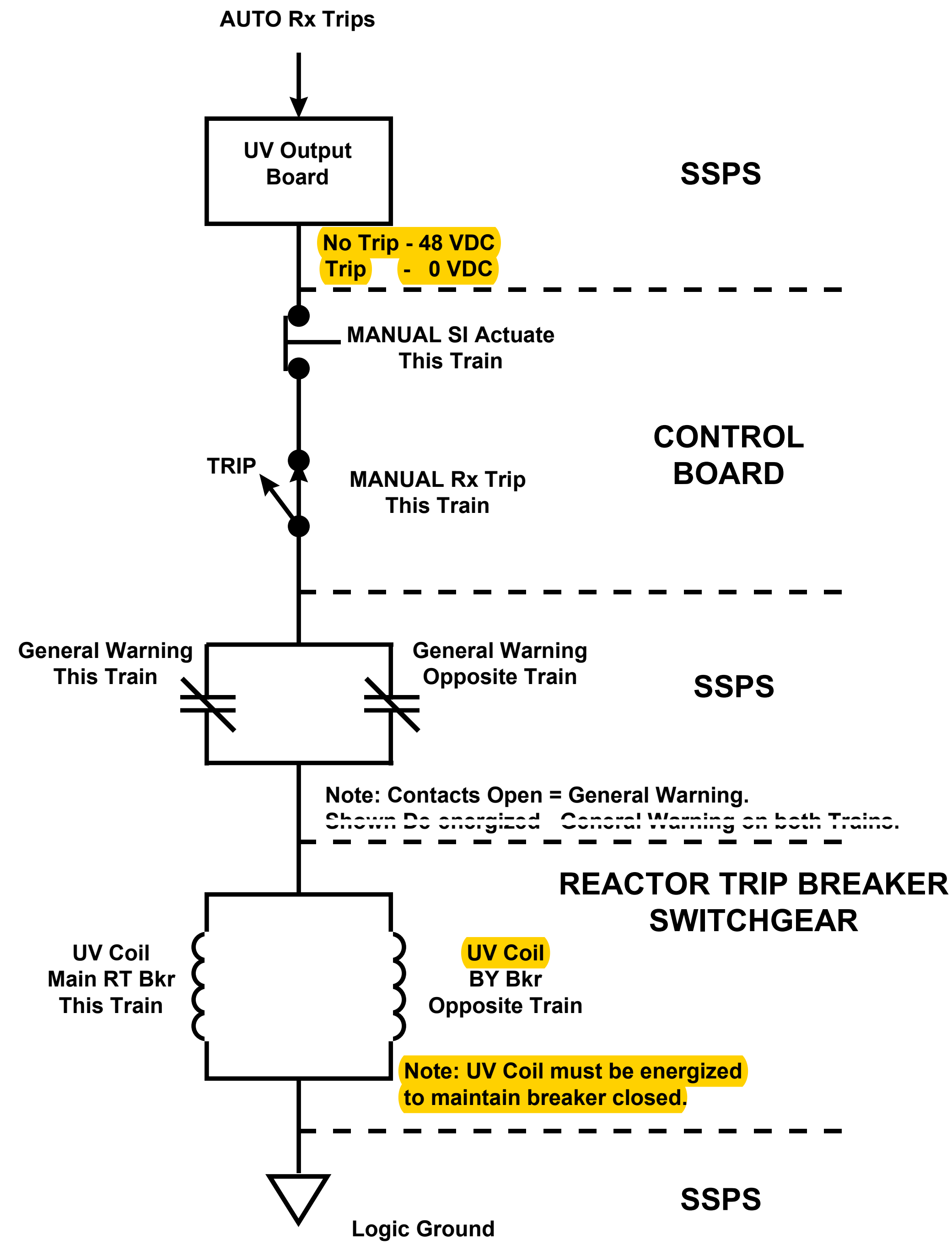
Objective 6, Licensed

Upon loss of a Vital 120 VAC Panelboards, all bistables on the affected channel will de-energize giving trip signal inputs while the control board bistable lights for the affected channel will be lit.

The trip signal for these bistables will change the trip logic. With one trip signal present, the 2/4 logic now becomes 1/3, and the 2/3 logic now becomes 1/2. The power loss should not cause a reactor trip unless another channel's bistable was in TRIP prior to the power loss.

Question 56

10.6 Figure 6: Reactor Trip Breaker Undervoltage Circuit - One Train



2. COMPONENT/SYSTEM DESCRIPTION

2.1 Pzr Level Channels

Objective 2A

Hot Calibrated Channels

Three Pzr Level Channels, 1, 2, and 3, are the “Hot” channels calibrated for the normal Pzr temperature of 653ºF. Their indicated level will be greater than actual level when Pzr temperature is less than 653ºF. See Attachment 4 for the process for determining actual level values. **Figures 1 and 2**

These Pzr level channels are used during normal operation for protection, control, and indication. Selected Pzr Levels are developed by median selector circuits in the Distributed Control System (DCS).

Two Selected Pzr level signals are developed to provide redundant control signals, designated Selected Pzr Level -1 and Selected Pzr Level -2. Looking at Attachment 4 Graphs you will see how the indicated level changes with Pzr water temperature changes. **Figures 3 and 4**

Cold Calibrated Channel

The “Cold Cal” channel is calibrated at 100ºF and 14.7 psia (~atmospheric pressure). It’s indicated level will be less than actual level when Pzr temperature is greater than 100ºF. See Attachment 4 for the process for determining values.

The cold cal channel is used during plant startup, shutdown, and refueling for indication and is considered a non-safety related indication.

Indication

The meter indication for all Pzr levels is 0 – 100% and is located in the Control Room (CR). The three Pzr Level Channels (Hot Cal) are on MC10 with the Pzr Reference Level indication next to them. The Pzr Cold Level indication is on MC5.

The DCS graphics page NC – Pressurizer and PRT has both selected Pzr levels and the setpoint (SP) or reference level. The DCS graphics page NV – Charging Flow Control has the Pzr Level Master with SP, PV (process variable) and O (output).

Separate transmitters feed both Auxiliary Shutdown Panels (ASP) and the Standby Shutdown Facility (SSF) indication. (3 transmitters)

On MC1 there is a 2-pen recorder providing indication of Selected Pzr Level-1 and Pzr Reference Level. These are normally used to trend Pzr level.

120 VAC Vital I&C Panelboards **ERPA** (N41), **ERPB** (N42), **ERPC** (N43), **ERPD** (N44), supply power to the Power Range detectors and circuitry.

For the PR detectors, instrument power supplies the meters, circuit processing components, high voltage and detector power. Instrument Power Fuses provide overcurrent protection for power supply circuits.

Control power supplies the lights on the drawer and 120 VAC to the bistable relay drivers to the plant relays. Control power fuses provide overcurrent protection for the control signal circuit transformers.

A trip signal will be generated when either instrument or control power fuses are removed. The instrument power fuse trip can be prevented by blocking or bypassing; the control fuse trip is prevented by blocking only.

Loss of ERPA, B, C or D

At 100% power, if <u>only</u> one power supply fails (ERPA, B, C or D), all of the bistables for the affected channels will go to the tripped condition. Since most of the actions that occur at 100% power require 2/4 channels, no automatic action should occur with the exception of OVERPOWER ROD STOP which requires only 1/4 channels.

If more than one power supply is lost to the Power Range detectors, the 2/4 Power Range logic for reactor trips will be met and the reactor will trip.

PR Instrument Power Failure

For a loss of instrument power only (Instrument Power Fuse failure), all light indications on the PR ‘A’ drawer will be present indicating that all of the bistables are in the tripped condition.

PR Control Power Failure

The bistables for the affected nuclear instruments will be in the tripped condition. The PR drawer bistable lights will NOT be lit.

Distractor Part 2

At 100% power, if two or more channels of power are affected, a reactor trip will occur.

Below P-10, if an Intermediate Range is affected by a loss of control power, the reactor will trip regardless of the affected channels bypass switch position.

Below P-6, if a Source Range is affected by a loss of control power, the reactor will trip regardless of the affected channels bypass switch position.

If the reactor trip from Source Range has been blocked on MC1 with the unit above P-6, then a Source Range control power failure will NOT result in a reactor trip. If unit goes below P-6, or the affected Source Range is manually unblocked below P-10, then the reactor will trip if the control power failure has not been corrected first.

If the reactor trip from Intermediate Range has been blocked on MC1 with the unit above P-10, then an Intermediate Range control power failure will NOT result in a reactor trip. If unit

Question 59

16.6 ENGINEERED SAFETY FEATURES

16.6-3 Inlet Door Position Monitoring System

COMMITMENT The Inlet Door Position Monitoring System shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, 3, and 4.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Inlet Door Position Monitoring System non-functional.	A.1.1 Verify the Ice Bed Temperature Monitoring System is FUNCTIONAL.	Immediately
	AND	
	A.1.2 Verify ice bed temperature is $\leq$ 27°F.	Once per 4 hours
	AND	
	A.1.3 Restore the Inlet Door Position Monitoring System to FUNCTIONAL status.	14 days
	OR	
	A.2 Restore the Inlet Door Position Monitoring System to FUNCTIONAL status.	48 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 4.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

The sump solution will have considerable depth, which inhibits the ready diffusion of hydrogen from solution. This retention of hydrogen in solution will have a significant effect in reducing the hydrogen yields to the containment atmosphere. The build-up of hydrogen concentration in solution will enhance the back reaction to formation of water and lower the net hydrogen yield.

Zirc - Water Reaction - The complete reaction of zirconium with steam can be characterized by the following:

$Zr + 2\ H_20 \rightarrow\ Zr0_2 + 2H_2 + Heat$

For every mole of zirconium reacted, one mole of zirconium dioxide and two moles of hydrogen are produced. Moreover, for each mole of zirconium reacted, two moles of steam are required. The reaction is exothermic, i.e., net energy (heat) is released.

Metal Water - Steel Steam Reaction

The presence of large amounts of stainless steel in a reactor vessel makes further consideration of the oxidation of steel by high temperature steam advisable. The mechanism of oxidation of steel is highly complex and several oxide forms are possible. As in the case of zirconium the reaction results in the generation of hydrogen and heat. The rate of oxidation of stainless steel is low at temperatures below approximately 1000°C but it becomes larger than that of zirconium at temperatures approaching the melting point of steel ~2550°F. During core heat-up and Zr clad reaction, high temperature (≈ 3000°F) steam and H_2 exit the core and heat-up thinner walled stainless steel components. Also heat radiation from the core contributes. During core heatup and zirc clad reaction, 150 pounds of hydrogen will be produced from metal-water reaction.

The DBA LOCA hydrogen analysis calculation (CNC-1552.08-00-0194, Revision 3) provides the technical details that support the Catawba UFSAR Section 6.2.5 (Combustible Gas Control in Containment). In this calculation it is documented that there are four potential sources of hydrogen during an accident -- (1) Zirc-Water reaction, (2) Dissolved primary system hydrogen, (3) Radiolysis - Core and Sump, and (4) Corrosion of metals - Zinc and Aluminum. Of these potential sources, dissolved primary system hydrogen is by far the least amount contributing to the overall hydrogen sources. In addition, from this calculation the **zirc-water reaction is assumed to contribute a maximum of 21,000 scf of hydrogen** throughout the accident based on the 10CFR50.46 criteria for evaluation of the ECCS, which requires zirc-water reaction be limited to 1% by weight of the total quantity of zirconium in the core.

HYDROGEN SOURCES

- Radiolysis of water
- Core
- Sump
- Zirc-Water reaction (highest production rate)
- Metal-Water (Stainless Steel/Aluminum)
- Paint
- Galvanized metal
- Concrete (decomposition if large core damage)
- Dissolved in coolant **Distractor Part 1**

HYDROGEN CONTROL

- Sampling - Hydrogen Analyzers
  - 2 trains
  - Read out in Control Room and Post Accident Sample Panel
- Hydrogen glow plugs
  - 72 plugs
  - 2 trains
- Hydrogen Recombiners
  - Maintain hydrogen concentrations <4% **Distractor Part 2**
  - Used in conjunction with skimmer fans
- Hydrogen purge
  - Used in conjunction with the VE system
  - Used only after consulting with the TSC

Question 60

Containment Hydrogen Control Systems

1. Purpose

To define the procedure for operation of the following Containment Hydrogen Control Systems:

- Hydrogen Skimmer System
- Containment Air Return System
- Containment Hydrogen Purge System
- Emergency Hydrogen Mitigation System (Glow Plugs)
- Containment Hydrogen Analyzers
- Hydrogen Recombiners

2. Limits and Precautions

2.1 Hydrogen concentrations greater than 3.5% are combustible.

2.2 The Containment Hydrogen Purge System shall **NOT** be in operation when H_2 concentration is below 3.0%, to prevent overloading the annulus ventilation filters.

2.3 Inadvertent operation of the Containment Air return fans may open the Ice Condenser inlet doors and cause containment pressure to fall below allowable limits.

2.4 After manual operation, maintenance, or packing adjustment of any safety related valve, it shall be cycled electrically to ensure reliable automatic operation.

2.5 When manually operating any motor operated valve, minimize the torque applied to the handwheel.

2.6 The maximum electric hydrogen recombiners heater temperature is 1400°F.

2.7 The Hydrogen Analyzer Control Unit shall **NOT** be in the "ON" position when either of following conditions exists to prevent suction pump damage:

- "HYDROGEN ANALYZER ISOLATION CONT VALVES" are in the "CLOSE" position.
- "POS 1 H2 ANALYZER POS 2 POST ACCIDENT SAMPLE PANEL" switch is in "POS 2".

2.8 Hydrogen Recombiners and Hydrogen Ignitors are **NOT** operated with hydrogen concentration ≥ 6% without TSC approval. **Correct Answer**

2.9 If VX Test Switches are used, (1ELMC0020, 1ELMC0021) the switches shall be rotated through a complete rotation and back to "OFF" to return relay alignment to normal.

Question: 70
(1 point)

Given the following Unit 1 timeline:

1000
- The Unit has experienced a runback, from 100% power, following a trip of 1A CFPT

1003
- Main Turbine target load has been reached
- 1AD-2 A/9 (Control Rod Bank Lo Limit) illuminates

1005
- 1AD-2 B/9 (Control Rod Bank Lo-Lo Limit) illuminates
- Steam Dumps have closed
- Temperature Error meter indicates (+) 1.8° F

Entry into the Action Statement of Tech Spec 3.1.6 (Control Bank Insertion Limits) is FIRST required at _____(1)_____ .

Per the conditions provided at **1005,** OMP 1-7 _____(2)_____ state that control rods should be placed in MANUAL.

Which ONE of the following completes the statements above?

A. 1. 1003
2. does

B. 1. 1003
2. does NOT

C. 1. 1005
2. does

D. 1. 1005
2. does NOT

Question 61

7.7.1.3.3 Control Bank Rod Insertion Monitoring

When the reactor is critical, the normal indication of reactivity status in the core is the position of the control bank in relation to reactor power (as indicated by the Reactor Coolant System loop ΔT) and coolant average temperature. These parameters are used to calculate insertion limits for the control banks. Two alarms are provided for each control bank.

Distractor Part 1

1. The "low" alarm alerts the operator of an approach to the rod insertion limits requiring boron addition by following normal procedures with the Chemical and Volume Control System.
2. The "low-low" alarm alerts the operator to take immediate action to stop any dilution in progress. Shutdown margin is subsequently verified above the required minimum or boron is added, and the control bank(s) is restored above its insertion limit setpoint.

The purpose of the control bank rod insertion monitor is to give warning to the operator of excessive rod insertion. The insertion limit maintains sufficient core reactivity shutdown margin following reactor trip and provides a limit on the maximum inserted rod worth in the unlikely event of the hypothetical rod ejection, and limits rod insertion such that acceptable nuclear peaking factors are maintained. Since the amount of shutdown reactivity required for the design shutdown margin following a reactor trip increases with increasing power, the allowable rod insertion limits must be decreased (the rods must be withdrawn further) with increasing power. Two parameters which are proportional to power are used as inputs to the insertion monitor. These are the ΔT between the hot leg and the cold leg, which is a direct function of reactor power, and T_{avg}, which is programmed as a function of power. The rod insertion monitor uses parameters for each control rod bank as follows:

Correct Answer Part 1

$$Z_{LL} = A(\Delta T)_{auct} + B(T_{avg})_{auct} + C$$

where:

Z_{LL}	=	Maximum permissible insertion limit for affected control bank
$(\Delta T)_{auct}$	2nd Highest ΔT of all loops	
$(T_{avg})_{auct}$	2nd Highest T_{avg} of all loops	
A,B,C	= Constants chosen to maintain $Z_{LL} \geq$ actual limit based on physics calculations	

The control rod bank demand position (Z) is compared to Z_{LL} as follows:

$$\text{If } Z - Z_{LL} \leq D \text{ a low alarm is actuated}$$

$$\text{If } Z - Z_{LL} \leq E \text{ a low} - \text{low alarm is actuated.}$$

Since the 2nd highest value of T_{avg} and ΔT are chosen by auctioneering, a conservatively high representation of power is used in the insertion limit calculation.

Actuation of the low alarm alerts the operator of an approach to a reduced shutdown reactivity situation. Administrative procedures require the operator to add boron through the Chemical and Volume Control System. Actuation of the low-low alarm requires the operator to stop any dilution in progress. Shutdown margin is subsequently verified above the required minimum or boron is added, and the control bank(s) is restored above its insertion limit setpoint. The value for "E" is chosen such that the low-low alarm would normally be actuated before the insertion limit is exceeded. The value for "D" is chosen to allow the operator to follow normal boration procedures. Figure 7-19 shows a block diagram representation of the control rod bank insertion monitor. The monitor is shown in more detail on the functional diagrams shown in Figure 7-2,

Attachment 11.1
General Statements of Philosophy

13. Manual Initiation of Safeguards Actions

ROs and SROs are expected to manually initiate safeguards actions if an automatic action setpoint is being approached, to avoid challenging the automatic safeguards function. An example of this is to manually initiate safety injection if pressure is decreasing in an uncontrolled manner to 1845 psig.

Exceptions to this philosophy are listed below:

- Do not initiate Phase B earlier than required. Procedural guidance is provided to initiate VX if containment pressure has not exceeded 3 psig and greater than 1 psig.

- During an ATWS, do not initiate S/I in "anticipation" of an S/I signal if the reactor will not trip, since this will cause a loss of CF flow to the S/Gs. This exception is stated in the APs that manually initiate S/I in "anticipation" of an S/I signal. (e.g. AP/1/A/5500/010 Enclosure 1)

The operator is expected to manually initiate any action which should have automatically occurred if the automatic function fails, such as the Safety Injection fails to initiate during an uncontrolled Reactor Coolant depressurization at 1845 psig (even during an ATWS) or an ECCS pump fails to start on a Safety Injection signal.

14. Placing Control Rods in manual following a load rejection

Following a load rejection/turbine runback, the control room crew should place control rods in manual once the steam dumps have closed. This will stop auto rod insertion with Tavg ~ 3°F higher than Tref. With the negative reactivity from the Xenon transient still causing Tavg to decrease, this should allow the crew the time to have a focus brief on borating the NC system and restoring rods above insertion limits (if necessary) without Tavg decreasing excessively below Tref.

Question 61

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. <u>Operator Actions</u>	
__ 1. **Verify turbine load - TRENDING DOWN IN AUTOMATIC.**	**Perform the following:** __ a. Select "MANUAL" on turbine control panel. __ b. Depress "CONTROL VALVES LOWER" pushbutton and reduce turbine load as required.
2. **Verify proper reactor response:** __ • Control rods - IN "AUTO" AND STEPPING IN __ • P/R neutron flux - TRENDING DOWN. Distractor Part 2	**<u>IF</u> <u>AT</u> <u>ANY</u> <u>TIME</u> T-Avg greater than 1.5°F higher than T-Ref, <u>THEN</u> perform the following:** __ a. Insert control rods as required to maintain T-Avg within 1°F of T-Ref. b. **<u>IF</u>** control rods will **<u>NOT</u>** insert, **<u>THEN</u>** perform the following: __ 1) Trip Reactor. __ 2) **<u>GO</u> <u>TO</u>** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Question 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- Reducing turbine generator load to stabilize vacuum is only effective when low vacuum is due to reduced RC cooling.

Correct Answer Part 1

- CA System is inoperable when available Condensate Storage System water temperature is greater than 136°F.

4. **Lower turbine load as required to stabilize vacuum as follows:**

Distractor Part 1

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ a. Verify vacuum loss due to reduced RC cooling.	a. Perform the following: __ 1) **IF AT ANY TIME** vacuum loss due to reduced RC cooling, **THEN RETURN TO** Step 4. __ 2) **GO TO** Step 5.

b. **IF** rapid power reduction required, **THEN** perform the following:

NOTE
- Any load reduction rate of greater than 25 MW/Min must be performed in manual mode.
- Unloading rates greater than 60 MW/Min (5%/minute) will meet C-7A interlock and may result in steam dump actuation.
- In manual mode, the control valves are capable of full travel within 3 minutes.

__ 1) Select "MANUAL" and "CONTROL VALVE LOWER" to reduce turbine load as required.

__ 2) **REFER TO** AP/1/A/5500/009 (Rapid Downpower).

__ 3) **GO TO** Step 5.

Question 62

NOTE If trend down in vacuum is rapid, using the reactor trip guidance for loss of CF pumps may be more appropriate.

1. **Reactor Trip Criteria:**

- **IF** reactor power greater than or equal to 69% **AND** main condenser vacuum trending down to 22 in. Hg in any condenser section imminent, **THEN** perform the following:

  a. Trip reactor. Correct Answer Part 2

  b. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

- **IF** in Mode 1 or 2 **AND** vacuum trending down to 16.9 in. Hg in both CF pump condensers imminent, **THEN** perform the following:

  a. Trip reactor.

  b. **WHEN** reactor trip verified, **THEN** trip CF pumps.

  c. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

2. **Turbine Trip Criteria:**

- **IF** reactor power less than 69% **AND** main condenser vacuum trending down to 22 in. Hg in any condenser section imminent, **THEN** perform the following:

  a. Trip turbine.

  b. **REFER TO** AP/1/A/5500/002 (Turbine Generator Trip).

- **IF** turbine impulse pressure less than or equal to 370 psig **AND** exhaust hood temperature trending up to 225°F imminent, **THEN** perform the following:

  a. Trip turbine.

  b. **REFER TO** AP/1/A/5500/002 (Turbine Generator Trip).

- **IF** turbine load less than or equal to 360 MWs **AND** main condenser vacuum trending down to 24.3 in. Hg in any condenser section imminent, **THEN** perform the following:

  a. Trip turbine. Distractor Part 2

  b. **REFER TO** AP/1/A/5500/002 (Turbine Generator Trip).

Question 63

A. Purpose

- **To verify proper response in the event of high activity in the NC System.**

B. Symptoms

- **Primary sample results indicate increasing activity level**
- **NC filter area monitors alarm - LIT:**
  - 1RAD-3, D/3 "1EMF 18 568 KK - 56 NC FILTER 1A"
  - 1RAD-3, D/4 "1EMF 19 568 KK,LL - 56 NC FILTER 1B".
- **1RAD-3, E/4 "1EMF 48 NC SAMPLE LINE REACTOR COOLANT" - LIT**
- **Dose Equivalent Iodine (DEI) has reached Action Level 3 limit as defined in AD-NF-ALL-0806, Nuclear Fuel Integrity.**

Part 2 Correct Answer

Distractor B.2 & D.2

5.2. 1EMF12 - Control Room

1EMF12 - Control Room	
Area Monitor - Measures the dose rates within the control room.	
Tech Spec or SLC	None
Power Supply	1RPA
Auto Actions	None
Notes	• Per the annunciator response immediate actions, all non-essential personnel should be evacuated because only 6 respirators are available for the control room personnel. • A valid reading of 15 mR/hr places the site in an Alert per AD-EP-ALL-0101.

5.3. 1EMF15 (2EMF4) - Spent Fuel Building Refueling Bridge **Distractor A.1 & B.1**

1EMF15 (2EMF4) - Spent Fuel Building Refueling Bridge	
Area Monitor **-** Measures activity over the fuel pool (for inadvertent criticality or fuel damage).	
Tech Spec or SLC	**SLC 16.7-10 (Radiation Monitoring for Plant Operations)** **Monitor 3: 1EMF-15, 2EMF-4) Fuel Storage Pool Area** **EMF Requirement:** 1 per unit with fuel in the fuel storage pool areas
Power Supply	1(2) RPA
Auto Actions	Automatically stops new fuel elevator from being raised
Notes	• Symptom for entry into AP/1(2)/A/5500/25 (Damaged Spent Fuel) • An unplanned lowering water level with corresponding rise in radiation levels places the site in an Unusual Event. • An unplanned valid Trip 2 alarm with damge to irradiated fuel resulting in a release of radioactivity places the site in an Alert

5.5. 1EMF18 and 19 (2EMF5 and 6) - Reactor Coolant Filter A and B

1EMF18 (2EMF5) - Reactor Coolant Filter A **1EMF19 (2EMF6) - Reactor Coolant Filter B**	Correct Answer
Area monitor - Measures dose rate next to the 1(2) A(B) reactor coolant filter housing which would result from activated particulate in the NCS or from particulate released from the NV demineralizers.	
Tech Spec or SLC	None
Power Supply	1(2) RPA
Auto Actions	None
Notes	• Symptom for entry into AP/1(2)/A/5500/18 (High Activity in Reactor Coolant) • Alarm setpoint set to ensure filter is removed from service and replaced before radiation level exceeds the shielding capacity of the transfer cask.

6.17. 1(2) EMF48- Reactor Coolant **Distractor A.1 & B.1**

1(2) EMF48- Reactor Coolant	
Process Monitor -Measures activity in the NC sample flow in the NM system. This EMF is NOT a process monitor depicted in Illustration 2 but is a GM detector mounted adjacent to the sample line. High activity would indicate a fuel cladding failure or crud burst**.**	
Tech Spec or SLC	**TS 3.3.3 (Post Accident Monitoring Instrumentation)** **EMF-48, NC System Monitor** **EMF Requirement:** 1 required channel in Modes 1, 2, and 3
Power Supply	1RPA
Auto Actions	None
Notes	• Dose Equivalent Iodine > 300µCi/cc is a Loss of the Fuel Clad Barrier per AD-EP-ALL-0101 Fission Product Barrier Matrix. During an event where fuel damage is possible, if 1(2) EMF-48 alarms, then the SRO should ensure Chemistry analysis is performed so the fission product barrier may be evaluated in a timely manner. • Symptom for entry into AP/1(2)/A/5500/018 (High Activity in Reactor Coolant)

Question 64

Given the following:

- Units 1 & 2 are at 100% RTP
- 1A RN pump is in service
- 1A and 2A KC trains are in service

Subsequently:

- The following Unit 1 annunciators are lit
  - 1AD-12 B/2 “RN PIT A Screen Hi D/P”
  - 1AD-12 B/1 “RN Pump Intake Pit A Level – LO”
  - 1AD-12 E/2 “RN Pit A Swap to SNSWP”
- The crew has entered AP/0/A/5500/020 (Loss of Nuclear Service Water), Case II (Loss of RN Pit Level)

Enclosure 2 (RN Valve Alignment for RN Swap to SNSWP) will direct the BOP to ensure _____(1)_____ is closed .

Following system stabilization, the BOP is directed to “Ensure KC Hx Outlet Mode Switches – Properly Aligned”. In response, the BOP _____(2)_____ required to reposition 1RN-291 (KC Hx 1A Outlet Throttle Valve) from its original alignment.

A. 1. 1RN-47A (RN Supply X-Over Isol)
   2. is

B. 1. 1RN-47A (RN Supply X-Over Isol)
   2. is NOT

C. 1. 1RN-48B (RN Supply X-Over Isol)
   2. is

D. 1. 1RN-48B (RN Supply X-Over Isol)
   2. is NOT

Original Question

MODIFIED

Question 64

2. **Ensure the following valves - CLOSED:**

__ • 1RN-1A (RN P/H Pit A Isol From Lake)

__ • 1RN-2B (RN P/H Pit A Isol From Lake)

__ • 1RN-5A (RN P/H Pit B Isol From Lake)

__ • 1RN-6B (RN P/H Pit B Isol From Lake)

__ • 1RN-53B (Station RN Disch Hdr X-Over)

__ • 1RN-54A (Station RN Disch Hdr X-Over)

__ • 1RN-57A (Station RN Disch To RL Sys)

__ • 1RN-843B (Station RN Disch To RL Sys)

__ • 1RN-49A (Non-Ess Supply Hdr Isol)

__ • 1RN-50B (Non-Ess Supply Hdr Isol)

__ • 2RN-49A (Non-Ess Supply Hdr Isol)

__ • 2RN-50B (Non-Ess Supply Hdr Isol)

__ • 1RN-847A (D/G 1A Hx Ret To Lake)

__ • 1RN-849B (D/G 1B Hx Ret To Lake)

__ • 2RN-847A (D/G 2A Hx Ret To Lake)

__ • 2RN-849B (D/G 2B Hx Ret To Lake).

__ 3. **IF RN supply header in single supply header operation, THEN return to procedure and step in effect.**

4. **IF RN Pit A initiated auto or manual swap to SNSWP, THEN ensure the following valves closed:**

__ • 1RN-48B (RN Supply X-Over Isol)

__ • 2RN-48B (RN Supply X-Over Isol).

15.2 Emergency Low Pit Level

Objective 16B, All

The Emergency Low Pit Level provides automatic transfer of the suction source for RN from Lake Wylie to the SNSWP. Upon receipt of an emergency low level actuation, the isolation valves from the SNSWP and Lake Wylie open and close, respectively.

The emergency low level setpoint only applies to the safety related instruments (three channels in each pit):

- 1/3 channels in either pit actuates a computer alarm.
- 2/3 channels in either pit actuates an annunciator alarm, actuates a computer alarm, and separates the RN loops (except through1RN-36A and 1RN-37B), isolates RN from Lake Wylie, and aligns RN to the SNSWP.

Actions interlocked to either pit:

- Start all 4 RN pumps (1A, 1B, 2A, and 2B). **Correct Answer Part 2**
- Opens
  - RN Pumphouse suction valves from SNSWP.
  - RN Return Header valves to the SNSWP
  - DG HX Returns to the SNSWP (Both units)
- Closes
  - RN Pumphouse Pit Isolations from the Lake
  - RN Non-essential Supply Isolations (both units)
  - Station RN Discharge Header Cross-Overs
  - Station RN Discharge to RL System
  - D/G HX Returns to RL System (Both units).

Actions interlocked to Pit A instruments only:

- Closes the opposite train RN Supply Cross-Over Isolations (Both units)

Actions interlocked to pit B instruments only: **Correct Answer Part 1**

- Closes the opposite train RN Supply Cross-Over Isolations (Both units)

L. Manual Preaction Systems (Turbine Bearing Deluge Systems)

1. The Main Turbine Bearings are protected by a Manual Preaction sprinkler system consisting of the following:
   a) Isolation (control) valve
   b) Mulsifyre clapper (deluge) valve
   c) Supervisory air supply (VS)
   d) Dry piping distribution system
   e) Closed sprinkler heads

Correct Answer Part 2

2. The purpose of this type of system is to deliver large quantities of Fire Protection (RF) water to totally engulf equipment while reducing the potential of equipment damage due to inadvertent discharge of water. (**Obj. #20**)
3. The Isolation and Mulsifyre Deluge valves are the same type used in the Mulsifyre (automatic water spray) systems discussed earlier; however, the actuation of the deluge valve is accomplished manually and not by the heat detectors. Also, the sprinkler heads are not open, they are the same as those used in the Automatic Wet Pipe Sprinkler Systems.
4. The thermal detectors associated with the Main Turbine Bearings fire protection system only provide an alarm to the Control Room. Upon receipt of the alarm it can be decided if the deluge valve should be opened. Opening the deluge valve can be accomplished from the remote pull stations (Control Room or next to MTOT), or from the local pull station at the valve. Opening the deluge valve allows water to enter the distribution piping, however water will not be discharged unless sufficient heat is present at the sprinkler heads to cause the heat sensitive elements to have been destroyed. (**Obj. #21, 22**)

Correct Answer Part 1

5. The distribution piping is normally pressurized by the Station Air (VS) System. Pressure is monitored, and an alarm is provided when pressure is low indicative of a fused sprinkler head or leak in the piping. Again, upon receipt of the alarm, it can be decided if the deluge valve should be opened (as described above). (**Obj. #23**)

M. Automatic Pre-action System (Waste Solidification Building)

1. The Waste Solidification Building is protected by an Automatic Preaction sprinkler system consisting of the following:
   a) Isolation (control) valve
   b) Mulsifyre clapper (deluge) valve
   c) Supervisory air supply (VS)
   d) Dry piping distribution system
   e) Closed sprinkler heads
2. In order to reduce the potential of freezing the water in the Waste Solidification Building fire protection system piping, an Automatic Preaction system is utilized.

Question 66

5.6 Standing Instructions (continued)

2. In a timely manner, all Operations personnel review and initial, acknowledging they understand the issued Standing Instruction, unless exempted (see below).

   a. Operations personnel **NOT** on shift will review new Standing Instructions within seven days from issuance or within seven following the return to work (e.g., vacation, extended time off).

   b. All shift Operations personnel reviews issued Standing Instructions before taking the watch. This may occur shortly after when allowed by the CRS or Shift Manager.

   c. If the Standing Instruction only impacts certain individuals in Operations, the SM can allow an exemption for that Standing Instruction review (e.g., a Standing Instruction that discusses EAL classification may not be applicable to AO's and the SM can waive their requirement to review).

   d. The required population or population exempted to review a Standing Instruction shall be documented in Attachment 1, Standing Instructions.

   e. A template for Standing Instruction review is provided in Attachment 16, Standing Instruction / OSIP Review Sheet

3. If any of the following are issued to provide additional monitoring criteria for plant operation, then communicate the guidance to the shifts through Standing Instructions:

   - External department guidance documents that will be in place for greater than 72 hours

   - Operational Decision Making (ODM) Evaluations that do not have an associated Adverse Condition Monitoring Plan (ACMP)

   - NCR immediate or interim corrective actions that warrant communication to the shift

4. Standing Instructions shall **NOT** be used to alter, change, or compensate for lack of appropriate procedures.

5. Standing Instructions shall **NOT** conflict with or be used as a replacement for Operating Procedures or other existing procedures.

Question 66

5.6 Standing Instructions (continued)

6. Standing Instructions shall **NOT** conflict with or be used in place of an incorrect or inaccurate procedure.

   a. If an Operating Procedure is technically inaccurate, then place the procedure on hold in accordance with AD-DC-ALL-0201, Development and Maintenance of Controlled Procedure Manual Procedures.

   b. If an EOP or AOP has an identified Deviation per AD-OP-ALL-1001, Conduct of Abnormal Operations, then a Standing Instruction can be used to inform the crew until the EOP or AOP is changed.

7. Standing Instructions are written and prepared by any knowledgeable person and are approved and signed by Operations management (SM or higher).

8. Use Attachment 1, Standing Instructions to develop a Standing Instruction.

   a. Attach supporting documentation as needed (e.g., NCRs, External Department Guidance) for clarification.

   b. Enter actions required by the shifts (e.g., additional monitoring, controls in manual) in the "Required Actions" section of Attachment 1.

9. Use Attachment 2, Standing Instruction Index to maintain an index of Standing Instructions.

10. Standing Instructions shall be numbered in consecutive order based on the current year (e.g., 17-001, 17-002, 17-003).

11. Attachment 1, Standing Instructions, shall indicate each unit to which the Standing Instruction is applicable.

    a. For a shared item (e.g., common system), a copy of the Standing Instruction shall be placed at each unit that is checked.

12. If a change to the information or required actions is necessary, then revise the Standing Instruction.

    a. Revisions to Standing Instructions can be made by a knowledgeable person.

    b. Indicate updates by adding the next revision number to the Standing Instruction Number (e.g., 07-01 Rev 1).

    c. If a revision to a Standing Instruction is issued, then Operations Management (SM or higher) reviews and signs the revision as if it were a new Standing Instruction.

Question 67

4.4.4 Senior Reactor Operators (continued)

7. Delegates implementation of reactivity plans to the Reactivity Manager when the Reactivity Manager position is staffed, but must maintain broad oversight of the plant.

4.4.5 Reactivity Manager

1. Provides dedicated oversight during Reactivity Manipulations to ensure that the expected reactor response is obtained, applicable error reduction tools are used and Control Room distractions are minimized. {7.1.2} {7.1.6} {7.1.7}

2. Attends pre-job brief and ensures it is conducted prior to manipulation of reactivity controls.

3. Ensures core parameters are maintained within prescribed limits.

4. [BWR] Ensures the reactivity effect of any reactor pressure and feedwater temperature adjustments are understood during control rod and recirculation pump manipulations. {7.1.2} {7.1.6} {7.1.7}

5. Ensures reactivity manipulations are performed by dedicated reactor operators with no concurrent duties in a careful, deliberate manner and expected results are verified by redundant indications. {7.1.2} {7.1.6} {7.1.7}

6. Assumes responsibility for receipt of alarms directly associated with the reactivity manipulation.

7. Oversees manipulation of control rods, Reactor Coolant System pressure, feedwater parameters and [BWR] recirculation flow.

4.4.6 Reactor Operators

1. [BWR] Maintain cognizance of plant conditions that are near the Scram Avoidance Region (OPRM operable) or Region II (OPRM inoperable).

2. Assume no other duties while supervising trainees in the operation of controls that affect reactivity. The trainee and reactor operator are considered one operator. A second licensed operator is required to perform peer checks for reactivity manipulations performed by a trainee/RO combination.

3. Are not involved in any potentially distracting activities during reactivity manipulations involving movement of control rods, [PWR] RCS Makeup System or [BWR] recirculation flow. Complete attention shall be given to proper setup and operation of the reactivity control system and monitoring reactor response. {7.1.1} {7.1.2} {7.1.6} {7.1.7} [7.3.4]

4. Inform the CRS of any unexpected core parameter or reactivity change.

Question 67

5.4.7 Abnormal Operations (continued)

5. Any time reactor power unexpectedly changes from steady state conditions, the OATC is expected to:

   a. Communicate to the CRS the initial alarm followed by reactor power level.

   b. Continuously monitor reactor power throughout the transient.

6. The OATC shall place appropriate control stations in manual if the plant response is not valid as indicated by any of the following:

   - NI power increasing above the pre-transient power level
   - Failed instrumentation is diagnosed
   - Invalid input exists and SRO directs control systems be placed in manual

7. Peer-checks during Abnormal Operations:

   a. Peer-checks are not required for reactivity manipulations during performance of Abnormal and Emergency Procedures unless additional manpower is available.

   b. The CRS is prohibited from performing peer-check of reactivity related manipulations.

   c. Peer-checks are not required when responding to a failure requiring manipulation of Reactivity Management related components.

8. [PWR] During abnormal conditions requiring manual control of the reactor, ROs shall inform the CRS of the following (notification to the CRS is not required prior to taking the action):

   - The first manual control rod insertion
   - All manual control rod withdrawals
   - The first manual adjustment made to the Turbine
   - The first manual adjustment made to Feedwater
   - Any water addition made to the VCT/LDST
   - No further communications of manual adjustments are necessary. When normal operation is resumed, communications return to normal operational requirements for reactivity changes.

Question 68

_____ 3.1.6 Slowly open 1AS-4 (Main Steam to Aux Steam Header Control Bypass) (TB-610, 1M-32).

_____ 3.1.7 Verify Unit 1 SM Header pressure is approximately equal to AS Header Pressure.

NOTE:	If turbine prewarming is **NOT** to begin at this time, continue with the procedure and sign off Steps 3.1.8 and 3.1.9 when ready to begin prewarming the high pressure turbine shell.

_____ 3.1.8 Verify turbine initial conditions contained in the turbine generator startup enclosure of OP/1/B/6300/001 (Turbine-Generator) have been signed off.

Question 68

CAUTION: Do **NOT** exceed 50 psig in the high pressure shell while doing Step 3.64 until PZR pressure is greater than 1945 psig (PZR Low Pressure Rx Trip Setpoint, P-11). This is to prevent P-13 (57 psig impulse pressure) enabling P-7 to unblock the Lo Power Reactor Trips (PZR High Level, PZR Low Pressure, Low NC Flow on 2/4 Loops, NC Pump Undervoltage, NC Pump Underfrequency).

NOTE: Step 3.64 may be done at anytime during plant heatup to 2235 psig/557°F. The purpose of this step is to use any extra steam available to start warming the high pressure shell of the turbine when the NC System heatup is being performed with the MSIV's open. The warming does **NOT** have to be at 60-100 psig as required by OP/1/B/6300/001 (Turbine-Generator). This warming is to reduce the amount of time required before placing the Turbine-Generator on the line.

_____ 3.64 **IF** MSIVs are open **AND** S/G pressure > 100 psig, begin warming of the turbine per OP/1/B/6300/001 (Turbine-Generator). (R.M.)

NOTE: Step 3.65 is designed to supplement the AS header being supplied by AEB(s) with Unit 1 SM to aid in maintaining both Unit 1 and 2 steam seals. This additional steam source will also aid in supplying AS to the Unit 1 CFPT that is in service. (R.M.)

_____ 3.65 **IF** AEB(s) are supplying Unit 1 AS **AND** it is desired to supplement this steam supply with Unit 1 SM during heatup perform following:

_____ 3.65.1 Ensure SM pressure is greater than or equal to 500 psig.

_____ 3.65.2 Verify 1AS-2 (Main Stm To Aux Stm) is closed. (1MC13)

_____ 3.65.3 Open 1AS-1 (SM To AS Inlet). (1MC13) **Distractor Part 1**

_____ 3.65.4 Notify Operator(s) at the AEB(s) that AS on Unit 1 will be supplemented by SM.

_____ 3.65.5 Adjust 1AS-2 (Main Stm To Aux Stm), in manual, as required to match steam pressure being supplied by the AEB(s) (140 to 165 psig on OAC Pt C1A0962 (As Crosstie Pressure)). (R.M.)

NOTE: Step 3.65.6 provides assurance that SM from Unit 1 is providing expected results.

_____ 3.65.6 Adjust 1AS-2 (Main Stm To Aux Stm) to obtain a slight increase in AS Header pressure as seen on OAC Pt C1A0962 (AS Crosstie Pressure)). (R.M.)

Question 68

5.1.2 Unplanned Reactivity Manipulations

1. A reactivity manipulation that does not allow time to establish the required controls for reactivity evolutions as outlined in this procedure (R1, R2, R3).

2. The required controls for planned reactivity evolutions are not applicable during emergency conditions.

3. Operators will place the plant in a stable, known safe condition and reestablish formal reactivity control as soon as practical following the event.

5.2 Reactivity Manipulation Significance

5.2.1 Major Reactivity Manipulation (R1 Activities)

1. A planned activity that significantly affects core power or reactivity requiring the highest level of operator attention and dedication.

2. Refer to Attachment 2, Planned Reactivity Evolution Category Examples, to assist with determination of the evolution category.

3. Approved Reactivity Manipulation Plan per AD-NF-ALL-0201, Reactivity Management Plan Development {7.1.2}

   a. A Reactivity Manipulation Plan is not required for Reactivity Manipulations made for Beginning-Of-Cycle (BOC) Startup and Power Escalation in accordance with site-specific procedures and AD-NF-ALL-0201, Reactivity Manipulation Plan Development.

4. Pre-job Briefing that includes pertinent OE

5. Operations Manager or designee present in the Control Room to provide management oversight (Shift Manager can not be the designee).

6. Shift Manager present during critical steps to provide oversight.

7. Shift Technical Advisor (STA) present in the Control Room, until all R1 activities are complete, the plant is stable, and SM concurrence is obtained.

8. Reactor Engineering support as follows:

   a. Development of Reactivity Manipulation Plan

   b. Control Room presence for power changes greater than 15%

9. Reactivity Manager oversight (Dedicated SRO, other than the CRS or STA, with no concurrent duties) {7.1.2} {7.1.6} {7.1.7}

Question 68

5.2.1 Major Reactivity Manipulation (R1 Activities) (continued)

10. Dedicated RO with no concurrent duties (can be one of the unit reactor operators) {7.1.2} {7.1.6} {7.1.7}

11. All unrelated parallel Control Room activities shall be evaluated for distractions to the Control Room Crew and approved by the Shift Manager and the Reactivity Manager prior to commencement (minimal impact to control room team, no excessive alarms). {7.1.2} {7.1.6} {7.1.7}

12. Specific monitoring requirements identified

13. Signage or barriers use to limit Control Room access {7.1.2}

14. Consider JITT {7.1.6}

5.2.2 Reactivity Manipulation (R2 Activities)

Distractor Part 2

1. A planned activity that involves routine changes to core reactivity requiring dedicated operator attention.

2. Refer to Attachment 2, Planned Reactivity Evolution Category Examples, to assist with determination of the evolution category.

3. Pre-job briefing that includes pertinent OE

4. Dedicated RO with no concurrent duties (can be one of the unit Reactor Operators) {7.1.2} {7.1.6} {7.1.7}

5. Reactivity Manager oversight (Can be CRS with no concurrent tasks) {7.1.2} {7.1.6} {7.1.7}

6. Reactor Engineering supports as follows:

   a. Development of a Reactivity Manipulation Plan (if requested by Operations)

   b. [BWR] Predicts of power changes

   c. Development of Control Rod Movement Sheets (if utilized)

7. Unrelated parallel Control Room activities shall be suspended during actual component manipulations or a dedicated SRO shall be established for the activity (activities may resume during R2 activity monitoring period) {7.1.2} {7.1.6} {7.1.7}

   a. Deviation to the unrelated parallel activity can be made on a case by case basis with the concurrence of the Reactivity Manager/CRS

Question 68

ATTACHMENT 2
Page 1 of 2

<< Planned Reactivity Evolution Category Examples >>

1. **R1 Evolutions:**

   a. Activity significantly affecting core power/reactivity which requires significant operator attention. Examples include but are not limited to:

      (1) Zero Power Physics Testing

      (2) Reactor startups

      (3) Power changes greater than or equal to [PWR] 10%, [BWR] 15%

      (4) Reactor Shutdown

      (5) [BWR] Control Rod Sequence Exchange

2. **R2 Evolutions:**

   a. Activity that affects Core Reactivity or the normal control of it, but is of minor effect. Examples include but are not limited to:

      (1) Removing/Restoring Steam Generator Blowdown

      (2) [PWR] Saturating an unsaturated demineralizer

      (3) Power changes less than [PWR] 10% [BWR] 15%

      (4) [PWR] Control Rod Drive Movement Tests

      (5) [PWR] Routine adjustments in boron concentration

      (6) [BWR] Adjustments to Reactor Recirculating pump including 'Preconditioning Ramp'

      (7) Evolution requiring reactivity adjustments while automatic controllers of reactivity components or systems are in manual

Question 69

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR for four loop operation; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.14 for the WRB-2M CHF correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080 degrees F, decreasing 58 degrees F for every 10,000 MWd/mtU of fuel burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

Question 69

BASES

BACKGROUND (continued)

The high and intermediate head subsystems of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a safety injection actuation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a small break LOCA and there is a high level of probability that the criteria are met following a large break LOCA:

Distractor Part 1

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

Question 70

5.5.7 Critical Parameters (continued)

4. Critical Parameters shall be monitored using multiple indications (if available).

   a. Backup or redundant instrumentation is available to monitor the Critical Parameter. {7.1.10}

5. As time permits, Critical Parameters should be displayed in the Control Room and contain the following information:

   - Parameter (Critical Parameter) being monitored
   - Control band or single parameter value
   - Reporting frequency

6. As time permits, announce Critical Parameter assignment in a Crew Update or brief to ensure that the entire crew understands the Critical Parameter and owner.

7. Critical Parameters under manual control shall receive heightened crew awareness and prioritization.

   a. The CRS shall consider reassigning any additional transient actions to other Operators not assigned to Critical Parameter manual control.

8. If a Critical Parameter has been or is expected to be exceeded, then inform the CRS (a Crew Update is preferred) and take actions as prescribed when the Critical Parameter was established.

5.5.8 Deleting Computer Alarms

1. Ensure deleted OAC computer alarms are logged in the eSOMS narrative log, unless being deleted per an approved procedure that will restore them at the conclusion of the procedure.

   a. Until the alarm is restored, the eSOMS log entry should be an open log entry.

2. All computer alarms are to be initially believed and a thorough review/response shall be performed to verify that indications are valid and proper actions are implemented.

Question 70

5.5.8 Deleting Computer Alarms (continued)

4. A weekly audit of deleted computer alarms will be performed to ensure that the issue is captured in an active Work Order/Work Request with appropriate priority assigned.

5.5.9 Guidelines for Bypassing of Safety Systems

1. Safety systems must be allowed to perform their automatic function when required for transient mitigation.

2. Safety systems must not be bypassed before automatic actuation, except as follows:

   a. Safety systems may be bypassed when directed by operating procedures for normal plant cooldown or when directed by procedures for testing.

   b. Safety systems may be bypassed when directed by Event Procedures for specific transients.

3. Equipment automatically actuated by a safety system must not be repositioned except as follows:

   a. Equipment may be overridden and repositioned when directed by Event Procedures for specific transients.

   b. If both of the following conditions are met and SRO approval is obtained, then equipment may be overridden and repositioned outside of the procedures:

      (1) The Safety System is not required to perform its intended safety function.

      (2) Continued operation of the Safety System could elevate the severity of the transient, damage equipment, or cause unnecessary operator burden.

   c. If a safety system has been bypassed or overridden, the operator assumes the responsibility to reactuate the system if necessary for transient mitigation.

Question 70

Saturday Day Shift Routine Activities for Unit 1

	Perform the following PT(s)/OP(s):
	• **Semi-Daily Surveillance (PT/1/A/4600/002) PIP C-94-0738, CSD 3.0.24 (Scope 2.1)**
	• **Drain System Analyzer Rack Traps (OP/0/A/6500/115)**
	• **Any AMT has been on HFS for >/= 3days Sample and conduct LWR (OP/0/B/6500/060)**
	• **Any AMT full and HFS is available then place AMT on HFS (OP/0/B/6500/061)**
	Perform the following weekly:
	• **Perform applicable schedule review (T-3 if on days after training, T-8 if on nights after training)**
	• **Review and Walk Down Online Clearances for the two weeks of weekday nights after training. Clearances are located in the OPS Online Office Bins.**
	• **Review the Active ODM's with the shift including trigger points and current trends.**
	Perform the Following as required:
	• **On first Saturday of the quarter, perform inspection of OPS Storage Areas, ensure all equipment properly stored.**
	• **Lower Personnel Airlock Leak Rate Test (PT/1/A/4200/001 F), if required to clear an active LCO tracking entry.**
	• **On the third Saturday of each month, perform a control board W/O, W/R Sticker audit. Print ST1613 OStick Audit Report from "Nuclear Reports – MS Reporting Services 2008" on the DAE. Ensure OOS stickers for Unit 0 & 1 match with this report (ie – remove sticker for completed WR/WO, ensure items on report have sticker). OAC points on the report do NOT need to be reviewed as part of this audit. Notify OWPG OPS Onlines of any discrepancies that cannot be resolved.**

Distractor Part 2

Individual items are to be initialed when complete.
Ensure any PT not performed an NCR is generated, LCO Tracking Entry Made, and notify SM (or designee).
Generate NCR (if needed) for item(s) not completed.
Reference SOMP 1-13

Question 71

5.1 Emergency Exposure Determination (continued)

NOTES

- Emergency exposure limits are exclusive of current occupational exposure.
- Only one emergency exposure is allowed per lifetime.
- Dose to the lens of the eye is limited to three times the listed value.
- Dose close to other organs, including skin and body extremities, is limited to ten times the listed value.

Table 1, Emergency Exposure Limits

TEDE Limit (Rem)	Activity
5	All activities during the emergency. Distractor Part 1
10	Protecting valuable property when lower dose is not practicable.
25	Lifesaving or protection of large populations when lower dose is not practical per EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.
Greater Than 25	Lifesaving or protection of large populations, only if individuals receiving exposure is a volunteer, and fully aware of risks involved.

6. If emergency exposure is needed and is estimated to be greater than 5 Rem but less than 25 Rem, then identify individual(s) who will perform the activity.

   a. Complete sections A and B of Attachment 1, Emergency Exposure Authorization.

   b. Brief the individual(s) on the activities to be performed and inform personnel of their assigned Emergency Exposure Limit.

Correct Answer Part 1

Question 72

5.3.5 Dose Monitoring Warning Flags and SRD Alarms (continued)

2. SRD Dose and Dose Rate Alarms:

a. SRD alarms are provided to help limit dose and keep dose ALARA.

b. SRDs are programmed to alarm at a predetermined dose and dose rate during the log-on process.

c. The alarm set-points are specified by the RWP.

d. The dose alarm consists of an audible alarm and a visual alarm.

(1) When the dose set-point is exceeded, then an audible alarm and a red light will flash on the SRD. Correct Answer Part 2

(2) When the worker logs out at the ACS, then the audible alarm and the flashing red light will **NOT** stop until the SRD is reset.

e. The dose rate alarm consists of an audible alarm and a visual alarm.

(1) Expect an audible alarm and a red light to flash on the SRD when the dose rate set-point is exceeded.

(2) The dose rate alarm automatically resets when the dose rate drops below 80% of the dose rate alarm set-point.

f. Other Alarms: Distractor Part 2

(1) The SRD is programmed to alarm when SRD is activated for 16 hours or when RWP specific stay time is exceeded.

(2) The SRD is programmed to alarm on low battery.

g. SRD Dose Alarm Response:

(1) Workers are expected to immediately inform co-workers, exit the RCA, log out at the ACS, and call RP upon receiving an SRD dose alarm. Re-entry is **NOT** permitted until the alarm is cleared by RP.

Question 72

5.3.5 Dose Monitoring Warning Flags and SRD Alarms (continued)

h. SRD Dose Rate Alarm Response:

(1) Work can continue following a travel path dose rate alarm providing the alarm clears prior to arriving at the work location.

Distractor Part 1

(a) RP will discuss this expected alarm during RWP briefings.

(2) Workers are expected to stop work, exit the area and notify RP upon receiving a third anticipated dose rate alarm unless otherwise directed by RP.

(3) Workers are expected to stop work, exit the area, and notify RP immediately for unexpected dose rate alarms (any dose rate alarm that is **NOT** briefed by RP prior to beginning work).

Correct Answer Part 1

i. SRD malfunctions:

(1) Workers are expected to immediately exit the RCA and notify RP about a malfunctioning SRD.

3. Alternate alarm indicators will be provided to those workers who self identify as having trouble hearing audible SRD alarms.

4. SRD dose-rate alarms may be expected by RP due to higher dose rates in the travel path to the work location or a worker being in close proximity to a radiation source.

5. Anticipated dose rate alarms are discussed during the RP brief prior to beginning work.

Question 73

During an emergency event:

The on-site emergency facility that assumes responsibility for communications with offsite agencies including the NRC once it is activated is the _____(1)_____.

The MINIMUM level of emergency classification that always REQUIRES an evacuation of all non-essential personnel from the site is a _____(2)_____.

Which ONE (1) of the following completes the statements above?

A. 1. Technical Support Center (TSC)
   2. Site Area Emergency

B. 1. Technical Support Center (TSC)
   2. General Emergency

C. 1. Operations Support Center (OSC)
   2. Site Area Emergency

D. 1. Operations Support Center (OSC)
   2. General Emergency

Original Question

MODIFIED

- Provide Protective Action Recommendations (PARs)
- Provide Offsite notifications
- Continued assessment of actual or potential consequences both on-site and off-site throughout the evolution of the emergency condition.
- Effective implementation of emergency measures in the environs including protective actions for affected areas, implementation of emergency monitoring teams and facilities to evaluate the environmental consequences of the emergency condition, prompt notification and communications with of-site authorities
- Continued maintenance of an adequate state of emergency preparedness until the emergency situation has been effectively managed and the station is returned to a normal or safe operating condition.

- The following EC responsibilities cannot be delegated:
  - Classification of the Event
  - Emergency Dose Extensions

  The following are the responsibility of the EC, until EOF activation:
  - Offsite Notifications
  - Protective Action Recommendations (PARs)

4. Emergency Classification

Guidance provided in AD-EP-ALL-0101 (Emergency Classification) and CSD-EP-CNS-0101-02 (EAL Wallcharts)

Objective #1 & 4, ALL

4.1 **Emergency Classification Levels (ECL):**

- Unusual Event (UE): (Least Severe) Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.
- Alert: Events are in progress, or have occurred, which involve an actual or potential substantial degradation of the level of safety of the plant OR a security event that involves probable life-threatening risk to site personnel or damage to site equipment because of hostile action. Any releases are expected to be small fractions of the EPA Protective Action Guideline exposure levels.
  - Requires activation of the OSC, TSC, EOF
- Site Area Emergency (SAE): Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the SITE BOUNDARY.

- Requires activation of the OSC, TSC, and EOF
- Requires a Site Assembly for personnel accountability
- Requires a Site Evacuation **Correct Answer Part 2**

- General Emergency (GE): (Most Severe) Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile actions that result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.
  - Requires activation of the OSC, TSC, and EOF
  - Requires a Site Assembly for personnel accountability
  - Requires a Site Evacuation **Distractor Part 2**

4.2 Event Classification

Objective #2, LPSO/LOR

The time between when indications exist that an EAL threshold has been exceeded and declaration of the event will not exceed 15 minutes, unless extraordinary conditions prevail. AD-EP-ALL-0101 (Emergency Classification) will direct classification using the EAL Wallchart.

The EAL classification information is formatted as follows:

The first letter indicates the EAL category:

- **R** – Abnormal Radiation Levels or Radiological Effluents
- **H** – Hazards
- **E** – ISFSI (Dry Cask Fuel Storage)
- **S** – System Malfunctions (Hot Conditions)
- **F** – Fission Product Barriers (Hot Conditions)
- **C** – Cold Shutdown / Refueling System Malfunctions (Cold Conditions)

The second letter indicates the emergency level:

- U – Unusual Event
- A – Alert
- S – Site Area Emergency
- G – General Emergency

Last is a number in #.# format, with the first being the subcategory and the second being the selected EAL in that subcategory.

HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

1	2	3	4	5	6	DEF

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

For example, the EAL HA1.2 indicates a classification in the Hazard Category at an Alert emergency level. Subcategory 1 represents a Security event and EAL 2 indicates an NRC-validated aircraft attack threat within 30 minutes of the site as indicated on the Wallchart.

- Complete accountability of site personnel within 30 minutes of a Site Area Emergency or General Emergency declaration or the decision to conduct accountability.
- Notification of event declaration to the NRC is required 'as soon as possible' following notification to State and Counties, but no later than 60 minutes after an event declaration.
- Activation of the NRC Emergency Response Data System (ERDS) data link is required within 60 minutes of an Alert or higher event declaration.
- Follow-up Notifications to the state(s) and local agencies are to occur within approximately 60 minutes from the first contact of the previous notification until a new time period is agreed upon by all offsite agencies

5.2 AD-EP-ALL-0111 INITIAL ACTIONS

Objective #5, LPSO/LOR

5.2.1 Immediate Actions

- If due to a Security threat or Hostile Action:
  - If an airborne threat, maintain continuous communications with NRC
  - If an EAL has been exceeded, then classify the event
  - If security-based event, notify the NRC and if <30 minutes away, perform a rapid evacuation of plant personnel
- Offsite Communicator initiates actions as directed

5.2.2 Action if EAL Exceeded

- Within 15 minutes: Classify and Declare the event
- If General Emergency, evaluate PARs
- If Site Area Emergency or General Emergency, take action to ensure onsite protective actions are complete within 30 minutes
  - Site Assembly required for Site Area Emergency or higher

5.2.3 Actions within 5 Minutes of Event Declaration

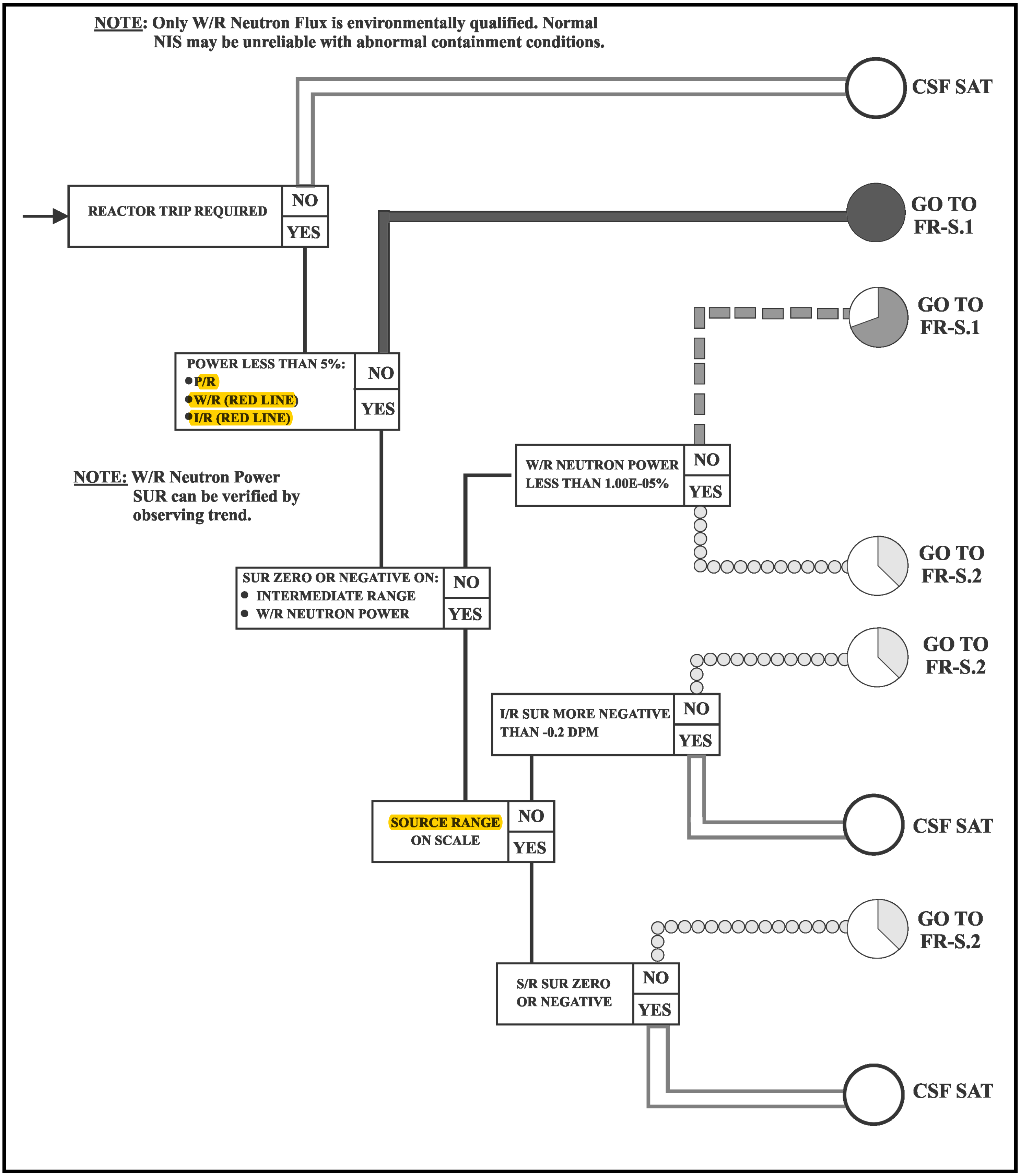
- If an Alert or Higher is declared, activate the ERO. **Correct Answer Part 1**
  - OSC, TSC, EOF activation
  - ERO can be activated for unusual event at EC discretion
- Make Plant Announcement
- Evaluate on-site protective actions **Distractor Part 1**

5.2.4 Actions within 15 Minutes of Event Declaration

- Fill out Emergency Notification Form and provide to Offsite Communicator for notification
- If release is in progress, direct performance of a dose assessment
  - If General Emergency, make follow-up notification within 30 minutes to include dose assessment and meteorological data

Question 74

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES SUBCRITICALITY - Page 1 of 1	PAGE NO. 3 of 11 Revision 9



Question 74

Table 3.3.3-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS
1.	Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	2	B,D,F,H
2.	RCS Cold Leg Temperature (Wide Range)	2	B,D,F,H
3.	RCS Pressure (Wide Range)	2	B,D,F,H
4.	Reactor Vessel Water Level	2	B,D,F,H
5.	Containment Sump Water Level (Wide Range)	2	B,D,F,H
6.	Containment Pressure (Wide Range)	2	B,D,F,H
7.	Containment Area Radiation (High Range)	1	B,D
8.	Not Used		
9.	Pressurizer Level	2	B,D,F,H
10.	Steam Generator Water Level (Narrow Range)	2 per steam generator	B,D,F,H
11.	Core Exit Temperature - Quadrant 1	2[(a)]	B,D,F,H
12.	Core Exit Temperature - Quadrant 2	2[(a)]	B,D,F,H
13.	Core Exit Temperature - Quadrant 3	2[(a)]	B,D,F,H
14.	Core Exit Temperature - Quadrant 4	2[(a)]	B,D,F,H
15.	Auxiliary Feedwater Flow	1 per steam generator	C,D,E,H
16.	RCS Radiation Level	1	B,D
17.	RCS Subcooling Margin Monitor	2	B,D,F,H
18.	Steam Line Pressure	2 per steam generator	B,D,F,H
19.	Refueling Water Storage Tank Level	2	B,D,F,H
20.	Neutron Flux (Wide Range) **Correct Answer**	2	B,D,F,H
21.	Steam Generator Water Level (Wide Range)	1 per steam generator	C,D,E,H

(a) A channel consists of two core exit thermocouples (CETs).

Question 75

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
32. (Continued)	
	5) **WHEN** all intact S/Gs pressure within 50 psig of steam header pressure, **THEN** perform the following:
	__ a) OPEN all MSIVs on intact S/Gs.
	__ b) CLOSE all MSIV bypass valves.
	__ c) Restore any alignments made using Enclosure 15 (Equalizing Across MSIVs).
	__ d) Dump steam to condenser while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour.
	__ e) **WHEN** condenser dumps established, **THEN** S/G PORVs may be closed.
	__ 6) **GO TO** Step 32.g to dump steam using S/G PORVs while pressure equalizing across MSIVs.
__ d. Verify steam dumps - IN PRESSURE MODE.	d. Place steam dumps in pressure mode as follows:
	__ 1) Place "STM DUMP CTRL" in manual.
	__ 2) Place steam dumps in pressure mode.
__ e. Dump steam to condenser while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour.	__ e. **IF** steam cannot be dumped to condenser, **THEN** **GO TO** Step 32.g.

Correct Answer Part 1

Question 75

Placing An ND Train in Service for Cooldown During Emergency Conditions

1. Limits and Precautions

1.1 When manually operating any motor operated valve, minimize the torque applied to the handwheel.

1.2 After manual operation, maintenance or packing adjustment of any motor operated safety related valve, it shall be cycled electrically to ensure reliable automatic operation.

Distractor Part 1

1.3 NC System cooldown rate of less than 80°F in any hour period is administratively recommended. The Tech Spec cooldown limit specified in TS 3.4.3 Figure 3.4.3-2 shall **NOT** be exceeded. (TS 3.4.3)

1.4 Avoid operation of an ND Pump with total pump flow rates (flow to Cold Legs, Letdown and PZR Spray as appropriate) between 2000 and 3000 gpm. If operation in this range is unavoidable, increased wear will occur on the pump motor lower bearings. The pump shall be evaluated for proper operation (no unusual flow noises or vibration) if operating in this range.

1.5 The ND pumps shall **NOT** be operated at flows less than 500 gpm for more than 3 continuous hours.

1.6 Due to the lack of positive seal seating surfaces, some minimal valve leak by may be experienced on valves 1ND-26, 1ND-27, 1ND-60, and 1ND-61 when they are in the full closed position.

2. Initial Conditions

_____ 2.1 Verify the KC System is in operation per OP/1/A/6400/005 (Component Cooling System).

_____ 2.2 Verify NC System pressure is less than 385 psig.

_____ 2.3 Verify NC System temperature is less than 350°F. Correct Answer Part 2

_____ 2.4 Verify the unit is presently under the control of an Emergency Procedure.

_____ 2.5 Verify the ND System has been injecting into the NC System.

2.6 Verify the following:

_____ • ND Pump suctions are aligned to the containment sump.

_____ • Both trains of ND are in operation.

_____ • The ND Train to remain in cold or hot leg recirculation is capable of supplying suction flow to the NV and NI pumps.

Question 76

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3. **Correct Answer Part 1**

ACTIONS

-----------------------------------NOTES-----------------------------------
Separate Condition entry is allowed for each PORV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One or two PORVs inoperable and not capable of being manually cycled.	B.1 Close associated block valves.	1 hour
	AND	
	B.2 Remove power from associated block valves.	1 hour
	AND	
	B.3 Restore PORV(s) to OPERABLE status.	72 hours

(continued)

Question 76

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of two pumps (charging pumps, safety injection pumps, or charging and safety injection pumps) capable of injecting into the RCS, the accumulators isolated, reactor coolant pump operation limited as specified in Table 3.4.12-1 and either a, b, or c below:

a. Two power operated relief valves (PORVs) with nominal lift setting = 400 psig (as left calibrated), allowable value ≤ 425 psig (as found), with RCS cold leg temperature ≥ 70°F; or

b. Two residual heat removal (RHR) suction relief valves with lift settings ≥ 417 psig and ≤ 509 psig with an indicated RCS cold leg temperature ≥ 70°F; or

c. A combination of any one PORV and one RHR suction relief valve, each with lift settings as described above.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ 210°F, **Distractor Part 2**
MODE 5,
MODE 6 when the reactor vessel head is on.

---NOTE---
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in Specification 3.4.3.

LCO (continued)

a. Two OPERABLE PORVs (NC-32B and NC-34A); or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the specified limit and testing proves its automatic ability to open at this setpoint, and motive power is available to the valve and its control circuit. The following restrictions are placed on PORV OPERABILITY for LTOP due to commonalities between the PORV power supplies and letdown isolation:

Distractor Part 2

- NC-32B is not OPERABLE for LTOP if excess letdown is in service.
- NC-32B is not OPERABLE for LTOP if normal letdown is in service and centrifugal charging pump B is in operation.
- NC-34A is not OPERABLE for LTOP if normal letdown is in service.

b. Two OPERABLE RHR suction relief valves (ND-3 and ND-38); or

An RHR suction relief valve is OPERABLE for LTOP when both of its RHR suction isolation valves are open, its setpoint is at or between 417 psig and 509 psig, and testing has proven its ability to open in this pressure range.

c. One OPERABLE PORV and one OPERABLE RHR suction relief valve.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 210°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 210°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 210°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows

Question 76

BASES

APPLICABLE SAFETY ANALYSES	Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for manual RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator. The PORVs are assumed to operate in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV automatic actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. Events that assume this condition include uncontrolled bank withdrawal at power, uncontrolled bank withdrawal from subcritical, and single rod withdrawal at power (Ref. 2). (This statement clarifies that worst case DNBR calculations are analyzed by assuming automatic PORV operation. This statement is to bound the worst case DNBR calculations based on all possible plant conditions and is not a requirement for automatic PORV operation for system OPERABILITY.) Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).
LCO	The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR. **Correct Answer Part 2** By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. Three PORVs are required to be OPERABLE to meet RCS pressure boundary requirements. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive leakage. Satisfying the LCO helps minimize challenges to fission product barriers.
APPLICABILITY	In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2.

Continued From Previous Page.

Question 77

- Thermal stresses on the steam generator tubes should also be considered before feeding a faulted steam generator. If the steam generator is dry, cold feed flow may stress the hot tubes causing tube failures. Further discussion on feeding a hot, dry steam generator is provided in the background document for FR-H.5 (Response To Steam Generator Low Level).

Distractor Part 1

- As an alternative cooldown method, one could steam a ruptured steam generator. In addition to increasing radiological releases, this will result in continued primary-to-secondary leakage. If the tube failure is large, the reactor coolant makeup supply could be depleted before ND system cooling can be established. This may also result in a steam generator overfill condition. Hence, before steaming a ruptured steam generator, one must consider potential radiological consequences, including availability of the condenser, reactor coolant activity, and meteorological conditions, and also the rate of accumulation of water in the ruptured steam generator and reactor coolant makeup supply.

STEP 42: Verify NC pump status as follows

PURPOSE:

Question 77

To alert the operator that NC pump seal damage may have occurred if NC pump cooling had previously been lost. In that case, starting the affected NC pump may further damage the seal and NC pump.

To establish forced coolant flow, if possible, or to verify natural circulation flow if NC pumps cannot be started.

APPLICABLE ERG BASIS:

Correct Answer Part 1

NC pump 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. This step provides guidance on establishing conditions for starting an NC pump to prevent NC pump damage and minimize any perturbations in NC system conditions.

Depressurization of the NC system may generate a steam bubble in the upper head region of the reactor vessel if no NC pump is running. This bubble could rapidly condense during pump startup, drawing liquid from the pressurizer and reducing reactor coolant subcooling. If pressurizer inventory is not sufficient, level may decrease off span. In addition, local flashing of reactor coolant could occur if NC system subcooling is not adequate. These conditions would require S/I reinitiation, thereby increasing leakage into the ruptured steam generator, and may confuse the operator if such behavior was unexpected.

Continued On Next Page.

Continued From Previous Page. **Question 77**

If an NC pump cannot be manually rotated due to containment access limitations or restrictions, the plant should be taken to Cold Shutdown conditions under natural circulation to permit pump disassembly and visual inspection as part of the status evaluation. An NC pump should not be started without this status evaluation since any seal misalignment or crud blockage could aggravate NC pump seal damage, potentially propagating into NC pump seal damage and increased seal leakage flow. However, there is an exception to this requirement for a status evaluation prior to NC pump start. The NC pump should be started even without a status evaluation if an extreme (red level) or severe (orange level) challenge to a Critical Safety Function is diagnosed via Status Tree monitoring and the operator is instructed to start an NC pump in the associated Function Restoration Guideline. Under these conditions, the NC pump support systems should be restored to as near normal conditions as possible and the NC pump started.

Correct Answer Part 2

A step has been added to evaluate RVLIS level prior to NC pump restart. After Safety Injection is terminated, it is possible voided conditions could occur in the upper head region at the time these steps are performed. If an NC pump restart is attempted under these conditions, pressurizer level could decrease off-scale low and/or NC system subcooling could be lost due to the collapse of the upper head void and filling of the upper head region with primary coolant. This would result in the need to reinitiate the S/I signal, consequently hindering plant recovery. In E-3, the RVLIS full range indication has always been used to determine if any voids exist in the upper head. If so, pressurizer level and NC system subcooling requirements are verified or established. These minimum values were intended to ensure plant conditions would be maintained following NC pump restart under voided conditions. If the RVLIS indication determines that a upper head void is not present, the restrictions on pressurizer level and NC subcooling do not apply. A second concern addressed for these steps is the need to provide guidance to use the pressurizer heaters to saturate the pressurizer prior to starting an NC pump with a void present in the upper head. Saturated conditions are desired to limit the pressure decrease upon NC pump restart under voided conditions but specific guidance was not originally provided. The original procedure relied on general operator training/knowledge to ensure saturated conditions existed prior to NC pump restart.

NC pumps should not be started prior to a status evaluation unless an extreme (red) or severe (orange) CSF challenge is diagnosed. Under such a CSF challenge, the "rules of usage" apply and an NC pump should be started if so instructed in the associated FRG. Under a CSF challenge, potential NC pump damage is an acceptable consequence if NC pump start is required to address a CSF challenge (e.g., to mitigate an inadequate core cooling condition). This is consistent with the intent of these FRGs which attempt to first establish support conditions to start an NC pump, but then start an NC pump whether or not the support conditions are established.

If NC pump seal cooling is lost for only a few minutes, the inventory of cold water in the seal area should prevent excessive seal heat up. For longer periods of time, seal and bearing temperatures may increase greater than 300°F. If excessive temperatures develop, the affected NC pump should not be restarted prior to a complete NC pump evaluation.

Continued On Next Page.

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 7. **Verify total CA flow - GREATER THAN 450 GPM.**	**Perform the following:** __ a. Ensure CA Pump #1 - RUNNING. __ b. **IF** flow less than 450 GPM due to operator action to control CA flow, **THEN GO TO** Step 8. __ c. Ensure all CA Pump #1 flow control valves - AT 100% DEMAND POSITION. __ d. **IF** flow less than 450 GPM, **THEN** dispatch operator to verify proper CA Pump #1 valve alignment. **REFER TO** Enclosure 2 (Local Valve Alignment for CA Pump #1).
8. **Attempt to restore power to 1ETA or 1ETB as follows:**	
__ a. Start both D/Gs from Control Room.	
__ b. Verify both D/Gs running.	b. Perform the following: __ 1) Initiate both trains S/I. __ 2) **IF** at least one D/G starts, **THEN GO TO** Step 8.c. __ 3) **GO TO** Step 9.

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

8. (Continued)

__ c. Verify both D/G load sequencers - AUTOMATICALLY LOADING BUS.

c. Perform the following for affected train(s):

- 1ETA:

1) **IF** 1ETA de-energized, **THEN** perform the following:

a) Ensure the following breakers - OPEN:

__ • "ETA NORM FDR FRM ATC"

__ • "ETA ALT FDR FRM SATA".

b) **IF** 1ETA still de-energized, **THEN** perform the following:

__ (1) **IF** D/G 1A running, **THEN** depress and hold D/G "OFF" pushbutton.

__ (2) Dispatch operator to open 1EDE-F01F (Diesel Generator Load Sequencer Panel 1DGLSA) (AB-577, BB-46, Rm 496).

__ (3) **WHEN** 1EDE-F01F open, **THEN** ensure D/G "OFF" pushbutton released.

__ 2) **IF** 1ETA energized, **THEN** ensure at least one A Train RN pump on.

(RNO continued on next page)

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

8. (Continued)

- 1ETB:

N/A

1) **IF** 1ETB de-energized, **THEN** perform the following:

a) Ensure the following breakers - OPEN:

__ • "ETB NORM FDR FRM ATD"

__ • "ETB ALT FDR FRM SATB".

b) **IF** 1ETB still de-energized, **THEN** perform the following:

__ (1) **IF** D/G 1B running, **THEN** depress and hold D/G "OFF" pushbutton.

__ (2) Dispatch operator to open 1EDF-F01F (Diesel Generator Load Sequencer Panel 1DGLSB) (AB-560, BB-46, Rm 372).

__ (3) **WHEN** 1EDF-F01F open, **THEN** ensure D/G "OFF" pushbutton released.

__ 2) **IF** 1ETB energized, **THEN** ensure at least one B Train RN pump on.

__ d. Verify 1ETA or 1ETB - ENERGIZED.

__ d. **GO TO** Step 9.

__ e. Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees).

__ f. **RETURN TO** procedure and step in effect.

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 4. **Verify 1ETA and 1ETB - ENERGIZED.**	**Perform the following:** __ a. **IF** 1ETA **AND** 1ETB de-energized, **THEN GO TO** EP/1/A/5000/ECA-0.0 (Loss of All AC Power). __ b. **WHEN** time allows, **THEN** attempt to restore power to de-energized switchgear while continuing with this procedure. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power).

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE The ESPS D/G can restore one essential bus within one hour. Energizing 1ETA or 1ETB through a shared transformer will require an operator in the 6.9KV switchgear room and require a second operator to rack in the standby breaker in the 4.16KV switchgear room.

19. **Restore power to 1ETA or 1ETB using any of the following while continuing with this procedure:**

- 1ETA:

  __ • **IF** desired, **THEN** energize 1ETA from D/G. **REFER TO** Enclosure 5 (ETA Power Restoration From D/G)

  OR

  __ • **IF** desired, **THEN** align Unit 1 normal power to 1ETA through 1ATC. **REFER TO** Enclosure 7 (Aligning Normal Power to 1ETA (1ATC))

  OR

  __ • **IF** desired, **THEN** align Unit 2 alternate power to 1ETA through SATA. **REFER TO** Enclosure 8 (Aligning Unit 2 Alternate Power to 1ETA (SATA))

  OR

  __ • **IF** desired, **THEN** align Unit 1 alternate power to 1ETA through SATA. **REFER TO** Enclosure 9 (Aligning Unit 1 Alternate Power to 1ETA (SATA))

  OR

  __ • **IF** desired **AND** ESPS D/G available, **THEN** align ESPS power to 1ETA through 1ATC. **REFER TO** Enclosure 49 (Aligning ESPS to 1ETA (1ATC)).

Distractors C.2 & D.2

Question 78

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE If NC pump seal cooling was previously isolated, further cooling of the NC pump seals will be established by natural circulation cooldown as directed in subsequent procedures.	
45. **Select recovery procedure as follows:**	
__ a. Verify NC subcooling based on core exit T/Cs - GREATER THAN 0°F.	__ a. **GO TO** EP/1/A/5000/ECA-0.2 (Loss of All AC Power Recovery With S/I Required).
__ b. Verify Pzr level - GREATER THAN 11% (30% ACC).	__ b. **GO TO** EP/1/A/5000/ECA-0.2 (Loss of All AC Power Recovery With S/I Required).
c. Verify the following valves - CLOSED: __ • 1NI-9A (NV Pmp C/L Inj Isol) __ • 1NI-10B (NV Pmp C/L Inj Isol).	__ c. **IF** any NV pump on, **THEN GO TO** EP/1/A/5000/ECA-0.2 (Loss of All AC Power Recovery With S/I Required).
__ d. **GO TO** EP/1/A/5000/ECA-0.1 (Loss of All AC Power Recovery Without S/I Required).	Distractors B.1 & D.1

END

Question 79

Given the following conditions on Unit 1:

- Unit is at 100% RTP
- 1ERPA has de-energized due to an inverter failure
- The crew is preparing to re-energize 1ERPA via 1VRD per AP/1/A/5500/029 (Loss of Vital or Aux Control Power)

Upon the loss of 1ERPA, _____(1)_____ VCT makeup capability was lost.

Once 1ERPA is aligned to 1VRD, the crew _____(2)_____ exit the action statement of Tech Spec 3.8.7 (Inverters – Operating).

Which ONE (1) of the following completes the statements above?

A. 1. auto ONLY
   2. will

B. 1. auto ONLY
   2. will NOT

C. 1. auto AND manual
   2. will

D. 1. auto AND manual
   2. will NOT

Original Question

MODIFIED

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

7. (Continued)

__ g. Ensure "MANUAL BYPASS SWITCH" on 1EME (Manual Bypass Switch For Swing Inverter 1EIE) - SELECTED TO "ALTERNATE AC SOURCE TO LOAD".

NOTE Padlock key interlock will provide a control to prevent closing of more than one output breaker on 1EME.

__ h. Unlock breaker 1EME B4 (1EME Output To 1EMA).

__ i. Ensure breaker 1EME B4 (1EME Output To 1EMA) - ON.

__ j. Ensure breaker 1EMA B4 (1EMA Output To 1ERPA) - ON.

__ k. Ensure "MANUAL BYPASS SWITCH" on 1EMA - SELECTED TO "ALTERNATE AC SOURCE TO LOAD".

Correct Answer Part 1

Question 79

Manual Bypass Switches

Distractor Part 1

Objective # 4F, ISS, LICENSED

Six Manual Bypass Switches - EMA, EMB, EMC, EMD, EME, and EMF - are utilized in this system to manually transfer the inverter to the alternate power source without power interruption (make-before-break contacts) to the 120 VAC panelboards (ERPA, ERPB, ERPC and ERPD). Reference **Figure 16** (Manual Bypass Switch)

The normal and alternate sources should be synchronized (in sync light lit) prior to transfer. However, there is no interlock to prevent transfer if the two sources are not synchronized. Each Manual Bypass Switch has an “In Sync” indication to let the operator know that the normal and alternate power supplies to the 120 VAC panelboards are synchronized.

The Manual Bypass Switch has two positions: “INVERTER TO LOAD” and “ALTERNATE SOURCE TO LOAD.” The power source for each of these positions for each inverter is listed below:

MANUAL BYPASS SWITCH	“INVERTER TO LOAD”	“ALTERNATE SOURCE TO LOAD”	MANUAL BYPASS SWITCH OUTPUT SUPPLIES:
EMA	EIA Output	EME Output	ERPA
EMB	EIB Output	EMF Output	ERPB
EMC	EIC Output	EME Output	ERPC
EMD	EID Output	EMF Output	ERPD

Question 79

ACTIONS (continued)

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC vital bus is powered from its voltage regulated transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC vital buses is the preferred source for powering instrumentation trip setpoint devices.

Correct Answer Part 2

If the channel-related inoperable inverter is replaced by its train's swing inverter, the 24 hour limit does not apply (unless the swing inverter is also inoperable).

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital bus energized from the inverter. The verification of proper indicated voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

Question 79

LCO (continued)

Distractor Part 2

by their associated battery powered inverters provide uninterruptible supply of AC electrical power to associated loads even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the AC vital bus be powered by its channel-related inverter, or swing inverter. When the redundant train of Class 1E AC vital bus electrical power distribution subsystem is required by LCO 3.8.10, the power source for these AC vital buses may consist of 1) the associated channel-related inverter powered by its associated battery; 2) the constant voltage source transformer; or 3) a swing inverter powered by one of the train-related batteries. This ensures the availability of sufficient power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, "Distribution Systems—Shutdown," the remaining OPERABLE Inverters may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances,

Question 80

	General Emergency	Site Area Emergency	Alert	Unusual Event
1 **Loss of Essential AC Power**	Prolonged loss of **all** offsite and **all** onsite AC power to essential buses 1 | 2 | 3 | 4 **SG1.1** Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB **AND** SSF fails to supply NC pump seal injection **OR** CA supply to SGs **AND EITHER**: - Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1) - Core Cooling RED PATH conditions met	Loss of **all** offsite and **all** onsite AC power to essential buses for 15 minutes or longer 1 | 2 | 3 | 4 **SS1.1** Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1) Unit 2 Condition at 1015 Correct Answer Distractor A.2 & C.2	Loss of **all but one** AC power source to essential buses for 15 minutes or longer 1 | 2 | 3 | 4 **SA1.1** AC power capability, Table S-1, to essential 4160V buses 1(2)ETA and 1(2)ETB reduced to a single power source for ≥ 15 min. (Note 1) **AND** **Any** additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS Unit 1 Condition at 1015 Distractor A.1 & B.1	Loss of **all** offsite AC power capability to essential buses for 15 minutes or longer 1 | 2 | 3 | 4 **SU1.1** Loss of **all** offsite AC power capability, Table S-1, to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1) Unit 2 has Not los ATC or ATD but has lost SATA and SATB since aligned via Unit 1 Power Supply
2 **Loss of Vital DC Power**	Loss of **all** essential AC and vital DC power sources for 15 minutes or longer 1 | 2 | 3 | 4 **S**[illegible]**2** Los[illegible] **all** offsite and **all** onsite AC power capability to essen[illegible] 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. **AN**[illegible] [illegible] **all** 1[illegible]VDC power based on battery bus voltage [illegible]VDC on **all** vital DC buses EDA, EDC, EDB and ED[illegible]r ≥ 1[illegible]n. (Note[illegible]	Loss of all vital DC power for 15 minutes or longer 1 | 2 | 3 | 4 **SS2.1** Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses EDA, EDC, EDB and EDD for ≥ 15 min. (Note 1)	None	None
3 **Loss of CR Indications**	Unit 2 Co[illegible]dition at 1[illegible] Correct [illegible]swer None	None	UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress 1 | 2 | 3 | 4 **SA3.1** An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1) **AND** Any significant transient is in progress, Table S-3	UNPLANNED loss of Control Room indications for 15 minutes or longer 1 | 2 | 3 | 4 **SU3.1** An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

Table S-1 AC Power Sources
Offsite
- ATC (Train A)
- SATA (Train A) (if already aligned)
- ATD (Train B)
- SATB (Train B) (if already aligned)
Onsite
- D/G A (Train A)
- D/G B (Train B)

Table S-3 Significant Transients
- Reactor trip

Question 81

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- After low steamline pressure main steam isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation.
- OAC graphic SMRATES to monitor S/G pressure rates can be accessed via a hot button in the center of the SM graphic.

11. **Initiate NC System cooldown to Cold Shutdown as follows:**

__ a. Verify "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT.

RNO: __ a. **GO TO** Step 11.g.

__ b. Verify MSIVs on all intact S/Gs - OPEN.

RNO: b. Perform the following:

__ 1) **IF** any S/G faulted, **THEN GO TO** Step 11.g.

__ 2) **IF** intact MSIVs required closed to isolate leak, **THEN GO TO** Step 11.g.

3) Reset Main Steam Isolation signal as follows:

__ a) Ensure manual loaders for all MSIV bypass valves - ADJUSTED TO 0%.

__ b) Reset SM Isolation.

__ c) Reset S/G PORVs.

(RNO continued on next page)

Question 81

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

11. (Continued)

4) Place steam dumps in pressure mode as follows:

__ a) Place "STM DUMP CTRL" in manual.

__ b) Adjust "STM DUMP CTRL to 0% demand.

__ c) Place steam dumps in pressure mode.

5) Perform the following to equalize pressure across MSIVs on intact S/Gs:

__ a) OPEN MSIV bypass valve on intact S/Gs.

__ b) **IF AT ANY TIME** pressure does not equalize as required, **THEN** isolate steam loads off main steam header. **REFER TO** Enclosure 7 (Equalizing Across MSIVs).

(RNO continued on next page)

Question 81

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
	6) **WHEN** all intact S/Gs pressure within 50 psig of steam header pressure, **THEN** perform the following:
	__ a) OPEN all MSIVs on intact S/Gs.
	__ b) CLOSE all MSIV bypass valves.
	__ c) **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK".
	__ d) Dump steam to condenser while maintaining cooldown rate based on NC T-Colds as close as possible without exceeding 100°F in an hour.
	__ e) **WHEN** condenser dumps established, **THEN** S/G PORVs may be closed.
	__ 7) **GO TO** Step 11.g to dump steam using S/G PORVs while pressure equalizing across MSIVs.
__ c. **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK".	
__ d. Verify steam dumps - IN PRESSURE MODE.	d. Place steam dumps in pressure mode as follows:
	__ 1) Place "STM DUMP CTRL" in manual.
	__ 2) Place steam dumps in pressure mode.

Question 81

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Monitor Enclosure 1 (Foldout Page).**	

Distractor Part 2

NOTE CSF should not be implemented until directed by this procedure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. **Verify at least one of the following annunciators - LIT:** __ • 1AD-20, B/2 "CONT. SUMP LEVEL >2.5 ft" OR __ • 1AD-21, B/2 "CONT. SUMP LEVEL >2.5 ft".	**<u>IF</u> both alarms dark, <u>THEN</u> perform the following:** a. Ensure S/I - RESET: __ 1) ECCS. __ 2) D/G load sequencers. __ b. Ensure ND pumps - OFF. c. <u>**IF**</u> either ND pump continues to run, <u>**THEN**</u> isolate affected trains discharge path as follows: 1) <u>**IF**</u> train "A" affected, <u>**THEN**</u> CLOSE: __ • 1NI-173A (ND Hdr 1A To Cold Legs C&D) __ • 1ND-32A (ND Train 1A Hot Leg Inj Isol). 2) <u>**IF**</u> train "B" affected, <u>**THEN**</u> CLOSE: __ • 1NI-178B (ND Hdr 1B To Cold Legs A&B) __ • 1ND-65B (ND Train 1B Hot Leg Inj Isol). (RNO continued on next page)

APPLICABLE SAFETY ANALYSES (continued)

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Analyses are performed in regard to static rod misalignment, single rod withdrawal, dropped rod, and dropped group of rods (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps. Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(X,Y,Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}(X,Y)$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(X,Y,Z)$ and $F^N_{\Delta H}(X,Y)$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in the safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36 (Ref. 6).

Question 88

In accordance with the following Technical Specification BASES:

(1) T.S. 2.1.2 (RCS Pressure SL)
Pressurizer Safeties, Reactor Trip Setpoints, and ______(1)______ are required in order to ensure Reactor Coolant System Pressure Safety Limits are not exceeded.

(2) T.S. 3.3.1 (Reactor Trip System Instrumentation)
Only three channels of pressurizer level (vs. four) are required because ___________(2)___________.

A. (1) Pressurizer PORVs
(2) pressurizer level does NOT provide a backup signal to any other reactor trips

B. (1) Pressurizer PORVs
(2) of the slow rate of charging that is available

C. (1) Steam Generator Safeties
(2) pressurizer level does NOT provide a backup signal to any other reactor trips

D. (1) Steam Generator Safeties
(2) of the slow rate of charging that is available

Original Question

MODIFIED

Question 83

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level-High

Distractor Part 2

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The setpoints are based on percent of instrument span. The LCO requires three channels of Pressurizer Water Level-High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

Correct Answer Part 2

10. Reactor Coolant Flow-Low

a. Reactor Coolant Flow-Low (Single Loop)

The Reactor Coolant Flow-Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48% RTP, a loss of flow in any RCS loop will actuate a reactor trip. The

APE033 2.4.50 - Loss of Intermediate Range Nuclear Instrumentation
APE033 GENERIC
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)

Given the following initial conditions on Unit 1:

- The Unit is conducting a startup at the POAH
- Intermediate Range Channel N-35 begins to operate erratically
- AP/1/A/5500/016 (Malfunction of Nuclear Instrumentation System) Case III (Intermediate Range Malfunction) is entered

Subsequently:

- Power is stable at the POAH
- AP/16 actions are complete

The "1/N-35A I/R CHANNEL 1 TRIP BYPASS" status light on 1SI-19 _____(1)_____ LIT.

Per Tech Spec 3.3.1 (RTS Instrumentation), the startup to MODE 1 _____(2)_____ continue.

Which ONE (1) of the following completes the statements above?

A. 1. is
2. may

B. 1. is
2. may NOT

C. 1. is NOT
2. may

D. 1. is NOT
2. may NOT

Question 84

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Verify reactor power - GREATER THAN 10%.**	__ **Stop any power increase.**
__ 2. **Verify 1AD-2, C/3 "I/R HI FLUX LEVEL ROD STOP" - DARK.**	__ **Adjust turbine load to maintain T-Avg at T-Ref.**
3. **Identify affected I/R channel:** __ • N-35 OR __ • N-36.	

NOTE 1AD-2, C/4 "N/I SYS S/R & I/R TRIP BYPASS" will actuate in the following step if malfunction is due to instrument power failure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 4. **At affected I/R drawer, place "LEVEL TRIP" switch for affected channel in "BYPASS".**	**Part 1 Correct Answer**
__ 5. **Verify affected I/R channel trip bypass status light (1SI-19) - LIT.**	**Perform the following:** __ a. Notify Shift Manager that due to loss of control power, I/R "HIGH LEVEL TRIP" cannot be bypassed. Reactor power reduction below P-10 I/R BLOCK PERMISSIVE will result in Reactor Trip due to "N/I HI FLUX IR RX TRIP". __ b. **GO TO** Step 7.
__ 6. **Verify 1AD-2, C/4 "N/I SYS S/R & I/R TRIP BYPASS" - LIT.**	

ACTIONS (continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux-Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-High; and
- SG Water Level-Low Low.

A known inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-three logic for actuation of the two-out-of-four trips. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 11.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 12 hours while performing routine surveillance testing of the other channels. The 12 hour time limit is justified in Reference 11.

F.1 and F.2

Distractor A.2 & C.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below

ACTIONS (continued)

Distractor A.2 & C.2

the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

<u>G.1 and G.2</u>

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint.

Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip. Required Action G.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this

ACTIONS (continued)

Part 2 Correct Answer

Action.

H.1

Condition H applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is below the P-6 setpoint and one or two channels are inoperable. Below the P-6 setpoint, the NIS source range performs the monitoring and protection functions. The inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. The NIS intermediate range channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10.

I.1

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately. Required Action I.1 is modified by a Note to indicate that normal plant control operations that individually add limited positive reactivity (e.g., temperature or boron fluctuations associated with RCS inventory management or temperature control) are not precluded by this Action.

J.1

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition and the unit exits this Condition.

Question 85

16.9 AUXILIARY SYSTEMS

16.9-5 Fire Rated Assemblies

COMMITMENT All required Fire Rated Assemblies (walls, floors/ceilings, cable enclosures and other fire barriers) and all sealing devices in fire rated assembly penetrations (fire doors, fire dampers, and penetration seals) as shown on the CN-1105 drawing series shall be FUNCTIONAL.

APPLICABILITY: At all times.

---NOTE---
Non-functional or breached fire barrier features (walls, floors, ceilings, doors, dampers, and penetration seals) in the diesel generator rooms and the auxiliary feedwater pump rooms may affect CO_2 System FUNCTIONALITY. See SLC 16.9-3, “CO_2 Systems”.

REMEDIAL ACTIONS

N/A IF the required Fire Rated Assembly sealing device is a Fire Door, see Table 16.9-5-1

N/A IF the required Fire Rated Assembly sealing device is a Fire Damper see Table 16.9-5-2

IF required Fire Rated Assembly is a Fire Barrier or Penetration Seal:

✓ 1. Identify the location of the impaired fire protection feature by elevation, column, and building

✓ 2. Verify the wall, floor/ceiling is a committed boundary on the CN-1105 drawing series (if not a committed boundary, SLC 16.9-5 does not apply)

✓ 3. Refer to CN-1209-10 series drawings to identify the Fire Area on both sides of the impaired feature

N/A 4. IF either of the Fire Areas is identified as High Safety Significant (HSS) (see Table 16.9-5-3) then implement the REQUIRED ACTION **CONDITION A**

✓ 5. IF the Fire Areas are not HSS, then identify the associated shutdown trains/methods of the Fire Areas on each side of the barrier using Table 16.9-5-4 and implement the REQUIRED ACTION as identified in the following Chart:

Shutdown Train (Side 1 & Side 2)	*A*	*B*	*SSS*	*A or B*	*A and B*
A	**CONDITION C**	**CONDITION B**	**CONDITION B**	**CONDITION C**	**CONDITION B**
B	**CONDITION B**	**CONDITION C**	**CONDITION B**	**CONDITION C**	**CONDITION B**
SSS	**CONDITION B**	**CONDITION B**	**CONDITION C**	**CONDITION B**	**CONDITION B**
A or B	**CONDITION C**	**CONDITION C**	**CONDITION B**	**CONDITION C**	**CONDITION B**
A and B	**CONDITION B**	**CONDITION B**	**CONDITION B**	**CONDITION B**	**CONDITION C**

Question 85

REMEDIAL ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
C.	One or more DID*** required Fire Rated Assemblies is non-functional.	C.1	Establish a once per shift fire watch on at least one side of the assembly.	1 hour
		OR		
		C.2	Verify at least one side of the assembly has FUNCTIONAL fire detection instrumentation.	1 hour
		OR		
		C.3	Complete an evaluation as permitted by NRC RIS 2005-07 to institute required action(s).	Prior to terminating Required Action C.1

*High Safety Significant (HSS) Fire Areas containing required Fire Rated Assemblies are defined in Table 16.9-5-3.

**Low Safety Significant (LSS) Fire Areas containing required Fire Rated Assemblies are defined as those areas with a boundary between redundant shutdown trains.

***Defense-in-Depth (DID) Fire Areas containing required Fire Rated Assemblies are defined as analysis compartment boundaries or PRA compartment boundaries that do not meet the HSS or LSS definitions.

Question 85

Table 16.9-5-2

REQUIRED FIRE DAMPERS

DAMPER NUMBER	BLDG	LOCATION	ELEVATION	FIRE AREA INTERFACE	RISK CRITERIA	REMEDIAL ACTION CONDITION
1TB-FD038	TB1	16-17/V	594+0	TB1/SRV	DID	C
1TB-FD039	TB1	16-17/V	594+0	TB1/SRV	DID	C
1TB-FD040	TB1	16/V	594+0	TB1/SRV	DID	C
1TB-FD043	TB1	30-31/1J-1K	568+0	TB1/OTT	DID	C
1TB-FD044	TB1	32/1J-1K	594+0	TB1/MTOT	DID	C
1TB-FD045	TB1	30/1J-1K	594+0	TB1/MTOT	DID	C
1TB-FD046	TB1	32/1K-1L	568+0	TB1/OTT	DID	C
2TB-FD013	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD014	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD015	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD016	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD017	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD018	TB2	21-22/P	594+0	TB2/SRV	DID	C
2TB-FD019	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD020	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD021	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD022	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD023	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD024	TB2	18-19/P	594+0	TB2/SRV	DID	C
2TB-FD031	TB2	32/2K-2L	568+0	TB2/OTT	DID	C
2TB-FD032	TB2	18/P	594+0	TB2/SRV	DID	C
2TB-FD036	TB2	16-17/P	594+0	TB2/SRV	DID	C
2TB-FD038	TB2	17-18/P	594+0	TB2/SRV	DID	C
2TB-FD039	TB2	32/2J-2K	594+0	TB2/MTOT	DID	C
2TB-FD040	TB2	30/2J-2K	594+0	TB2/MTOT	DID	C
2TB-FD041	TB2	30-31/2J/2K	568+0	TB2/OTT	DID	C

*2VA-FD070 is exempt from inspection requirements (SLC TR 16.9-5-5) for ALARA reasons

Table 16.9-5-3

HIGH SAFETY SIGNIFICANT (HSS) FIRE AREAS*

Distractor Part 1

FIRE AREA	BLDG	ELEVATION	DESCRIPTION
6	AUX	560+0	Unit 1 Electrical Pen Room El 560
12	AUX	577+0	Unit 2 Electrical Pen Room El 577
13	AUX	577+0	Unit 1 Electrical Pen Room El 577
14	AUX	577+0	Unit 2 4160V Essential Swgr Room (2ETA)
15	AUX	577+0	Unit 1 4160V Essential Swgr Room (1ETA)
16	AUX	574+0	Unit 2 Cable Room El 574
17	AUX	574+0	Unit 1 Cable Room El 574
21	AUX	594+0	Main Control Room El 594

*High Safety Significant (HSS) Fire Areas are defined as the areas with HSS fire barrier features in accordance with the Catawba NFPA 805 Monitoring Program.

Question 85

Table 16.9-5-4

FIRE AREAS AND SHUTDOWN TRAIN / METHOD

FIRE AREA	FIRE AREA DESCRIPTIONS	ASSURED SHUTDOWN TRAIN / METHOD
1	ND & NS Pump Room El 522 (Common)	SSS
2	Unit 2 CA Pump Room El 543	SSS
3	Unit 1 CA Pump Room El 543	SSS
4	Aux Bldg. Gen Area & NV Pump Room El 543 (Common)	SSS
5	Unit 2 Electrical Pen Room El 560	A
6	Unit 1 Electrical Pen Room El 560	A
7	Unit 2 4160V Essential SWGR Room El 560	A
8	Unit 1 4160V Essential SWGR Room El 560	A
9	Unit 2 Battery Room El 554	SSS
10	Unit 1 Battery Room El 554	SSS
11	Aux Bldg. Gen Area & U1 KC Pump Room El 560 (Common)	SSS
12	Unit 2 Electrical Pen Room El 577	B
13	Unit 1 Electrical Pen Room El 577	B
14	Unit 2 4160V Essential SWGR Room El 577	B
15	Unit 1 4160V Essential SWGR Room El 577	B
16	Unit 2 Cable Room El 574	SSS
17	Unit 1 Cable Room El 574	SSS
18	Aux Bldg. Gen Area & U2 KC Pump Room El 577 (Common)	SSS
19	Unit 2 Electrical Pen Room El 594	A
20	Unit 1 Electrical Pen Room El 594	A
21	Control Room El 594 (Common)	SSS
22	Aux Bldg. Gen Area El 594 (Common)	SSS
23	Unit 2 Fuel Storage Area El 605	A
24	Unit 1 Fuel Storage Area El 605	A
25	Diesel Generator Bldg. 1A El 556	B
25A	Diesel Generator Bldg. 1A Stairwell	B
26	Diesel Generator Bldg. 1B El 556	A
26B	Diesel Generator Bldg. 1B Stairwell	A
27	Diesel Generator Bldg. 2A El 556	B
27A	Diesel Generator Bldg. 2A Stairwell	B
28	Diesel Generator Bldg. 2B El 556	A
28B	Diesel Generator Bldg. 2B Stairwell	A
29	Train A RN Pump Structure El 600 (Common)	B
30	Train B RN Pump Structure El 600 (Common)	A
31	Unit 2 Train A Aux Shutdown Panel El 543	B
32	Unit 1 Train A Aux Shutdown Panel El 543	B
33	Unit 2 Train B Aux Shutdown Panel El 543	A
34	Unit 1 Train B Aux Shutdown Panel El 543	A
35	Control Room Tagout Area El 594	A or B
36	Unit 2 Turbine Driven CA Pump Control Panel Room El 543	B
37	Unit 1 Turbine Driven CA Pump Control Panel Room El 543	B
38	Unit 1 Fuel Storage Area HVAC Room El 631	A or B

(continued)

Catawba Nuclear Station

ILT-17 NRC Written Exam CNS SRO NRC Examination

Question: 86
(1 point)

Given the following conditions on Unit 1:

- The Unit is stable at 100% RTP
- Operators are performing PT/1/A/4150/001 A (NC Pump Seal Injection Flow Verification)
- 1NV-294 is fully open
- Seal Injection flow is 42 gpm and stable

Based on the conditions above, the Action Statement of Tech Spec 3.5.5 (Seal Injection Flow) ______(1)______ required to be entered.

In accordance with the Bases of Tech Spec 3.5.5, the Seal Injection Flow Limit ______(2)______ based on the safety analysis assumptions for minimum ECCS Injection flow.

Which ONE of the following completes the statements above?

A. 1. is
 2. is

B. 1. is NOT
 2. is

C. 1. is
 2. is NOT

D. 1. is NOT
 2. is NOT

Original Question

Question 86

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be ≤ 40 gpm with centrifugal charging pump operating and the charging flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow not within limit.	A.1 Adjust manual seal injection throttle valves to give a flow within limit with centrifugal charging pump operating and the charging flow control valve full open.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

Question 86

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

This LCO is applicable only to those units that utilize the centrifugal charging pumps for safety injection (SI). The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the centrifugal charging pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI.

APPLICABLE SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps. The centrifugal charging pumps are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the centrifugal charging pumps. The steam generator tube rupture and main steam line break event analyses also credit the centrifugal charging pumps, but do not set the limits on their flow requirements. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

This LCO ensures that seal injection flow of ≤ 40 gpm, with centrifugal charging pump operating and charging flow control valve full open, will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the centrifugal charging pumps will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical for a large LOCA. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory. Seal injection flow satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

Catawba Nuclear Station

ILT-17 NRC Written Exam CNS SRO NRC Examination

Question: 87
(1 point)

Given the following Unit 2 conditions:

- The Unit is at 100% RTP
- A loss of 2ERPD has occurred
- The crew has entered AP/2/A/5500/029 (Loss of Vital or Aux Control Power)
- NO Tech Spec actions have been addressed

The current Containment Pressure channel logic for the remaining Containment Pressure channels which will cause a **Phase B** actuation is _____(1)_____ .

In accordance with Tech Spec 3.3.2 (ESFAS Instrumentation) LCO Actions, when the failed channel is removed from service, I&E will place the Containment Pressure **Hi-Hi** Bistable in _____(2)_____ .

Which ONE of the following completes the statements above?

A. 1. 2/3
   2. Trip

B. 1. 2/3
   2. Bypass

C. 1. 1/3
   2. Trip

D. 1. 1/3
   2. Bypass

Original Question

Question 87

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

containment isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase B containment isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase B Isolation-Containment Pressure - High-High

Containment Pressure - High-High uses four channels in a two-out-of-four logic configuration. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip.

Containment Pressure - High-High must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to pressurize the containment and reach the Containment Pressure - High-High setpoints.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

Question 87

ACTIONS (continued)

D.1, D.2.1, and D.2.2

Condition D applies to:

- Containment Pressure-High;
- Pressurizer Pressure-Low;
- Steam Line Pressure-Low;
- Steam Line Pressure-Negative Rate-High;
- Loss of offsite power (refer to Condition D footnote);
- SG Water level—Low Low; and
- SG Water level—High High (P-14) for the Feedwater Isolation Function.

Distractor Part 2

If one channel is inoperable, 72 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements. The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition is justified in Reference 13.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 12 hours for surveillance testing of other channels. The 12 hours allowed for testing is justified in Reference 13.

Question 87

ACTIONS (continued)

E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Phase B Isolation Containment Pressure-High High; and

Correct Answer Part 2

- Steam Line Isolation Containment Pressure - High High.

Neither of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious isolation initiation. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate isolation.

To avoid the inadvertent actuation of Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 72 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 72 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 12 hours for surveillance testing. Placing a second channel in the bypass condition for up to 12 hours for testing purposes is acceptable based on the results of Reference 13.

Question 88

A. Purpose

- **To verify proper response to a loss of feedwater supply to the S/Gs.**
- **To verify proper response to a loss of normal supply of auxiliary feedwater.**
- **To verify proper response to an abnormal S/G feedwater control condition.**

B. Symptoms

Case I. Loss Of CF Supply To S/Gs:

- CFPT A and B - TRIPPED
- 1AD-3, C/6 "CF ISOL TRN A" - LIT
- 1AD-3, D/6 "CF ISOL TRN B" - LIT
- Any S/G lo level alert alarm on 1AD-4 - LIT
- Any S/G flow mismatch lo CF flow alarm on 1AD-4 - LIT.

Case II. Loss Of Normal CA Supply:

- Any CA start resulting from abnormal plant conditions (auto or manual)
- 1AD-8, B/1 "UST LO LEVEL" lit **AND** 1CA-4 (CA Pmps Suct From UST) open
- CACST less than 50% **AND** 1CA-6 (CA Pmps Suct From CACST) open.

Case III. CF Control Not in Auto:

- Any unanticipated CF controller in manual
- Any S/G lo level alert alarm on 1AD-4 - LIT
- Any S/G flow mismatch lo CF flow alarm on 1AD-4 - LIT
- 1AD-2, F/9 "DCS ALTERNATE ACTION" - LIT.

Question 88

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Verify reactor power - LESS THAN 5%.** Correct Answer	**IF AT ANY TIME all CF supply to S/G(s) lost, THEN perform the following:** __ a. Trip reactor. __ b. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ 2. **Verify all S/G hi-hi level alert alarms (1AD-4) - DARK.**	**IF 2/4 S/G N/R levels on any one S/G greater than 83%, THEN perform the following:** __ a. Verify all Feedwater Isolation status lights (1SI-5) - LIT. b. **IF** any Feedwater Isolation status light not lit, **THEN** perform the following: __ 1) Initiate Feedwater Isolation. __ 2) **IF** proper status light indication not obtained, **THEN** CLOSE affected valve(s).
__ 3. **Verify 1AD-2, F/9 "DCS ALTERNATE ACTION" - DARK.**	**IF all the following conditions exist:** __ • 1AD-2, F/9 in alarm for CF control function in alternate action __ • At least one CF pump - IN SERVICE __ • 1AD-3, C/6 "CF ISOL TRN A" - DARK __ • 1AD-3, D/6 "CF ISOL TRN B" - DARK, __ **THEN GO TO** Case III (CF Control Not In Auto).

Question 88

A. Purpose

- **To verify the proper response in the event of a turbine generator trip.**
- **To place the plant in a stable condition.**

B. Symptoms

- **Any Turbine Trip alarm on 1AD-1 - LIT**
- **Turbine stop valves closed status lights (1SI-2) - LIT**
- **Zone G Lockout Trip alarm on 1AD-11 (F/4) - LIT.**

Question 88

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

1. __ **Verify reactor power - LESS THAN 69%.**

   Distractor

   Perform the following:

   __ a. Ensure reactor - TRIPPED.

   __ b. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

2. **Verify Turbine Trip:**

   __ • All turbine stop valves - CLOSED.

   Perform the following:

   __ a. Trip turbine.

   b. **IF** turbine will not trip, **THEN** perform the following:

   __ 1) Depress "MANUAL" pushbutton on turbine control panel.

   __ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons.

   3) **IF** turbine will not runback, **THEN** perform the following:

   __ a) Trip reactor.

   b) CLOSE the following valves:

   __ • All MSIVs

   __ • All MSIV bypass valves.

   __ c) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Steam Line Pressure-Negative Rate-High must be OPERABLE in MODE 3 when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure-Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, stop the excessive flow of feedwater into the SGs, and to limit the energy released into containment. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows. Feedwater Isolation serves to limit the energy released into containment upon a feedwater line or steam line break inside containment.

The Functions are actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine;

Distractor

- Trips the MFW pumps;
- Initiates feedwater isolation; and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves.

Turbine Trip and Feedwater Isolation signals are both actuated by SG Water Level-High High, or by an SI signal. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. A Feedwater Isolation signal is also generated by a

Question 88

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

(1) Feedwater Isolation-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Distractor

(2) Feedwater Isolation-Steam Generator Water Level-High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic. The setpoints are based on percent of narrow range instrument span.

(3) Feedwater Isolation-Safety Injection

Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements. Item 5.b.(1) is referenced for the applicable MODES.

Distractor

(4) Feedwater Isolation - RCS T_{avg} - Low coincident with Reactor Trip (P-4)

This signal provides protection against excessive cooldown, which could subsequently introduce a positive reactivity excursion after a plant trip. There are four channels of RCS T_{avg} - Low (one per loop), with a two-out-of-four logic required coincident with a reactor trip signal (P-4) to initiate a feedwater isolation. The P-4 interlock is discussed in Function 8.a.

Question 88

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Engineered Safety Feature Actuation System Interlocks—Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker is open. Operators are able to reset SI 60 seconds after initiation. If a P-4 is present when SI is reset, subsequent automatic SI initiations will be blocked until the RTBs have been manually closed. This Function allows operators to take manual control of SI systems after the initial phase of injection is complete while avoiding multiple SI initiations. The functions of the P-4 interlock are:

Correct Answer

- Trip the main turbine;
- Isolate MFW with coincident low T_{avg};
- Prevent reactuation of SI after a manual reset of SI;
- Transfer the steam dump from the load rejection controller to the unit trip controller; and
- Prevent opening of the MFW isolation valves if they were closed on SI or SG Water Level—High High.

Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated power. To avoid such a situation, the noted Functions have been interlocked with P-4 as part of the design of the unit control and protection system.

None of the noted Functions serves a mitigation function in the unit licensing basis safety analyses. Only the turbine trip Function is explicitly assumed since it is an immediate consequence of the reactor trip Function. Neither turbine trip, nor any of the other four Functions associated with the reactor trip signal, is required to show that the unit licensing basis safety analysis acceptance criteria are not exceeded.

The RTB position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable

Question 89

F Fission Product Barriers	FG1.1 1 2 3 4	FS1.1 1 2 3 4	FA1.1 1 2 3 4
	Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1)	Loss or potential loss of **any** two barriers (Table F-1)	**Any** loss or **any** potential loss of either Fuel Clad or NCS (Table F-1)

Time After S/D	FC Loss	NCS Loss	Loss
0 - 1	550	8.8	5500
1 - 2	400	8.4	4000
2 - 8	160	7.0	1600
>8	100	6.2	1000

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (NCS) Barrier		Containment (CMT) Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A. NCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER**: • UNISOLABLE NCS leakage • SG tube RUPTURE	1. CSFST Integrity-RED PATH conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B. Inadequate Heat Removal	1. CSFST Core Cooling-RED PATH conditions met	1. CSFST Core Cooling-ORANGE PATH conditions met 2. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED PATH conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C. CMT Radiation / NCS Activity	1. EMF53A/B > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 μCi/gm	None	1. EMF53A/B > Table F-2 column "NCS Loss"	None	None	1. EMF53A/B > Table F-2 column "CMT Potential Loss"
D. CMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of NCS leakage outside of containment	1. CSFST Containment-RED PATH conditions met 2. Containment hydrogen concentration > 6% 3. Containment pressure > 3 psig with < one full train of containment cooling operating per design for > 15 min. (Notes 1, 10)
E. EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Question 89

<< Emergency Coordinator Checklist >>

1.5 Actions to Complete Within 30 Minutes of Site Area Emergency (or Higher) Declaration

1. **IF** Site Area Emergency or General Emergency, **THEN evaluate** concurrently Attachment 1 Section 3.1, Site Assembly and Accountability and Attachment 1 Section 3.2, Site Evacuation, to ensure onsite protective actions are taken. ...._____

1.6 Actions to Perform As Soon As Possible, But No Later Than 60 Minutes After Event Declaration

NOTE
Initial notification of the NRC is to occur as soon as possible following State and Counties notifications, but not to exceed one hour, following the declaration of an emergency or a change in the classification level. .... ☐

1. Initial notification of the NRC is to occur as soon as possible following State and Counties notifications, but not to exceed 1 hour, following the declaration of an emergency or a change in the classification level. ....__________

   a. **Obtain** copy of the completed and approved ENF. .... ☐

   b. Using the ENF, **direct** Offsite Communicator to notify NRC in accordance with Attachment 3, NRC Communications Checklist. .... ☐

1.7 Actions to Perform Within 60 Minutes of Alert or Higher Event Declaration

1. **Ensure** the NRC Emergency Response Data System (ERDS) data link is activated or transmitting data. ....__________

   a. **IF** ERDS is **NOT** currently activated, **THEN activate** ERDS. ....__________

Question 89

<< NRC Communications Checklist >>

1.1 Mobilization (continued)

2. **Notify** Shift Manager/Emergency Coordinator (SM/EC) of arrival on station. .... ☐

   a. **IF** directed to initiate ERONS, **THEN initiate** ERONS to activate the ERO in accordance with AD-EP-ALL-0301, Activation of the Emergency Response Organization Notification System (ERONS), using the appropriate Template ID and Message Title. .... ____/**TAB 2**

   b. **IF** directed to initiate ERONS **AND** ERONS is **NOT** available, **THEN activate** the ERO using the back-up system per CSD-EP-ALL-0111-01, ERONS Backup Job Aid. .... ____/**TAB 5**

1.2 Establish Control Room / NRC interface

> **NOTE**
>
> Emergency Response Data System (ERDS) is required to be activated within 1 hour of declaration of an Alert or higher emergency classification level. .... ☐

1. **Ensure** ERDS activation. .... ☐

   a. **IF** ERDS has **NOT** been activated, **THEN activate** ERDS using station specific procedure. .... ____/**TAB 3**

2. **Obtain** the following: .... ☐

   - NRC authentication code .... ☐

> **NOTE**
>
> Emergency Notification System (ENS) phone numbers are provided on ENS phones and in CSD-EP-ALL-0104-01, Emergency Telephone Directory. .... ☐

3. **IF** initial notification from the Control Room due to declaration of any emergency classification, **THEN** perform the following: .... ________

   a. **Obtain** copy of completed and approved Emergency Notification Form (ENF). .... ☐

Question 89

<< NRC Communications Checklist >>

1.2 Establish Control Room / NRC interface (continued)

b. Using the ENF, **notify** the NRC Operations Center by one of the following means: .... ☐

Correct Answer Part 2

- Primary: Emergency Notification System (ENS) Phone .... ☐
  - ◊ 1-301-816-5100

OR

- Alternate: Commercial Telephones .... ☐
  - ◊ 1-800-532-3469
  - ◊ 1-301-951-0550
  - ◊ 1-301-415-0550
  - ◊ 1-301-415-0553
  - ◊ 1-800-449-3694

AND

- Facsimile .... ☐
  - ◊ 1-301-816-5151

4. **IF** requested by the NRC Operations Center, **THEN maintain** continuous communications with the NRC Operations Center. .... _________

5. **Notify** the NRC of an upgrade in classification as soon as possible and within one hour. .... ☐

6. **WHEN** TSC is activated, **THEN provide** information necessary for turnover and **notify** Shift Manager/Emergency Coordinator (SM/EC) that TSC is activated and assuming NRC communication responsibilities. .... _________

7. **WHEN** TSC is activated, **THEN notify** NRC that the TSC is activated, assuming NRC communication responsibilities and provide contact information. .... _________

Question 89

<< NRC Communications Checklist >>

2.0 ONGOING ACTIONS

2.1 NRC Communications

1. **Provide** plant condition and event information to NRC as requested. .... ☐

2. **Notify** SM/EC of planned NRC activities (e.g., dispatch of a site team, specific requests). .... ☐

3. **Consult** with SME/EC for information about emergency and planned or in-progress mitigating actions to communicate to NRC. .... ☐

2.2 NRC Notifications

1. **Notify** the NRC of an upgrade in classification as soon as possible and within one hour. .... ☐

2. **Complete** NRC Event Notifications using site-specific procedure. .... ____/**TAB 4**

Distractor Part 2

   a. **Obtain** SM/EC review and approval of completed form. .... ☐

   b. **Transmit** form to the NRC. .... ☐

NOTE

The NRC Headquarters Operation Officer (HOO) receives information about an upgrade in classification. .... ☐

3. **IF** the HOO has left ENS Bridge Line,
THEN request for HOO to rejoin Bridge Line. ....________

Question 90

1.15 When paralleling a D/G to the Essential Switchgear, the in-service KC and/or NV Pumps are aligned to the opposite train to prevent the simultaneous loss of NC Pump Seal Injection and Thermal Barrier Cooling if a D/G failure results in the loss of the Essential Switchgear. If both KC and NV in-service Pumps are powered from the same bus as the D/G being tested, KC is swapped to the opposite train. (PIP 99-3510)

1.16 Do **NOT** operate the D/G at no load or light loads for long periods of time to prevent buildup of carbon and sludge in the engine.

Part 1 Correct Answer

1.17 LD heat trace is designed to only maintain the LD stagnant loop temperature after the stagnant loop is heated to the Tech Spec range. The LD heat trace **CANNOT** be used to heatup the LD system.

2. Initial Conditions

_____ 2.1 Verify D/G 1A is aligned per Enclosure 4.1 (Diesel Alignment for ES Actuation).

NOTE:
- Verifying lube oil temperature and jacket water temperatures at the engine inlets and outlets are 140-190°F helps ensure long term reliability of the engine. Under normal starting conditions lube oil inlet and outlet temperatures and jacket water inlet and outlet temperatures of 140-190°F is the preferred starting temperature. Operability temperature is 120-190°F. (PIP C-14-02352)
- The following readings can be accessed from the OAC (e.g., using OAC graphic DGAKDLD).

2.2 Verify lube oil temperature and jacket water temperatures at the engine inlets and outlets per one of the following conditions:

_____ • **IF** starting under normal plant conditions verify lube oil temperature and jacket water temperatures at the engine inlets and outlets are 140-190°F.

_____ • **IF** starting under abnormal or emergency plant conditions verify lube oil temperature and jacket water temperatures at the engine inlets and outlets are 120-190°F.

_____ 2.3 Verify the Control Room has been notified that a LOCAL START of D/G 1A is being performed.

Question 90

_____ 3.9 **IF** starting D/G 1A for Aux Safeguards Testing, start D/G 1A per PT/1/A/4200/009A (Auxiliary Safeguards Test Cabinet Periodic Test) to simulate an emergency signal.

_____ 3.10 **IF** performing a manual slow start, perform the following:

_____ 3.10.1 Establish a dedicated operator for the "SLOW START ENABLED" switch on 1DECPA.

Dedicated Operator ______________________________

_____ 3.10.2 **IF AT ANYTIME** while the "SLOW START ENABLED" switch is in the "ON" position, an emergency start signal is received, place " SLOW START ENABLED" switch in "OFF" position.

_____ 3.10.3 Place the "SLOW START ENABLED" switch on 1DECPA in the "ON" position.

_____ 3.10.4 Verify the "SLOW START ENABLED" light on 1DECPA is lit.

NOTE:
1. The "Turbo Oil Sol Vlv" will automatically close at 95% D/G speed.
2. Step 3.11 shall be performed during routine tests and starts where time permits. It shall be omitted during emergency situations due to the nature of the start. This step is to allow oil to be supplied to the turbochargers to prolong bearing life.
3. The following step will result in D/G 1A Annunciator Panel, B/6 "TURBO OIL SOL VLV OPEN" alarming.

CAUTION: The "Turbo Oil Sol Vlv" shall **NOT** be open for more than 5 minutes to prevent oil overflowing from the turbocharger bearing to the exhaust manifold potentially causing a turbocharger fire.

_____ 3.11 Unless directed otherwise, 3-5 minutes before engine start, place the "Turbo Oil Sol Vlv" switch on 1DECPA in the "OPEN" position.

_____ 3.12 Direct the Control Room Operator to announce the start of 1A Diesel Generator per AD-OP-ALL-1000 (Conduct of Operations).

NOTE: The following step will result in D/G 1A Annunciator Panel, F/2 "AUX EQUIP NOT IN AUTO".

_____ 3.13 Place "L. O. Pump & Heater" switch on 1DECPA in "RUN" position.

Distractor A.1 & B.1

Question 90

BACKGROUND (continued)

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Approximately 1 minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Part 2 Correct Answer

Ratings for Train A and Train B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref. 3). The continuous service rating of each DG is 7000 kW with 10% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

7000 KW + 10% = 7700 KW

APPLICABLE SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the UFSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the Accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of the onsite or offsite AC sources OPERABLE during Accident conditions in the event of:

a. An assumed loss of all offsite power or all onsite AC power; and

b. A worst case single failure.

The AC sources satisfy Criterion 3 of 10 CFR 50.36 (Ref. 6).

LCO

Two qualified circuits between the offsite transmission network and the onsite Essential Auxiliary Power System and separate and independent DGs for each train ensure availability of the required power to shut down

- Low Suction Pressure Protection for Manual Start of CA

Should the CA system be operating in manual or "CA SYS VLV" be RESET, the auto open feature is blocked and the CA pump that is running will trip. This trip feature operates on a time delay of five (5) seconds. The low pressure condition must exist continuously for 5 seconds to initiate a trip.

The Low Suction Pressure trips can be defeated by use of switches on the main control board. This action is done when aligning the CA suctions to the Hotwell per AP/1/A/5500/006 (Loss of S/G Feedwater).

The CA pumps' trip on low suction pressure is blocked in Case II of AP/1/A/5500/006 (Loss of Normal CA Supply) so that the hotwell can be used as a CA suction source. The following switches on MC10 are used to perform that function:

CA TD PMP Lo Suct Press Trip Block

CA PMP 1A Lo Suct Press Trip Block

CA PMP 1B Lo Suct Press Trip Block

7.2. CAPT OVERSPEED PROTECTION

Objective 9B, Licensed

- Electronic Overspeed

At 115% of Rated Speed, the CAPT Trip Throttle valve trips closed to stop the pump. The operator must re-open the valve by depressing the OPEN pushbutton on MC10 or at the Local Panel to reset the pump. The electronic overspeed indicating light will stay lit until the condition clears and the CAPT Trip Throttle valve is re-opened. Annunciator AD-5, E/3 "CAPT STOP VLV CLSD" will alarm when the CAPT Trip Throttle valve is closed.

- Mechanical Overspeed **Distractor B.2 & C.2**

At 125% of Rated Speed, the CAPT Trip Throttle valve trips closed and the linkage "opens" to stop the pump. For the mechanical overspeed trip the operator must at the CAPT manually move the trip linkage towards the CAPT Trip Throttle valve (away from the turbine); reference OP/1/A/6250/002 (Auxiliary Feedwater System).

The CAPT Trip Throttle valve may then be reopened manually by operating the valve's handwheel or locally at the CAPT control panel (located behind the turbine driven pump auxiliary shutdown panel) by depressing the OPEN pushbutton, or from the Control Room using the OPEN pushbutton on MC10.

7.3. LOSS OF VI

Objective 9C, Licensed

The discharge flow control valves are pneumatically operated valves provided with air accumulators for each valve to allow the valves to be remotely operated on a loss of VI. The accumulators are sized so the flow control valves can be operated for one hour to allow Operators enough time to manually secure flow to a S/G following a S/G Tube Rupture (SGTR) Design Basis Event in order to prevent overfill of the S/G.

RECOVERY/RESTORATION TECHNIQUE:

The objective of the recovery/restoration technique incorporated into guideline FR-H.1 is to restore and/or maintain adequate secondary heat removal capability and to establish NC system feed and bleed heat removal if secondary heat removal capability cannot be maintained.

The following subsections provide a summary of the major categories of operator actions and key utility decision points for guideline FR-H.1, Response To Loss Of Secondary Heat Sink.

MAJOR ACTION CATEGORIES IN E-0:

- Attempt Restoration of Feed Flow to Steam Generators

The operator attempts to restore or establish auxiliary feedwater flow, main feedwater flow, and condensate flow (in that order) while checking symptoms for a loss of secondary heat sink. Auxiliary feedwater flow restoration is attempted first and, if unsuccessful, NC pumps are tripped to extend the available time to establish feed flow from the main feedwater and condensate systems.

- Initiation of NC System Feed and Bleed Heat Removal

If symptoms for loss of secondary heat sink are reached, NC system feed and bleed heat removal is initiated through S/I actuation (feed path) and opening the pressurizer PORVs (bleed path). Feed and bleed heat removal is maintained until the secondary heat sink is reestablished and verified.

- Restore and Verify Secondary Heat Sink

After NC system feed and bleed heat removal is established, the operator continues attempts to restore narrow range level in at least one S/G. After level is established, the effectiveness of the secondary heat sink is verified by decreasing NC system temperatures.

- Termination of NC System Feed and Bleed Heat Removal

With a verified secondary heat sink, the operator performs a coordinated sequence for S/I flow reduction and closing of pressurizer PORVs. After the completion of the sequence, the operator is transferred to ES-1.1, SI Termination, for plant recovery.

Continued From Previous Page.

Question 91

KNOWLEDGE/ABILITY:

Step was simplified to quickly verify NV S/I flow, prior to opening Pzr PORVs. ECCS pumps and valve positions are not checked at this time. If NV S/I flow is indicated, pumps and valves are expected to be aligned to establish S/I flow. If NV S/I flow is not indicated then the RNO provides actions necessary to restore ECCS flow.

- Initiating Feed and Bleed is time critical. CNC-1552.08-00-0285 (McGuire and Catawba Nuclear Stations- All Units, Loss of Feedwater Feed and Bleed Analysis) assumes that feed and bleed will be initiated within 4 minutes of meeting initiation criteria, or within 8 minutes of Rx trip on Lo-Lo S/G Level. The intent of this step is to ensure that ECCS flow is established before Pzr PORVs are opened. This intent will be met by either NV S/I flow indication or, if initiation criteria occurs after 90 minutes, one NI pump running. NI pumps are normally aligned for S/I mode, so just checking the pump on is adequate.

- Note that this step is also intended to obtain maximum available ECCS flow. This intent is met by the previous step to initiate S/I (initiating S/I will start all available ECCS pumps and align valves), and by STEP 29, which will check all ECCS pumps and alignments.

- Note that it is not justified to delay opening Pzr PORVs (required to establish alternate decay heat removal) to first check all ECCS alignments. As long as ECCS flow from NV (or NI if 90 minute requirement met) has been established, it is acceptable to open Pzr PORVs (per ERG background document). After the time critical action to establish feed and bleed is completed, it is then appropriate to check for maximum ECCS capability. Delaying time critical actions to address the possibility of multiple failures on both ECCS trains during an event involving loss of both trains of aux feedwater is not appropriate. This is based on operator response time assumed in plant specific analysis.

Time Critical Operator Action sequence - See STEP 26.

Distractor Part 1

Correct Answer Part 1

Question 91

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 36. **Verify KC flow to ND heat exchangers - INDICATING FLOW.**	__ **IF AT ANY TIME ND pump operating with flow less than 1000 GPM to NC loops AND KC to associated ND HX isolated, THEN stop affected ND pump within 3 hours.**
37. **Align CA to establish control of S/G feed as follows:** __ a. Ensure CA System valve control - RESET. __ b. CLOSE CA flow control valves on S/Gs not presently being fed.	
38. **Continue attempts to establish secondary heat sink in at least one S/G as follows:** __ • CA flow. **REFER TO** Steps 6 through 7 __ • CF or CM flow. **REFER TO** Steps 10 through 18	

NOTE RY System should not be used unless all other feedwater sources are unavailable.

__ • RY flow. **REFER TO** Step 19.	
__ 39. **Verify N/R level in at least one S/G - GREATER THAN 11% (29% ACC).**	__ **RETURN TO Step 38.**
40. **Verify NC System temperatures as follows:** __ • Core exit T/Cs - TRENDING DOWN __ • NC T-Hots associated with S/Gs being fed - TRENDING DOWN.	__ **RETURN TO Step 38.**

Catawba Nuclear Station

ILT16 CNS SRO NRC Examination

Question: 76
(1 point)

Given the following conditions on Unit 2:

- Unit initially operating at 100% RTP
- Containment pressure rapidly increased to 3.4 PSIG
- 2B S/G completely depressurized
- Automatic reactor trip did not occur and manual trip from the control room was unsuccessful
- OATC is manually inserting control rods per EP/2/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS)

Current conditions:

- While inserting rods, the OATC notes the following:
  - DRPI indication for Control Bank ‘D’ rod M4 is YELLOW with a DATA ‘B’ indication above the rod
  - DRPI indication for rod M4 indicates 198 steps
  - Demand position counters for Groups 1 & 2 are blinking with position indication of 210 steps

In accordance with T.S. 3.1.7 (Rod Position Indication) bases, LCO 3.1.7 _____(1)_____ met.

When reactor shutdown has been verified, the NEXT required procedure transition from FR-S.1 is to GO TO _____(2)_____.

Which ONE (1) of the following completes the statements above?

A. 1. is
2. E-0 (Reactor Trip or Safety Injection)

B. 1. is NOT
2. E-0 (Reactor Trip or Safety Injection)

Original Question

C. 1. is
2. E-2 (Faulted Steam Generator Isolation)

MODIFIED

D. 1. is NOT
2. E-2 (Faulted Steam Generator Isolation)

Question 92

APPLICABLE SAFETY ANALYSES (continued)

worth, and with at least minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that one DRPI System and one Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

a. The DRPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits";

b. For the DRPI System either Data A or Data B is OPERABLE for each rod; and

c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore,

Question 92

6.4 **IF** a control rod fails to withdraw or a single rod is dropped during approach to criticality perform one of the following:

Correct Answer Part 2

- **IF** malfunction is in Control Bank, insert all Control Banks.
- **IF** malfunction is in Shutdown Bank, insert all Control and Shutdown Banks.

This guidance is more conservative than that given in AP/1(2)/5500/015, Rod Control Malfunction, and therefore shall take precedence. {PIP C-06-4287} Distractor Part 2

6.5 **IF** more than one control rod drops during the approach to criticality, **MANUALLY TRIP** the reactor per AP/1(2)/A/5500/14, Control Rod Misalignment.

6.6 **IF** an alarm is received on the Rod Control System or DRPI requiring rod withdrawal to be halted **AND** IAE cannot determine cause for alarm and repair problem or determine if further rod withdrawal is permissible, reinsert all Control Banks.

- **IF** Startup is **NOT** xenon-free (xenon worth ≥ 100 pcm), take action within 30 minutes of malfunction.
- **IF** startup is xenon-free (xenon worth < 100 pcm), take action within 60 minutes of malfunction.
- Obtain ICRR data at 10-minute intervals for the duration of the delay.

6.7 To reduce uncertainties in achieving criticality, T-AVG shall be maintained between 555 and 559 °F during approach to criticality.

6.8 **IF** diluting with BDMS enabled, periodically monitor and reset the BDMS actuation setpoint.

6.9 **IF** abnormal changes in count rate (i.e., irregular count rates, instrument drift, etc.) are observed on either Source Range or BDMS detector, rod withdrawal shall be suspended. Rod withdrawal may be resumed only after the source of the abnormality has been identified and it has been determined that it will not jeopardize plant safety.

6.10 **IF** it is expected that criticality will be achieved above the Upper Allowable Limit (UAL)/Rod Withdrawal Limits **OR** below the Lower Allowable Limit (LAL)/Rod Insertion Limits, insert all control banks and contact Reactor Systems Engineering Supervisor or designee.

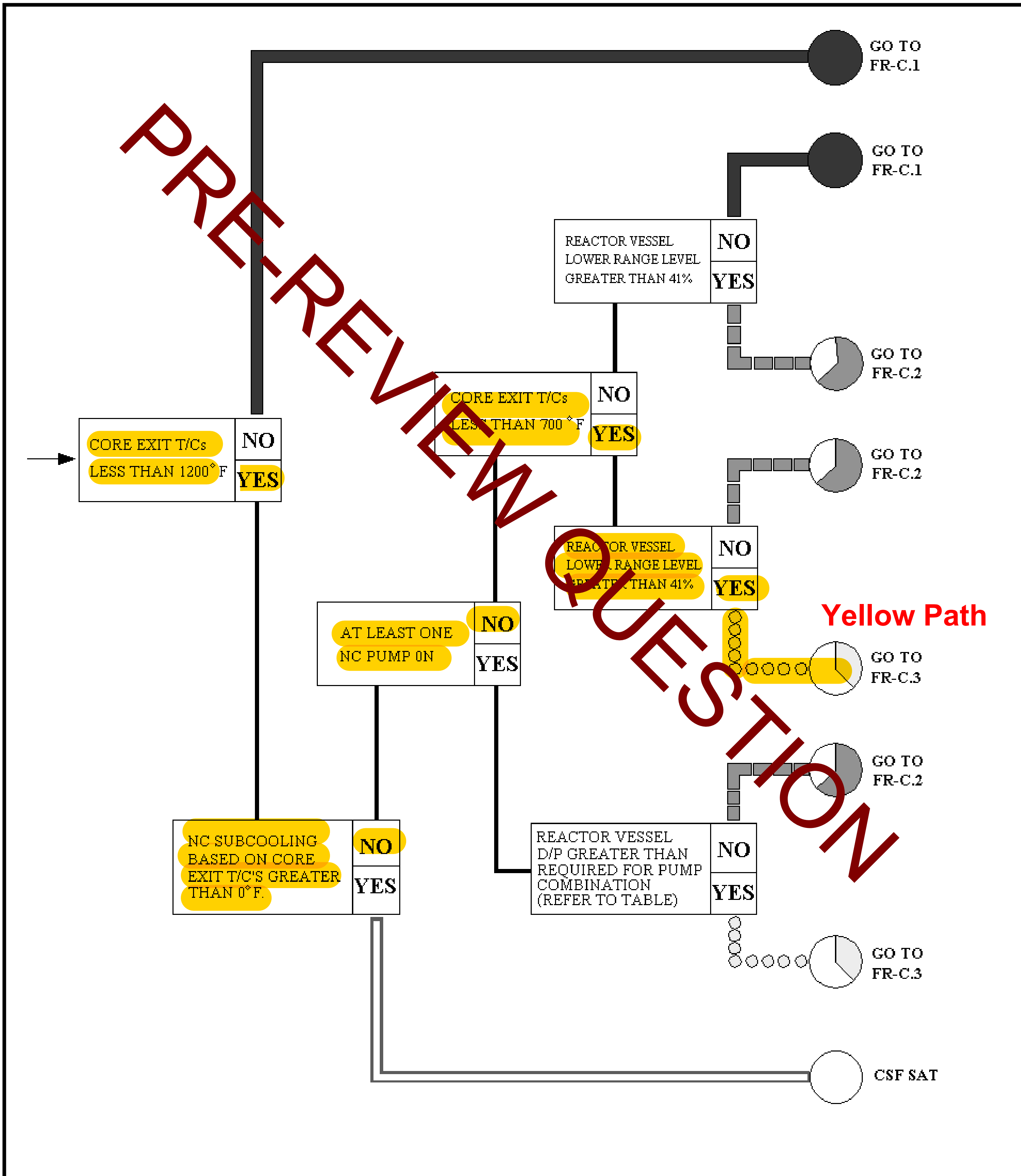
6.11 Ensure that the NC boron sample used for reactivity balance calculations is representative of current NC system boron (i.e. taken with all four NC pumps operating, sufficient time allowed for mixing after last boron change, etc.).

6.12 Prerequisite steps in Sections 4, 7, and 8 may be signed off in any order.

Question 93

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES CORE COOLING - Page 1 of 1	PAGE NO. 4 of 11 Revision 9

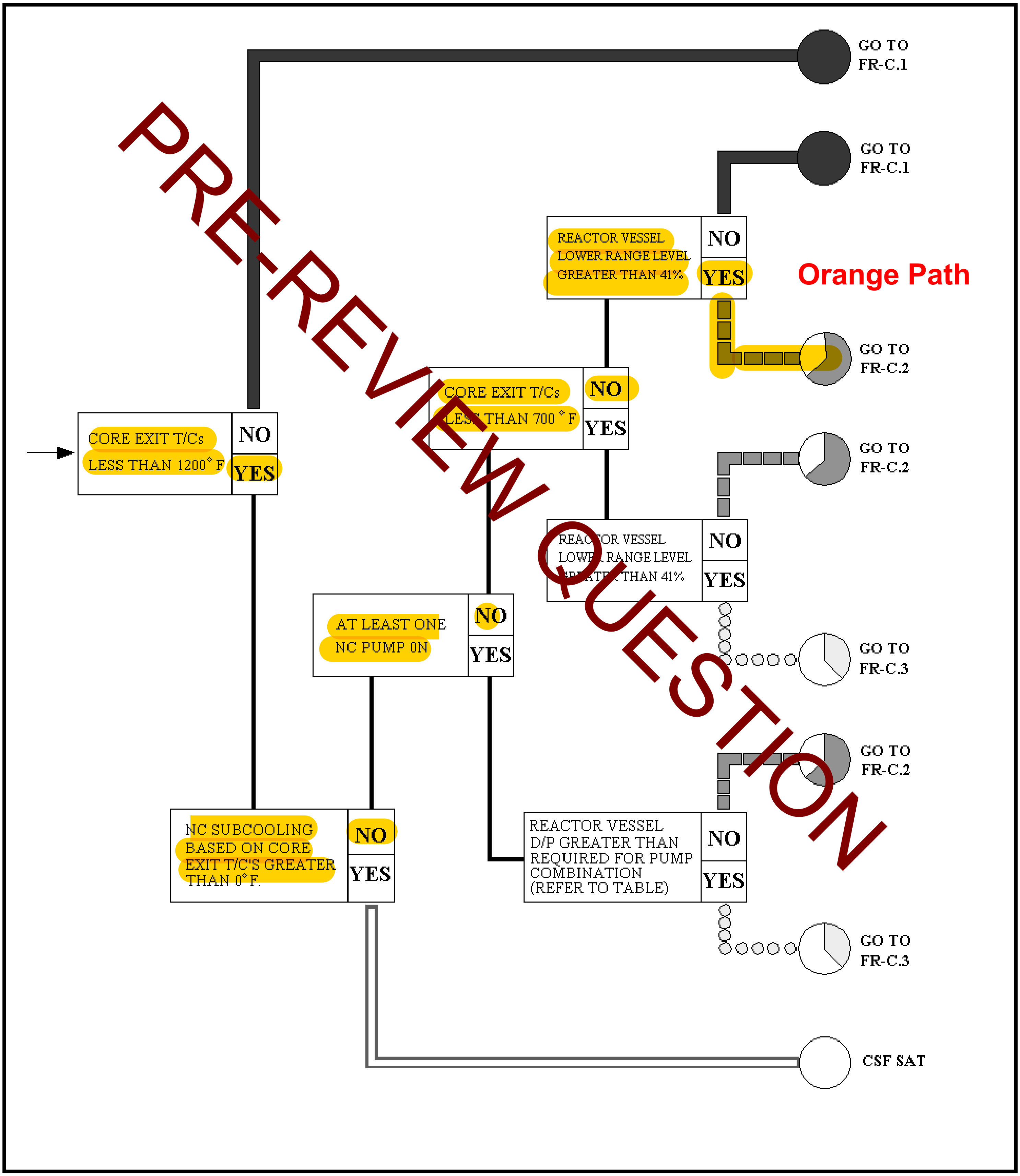
Given Indications at 0930



Question 93

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES CORE COOLING - Page 1 of 1	PAGE NO. 4 of 11 Revision 9

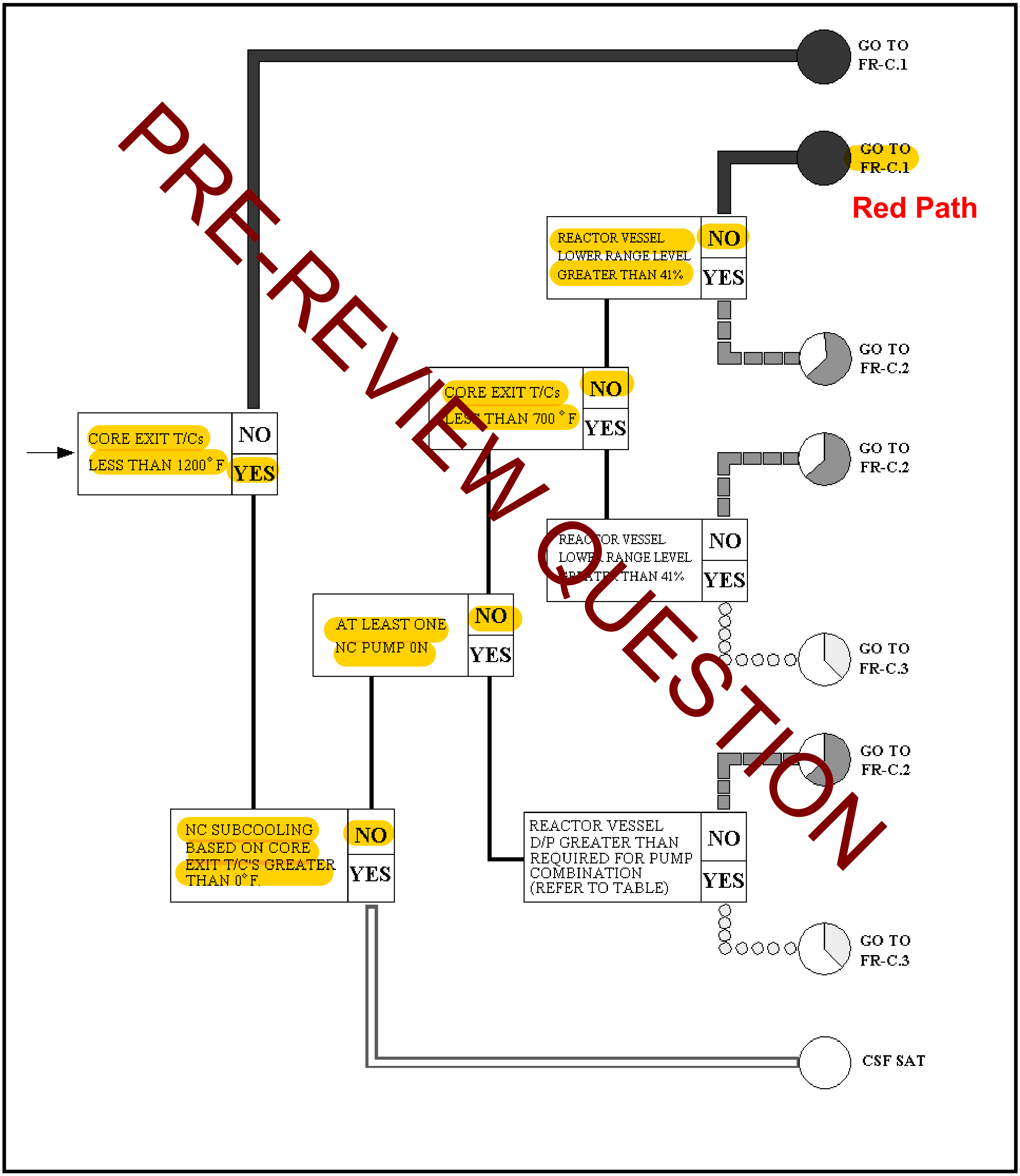
Given Indications at 0945



Question 93

CNS EP/1/A/5000/F-0	CRITICAL SAFETY FUNCTION STATUS TREES CORE COOLING - Page 1 of 1	PAGE NO. 4 of 11 Revision 9

Given Indications at 1015



Question 93

F Fission Product Barriers	FG1.1 1 2 3 4	FS1.1 1 2 3 4	FA1.1 1 2 3 4
	Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1)	Loss or potential loss of **any** two barriers (Table F-1)	**Any** loss or **any** potential loss of either Fuel Clad or NCS (Table F-1)
	Correct Classification at 1015	Correct Classification at 0945	Initial Alert Classification 0910

Time After S/D	FC Loss	NCS Loss	CMT Potential Loss
0 - 1	550	8.8	5500
1 - 2	400	8.4	4000
2 - 8	160	7.0	1600
>8	100	6.2	1000

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (NCS) Barrier		Containment (CMT) Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A. NCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER**: • UNISOLABLE NCS leakage • SG tube RUPTURE	1. CSFST Integrity-RED PATH conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B. Inadequate Heat Removal	1. CSFST Core Cooling-RED PATH conditions met	1. CSFST Core Cooling-ORANGE PATH conditions met 2. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED PATH conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C. CMT Radiation / NCS Activity	1. EMF53A/B > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. EMF53A/B > Table F-2 column "NCS Loss"	None	None	1. EMF53A/B > Table F-2 column "CMT Potential Loss"
D. CMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of NCS leakage outside of containment	1. CSFST Containment-RED PATH conditions met 2. Containment hydrogen concentration > 6% 3. Containment pressure > 3 psig with < one full train of containment cooling operating per design for > 15 min. (Notes 1, 10)
E. EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Question 94

5.2 Procedure Compliance Standards During Abnormal and Emergency Operations (continued)

e. If a Variance is utilized, then at a minimum, enter the following relevant information into the Operations Narrative Logs:

- Variance description
- Any SRO concurrence obtained
- Basis for the Variance (e.g., public health and safety, personnel injury, or plant equipment damage)

f. Ensure necessary measures to maintain Plant Status Control are performed.

g. Initiate a NCR to evaluate the effectiveness of the transient response procedures, including Variances implemented during simulator scenarios.

(1) Ensure the word "Variance" is in the NCR subject to support NCR trending.

7. Deviations - 10 CFR 50.54x Departure from Licensed Condition for Protection of Public Health and Safety

a. Deviations from Technical Specifications or License Condition during normal plant operation are **NOT** allowed.

(1) In the event of an emergency, the SM has the authority and responsibility to take action necessary to protect the health and safety of the public as allowed by 10 CFR 50.54(x), 10 CFR 50.54(y), and 10 CFR 72.

b. In order to invoke a Deviation, no action consistent with License Conditions and Technical Specifications that can provide adequate or equivalent protection is immediately apparent.

c. If the TSC is activated, then the TSC Emergency Coordinator determines to depart from license conditions.

(1) If emergency responsibilities are assumed, then the TSC Emergency Coordinator has higher authority for command and control.

(2) Prior to exiting the license conditions, the TSC Emergency Coordinator should consult with a SRO, but the TSC Coordinator is **NOT** required to be a SRO and does **NOT** require SRO concurrence for departing license conditions

Question 94

5.2 Procedure Compliance Standards During Abnormal and Emergency Operations (continued)

d. If the TSC is **NOT** activated, then a SRO must approve a Deviation prior to taking action. If time allows, then concurrence from a second SRO should be obtained prior to taking action.

e. Enter the relevant information concerning the Deviation into the Unit Log as soon as conditions allow.

f. Notify the NRC per 10 CFR 50.72 requirements due to invoking 50.54(x).

(1) Notify the NRC Operations Center using the Emergency Notification System (ENS).

(2) When time permits, then notify the NRC Operations Center prior to taking the action.

(3) Otherwise, notify the NRC Operations Center as soon as possible and in all cases, within 1 hour after taking the action.

g. Notify the Resident NRC Inspector as soon as practical.

h. Submit an NCR to review the circumstances and determine whether a change to the procedure is required to prevent a Deviation in the future.

(1) Include the word 'Deviation' in the NCR title to support trending.

8. Deviation - 10 CFR 50.54(dd) Departure from Licensed Condition for National Security

a. In the event of a national security emergency, the SM has the authority to take action necessary to implement national security objectives as stated in 10 CFR 50.54(dd) which reads as follows:

(1) A licensee may take reasonable action that departs from license condition or a technical specification in a national security emergency when this action is immediately needed to implement national security objectives as designated by the national command authority through the NRC.

AND

(2) No action consistent with license conditions and technical specifications that can meet national security objectives is immediately apparent.

Question 95

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	$\geq$ 0.99	> 5	NA
2	Startup	$\geq$ 0.99	$\leq$ 5	NA
3	Hot Standby	< 0.99	NA	$\geq$ 350
4	Hot Shutdown[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	$\leq$ 200
6	Refueling[c]	NA	NA	NA

Distractor Part 1

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

Question 95

Catawba 1 Cycle 26 Core Operating Limits Report

2.16 Standby Shutdown System - (SLC-16.7-9)

2.16.1 Minimum boron concentration limit for the spent fuel pool required for Standby Makeup Pump Water Supply. Applicable for MODES 1, 2, and 3.

Parameter	Limit
Spent fuel pool minimum boron concentration for TR 16.7-9-3.	2,700 ppm

2.17 Boration Systems Borated Water Source – Shutdown (SLC 16.9-11)

2.17.1 Volume and boron concentrations for the Boric Acid Tank (BAT) and the Refueling Water Storage Tank (RWST) during MODE 4 with any RCS cold leg temperature ≤ 210 °F, and MODES 5 and 6.

Distractor Part 2

NOTE: When cycle burnup is ≥ 445 EFPD, Figure 6 may be used to determine the required BAT Minimum Level.

Parameter	Limit
BAT minimum boron concentration	7,000 ppm
Volume of 7,000 ppm boric acid solution required to maintain SDM at 68°F	2,000 gallons
BAT Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-11)	Correct Answer Part 2 13,086 gallons (14.9%)
RWST minimum boron concentration	2,700 ppm
Volume of 2,700 ppm boric acid solution required to maintain SDM at 68 °F	7,000 gallons
RWST Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-11)	48,500 gallons (8.7%)

Question 95

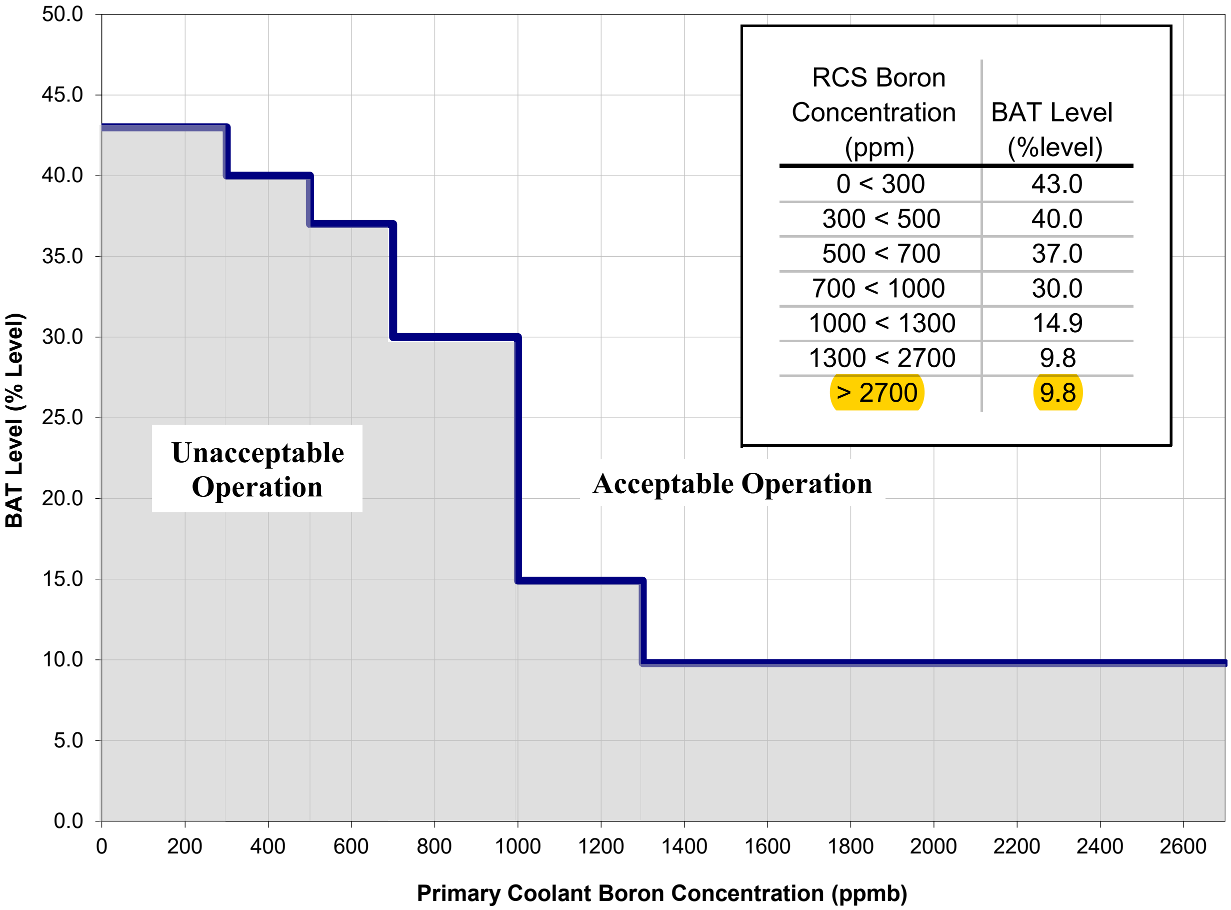
Catawba 1 Cycle 26 Core Operating Limits Report

Figure 6

**Boric Acid Storage Tank Indicated Level Versus
Primary Coolant Boron Concentration**

(Valid When Cycle Burnup is ≥ 445 EFPD)

This figure includes additional volumes listed in SLC 16.9-11 and 16.9-12



RCS Boron Concentration (ppm)	BAT Level (%level)
0 < 300	43.0
300 < 500	40.0
500 < 700	37.0
700 < 1000	30.0
1000 < 1300	14.9
1300 < 2700	9.8
> 2700	9.8

Distractor Part 2

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding to approximately 75 percent of its required level and then using the refueling water pump to complete the filling process or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

Question 96

4.6 Shift Manager

1. Determines appropriateness of the use of clearances in times of emergency.

2. Ensures plant status control is maintained in accordance with AD-OP-ALL-0204, Plant Status Control, and this procedure.

Correct Answer Part 2

3. Approves Exceptional Clearances along with a Work Group Supervisor.

4.7 Clearance Approver

1. Maintains an active or current SRO license.

Distractor Part 2

2. Approves EEI's, and the hang, temp lifts, removal of clearances.

3. Maintains Control Room personnel informed of all plant status changes prior to establishing or removing a clearance.

4. Ensures closeout of clearances.

5. Ensures the impact of clearances on the station is understood.

4.8 Clearance Requestor

1. Requests a clearance in accordance with the guidance of this procedure and AD-MN-ALL-0005, Nuclear Planning.

2. Typically this role is performed by the Maintenance Planner, with support from the craft supervisor as described in the Work Order Planning Process. However, other personnel may request a clearance.

4.9 Clearance Preparer

1. Designs the protection boundary for the work.

2. Verifies the clearance request contains a scope statement of sufficient detail to communicate the exact work scope to be included in the clearance boundary.

3. Applies systems and component knowledge, as well as knowledge of procedures, operating experience, and regulations to develop the clearance boundary.

4. Contacts the Clearance Requestor or Work Group Supervisor as necessary to ensure understanding of the work scope and clearance boundary needs.

Question 96

5.6 Clearance Reviewer

1. For Active Clearances being reviewed from a Model Clearance, use the guidance in Section 1.0 of Attachment 5, Clearance Hang Preparation Checklist. For all other Clearances or new Model Clearance generation continue in this section.

   a. If the Active Clearance is being reviewed from a Model and does not meet any of the reviewer criteria in Section 1.0 of Attachment 5, Clearance Hang Preparation Checklist, then the Active Clearance shall be deleted or prepared and reviewed using section 2.0 if Attachment 5. Additionally, action shall be taken to revise the Model.

2. The purpose of the clearance review is to ensure that the design of the clearance is technically correct, that good isolation practices have been used, and that appropriate procedural requirements have been met.

   a. The clearance reviewer is charged with making an informed decision about the technical accuracy and safety adequacy, in all respects, of the clearance that is presented for review.

   b. The clearance reviewer shall maintain an independent perspective on the clearance.

   Distractor Part 1

   c. For all clearances (except Keowee), the review shall be performed by a licensed operator or a previously licensed operator at that station.

      (1) For Keowee Operations clearances, the review shall be performed by a Keowee Operations Tech 3 or exempt Keowee personnel.

3. A detailed understanding of the system is needed by the clearance reviewer to identify parallel flowpaths, remote controlling locations, electrical backfeed sources, proximity to energized sources, multiple power sources, and so on. When in doubt, the appropriate technical help should be sought. In making an informed decision about the clearance, the following techniques are ways to ensure an adequate technical review by improving the independence of the clearance reviewer's perspective:

   a. Do not discuss the clearance with the clearance preparer until the clearance reviewer has developed his or her own idea of the clearance boundary.

   b. References used by the clearance preparer may be used, but no marked up items are addressed until after a review of the task and the creation of an independent boundary has been completed. Previously marked up prints may be used for review of outage block/master clearances.

   c. Independently assess and understand the hazards.

Question 97

ATTACHMENT 4
Page 1 of 3

<< Defense-In-Depth (DID) Status Sheet Guidelines >>

NOTE

The Defense In Depth Sheets combine the generic needs and risk analysis methods common to all sites. The site specific DEFENSE-IN-DEPTH Basis documents give site specific details to guide Operators and Work Control personnel to correctly assess and manage risk for that site. Compliance with the defense in depth plan is verified at least once per shift and before major safety systems or components are removed from service that affects the Defense-in-Depth sheet. {7.1.5}

1.0 GUIDANCE FOR DEFENSE-IN-DEPTH (DID) STATUS SHEETS

1. During outage development, Site Outage Management will develop DID sheets for unique configurations for review during the Independent Review Team outage risk assessment using Attachment 2, [BWR] Defense-in-Depth (DID) Status Sheet.

2. When entering Mode 4 (PWR) or Mode 3 (BWR), the risk will be analyzed using the DID sheet.

3. During an outage an Outage Risk Analyst will generate a DID sheet for each of the following:

   a. Prior to entering a new configuration to ensure risk will be acceptable.

   b. When an un-scheduled change occurs such as equipment failure or new external event.

   c. Once every 12 hours (per shift).

4. During Outage planning and before entering Mode 4 (PWR) or Mode 3 (BWR), the Shift Manager and Work Control Outage Manager shall review the configuration requirements and notes. {7.1.7}

   a. The Shift Manager and Work Control Outage Manager then concur that the status of plant equipment agrees with the planned shutdown Risk Management configuration and the Shift Manager will direct mode entry. {7.1.10}

5. Before manipulating the plant into the next configuration, the Shift Manager will review and approve the DID status. {7.1.7} {7.1.10} {7.1.11}

Question 97

5.4 Plant Condition Mode Changes (PCMC) Coding

1. The purpose of PCMC coding is to aid tracking and review of the status of equipment required by Technical Specifications (TSs), Selected Licensee Commitments (SLCs), or other mode dependent commitments for specific plant conditions or modes to assist Operations in verifying conditions to make a Mode Change. [7.3.2]

   Distractor Part 2

2. During schedule development and revision all WOTs shall be PCMC coded.

3. The following are the required minimum conditions to apply PCMC codes. Additional PCMC codes can be used in conjunction with these if needed.

   - **MODE / Plant Condition Not Applicable:** The value entered in CAS is 0 to show the WO had been reviewed as N/A to Mode or Plant Condition requirements.

   - **MODE 1:** The work must be completed before Mode 1 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 1.

   - **MODE 2:** The work must be completed before Mode 2 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 2.

   - **MODE 3:** The work must be completed before Mode 3 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 3.

   - **MODE 4:** The work must be completed before Mode 4 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 4.

   - **MODE 5:** The work must be completed before Mode 5 up. The equipment affected by this WO is required for operability by TSs or SLCs in MODE 5.

   - **MODE 6 Fuel Movement:** Work that must be completed before initiating fuel movement.

   - **MODE 6 Core Alterations:** The work must be completed prior to beginning core alterations in MODE 6.

   - **Less than 23 Feet in the Refueling Cavity:** The work must be completed before draining below 23 feet in the refueling cavity.

   - **Low Temperature Overpressure Protection (LTOP):** The work must be completed before entering LTOP.

Question 97

5.4 Plant Condition Mode Changes (PCMC) Coding (continued)

- **Mode 3T:** Turbine Driven Auxiliary Feedwater Pump-Testing that must be completed in MODE 3 at the appropriate steam pressure before declaring the Turbine Driven Auxiliary Feedwater Pump fully operable. All other work and testing on the Turbine Driven Auxiliary Feedwater pump that can be completed independent of the pump IWP must be completed prior to MODE 3.

- **Mode 5T - Rod Control:** The work must be completed prior to making the Control Rods capable of being withdrawn. This will include work on rod position indication, Source Range Instrumentation, Automatic Trip and Interlock Logic, Manual Reactor Trip, and Reactor Trip Breakers.

4. The Operations SOCs, a Senior Reactor Operator (SRO), or individuals designated by Operations Management shall review all required WOTs to be worked during an outage and apply proper PCMC codes. [7.3.1]

   a. PCMC is not intended to track operability of equipment required for the current plant condition or mode. [7.3.2] **Correct Answer Part 2**

   b. The AOM-Work Control, a SRO, or designated persons by Operations Management shall review all WOTs to be work worked during an outage to ensure proper PCMC coding has been applied.

   c. This applies to WOTs coded prior to the outage start as well as scope identified during the outage time period.

5. Operations SOCs or an SRO shall perform a second review of PCMC codes of WOTs that are amended after the initial review to ensure proper PCMC coding. [7.3.1]

5.5 <u>Outage Window/Segment Templates</u>

1. Establish site refueling outage window/segment templates at each site with a goal of standardizing window/segment durations and work practices across the fleet to the greatest extent feasible to support fleet outage coordination.

2. Schedule templates will be the minimum Critical Path schedule to shutdown, offload fuel, shuffle fuel, reload fuel, heat up, and perform required Technical Specification testing. No other work is assumed.

3. Base schedule templates and activity durations with plants of similar design on best times achieved across the fleet verses best times at a particular site.

Question 98

TESTING REQUIREMENTS

---NOTE---
The Liquid Radwaste Treatment System shall be demonstrated FUNCTIONAL by meeting SLC 16.11-1 and SLC 16.11-3.

TEST		FREQUENCY
TR 16.11-4-1	Project liquid release doses from each unit to UNRESTRICTED AREAS, in accordance with the methodology and parameters in the ODCM, when the Liquid Radwaste Treatment System is not being fully utilized.	31 days

BASES

The FUNCTIONALITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept “as low as is reasonably achievable”. This COMMITMENT implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

This SLC applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared radwaste treatment systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the radwaste treatment system. For determining conformance to COMMITMENTS, these allocations from shared radwaste treatment systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

REFERENCES

1. Catawba Offsite Dose Calculation Manual.
2. 10 CFR Part 50, Appendix A.

Question 98

2.8 Release flowrates will be limited to ensure the Boron concentration at outfall 001 is less than or equal to 12 ppm boron. (PIP C-99-1682)

2.9 If pH is greater than or equal to 9.3 AMT **CANNOT** be released. (PIP C-03-1508)

2.10 If pH is greater than 9.0 but less than or equal to 9.3 release flow rates will be limited so as **NOT** to exceed the pH limit at outfall 001. (PIP C-03-1508)

2.11 Releases with greater than or equal to 500 ppm Boron shall be limited to 3 hours in a 24 hour period.

2.12 Liquid waste release tanks containing boron **CANNOT** be discharged for more than three hours in any 24-hour period without special approval. (PIP C-99-1682)

2.13 Release flow rates shall be greater than 100 gpm and less than 250 gpm.

2.14 If a Site Assembly occurs during a release, the release shall be secured for the duration of the assembly, unless dispatched from the OSC to make release.

2.15 All samples should be evaluated for excessive foaming in accordance with LM/C/P061 (Foam in Aqueous Media (Blender Method)). There is an NPDES requirement of **NO** visible foam on the discharge canal. (PIP-C-94-1299)

2.16 Manipulation of MTB CTRL Mode Sel Switch (0ELCP0022) remote/local control is allowed anytime during procedure.

2.17 Cooling Tower Blowdown can affect dilution flow during release. Environmental Chemistry shall be contacted prior to each release.

2.18 Releases that are interrupted by 0EMF-57 "Hi-Rad Discharge Trips" may be initiated a maximum of three times (including original initiation) before terminating the release. without resampling per HP/0/B/1004/004 (Radioactive Liquid Waste Release).

3. Procedure

Distractor Part 1

Refer to Section 4 (Enclosures)

Question 99

5.1.5 Reactor Building Entry (Modes 3 & 4) Distractor Part 1

- Access is normally controlled by the WCCSRO.
- Control may be transferred to another responsible group if that group posts a person at the personnel hatch.
  - This person shall be responsible for maintaining the Containment Access Log Book and to account for persons inside Containment.
- Each time responsibility for Containment access is transferred to another group from or to Operations, the date, time and name of the other group shall be entered in the Unit Unified Logbook.

5.1.5.1 Upper Containment

A. Entry into Upper Containment requires pre-job authorization for each job from the Radiation Protection Supervisor or designee.

B. Dependent on location of the Containment Access Log Book, personnel entering Upper Containment shall notify the WCCSRO or Personnel Hatch Watch.

C. A Task Preview shall be given by WCCSRO or Personnel Hatch Watch.

5.1.5.2 Lower Containment

A. Entry into Lower Containment requires pre-job authorization for each job from Radiation Protection Management dependent upon Radiation Protection postings/survey results.

B. Dependent on location of Containment Access Log Book, personnel entering Lower Containment shall notify the WCCSRO or Personnel Hatch Watch.

C. A Task Preview shall be given by WCCSRO or Personnel Hatch Watch.

Question 99

BASES

APPLICABLE SAFETY ANALYSES (continued)

containment was designed with an allowable leakage rate of 0.30% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.30% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 14.68 psig following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is

ACTIONS (continued)

preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

Note that for the purpose of Required Action A.1, A.2 and A.3, the bulkhead associated with an air lock door is considered to be part of the door. For example, an air lock door may be declared inoperable if the equalizing valve becomes inoperable or if it is replaced. It is appropriate to treat the associated bulkhead as part of the door because a leak path through the bulkhead is no different than a leak path past the door seals.

- If a valid orange path is encountered, the operator is expected to scan all of the remaining trees, and then, if no red path is encountered, to promptly implement the corresponding EP. If during the performance of an orange path procedure, any red condition or higher priority orange condition arises, then the red or higher priority orange condition shall be addressed first, and the original orange path procedure suspended.

- Once a procedure is entered due to a valid red or orange condition, that procedure shall be performed to completion unless preempted by some higher priority condition. It is expected that the actions in the procedure will clear the red or orange condition before all the operator actions are complete. However, these procedures shall be performed to the point of the defined transition to a specific procedure. At this point, any lower priority red or orange paths currently indicating or previously started but not completed shall be addressed.

- If a CSF procedure directs the operator to return to the procedure and step in effect and the corresponding status tree continues to display the off normal condition, then the corresponding CSF procedure does not have to be implemented again since all recovery actions have already been completed. However, if the same status tree subsequently changes to a valid higher priority condition, then the corresponding CSF procedure shall be implemented as required by its priority.

- Certain CSF procedures are used to address both orange and red path conditions for the same parameters. If the procedure is already in progress due to the orange path condition, it is not required to return to the first step if the condition becomes red. Also, at the completion of the procedure, the procedure does not have to be implemented again, since all recovery actions have already been implemented.

2021 INITIAL LICENSE NRC EXAM
SCENARIO # 1

Facility: Catawba NRC Exam 2021	Scenario No.: 1	Op Test No.: 2021301	
Examiners:		Operators:	SRO
			RO
			BOP
Initial Conditions: Unit 1 is at 50% power at the MOL. Unit 2 is at 100% power.			
Turnover: Unit 1 is at 50% power at the MOL. Unit 2 is at 100% power. Direction for the crew is to raise reactor power to 100%. A reactivity plan has been provided by Reactor Engineering for raising reactor power.			

Event No.	Malf. No.	Event Type*	Event Description
1		R – RO N – BOP N – SRO	Raise Reactor Power
2		C – BOP C – SRO	1KC-132 Fails closed
3		C – BOP C – SRO TS – SRO	1KC-C37A Fails Open
4		C – RO C – SRO	Loss of both KG pumps with failure of auto turbine runback
5		TS – SRO	1A NS Pump Loss of Power
6		C – RO C – SRO	Steam Leak
7		M – ALL	Steam Line Break Inside Containment
8		C – RO C – SRO	MSIVs fail to auto close on Main Steam Isolation
9		C – BOP C – SRO	VA Unfiltered Exhaust Fans fail to secure on S/I

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Scenario 1 – Summary

Initial Condition
Unit 1 is at 50% power at the MOL. Unit 2 is at 100% power.

Turnover:
Unit 1 is at 50% power at the MOL. Unit 2 is at 100% power. Direction for the crew is to raise reactor power to 100%. A reactivity plan has been provided by Reactor Engineering for raising reactor power.

Event 1
BOP will perform an initial dilution and RO will input desired load rate and target load into the main turbine. RO may also withdraw control rods prior to placing the main turbine in 'GO' to initiate the raise in power.

Event History: 50% power increase at MOL last used 13 (2).

Event 2
Letdown Heat Exchanger Temperature Control Valve (1KC-132) setpoint increases causing 1KC-132 to close and causing a letdown heat exchanger outlet high temperature alarm. Crew will refer to Annunciator Response Procedure for 1AD-7 F/3 (Letdn HX Outlet Hi Temp), place 1KC-132 in Manual, and adjust CCW flow to restore letdown cooling.

Verifiable Action – BOP will manually control 1KC-132 as required.

Event History: This failure last used 19 (1).

Event 3
The Unit 1 'A' train KC miniflow valve 1KC-C37A will fail open. The crew will enter AP/1/A/5500/021 (Loss of Component Cooling Water) to start an additional KC pump. TS evaluation by the SRO is required.

Verifiable Action – The BOP will start an additional KC pump as directed.

Event History: KC miniflow valve failure last used 16 (2).

Event 4
The running KG (Generator Stator Cooling Water) pump will trip and the standby KG pump will fail to start requiring a turbine runback. Automatic runback will be failed requiring the RO to manually runback the main turbine. Crew will enter AP/1/A/5500/003 (Load Rejection) to address the runback. Once the manual runback is complete, the standby KG pump will start.

Verifiable Action – RO will manually runback the turbine to ~ 275 MWe. Crew will stabilize the plant at ~ 25% reactor power, and place control rods to manual once the steam dumps have been closed (Tavg ~ 3°F higher than Tref).

Event History: This failure last used 14 (3).

Event 5
A loss of power will occur to the 1A NS Pump. The SRO will determine appropriate TS.

Event History: This failure has not previously been used on an NRC exam.

Event 6
A steam leak inside containment will develop from the 1C S/G. The crew will enter AP/1/A/5500/028 (Steam Leak) to address.

Verifiable Action – RO will adjust turbine load to maintain reactor power stable. Crew will initiate makeup to Unit 1 Upper Surge Tank. BOP will start all Lower Containment Ventilation Units in low speed, start all Upper Containment Ventilation Units, and place all Lower and Upper Containment Ventilation Units in "MAX" cooling mode.

Event History: This steam leak malfunction allowing use of AP-28 is new.

Event 7
Steam leak size will increase requiring the crew to enter EP/1/A/5000/E-0 (Reactor Trip or Safety Injection) and eventually transition to EP/1/A/5000/E-2 (Faulted Steam Generator Isolation). Crew will isolate the faulted S/G. Once 1C S/G has completely blown down, the RO will throttle S/G PORVs on intact S/Gs to stabilize NC System Thots.

Verifiable Action – BOP will isolate feed flow to the faulted S/G. RO will throttle S/G PORVs on intact S/Gs to stabilize NC System Thots.

Event History: The last steam line break inside containment was performed in 19 (3) but was on 1B S/G.

Event 8
Once steam generator pressures lower to 775 PSIG, a Main Steam Isolation signal will be generated, but the Main Steam Isolation valves will fail to automatically close, requiring the RO to manually close them.

Verifiable Action – RO will manually close the MSIVs on Unit 1.

Event History: CNS has had single MSIVs failing to automatically close on previous exams (last one in 19 (3) – 1SM-5), but the failure of all MSIVs to close automatically on the MSI signal with manual closure of these valves possible has not been used before.

Event 9
Auxiliary Building Ventilation Unfiltered Exhaust Fans will fail to automatically secure following the Safety Injection.

Verifiable Action – BOP will manually secure the Unfiltered Exhaust Fans.

Event History: This failure was last used on 16 (3).

Manual Control of Automatic Functions		
Event	**Position**	**Description**
2	BOP	Manually control Automatic Letdown HX Temperature Control Valve (1KC-132)
4	RO	Manual turbine runback to ~ 275 MWe

<u>Critical Task 1</u> – Runback the Main Turbine prior to Main Turbine Trip (must be below amps for 332 MW @ 0.9 Power Factor within 3.5 minutes).

<u>Critical Task 2</u> – Close MSIVs prior to a severe challenge (Orange Path) on NC system Integrity CSF.

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Total malfunctions (5–8)	8
2. Malfunctions after EOP entry (1–2)	2
3. Abnormal events (2–4)	4
4. Major transients (1–2)	1
5. EOPs entered/requiring substantive actions (1–2)	1
6. EOP contingencies requiring substantive actions (0–2)	0
7. Critical tasks (2–3)	2

EXERCISE GUIDE WORKSHEET

1. INITIAL CONDITIONS:

   1.1 Reset to IC # 13 and load schedule file for NRC Scenario 1

START TIME:__________

✓	✓	Trigger	Instructor Action	Final	Delay	Ramp	Delete In	Event
			ANN-AD11-B03 (TRANSFORMER A TROUBLE)	ON				
			ANN-AD11-E03 (TRANSFORMER B TROUBLE)	ON				
		2	OV_SLIM16SpIncPB (KC-132 Setpoint Increment pushbutton)	PRESS-ED			:10	2
		2	OV_SLIM16manPB (KC-132 Man Pushbutton)	PRESS-ED	:11		:01	2
			VLV-NV035A (NV153A L/D HX DIVERSION FAIL AUTO ACTIONS)	ACTIVE				2
		3	VLV-KC027F (KCC37A TRAIN A RECIRC LINE ISOL VLV FAIL TO POSITION)	1		:02		3
			MAL-EHC003F (ALL TURBINE AUTO RUNBACK FAILURE)	BLOCK				4
		4	LOA-EGB003 (KG PUMPS)	STOP BOTH				4
		4	LOA-EGB003 (KG PUMPS)	RUN B/A RES	5:00			4
		5	LOA-NS003 (RACKOUT NS PMP 1A)	RACK-OUT				5
		5	MAL-SM007C (STM BRK INSIDE CONTAINMENT LOOP C)	3000		:15		6
		7	MAL-SM007C (STM BRK INSIDE CONTAINMENT LOOP C)	2e+6		:15		7
			MAL-SM006A (SM7 MSIV A FAILURE)	ACTIVE				8
			MAL-SM006B (SM5 MSIV B FAILURE)	ACTIVE				8
			MAL-SM006C (SM3 MSIV C FAILURE)	ACTIVE				8
			MAL-SM006D (SM1 MSIV D FAILURE)	ACTIVE				8
		10	MAL-SM006A (SM7 MSIV A FAILURE)	ACTIVE			:01	8
		11	MAL-SM006B (SM5 MSIV B FAILURE)	ACTIVE			:01	8
		12	MAL-SM006C (SM3 MSIV C FAILURE)	ACTIVE			:01	8
		13	MAL-SM006D (SM1 MSIV D FAILURE)	ACTIVE			:01	8
			MAL-ISE011B (AUX BLDG FANS FAIL TO AUTO STOP)	ACTIVE				9

		17	**LOA-CNT002 (H2 ANALYZERS)**	**BOTH**	**10:00**			
		Ensure Event 10 = x02i071c (1SM-7 Close PB)						
		Ensure Event 11 = x02i074c (1SM-5 Close PB)						
		Ensure Event 12 = x02i079c (1SM-3 Close PB)						
		Ensure Event 13 = x02i082c (1SM-1 Close PB)						

2. SIMULATOR BRIEFING

2.1 Control Room Assignments:

Position	Name
CRS	
RO	
BOP	

2.2 Give a copy of Attachment 2 (Shift Turnover Information) to the CRS.

3. EXERCISE PRESENTATION

3.1 Familiarization Period

A. Allow examinees time to familiarize themselves with the Control Board alignments.

3.2 **Scenario EVENT 1**, Increase Reactor Power

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as DEC-BA (Balancing Authority / SOC) by the crew to inform of commencing power increase, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	WHEN contacted as Secondary Chemistry to obtain maximum blowdown for appropriate load (step 3.2.3.9), **REPORT** "Maintain blowdown at current flow rates."

3.3 **Scenario EVENT 2**, Letdown Heat Exchanger Temperature Control Valve (1KC-132) Fails Closed

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 2** to cause 1KC-132 to slowly close.

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for 1KC-132 or 1NV-153A, **REPEAT** the information.

3.4 **Scenario EVENT 3**, KC Miniflow Valve 1KC-C37A fails open

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 3** to cause KC miniflow valve 1KC-C37A to fail open.

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for 1KC-C37A, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as an AO to perform a post start check out of the started KC pump, after 5 minutes **REPORT**: • "KC pump 1A1, 1B1, 1B2 looks good for continued operation."

3.5 **Scenario EVENT 4**, Loss of both KG pumps / Automatic Turbine Runback Failure

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 4** to cause the running KG pump to trip and failure of the standby KG pump to start.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as an AO to investigate the loss of KG, after 5 minutes **REPORT**: • "I do not see any reason that the running KG pump tripped, and I have the 1B KG pump in service. I will contact SPOC to help with troubleshooting the reason for the 1A KG pump trip."

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for loss of KG or automatic turbine runback failure, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as DEC BA (Balancing Authority) to notify of Unit status, **REPEAT** the information.

3.6 **Scenario EVENTS 5 & 6**, 1A NS Pump Loss of Power / Steam Leak Inside Containment

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 5** to cause a loss of power to 1A NS pump.

✓	BOOTH INSTRUCTOR ACTION
	IF dispatched as an AO to the breaker for 1A NS pump, **REPEAT** the information.
	After 5 minutes, contact the control room crew and **REPORT**: • "All indicating lights on the front of the breaker for 1A NS pump are dark and there is an acrid smell from the upper compartment."

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for 1A NS pump loss of power, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as RP to notify of the steam leak, **REPEAT** the information.

3.8 **Scenario EVENTs 7, 8, and 9**, Steam Line Break Inside Containment / MSIVs fail to close on MSI signal / Auxiliary Building Unfiltered Exhaust Fans fail to secure on S/I

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 7** to initiate a steam line break inside containment from the 1C S/G.

✓	BOOTH INSTRUCTOR ACTION
	IF dispatched as an AO to secure all ice condenser air handling units per G-1 Enclosure 11 and to place the Hydrogen Analyzers in service per OP/1/A/6450/010, **REPEAT** the information and **INSERT SIMULATOR Trigger 17**.
	After 10 minutes **REPORT**: • "Ice condenser air handling units have been secured per G-1 Enclosure 11 and the Hydrogen Analyzers have been placed in service per OP/1/A/6450/010."

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as chemistry to sample all S/Gs for activity, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as RP to frisk all S/G cation columns for activity, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as an AO to unlock and close 1SA-4, **REPEAT** the information.

Op Test No.:	301	Scenario #	1	Event #	1	Page	12	of	153
Event Description:	Increase Reactor Power								

Note To Evaluator:

The scenario begins with a power increase from 50% by the crew. This will involve several procedures to accomplish. The following procedures are included in this guide:

- **OP/1/A/6150/009 Enclosure 4.3 (Dilution)**
- **OP/1/A/6150/008 Enclosure 4.16 (Control Bank Manual Operation At Power)**
- **OP/0/B/6300/001 Enclosure 4.2 (Load Changing)**

These procedures may be performed in any order by the crew. Instructions for continuing to the next Event are included at the end of OP/0/B/6300/001 Enclosure 4.2.

Enclosure 4.3
Dilution

OP/1/A/6150/009
Page 2 of 5

2. Initial Conditions

_____ 2.1 Ensure R2 reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 2.2 Verify Unit 1 is in Mode 1 or 2.

_____ 2.3 Verify the NV System is in operation per OP/1/A/6200/001 (Chemical and Volume Control System).

_____ 2.4 Verify sufficient RHT volume is available to receive the reactor coolant displaced during the planned dilution operation.

_____ 2.5 Verify the NB System is in operation per OP/1/A/6200/012 (Reactor Makeup Water).

_____ 2.6 <u>IF</u> NC System boron concentration will be changed by ≥ 50 ppm, initiate PZR spray to equalize the boron concentration throughout the system by operating backup heaters per OP/0/A/6200/055 (Miscellaneous Component Operation).

3. Procedure

NOTE:	This enclosure will affect reactivity of the core and is therefore designated important to Reactivity Management per the guidelines of AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 3.1 Ensure valves are aligned per Enclosure 4.8 (Valve Checklist).

_____ 3.2 <u>IF</u> the blender is set for automatic makeup per Enclosure 4.1 (Automatic Makeup), record the setpoint on 1NV-242A (RMWST To B/A Blender Ctrl): __________ gpm

NOTE:	The purpose of the following step is to minimize the pressure fluctuation caused by manually diverting to the RHT and effects on Reactor Water Makeup flow.

_____ 3.3 <u>IF</u> desired, adjust the setpoint for 1NV-172A (VCT-LEVEL CTRL) (1NVSS5760) to 55%.

3.4 Ensure the following valve control switches in "AUTO":

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ 3.5 Ensure 1NV-242A (RMWST To B/A Blender Ctrl) controller in auto.

_____ 3.6 Ensure at least one reactor makeup water pump is in "AUTO" or "ON".

Enclosure 4.3

Dilution

OP/1/A/6150/009
Page 3 of 5

☐ 3.7 Record the desired volume of reactor makeup water to be added. _________ gallons

_____ 3.8 Adjust the total makeup counter to the desired volume of reactor makeup water to be added. (R.M.)

_____ 3.9 Place the "NC MAKEUP MODE SELECT" switch to the "DILUTE" position.

NOTE:	High letdown flow rates result in increased backpressure on the letdown line. If letdown flow is ≥ 90 gpm, it may be desirable to reduce the dilution flow rate to 80 gpm to avoid the Rx Make-up Flow Deviation alarm and associated automatic actions.

_____ 3.10 __IF__ required, adjust the setpoint for 1NV-242A (RMWST To B/A Blender Ctrl) to the desired flow.

_____ 3.11 __IF AT ANY TIME__ it is desired to divert letdown to the RHT manually operate 1NV-172A (3-Way Divert To VCT-RHT) as follows:

_____ 3.11.1 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) to the "RHT" position.

_____ 3.11.2 Ensure VCT level is monitored continuously while diverting to the RHT.

NOTE:	Procedure may continue while performing the following step.

3.11.3 __WHEN__ desired VCT level is reached return 1NV-172A (3-Way Divert To VCT-RHT) to auto as follows:

_____ 3.11.3.1 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) in the "VCT" position.

_____ 3.11.3.2 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) in the "AUTO" position.

_____ 3.12 __IF AT ANY TIME__ during the makeup it becomes necessary to change the makeup flow rate, adjust the setpoint for 1NV-242A (RMWST To B/A Blender Ctrl) as necessary to achieve the desired flow.

Op Test No.:	301	Scenario #	1	Event #	1	Page	15	of 153
Event Description:	Increase Reactor Power							

Enclosure 4.3 OP/1/A/6150/009

Dilution Page 4 of 5

_____ 3.13 **IF AT ANY TIME** while dilution is in progress it becomes necessary **OR** it is desired to stop the dilution (for example: loss of all NC Pumps, unexpected results observed or a large makeup being made in multiple batches), perform the following:

_____ 3.13.1 Place the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.13.2 Ensure the following valves close:

- _____ • 1NV-242A (RMWST To B/A Blender Ctrl)
- _____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ 3.13.3 **IF** in "AUTO", verify the reactor makeup water pump stops.

☐ 3.13.4 Record reactor makeup water volume added as indicated on the total makeup counter. ____________ gallons

3.13.5 **WHEN** conditions allow resuming the dilution, perform the following:

☐ 3.13.5.1 Determine remaining volume to be added by subtracting the amount previously added (Step 3.13.4) from the desired volume to be added (Step 3.7).

____________ - ___________ = ___________ gallons
(Step 3.7) (Step 3.13.4)

_____ 3.13.5.2 Adjust total makeup counter to the volume of reactor makeup water determined in Step 3.13.5.1. (R.M.)

_____ 3.13.5.3 Place the "NC MAKEUP CONTROL" switch in the "START" position. (R.M.)

_____ 3.13.5.4 Verify the following:

☐ 1NV-242A (RMWST To B/A Blender Ctrl) modulates to establish desired flow
☐ 1NV-181A (B/A Blender Otlt To VCT) opens

_____ 3.13.5.5 **IF** in "AUTO", verify the reactor makeup water pump starts.

_____ 3.14 **WHILE** makeup is in progress, monitor the following for expected results:

☐ Control rod motion
☐ NC System Tavg
☐ Reactor Power

Enclosure 4.3
Dilution

OP/1/A/6150/009
Page 5 of 5

> **NOTE:** If a small makeup is being performed, placekeeping for Steps 3.15 through 3.17 may be performed after Step 3.18 is performed.

_____ 3.15 Place the "NC MAKEUP CONTROL" switch in the "START" position. (R.M.)

_____ 3.16 Verify the following:

- ☐ 1NV-242A (RMWST To B/A Blender Ctrl) modulates to establish desired flow
- ☐ 1NV-181A (B/A Blender Otlt To VCT) opens

_____ 3.17 <u>IF</u> in "AUTO", verify the reactor makeup water pump starts.

> **NOTE:** The total makeup counter may count up 1 - 5 gallons after termination.

3.18 <u>WHEN</u> the desired volume of reactor makeup water is reached on the total makeup counter, ensure the following valves close. (R.M.)

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ 3.19 <u>IF</u> Step 3.3 was performed, return the setpoint for 1NV-172A (VCT LEVEL CRL) (1NVSS5760) to 75.0 %.

_____ 3.20 <u>IF</u> automatic makeup is desired, perform one of the following:

_____ 3.20.1 <u>IF</u> it is desired to change the blender outlet boron concentration, refer to Enclosure 4.1 (Automatic Makeup).

OR

_____ 3.20.2 <u>IF</u> makeup at the previous concentration is acceptable <u>AND</u> the system was previously aligned per Enclosure 4.1 (Automatic Makeup), perform the following:

_____ 3.20.2.1 Ensure the controller for 1NV-242A (RMWST To B/A Blender Ctrl) is set to the value recorded in Step 3.2. (R.M.)

_____ 3.20.2.2 Place the "NC MAKEUP MODE SELECT" switch in "AUTO".

_____ 3.20.2.3 Place the "NC MAKEUP CONTROL" switch to the "START" position. (R.M.)

3.21 Do <u>NOT</u> file this enclosure.

Enclosure 4.16

Control Bank Manual Operation At Power

OP/1/A/6150/008
Page 1 of 2

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can affect core reactivity by changing control rod position. (R.M.)

1.2 The following Limits and Precautions are Reactivity Management related: (R.M.)

1.2.1 When rods are being moved, observe "RODS IN/RODS OUT" light for proper direction.

1.2.2 When rods are being moved, observe the demand position and actual (digital) position to verify proper operation of the Rod Control System.

1.2.3 Adjusting T-Avg ± 1°F of T-Ref before transferring rod control to "AUTO" will prevent undesired rod movement.

1.2.4 Monitor startup rate continuously during any rod motion to ensure < 0.5 DPM stable startup rate.

1.3 Automatic rod control shall **NOT** be used when less than 15% (184 MW_e) turbine power.

1.4 Individual control bank positions on "CRD BANK SELECT" switch shall not be used to position rods manually. (The automatic overlap feature is disabled.)

1.5 After releasing Rod Motion lever, waiting 2 seconds before attempting to move rods again will allow all signals to clear the firing cards.

1.6 A rod motion demand below zero steps may result in the movable grippers **NOT** properly engaging the drive shaft.

2. Initial Conditions

AA 2.1 Ensure Reactivity Management controls established per AD-OP-ALL-0203 (Reactivity Management. (RM)

AA 2.2 Verify Unit 1 is **NOT** in an EP or AP.

AA 2.3 Verify one of the following exist:

- ☑ Control Bank movement required to increase/decrease Reactor Power
- ☐ Control Bank movement required to increase/decrease Tavg
- ☐ Control Bank movement required to maintain AFD
- ☐ Control Bank manual control required to support testing/maintenance activity

Enclosure 4.16
Control Bank Manual Operation At Power

OP/1/A/6150/008
Page 2 of 2

3. Procedure

> **NOTE:** Steps 3.1 through 3.6 may be signed off as time allows ensuring operator maintains proper focus on reactivity management.

____ 3.1 Monitor the following:

- Tavg/Tref
- Demand Counter positions
- DRPI rod positions
- ROD MOTION RODS-IN/RODS-OUT Light
- ROD MOTION DEMAND SIGNALS - TEMP ERROR/POWER MISMATCH
- Power Range instruments
- IR SUR (Startup Rate)

AA 3.2 IF MANUAL ROD movement is desired, perform the following:

✓ Verify the "ALM" LED on circuit card A206 in the left side of 1ERCC0006 (Rod Control Logic Cabinet) is NOT illuminated.

✓ Verify one GRP select light is illuminated on each power cabinet.

AA 3.3 IF plant conditions require, place the "CRD BANK SELECT" switch in "MAN".

____ 3.4 IF withdrawing Control Banks, pull and hold the "ROD MOTION" lever "OUT" as required until control rods are in the desired position. (R.M.)

____ 3.5 IF inserting Control Banks, push and hold the "ROD MOTION" lever "IN" as required until control rods are in the desired position. (R.M.)

____ 3.6 IF automatic rod control is desired, perform the following:

____ 3.6.1 Verify Unit 1 Reactor Power is ≥ 15% RTP.

____ 3.6.2 WHEN Tavg is within 1°F of Tref, place "CRD BANK SELECT" in "AUTO".

3.7 Do NOT file this enclosure.

Enclosure 4.2
Load Changing

OP/1/B/6300/001
Page 1 of 6

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can effect reactivity. (R.M.)

1.2 Low load operation limits:

1.2.1 The unit can be operated continuously at low loads when exhaust hood temperature is < 175°F. The load shall, however, be increased slowly until the temperature decreases below 125°F before increasing load at normal rate (Multipoint Recorder on 1MC3).

1.2.2 Excessive use of the exhaust hood sprays shall be avoided to prevent accelerated blade erosion.

1.3 Do **NOT** exceed the load, hydrogen pressure, and power factor limits per the Unit One Revised Data Book Figure 43.

1.4 If the limits of the Unit One Revised Data Book Figure 43 (Generator Capability Curves) are exceeded, the Turbine Generator shall be tripped.

1.5 Under certain environmental conditions, indicated condenser vacuum less than 24.3 inches Hg may be reached at full load. Exhaust hood temperatures are a more accurate indicator of true vacuum. It is recommended the turbine **NOT** be operated under the following conditions at full load:

- Exhaust Hood 1A temperature ≥ 136°F
- Exhaust Hood 1B temperature ≥ 129°F
- Exhaust Hood 1C temperature ≥ 124.5 °F

1.6 The maximum differential pressure between adjacent LP shell pressures shall **NOT** exceed 2.0 inches Hg. (main condenser vacuum gauges on 1MC13, OAC points C1P1669 (D/P between A & B Condensers) and C1P1670 (D/P between B & C Condensers) or Main Condenser graphic (CMCOND)).

1.7 A sudden downward trend on an LP turbine's lower extraction temperature shall be investigated as a possible indication of water induction into the turbine. This is indicated on the recorder on the rear of 1MC8 labeled "TURBINE WATER DETECTION", using any of the LP 8th stage lower temperatures.

1.8 A "LOAD RATE" > "6.2 MW/MIN" shall **NOT** be used during normal load changes.

Enclosure 4.2 OP/1/B/6300/001

Load Changing Page 2 of 6

1.9 Differential temperature between adjacent exhaust hoods shall **NOT** exceed 30°F unless evaluated and approved by the responsible engineer (Turbine Generator System Expert). (OAC points C1P1667 (A & B Exhaust Hoods Metal Delta Temp) and C1P1668 (B & C Exhaust Hoods Metal Delta Temp) or Main Condenser graphic (CMCOND)).

1.10 The Main Turbine OIU Work Station has the capability to perform control functions for the Main Turbine, including tripping and resetting of the turbine. If a control function window is inadvertently selected while manipulating the Main Turbine OIU Work Station, the window shall be closed to prevent actuation of the control function.

1.11 To reduce potential for Turbine rubs at low power levels (< 30% Turbine Load) observe the following:

- Steam Seal Header Pressure between 3 and 5 psig.
- Gland Steam Condenser Header vacuum of 10 - 12" H_2O vacuum.
- Condenser Vacuum < 28.0" Hg
- Minimize time that Turbine is at speed no load.
- Minimize time between Turbine Shell Warming and rolling the Turbine.

1.12 Exhaust hood water sprays are used to cool the last-stage buckets and to minimize temporary distortion of the low-pressure hood and shell structures. These sprays have a significant potential for quenching the LP turbine structure, and if they are applied manually should be undertaken very gradually. Large and rapid changes in the temperature of the exhaust hood can also have an impact on bearing alignment and may cause a rub to develop. Excessive use of the sprays may cause unnecessary erosion of the long last-stage buckets during low flow conditions.

2. Initial Conditions

_____ Verify Turbine Generator is On Line per Enclosure 4.1 (Turbine Generator Startup).

Op Test No.:	301	Scenario #	1	Event #	1	Page	21	of	153
Event Description:	Increase Reactor Power								

Enclosure 4.2 OP/1/B/6300/001

Load Changing Page 3 of 6

3. Procedure

CAUTION:
1. The load, hydrogen pressure and power factor limits per the Unit One Revised Data Book Figure 43 shall **NOT** be exceeded.
2. Rate of change of First-Stage Bowl Inner Surface Temperature shall **NOT** exceed 150°F/hr (OAC point C1P1283 (First Stage Metal Temp Rate)).
3. Control valve casing difference, OAC point C1A0961 (Turb Valve Chest Inner Surface Metal Temp) minus C1A0967 (Turb Valve Chest Outer Surface Metal Temp), shall **NOT** exceed curve "Allowable Temp Difference on Turbine Valve Chest" in the Unit 1 OAC Databook.
4. OAC point C1A1140 (Turbine Lower Inner Shell Temp) vs. Percent Steam Flow (OAC point C1P1588 (Design Total Main Steam Flow, Measured (%))) shall be maintained above and to the left of the curve in the Unit One OAC Databook "Load-Changing Recommendations".

NOTE:
- Several of the parameters required for this procedure can be found on OAC graphics, and a list of all OAC points are found on Enclosure 4.8 (Turbine Generator Roll Computer Points).
- Step 3.1 and Step 3.2 may be performed in any order.

_____ 3.1 **IF** increasing turbine generator load, perform the following:

_____ 3.1.1 Ensure the proper reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

AA 3.1.2 **WHILE** increasing Turbine Generator load, perform the following:

N/A 3.1.2.1 **IF AT ANY TIME** Turbine load is < 30%, operate the RC system pumps and fans as required per OP/1/B/6400/001 A (Condenser Circulating Water System) to maintain vacuum in Condenser C < 28" Hg.

N/A 3.1.2.2 **IF** applicable, verify Groups B and C valves on Enclosure 4.6 (Valve Checklist) close at 15% of full load (184 MWe, 107 psig Turbine Impulse Pressure).

N/A 3.1.2.3 **IF** applicable, verify the following valves close at 15% of full load (184 MWe, 107 psig Turbine Impulse Pressure):

- ☐ 1SM-21 (Ctrl Vlv #2 Stm Lead Drn)
- ☐ 1SM-29 (Ctrl Vlv #1 Stm Lead Drn)

Enclosure 4.2 OP/1/B/6300/001

Load Changing Page 4 of 6

_____ 3.1.2.4 **IF** applicable, **WHEN** CV3 comes off of its fully closed seat (65% of full load, 796 MWe), verify 1SM-25 (Ctrl Vlv #3 Stm Lead Drn) closes.

_____ 3.1.2.5 **IF** applicable, **WHEN** CV4 comes off of its fully closed seat (92% of full load, 1127 MWe), verify 1SM-33 (Ctrl Vlv #4 Stm Lead Drn) closes.

CAUTION:
1. Until it is recognized that the first stage shell metal temperature change rate stays below the allowable limit (150°F/hr), the following loading rate shall **NOT** be exceeded:
   - 1/2%/min - First Stage Inner Shell Temperature (1MC3 or OAC point C1A1140 (Turbine Lower Inner Shell Temp)) ≤ 350°F
   - 1%/min - First Stage Inner Shell Temperature (1MC3 or OAC point C1A1140 (Turbine Lower Inner Shell Temp)) > 350°F
2. Normal steady-state load changes shall be made without exceeding the limits shown on Enclosure 4.7 (Generator Operating Limits) and in the Unit 1 OAC Databook "Recommended Startup and Loading Curves".
3. Unit One Reactor Operating Data, Section 2.4 shall be referred to for allowable ramp rates. A "LOAD RATE" > 6.2 MW/MIN shall **NOT** be used during normal load changes.

3.1.3 Increase turbine generator load by performing the following:

_____ 3.1.3.1 Select "LOAD RATE" and verify it illuminates.

_____ 3.1.3.2 Input the desired load rate.

_____ 3.1.3.3 Select "ENTER" and verify "LOAD RATE" goes dark.

_____ 3.1.3.4 Select "TARGET" and verify it illuminates.

_____ 3.1.3.5 Input the desired load target.

_____ 3.1.3.6 Select "ENTER" and verify "TARGET" light goes dark.

_____ 3.1.3.7 Verify new load target appears on Target Display.

_____ 3.1.3.8 Select "GO" and verify it illuminates to start load increase.

_____ 3.1.3.9 Coordinate with Secondary Chemistry to adjust S/G blowdown flowrates to obtain maximum blowdown for the appropriate load.

Note to Evaluator:

At this point, the power increase has begun. At the discretion of the Lead Evaluator, the scenario may continue to the next event by instructing the booth operator to INSERT Trigger 2 (1KC-132 Failure).

Op Test No.: 301	Scenario # 1	Event # 2	Page 23 of 153				
Event Description:	Letdown Heat Exchanger Temperature Control Valve (1KC-132) Fails Closed						

Control Room Indications
1AD-7, F/3 “LETDN HX OUTLET HI TEMP” - LIT
1KC-132 (Letdn Hx Otlt Temp Ctrl) closing
Letdown temperature on 1NVPT5590 increasing
DCS Alarm “Letdown HX TEMP CTRL IN MAN”

Note To Evaluator:

The crew response for this failure can be found in the annunciator response for 1AD-7, F/3 on the following page. The failure is on the setpoint increase button on the controller for 1KC-132 and will delete after 10 seconds and transfer the controller to Manual. The crew will manually control 1KC-132 to increase cooling flow to the Letdown Heat Exchanger. If letdown temperature exceeds 136°F, then letdown 3-way valve 1NV-153A should bypass the mixed bed demineralizers.

Op Test No.:	301	Scenario #	1	Event #	2	Page	24	of 153

Event Description: Letdown Heat Exchanger Temperature Control Valve (1KC-132) Fails Closed

OP/1/B/6100/010 H
Page 35 of 64

PANEL: 1AD-7

F/3

LETDN HX OUTLET HI TEMP

SETPOINT: 128°F

ORIGIN:

Instrument	DCS	Description
1NVPT5590	1NVAA5590	LETDOWN HX OTLT TEMP

PROBABLE CAUSE:

1. Letdown flow too high
2. 1KC-132 (Letdn Hx Otlt Temp Ctrl) (controlled by 1NVSS5590) malfunction

AUTOMATIC ACTIONS: **IF** letdown temp. continues to rise, at 136°F 1NV-153A (Ltdn Hx Otlt 3-Way Vlv) will divert Letdown to volume control tank.

IMMEDIATE ACTIONS:

1. **IF** due to hi letdown flow, reduce flow rate by removing orifices from service and/or taking manual control of 1NV-148 (Letdn Press Control Valve) as necessary.
2. **IF** due to a low KC flow, attempt to restore normal flow, by taking manual control of 1KC-132 (Letdn Hx Otlt Temp Ctrl).
3. **IF** KC flow **CANNOT** be restored to Letdown Heat Exchanger, refer to AP/1/A/5500/021 (Loss of Component Cooling).

SUPPLEMENTARY ACTIONS:

1. Ensure letdown flow does **NOT** exceed 120 gpm.
2. Ensure ND letdown flow in Modes 5, 6 or No Mode does **NOT** exceed 185 gpm.
3. Verify that 1NV-148 (Letdn Press Control) is maintaining proper back pressure of 350 psig.
4. **IF** letdown temperature exceeds 136°F, ensure 1NV-153A (Ltdn Hx Otlt 3-Way Vlv) diverts flow to the VCT.
   - 4.1 **WHEN** letdown temperature decreases below 136°F, ensure 1NV-153A (Ltdn Hx Otlt 3-Way Vlv) directs letdown flow to the NV demineralizers.

NOTE: Completion of the evaluation/inspection in the following step shall **NOT** delay a return to normal operation.

5. **IF** KC flow is lost to the Letdown Hx for greater than 30 seconds, contact Engineering to evaluate/inspect for any possible damage due to water hammer.

REFERENCES:

1. CN-1554-01.06
2. CN-1573-01.02

Note to Evaluator:

This completes Event 2. At Lead Evaluator discretion, the scenario may continue by directing the booth operator to insert Trigger 3 (KC Miniflow Valve 1KC-C37A Fails Open).

Op Test No.:	301	Scenario #	1	Event #	3	Page	25 of 153
Event Description:	KC Miniflow Valve 1KC-C37A Fails Open						

Control Room Indications
1AD-9, F/5 “KC TRAIN A SINGLE PUMP RUNOUT” – LIT
RED OPEN indicating light for 1KC-C37A – LIT
Various KC low flow alarms - LIT

Op Test No.:	301	Scenario #	1	Event #	3	Page	26 of 153
Event Description:	KC Miniflow Valve 1KC-C37A Fails Open						

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 2 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

CAUTION **Failure to restore NC pump seal cooling via thermal barrier cooling or NV seal injection within 10 minutes will cause damage to the NC pump seals resulting in NC inventory loss.**

Note to Evaluator:

Enclosure 1 can be found as Attachment 3 in the back of this document.

__ 1. **Monitor Enclosure 1 (Foldout Page).**

2. **Verify the following:**

__ • At least one KC pump - ON

AND

__ • Adequate flow to KC loads presently in service.

__ a. Start additional KC pump(s) as necessary.

N/A b. **IF** KC Train HX aligned for maintenance and cross train alignment desired, **THEN** perform the following:

1) **IF** Unit 1 in Mode 5, 6 or No Mode, **THEN** perform the following:

__ • Continue in this procedure

AND

__ • Place KC in cross train cooling alignment. **REFER TO** Enclosure 7 (KC Cross Train Cooling).

N/A c. **IF** no KC pump can be started, **THEN** perform the following:

__ 1) **IF** S/I has actuated on either unit, **THEN GO TO** Step 4.

CAUTION **YD can only supply one Unit's NV pump at a time.**

__ 2) Determine which unit will receive alternate NV pump cooling from YD.

(RNO continued on next page)

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 3 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	
	__ 3) **IF** Unit 2 selected to receive YD cooling to 2A NV pump, **THEN GO TO** Step 4.
	NOTE • NV pumps may be started without regard to cooling water alignment. • Operating NV Pump will reach high temperature conditions in approximately 15 minutes with no cooling water.
	__ 4) Dispatch operator to align YD cooling to NV pump 1A. **REFER TO** Enclosure 2 (Alternate Cooling To NV Pump 1A).
	__ 5) Maximize NV pump run time. **REFER TO** Enclosure 5 (Maximize NV Pump Run Time).
	__ 6) **IF AT ANY TIME** S/I occurs on either unit, **THEN** notify dispatched operator to realign NV Pump 1A cooling to normal. **REFER TO** Enclosure 2 (Alternate Cooling To NV Pump 1A).
	__ 7) **GO TO** Step 4.
	(RNO continued on next page)

Op Test No.:	301	Scenario #	1	Event #	3	Page	28 of 153
Event Description:	KC Miniflow Valve 1KC-C37A Fails Open						

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 4 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	
	N/A d. **IF** NV pump operating with no KC cooling, **THEN** perform the following:
	__ 1) **IF** S/I has actuated on either unit, **THEN GO TO** Step 4.
	CAUTION **YD can only supply one Unit's NV pump at a time.**
	__ 2) Determine which unit will receive alternate NV pump cooling from YD.
	__ 3) **IF** Unit 2 selected to receive YD cooling to 2A NV pump, **THEN GO TO** Step 4.
	NOTE • NV pumps may be started without regard to cooling water alignment. • Operating NV Pump will reach high temperature conditions in approximately 15 minutes with no cooling water.
	__ 4) Dispatch operator to align YD cooling to NV pump 1A. **REFER TO** Enclosure 2 (Alternate Cooling To NV Pump 1A).
	__ 5) Maximize NV pump run time. **REFER TO** Enclosure 5 (Maximize NV Pump Run Time).
	(RNO continued on next page)

Event Description: KC Miniflow Valve 1KC-C37A Fails Open

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 5 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	
	__ 6) **IF AT ANY TIME** S/I occurs on either unit, **THEN** notify dispatched operator to realign NV Pump 1A cooling to normal. **REFER TO** Enclosure 2 (Alternate Cooling To NV Pump 1A).
__ 3. **IF AT ANY TIME all KC pumps lost, THEN RETURN TO STEP 2.**	

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 6 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE Uncooled letdown may result in loss of NV pumps within a matter of minutes.

4. **Verify the following:**
   - __ • 1AD-7, F/3 "LETDN HX OUTLET HI TEMP" - DARK

     AND
   - __ • At least one KC pump - ON.

IF KC flow unavailable to letdown HX, THEN isolate letdown as follows:

a. Ensure the following valves - CLOSED:
   - __ • 1NV-10A (Letdn Orif 1B Otlt Cont Isol)
   - __ • 1NV-11A (Letdn Orif 1C Otlt Cont Isol)
   - __ • 1NV-13A (Letdn Orif 1A Otlt Cont Isol).

__ b. Control charging to stabilize Pzr level at program level while maintaining seal injection flow.

__ c. Ensure 1NV-153A (Letdn Hx Otlt 3-Way Valve) - ALIGNED TO VCT.

__ d. Ensure 1NV-172A (3-Way Divert To VCT-RHT) - ALIGNED TO RHT.

__ e. Ensure VCT makeup - IN AUTO.

__ f. **WHEN** time and manpower permit, **THEN REFER TO** AP/1/A/5500/012 (Loss of Charging or Letdown).

g. **IF AT ANY TIME** the following conditions exist:
   - __ • VCT level - LESS THAN 23%

     OR
   - __ • PZR level - GREATER THAN 85% **AND** TRENDING UP,

   __ **THEN GO TO** Enclosure 6 (Rx Trip Sequence).

__ h. **GO TO** Step 6.

Op Test No.:	301	Scenario #	1	Event #	3	Page	31 of 153
Event Description:	KC Miniflow Valve 1KC-C37A Fails Open						

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 7 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 5. **IF AT ANY TIME 1AD-7, F/3 "LETDN HX OUTLET HI TEMP" lit, THEN perform Step 4 RNO.**	
__ 6. **Verify both KC surge tank levels - 50% - 90% AND STABLE.**	__ **Observe Caution prior to Step 8 and GO TO Step 8.**
__ 7. **Start additional KC pump(s) as necessary to supply any KC loads presently in service.**	__ **IF KC pump(s) damaged by fire, THEN notify IAE to repair cables to pumps needed for recovery. REFER TO IP/1/A/3890/027A (Fire Damage Control Procedure).**

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 8 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION **A loss of KC cooling to the NC pumps results in a gradual approach to an overheated condition in approximately 10 minutes which will result in shaft seizure.**

8. **Verify KC flow to NC pumps as follows:**

__ • 1AD-20, A/1 "KC SUPPLY HDR FLOW TO NCP BRGS LOW" - DARK

__ • 1AD-21, A/1 "KC SUPPLY HDR FLOW TO NCP BRGS LOW" - DARK.

Perform the following:

a. Ensure the following valves - OPEN:

__ • 1KC-425A (NC Pumps Ret Hdr Cont Isol)

__ • 1KC-338B (NC Pumps Sup Hdr Cont Isol)

__ • 1KC-424B (NC Pumps Ret Hdr Cont Isol).

b. **IF AT ANY TIME** any of the following conditions met:

__ • Time since loss of KC - GREATER THAN 10 MINUTES

OR

__ • Any NC pump trip criteria from Enclosure 1 (Foldout Page) met,

__ **THEN GO TO** Enclosure 6 (Rx Trip Sequence).

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 9 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Verify KC available as follows:**	
	NOTE The KC non-essential header valves can be reopened when the appropriate train's level switch is reset. This should occur between 40% and 48% KC surge tank level.
a. Verify the following Train A KC non-essential header isolation valves - OPEN: __ • 1KC-230A (Rx Bldg Non-Ess Hdr Isol) __ • 1KC-3A (Rx Bldg Non-Ess Ret Hdr Isol) __ • 1KC-50A (Aux Bldg Non-Ess Hdr Isol) __ • 1KC-1A (Aux Bldg Non-Ess Ret Hdr Isol).	__ a. **WHEN** OAC alarm C1D2215 (KC Train A Low-Low Level Surge Tank Isol) "**NOT** ACTUATED" **AND** cause of valve closure known, **THEN** ensure affected valve(s) open.

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 10 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
	NOTE The KC non-essential header valves can be reopened when the appropriate train's level switch is reset. This should occur between 40% and 48% KC surge tank level.
b. Verify the following Train B KC non-essential header isolation valves - OPEN: __ • 1KC-228B (Rx Bldg Non-Ess Hdr Isol) __ • 1KC-18B (Rx Bldg Non-Ess Ret Hdr Isol) __ • 1KC-53B (Aux Bldg Non-Ess Hdr Isol) __ • 1KC-2B (Aux Bldg Non-Ess Ret Hdr Isol).	__ b. **WHEN** OAC alarm C1D2214 (KC Train B Low-Low Level Surge Tank Isol) "**NOT** ACTUATED" **AND** cause of valve closure known, **THEN** ensure affected valve(s) open.
__ c. Start additional KC pump(s) as necessary to supply any KC loads presently in service.	__ c. **IF** KC pump(s) damaged by fire, **THEN** notify IAE to repair cables to pumps needed for recovery. **REFER TO** IP/1/A/3890/027A (Fire Damage Control Procedure).

Event Description: KC Miniflow Valve 1KC-C37A Fails Open

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 11 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Verify KC surge tank levels normal as follows:**	
__ a. Verify both KC surge tank levels - 50% - 90% AND STABLE.	a. Perform the following: 1) Dispatch operator to initiate makeup to surge tank(s) by opening appropriate valve(s): __ • 1KC-107 (1A KC Surge Tank YM M/U) (AB-601, PP-59, Rm 500) OR __ • 1KC-111 (1B KC Surge Tank YM M/U) (AB-601, PP-59, Rm 500). __ 2) Dispatch operators to locate and isolate KC System leakage. **NOTE** Normal makeup level is approximately 75%. __ 3) **WHEN** affected KC surge tank(s) level at desired level, **THEN** notify dispatched operator to secure makeup. __ 4) **IF AT ANY TIME** additional YM makeup required and Enclosure 3 (Surge Tank Makeup) **NOT** in progress, **THEN** perform Step 10.a RNO as required. __ 5) **GO TO** Step 11.
__ b. **GO TO** Step 14.	

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 12 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **Verify at least one KC surge tank above lo-lo level as follows:** __ • 1AD-10, A/1 "KC SURGE TANK A LO-LO LEVEL" - DARK OR __ • 1AD-10, A/2 "KC SURGE TANK B LO-LO LEVEL" - DARK.	**Perform the following:** a. Verify the following: __ 1) Both Unit 1 RN essential headers - PRESSURIZED. __ 2) **IF** only one RN essential header pressurized, **THEN** use it for surge tank makeup. __ 3) **IF** at any time RN essential header being used for makeup becomes depressurized, **THEN** notify dispatched operator to secure makeup from RN. **NOTE** Preference should be given to the surge tank with the highest stable level and available pumps. __ b. Dispatch operator to makeup to available train of KC from YM and RN. **REFER TO** Enclosure 3 (Surge Tank Makeup). __ c. Dispatch operators to locate and isolate KC System leakage. __ d. Notify Chemistry of RN makeup to KC System. (RNO continued on next page)

Op Test No.:	301	Scenario #	1	Event #	3	Page	37 of 153
Event Description:	KC Miniflow Valve 1KC-C37A Fails Open						

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 13 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	e. **WHEN** KC surge tank level above lo-lo level setpoint, **THEN** perform the following: __ 1) Ensure KC pumps on affected train - ON. **NOTE** The KC non-essential header valves can be reopened when the appropriate train's level switch is reset. This should occur between 40% and 48% KC surge tank level. 2) OPEN non-essential header isolation valves for affected train as follows: • Train A: __ • 1KC-230A (Rx Bldg Non-Ess Hdr Isol) __ • 1KC-3A (Rx Bldg Non-Ess Ret Hdr Isol) __ • 1KC-50A (Aux Bldg Non-Ess Hdr Isol) __ • 1KC-1A (Aux Bldg Non-Ess Ret Hdr Isol). (RNO continued on next page)

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 14 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	OR • Train B: __ • 1KC-228B (Rx Bldg Non-Ess Hdr Isol) __ • 1KC-18B (Rx Bldg Non-Ess Ret Hdr Isol) __ • 1KC-53B (Aux Bldg Non-Ess Hdr Isol) __ • 1KC-2B (Aux Bldg Non-Ess Ret Hdr Isol). __ f. **WHEN** one train's non-essential header isolation valves open, **THEN** perform Steps 12 and 13. __ g. **GO TO** Step 14.

Event Description: KC Miniflow Valve 1KC-C37A Fails Open

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 15 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 12. **Verify 1AD-10, A/1 "KC SURGE TANK A LO-LO LEVEL" - DARK.**	**Perform the following:** a. Ensure the following valves - CLOSED: __ • 1KC-230A (Rx Bldg Non-Ess Hdr Isol) __ • 1KC-3A (Rx Bldg Non-Ess Ret Hdr Isol) __ • 1KC-50A (Aux Bldg Non-Ess Hdr Isol) __ • 1KC-1A (Aux Bldg Non-Ess Ret Hdr Isol). __ b. Ensure both Train B KC pumps - ON. c. **IF** KC Surge Tank 1A level continues to trend down **OR** is empty, **THEN** perform the following: 1) Ensure the following Train B essential equipment - IN SERVICE AS NEEDED: __ • NV Pump 1B __ • NI Pump 1B __ • ND Pump 1B __ • ND Hx 1B __ • CA Pump 1B __ • NS Pump 1B. (RNO continued on next page)

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 16 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. (Continued)	
	2) Ensure the following Train A essential equipment - OFF: __ • NV Pump 1A __ • NI Pump 1A __ • ND Pump 1A __ • CA Pump 1A __ • NS Pump 1A.
	__ 3) Ensure both Train A KC pumps - OFF.
	__ 4) Locate and isolate leak on Train A essential header.
__ 13. **Verify 1AD-10, A/2 "KC SURGE TANK B LO-LO LEVEL" - DARK.**	**Perform the following:** a. Ensure the following valves - CLOSED: __ • 1KC-228B (Rx Bldg Non-Ess Hdr Isol) __ • 1KC-18B (Rx Bldg Non-Ess Ret Hdr Isol) __ • 1KC-53B (Aux Bldg Non-Ess Hdr Isol) __ • 1KC-2B (Aux Bldg Non-Ess Ret Hdr Isol). __ b. Ensure both Train A KC pumps - ON. (RNO continued on next page)

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 17 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. (Continued)	c. IF KC Surge Tank 1B level continues to trend down **OR** is empty, **THEN** perform the following: 1) Ensure the following Train A essential equipment - IN SERVICE AS NEEDED: __ • NV Pump 1A __ • NI Pump 1A __ • ND Pump 1A __ • ND Hx 1A __ • CA Pump 1A __ • NS Pump 1A. 2) Ensure the following Train B essential equipment - OFF: __ • NV Pump 1B __ • NI Pump 1B __ • ND Pump 1B __ • CA Pump 1B __ • NS Pump 1B. __ 3) Ensure both Train B KC pumps - OFF. __ 4) Locate and isolate leak on Train B essential header.
__ 14. **Ensure KC heat exchanger outlet mode switches - PROPERLY ALIGNED.**	
__ 15. **Determine and correct cause of loss of KC.**	

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 18 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16. **Ensure compliance with appropriate Tech Specs and Selected Licensee Commitments Manual:** __ • SLC 16.9-7 (Boration Systems Flow Path - Shutdown) __ • SLC 16.9-8 (Boration Systems Flow Path - Operating) __ • SLC 16.9-9 (Boration Systems Pumps - Shutdown) __ • SLC 16.9-10 (Boration Systems Charging Pumps - Operating) __ • 3.5.2 (ECCS - Operating) __ • 3.5.3 (ECCS - Shutdown) __ • 3.6.6 (Containment Spray System) __ • 3.7.5 (Auxiliary Feedwater (AFW) System) __ • 3.7.7 (Component Cooling Water (CCW) System).	**TECH SPEC EVALUATION** ***See Attachment 12 for applicable Tech Specs.*** **T.S. 3.7.7** **Condition A: Restore Component Cooling Water Train to OPERABLE in 72 hours.**
17. **Determine required notifications:** __ • **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO) __ • **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation).	
N/A 18. **IF KC Hx leak to RN suspected, THEN perform the following:** __ • Notify Radiation Protection that a potential unmonitored release may have occurred. __ • Notify Station Management to evaluate a KC Hx to RN leak.	

Op Test No.:	301	Scenario #	1	Event #	3	Page	43 of 153

Event Description: KC Miniflow Valve 1KC-C37A Fails Open

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 19 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. **Verify KC surge tanks level as follows:**	__ **RETURN TO Step 9.**
__ • Greater than 50%	
__ • Stable or trending up.	
20. **WHEN plant conditions permit, THEN perform the following:**	
__ • Return KC pumps to normal operation. **REFER TO** OP/1/A/6400/005 (Component Cooling Water System).	
__ • Return NV Pump 1A to normal cooling as applicable. **REFER TO** Enclosure 2 (Alternate Cooling To NV Pump 1A).	
21. **Verify the following:**	**Perform the following:**
__ • 1AD-7, F/3 "LETDN HX OUTLET HI TEMP" - DARK	__ a. **IF** letdown isolated, **THEN REFER TO** AP/1/A/5500/012 (Loss of Charging or Letdown).
__ • 1AD-7, H/3 "VCT HI TEMP" - DARK	__ b. Do not continue in this procedure until Step 21 conditions met.
__ • Normal letdown - IN SERVICE.	

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 20 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22. **Ensure VCT and letdown path aligned as follows:**	
a. **IF** desired to align NV pump suction to VCT, **THEN** perform the following:	
1) OPEN the following valves: __ • 1NV-188A (VCT Otlt Isol) __ • 1NV-189B (VCT Otlt Isol).	
2) Verify both the following valves - OPEN: __ • 1NV-188A (VCT Otlt Isol) __ • 1NV-189B (VCT Otlt Isol).	2) Perform the following: __ a) **IF** either valve in intermediate position, **THEN** allow 10 seconds for valve to open. __ b) **IF** both valves open, **THEN GO TO** 22.a.3. c) **IF** either valve closed **OR** in intermediate position, **THEN** perform the following: (1) CLOSE the following valves: __ • 1NV-188A (VCT Otlt Isol) __ • 1NV-189B (VCT Otlt Isol). __ (2) Notify Station Management that NV pump suction remains aligned to FWST. __ (3) **GO TO** Step 22.b.
3) CLOSE the following valves: __ • 1NV-252A (NV Pumps Suct From FWST) __ • 1NV-253B (NV Pumps Suct From FWST).	

Op Test No.:	301	Scenario #	1	Event #	3	Page	45 of 153
Event Description:	KC Miniflow Valve 1KC-C37A Fails Open						

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING	PAGE NO. 21 of 59 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

22. (Continued)

__ b. **WHEN** NV suction aligned to VCT, **THEN** momentarily place 1NV-172A (3-Way Divert To VCT-RHT) to "VCT" position and return to "AUTO".

c. **IF** desired to restore letdown flow through NV demineralizers, **THEN** perform the following:

__ • **WHEN** letdown temperature stable, **THEN** momentarily place 1NV-153A (Letdn Hx Otlt 3-Way Valve) to "DEMIN" position and return to "AUTO".

__ 23. **Determine long term plant status. RETURN TO procedure in affect.**

END

Note to Evaluator:

At lead evaluators discretion, the scenario may continue by having the booth operator insert Trigger 4 (Loss of KG / Automatic Turbine Runback Failure).

Op Test No.:	301	Scenario #	1	Event #	4	Page	46	of	153
Event Description:	Loss of KG / Automatic Turbine Runback Failure								

Control Room Indications
1AD-1, F/6 "H2-KG PANEL TROUBLE" – LIT
OAC indications of both KG pumps tripped

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 3 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Verify turbine load - TRENDING DOWN IN AUTOMATIC.**	**Perform the following:** __ a. Select "MANUAL" on turbine control panel. __ b. Depress "CONTROL VALVES LOWER" pushbutton and reduce turbine load as required. Critical Task #1
Note to Evaluator: **RO should take Turbine Control to manual and lower Turbine load to ~275 MWe. If the crew fails to get below the amps for 332 MW @ 0.9 power factor within 3.5 minutes, a turbine trip will occur.**	
2. **Verify proper reactor response:** __ • Control rods - IN "AUTO" AND STEPPING IN __ • P/R neutron flux - TRENDING DOWN.	**IF AT ANY TIME T-Avg greater than 1.5°F higher than T-Ref, THEN perform the following:** __ a. Insert control rods as required to maintain T-Avg within 1°F of T-Ref. b. IF control rods will NOT insert, THEN perform the following: __ 1) Trip Reactor. __ 2) GO TO EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 4 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **Verify proper steam dump operation as follows:**	
__ a. Verify T-Ref instrumentation - AVAILABLE.	__ a. **IF** T-Avg Coastdown in progress, **THEN** determine T-Ref from table. **REFER TO** Enclosure 4 (T-Ref Value Following Runback/Power Reduction).
__ b. "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT.	b. Perform the following: __ 1) Operate S/G PORVs as necessary to maintain T-Avg at T-Ref. __ 2) **GO TO** Step 4.
c. Verify the following: __ • "C-7A LOSS OF LOAD INTLK COND DUMP" status light (1SI-18) - LIT. __ • Steam dump valves - MODULATING.	c. **IF** steam dump valves closed **AND** T-Avg 3°F greater than T-Ref, **THEN** perform the following: __ 1) Place "STM DUMP CTRL" in manual. __ 2) Place steam dumps in pressure mode. __ 3) Operate condenser steam dump valves to maintain T-Avg at T-Ref. __ 4) **IF** steam dump valves fail to operate, **THEN** dump steam as necessary from available S/G PORVs to maintain T-Avg at T-Ref.

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 5 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. (Continued)	
__ d. T-Avg - TRENDING DOWN TO T-REF.	d. Perform the following: __ 1) Place "STM DUMP CTRL" in manual. __ 2) Place steam dumps in pressure mode. __ 3) Operate condenser steam dump valves to maintain T-Avg at T-Ref. __ 4) **IF** steam dump valves fail to operate, **THEN** dump steam as necessary from available S/G PORVs to maintain T-Avg at T-Ref.

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 6 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. **Verify Pzr PORV and Pzr spray valve status as follows:**	
__ a. All Pzr PORVs - CLOSED.	a. **IF** Pzr pressure less than 2315 PSIG, **THEN** perform the following: __ 1) CLOSE Pzr PORV(s). __ 2) **IF** any Pzr PORV cannot be closed, **THEN** close its isolation valve. 3) **IF** Pzr PORV isolation valve cannot be closed, **THEN** perform the following: __ a) Trip Unit 1 reactor. __ b) **WHEN** reactor tripped **OR** S/I setpoint reached, **THEN** ensure S/I initiated. __ c) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ b. Normal Pzr spray valves - CLOSED.	b. **IF** Pzr pressure less than 2150 PSIG, **THEN** perform the following: __ 1) CLOSE affected spray valve(s). __ 2) **REFER TO** AP/1/A/5500/011 (Pressurizer Pressure Anomolies).

Op Test No.:	301	Scenario #	1	Event #	4	Page	51 of 153
Event Description:	Loss of KG / Automatic Turbine Runback Failure						

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 7 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify proper CM System operation as follows:**	
__ a. **WHEN** reactor power less than 75%, **THEN** ensure both C-htr drain pumps - OFF.	
__ b. Verify reactor power - GREATER THAN 56% PRIOR TO THE EVENT.	__ b. **GO TO** Step 6.
__ c. Verify standby hotwell pump(s) - ON.	__ c. Start standby hotwell pump(s) as necessary.
__ d. Verify standby condensate booster pump(s) - ON.	__ d. Start standby condensate booster pump(s) as necessary.
6. **Verify the following generator alarms - DARK:**	__ **Ensure turbine generator load - REDUCED TO APPROXIMATELY 48% AND THE ALARM CLEARS.**
__ • 1AD-11, C/1 "GEN BKR A OVER CURRENT"	
__ • 1AD-11, F/1 "GEN BKR B OVER CURRENT".	
7. **Verify S/G levels adequate as follows:**	**Perform the following:**
__ • All S/G low level alert alarms (1AD-4) - DARK	__ a. Ensure feedwater regulating valves - MODULATING TO CONTROL S/G LEVELS AT PROGRAM SETPOINT.
__ • All S/G low CF flow alarms (1AD-4) - DARK.	b. **IF** any S/G(s) N/R level trending down in uncontrolled manner, **THEN** perform the following: __ 1) Trip reactor. __ 2) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 8 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 8. **Verify reactor power - GREATER THAN 20%.**	**Perform the following:** __ a. Place "CRD BANK SELECT" switch in manual. __ b. Maintain control rods above insertion limits. __ c. Operate control rods to stabilize unit at appropriate power level. d. **IF AT ANY TIME** reactor power less than or equal to 5%, **THEN** perform the following: __ 1) Ensure steam dumps in pressure mode. __ 2) Ensure turbine - TRIPPED. __ 3) Ensure steam dumps maintain steam header pressure at 1090 PSIG. __ 4) Concurrently insert control rods to shutdown the reactor. **REFER TO** OP/1/A/6150/008 (Rod Control). __ 5) **GO TO** AP/1/A/5500/002 (Turbine Generator Trip). __ e. **GO TO** Step 10.
__ 9. **IF AT ANY TIME reactor power less than or equal to 20%, THEN perform Step 8 RNO.**	

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 9 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Maintain AS header pressure as follows:**	
__ a. Verify runback target load less than 85%.	__ a. **GO TO** Step 10.d.
__ b. Adjust 1AS-2 (Main Stm To Aux Steam) setpoint to maintain AS header pressure at 165 PSIG.	__ b. Adjust 1AS-2 as required to maintain AS header pressure at 165 PSIG.
__ c. **GO TO** Step 11.	
__ d. Verify AS header pressure maintained between 140 PSIG and 165 PSIG.	__ d. Adjust 1AS-2 as required to maintain AS header pressure between 140 PSIG and 165 PSIG.
__ 11. **Adjust 1TL-4 (Stm Seal Reg Byp) as necessary to maintain steam seal pressure between 3 PSIG - 5 PSIG.**	
__ 12. **Monitor Enclosure 3 (Rod Insertion Limit Boration).**	

Note to Evaluator:

Enclosure 3 can be found as Attachment 4 in the back of this document.

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 10 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 13. **Verify reactor power - LESS THAN 30%.**	**Perform the following:** a. IF runback target load less than 30%, THEN perform the following: __ 1) WHEN time and manpower permit, THEN perform applicable steps of OP/1/A/6100/003 (Controlling Procedure For Unit Operation). __ 2) Do not continue in this procedure until reactor power less than 30%. __ 3) WHEN reactor power less than 30%, THEN GO TO Step 14. b. WHEN appropriate runback target load reached, THEN perform the following: __ 1) Stabilize unit at appropriate power level. __ 2) Maintain control rods above insertion limits. 3) Adjust the following as required to maintain T-Avg within 1°F of T-Ref: __ • Turbine load __ • Control rods __ • Boron concentration. __ c. GO TO Step 15.

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 11 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 14. **Verify "RESET" light on "AMSAC FOR CF VALVES" switch - DARK.**	**Perform the following:** __ a. **IF** turbine impulse pressure less than 190 PSIG, **THEN** notify IAE to correct cause of AMSAC failure to deactivate. __ b. Depress "BYPASS" pushbutton on "AMSAC FOR CF VALVES" switch. __ c. **WHEN** 2 minutes elapsed, **THEN** verify "RESET" light on "AMSAC FOR CF VALVES" switch remained dark.
__ 15. **Adjust power factor as necessary. REFER TO Unit 1 Revised Data Book, Figure 43 Generator Capability Curves.**	
16. **WHEN appropriate runback target load reached, THEN perform the following:** __ • Stabilize unit at appropriate power level. __ • Maintain control rods above insertion limits. • Adjust the following as required to maintain T-Avg within 1°F of T-Ref: __ • Turbine load __ • Control rods __ • Boron concentration.	

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 12 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 17. **Verify at least one KG pump - RUNNING.**	**Perform the following:** **NOTE** Maximum generating time without KG flow is 1 hour at less than 23%. __ a. Restore at least one KG pump to service. __ b. **IF** flow from at least one KG pump cannot be established within 1 hour, **THEN** shutdown turbine/generator. **REFER TO** OP/1/B/6300/001 (Turbine Generator).
18. **Verify the following PCBs - CLOSED:** __ • Generator breaker 1A __ • Generator breaker 1B __ • PCB 14 __ • PCB 15 __ • PCB 17 __ • PCB 18.	**Perform the following:** a. **IF** any generator connection with switchyard lost, **THEN** perform the following: 1) Notify DEC TOP (Transmission Operations) to investigate and repair loss of generator connection with switchyard: • Outside line: __ • 800-326-6534 __ • 800-326-6537. • TOP satellite phone: __ • Local: 1-828-490-7791 __ • International: 011-8816-234-50494. __ • Two-way radio. (RNO continued on next page)

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 13 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

18. (Continued)

2) **IF** any busline switchyard connection lost as follows:

__ • PCB 14 **AND** PCB 15 OPEN

OR

__ • PCB 17 **AND** PCB 18 OPEN,

__ **THEN** notify DEC TOP (Transmission Operations) to calculate Catawba RTCA (Real Time Contingency Analysis).

__ 3) **WHEN** time and manpower permit, **THEN** restore affected generator connection with switchyard. **REFER TO** Enclosure 1 (Offsite Power Restoration) .

b. **IF AT ANY TIME** the following condition exists:

__ • Any switchyard bus energized

AND

• Any Unit Tie PCB will be open greater than 1 hour,

__ **THEN** coordinate with Station Management to evaluate isolating affected Unit Tie PCB(s) to prevent damage to PCB capacitors. **REFER TO** OP/0/A/6350/010 (Operation of Station Breakers and Disconnects).

CNS AP/1/A/5500/003	LOAD REJECTION Case I Generator Connected To Switchyard	PAGE NO. 14 of 55 Revision 47

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. **Notify DEC BA (Balancing Authority) of Unit 1 status using one of the following:** __ • Red dispatcher phone __ • 800-943-7586 __ • BA satellite phone: __ • Local: 1-828-490-9313 __ • International: 011-8816-234-60905.	
__ 20. **Determine and correct cause of load rejection.**	

Note to Evaluator:

Once reactor power is stable, and at the lead evaluators discretion, the scenario may continue by having the booth operator insert Trigger 5 (1A NS Pump Loss of Power / Steam Leak Inside Containment).

Op Test No.:	301	Scenario #	1	Event #	5	Page	59 of 153
Event Description:	1A NS Pump Loss of Power						

Control Room Indications
1AD-11, A/1 “4KV ESS PWR TRAIN A TROUBLE” – LIT
‘A’ Train 1.47 Bypass alarm for 1A NS

Note To Evaluator:

This event does not have any specific crew response. This is an SRO Tech Spec call only.

See Attachment 12 for applicable Tech Specs.

Tech Spec 3.6.6 (Containment Spray System) Condition A (One containment spray train inoperable) – Restore containment spray train to OPERABLE status in 72 hours.

Op Test No.:	301	Scenario #	1	Event #	6	Page	60	of 153
Event Description:	Steam Leak Inside Containment on 1C S/G							

Control Room Indications
Containment Pressure – RISING
Containment Temperature – RISING
Containment Humidity – RISING

Op Test No.:	301	Scenario #	1	Event #	6	Page	61 of 153
Event Description:	Steam Leak Inside Containment on 1C S/G						

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 2 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 5 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
__ 2. **Verify turbine - ONLINE.**	__ **GO TO Step 6.**
3. **Verify the following:** __ • Reactor power - LESS THAN OR EQUAL TO 100% POWER __ • T-Avg - WITHIN 1.5°F OF T-Ref.	**Perform the following:** __ a. Select "MANUAL" on turbine control panel. b. Depress "CONTROL VALVES LOWER" pushbutton and reduce turbine load to maintain: __ • Reactor power - LESS THAN OR EQUAL TO 100% POWER __ • T-Avg - WITHIN 1.5°F OF T-Ref.
4. **Verify proper reactor response as follows:** __ • Control rods - IN AUTO AND STEPPING IN __ • P/R neutron flux - TRENDING DOWN.	__ **IF T-Avg greater than 1.5°F higher than T-Ref, THEN insert control rods as required to maintain T-Avg within 1°F of T-Ref.**
__ 5. **IF AT ANY TIME reactor power greater than 100%, THEN perform Step 3 RNO.**	

Event Description: Steam Leak Inside Containment on 1C S/G

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 3 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 6. **Verify Pzr level - STABLE OR TRENDING UP.**	**Perform the following:**

__ a. Maintain charging flow less than 180 GPM.

__ b. THROTTLE 1NV-294 (NV Pmps A&B Disch Flow Ctrl) to stabilize Pzr level.

__ c. **IF** Pzr level stable **OR** trending up, **THEN GO TO** Step 7.

d. **IF** Pzr level continues to trend down, **THEN** perform the following:

1) Reduce letdown flow to 45 GPM as follows:

a) **IF** 1NV-10A (Letdn Orif 1B Otlt Cont Isol) open, **THEN** perform the following:

__ (1) Control 1NV-148 (Letdn Press Control) to establish letdown pressure between 375 - 400 PSIG.

__ (2) THROTTLE 1NV-849 (Letdn Flow Var Orif Ctrl) for 45 GPM letdown flow.

__ (3) **WHEN** 45 GPM letdown flow established, **THEN** adjust 1NV-148 (Letdn Press Control) to maintain letdown pressure at 350 PSIG.

__ (4) **WHEN** letdown pressure stable at 350 PSIG, **THEN** place 1NV-148 (Letdn Press Control) in "AUTO".

(RNO continued on next page)

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 4 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
	b) **IF** 1NV-13A (Letdn Orif 1A Otlt Cont Isol) open, **THEN** perform the following:
	__ (1) Control 1NV-148 (Letdn Press Control) to establish letdown pressure between 150 - 200 PSIG.
	__ (2) OPEN 1NV-11A (Letdn Orif 1C Otlt Cont Isol).
	__ (3) Adjust 1NV-148 (Letdn Press Control) to establish letdown pressure between 375 - 400 PSIG.
	__ (4) CLOSE 1NV-13A (Letdn Orif 1A Otlt Cont Isol).
	__ (5) Adjust 1NV-148 (Letdn Press Control) to maintain letdown pressure at 350 PSIG.
	__ (6) **WHEN** letdown pressure stable at 350 PSIG, **THEN** place 1NV-148 (Letdn Press Control) in "AUTO".
	__ 2) **IF** Pzr level stable **OR** trending up, **THEN GO TO** Step 7.
	(RNO continued on next page)

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 5 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	3) **<u>IF</u>** Pzr level continues to trend down **<u>OR</u>** Pzr level cannot be maintained greater than 11%, **<u>THEN</u>** perform the following: __ a) Trip Unit 1 reactor. b) CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves. __ c) Initiate S/I. __ d) **<u>GO TO</u>** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ 7. **<u>IF AT ANY TIME</u> while in this procedure Pzr level trending down in uncontrolled manner, <u>THEN RETURN TO</u> Step 6.**	
8. **<u>IF AT ANY TIME</u> VCT level goes below 23%, <u>THEN</u> align NV pump suction to FWST as follows:** a. OPEN the following valves: __ • 1NV-252A (NV Pumps Suct From FWST) __ • 1NV-253B (NV Pumps Suct From FWST). b. CLOSE the following valves: __ • 1NV-188A (VCT Otlt Isol) __ • 1NV-189B (VCT Otlt Isol).	

Event Description: Steam Leak Inside Containment on 1C S/G

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 6 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Attempt to identify and isolate leak as follows:**	
a. Verify the following conditions - NORMAL: __ • Containment temperature __ • Containment pressure __ • Containment humidity __ • Containment floor & equipment sump level.	a. Perform the following: __ 1) Evacuate containment. 2) Perform the following: __ a) Start all lower containment ventilation units in low speed. __ b) Start all upper containment ventilation units. __ c) Place all upper and lower containment ventilation units in "MAX" cooling. 3) **IF AT ANY TIME** containment pressure reaches 1.2 PSIG, **THEN** perform the following: __ a) Ensure Unit 1 reactor tripped. __ b) Ensure S/I initiated. c) CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves. __ d) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). __ 4) **GO TO** Step 10.
__ b. Dispatch operators to locate and identify source of steam leak.	

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 7 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
__ c. Verify S/G PORVs - CLOSED.	c. **IF** S/G pressure less than 1090 PSIG, **THEN** perform the following: __ 1) CLOSE affected S/G PORV. 2) **IF** S/G PORV still open, **THEN** perform the following: __ a) CLOSE affected S/G PORV isolation valve. __ b) **IF** S/G PORV isolation valve still open, **THEN** dispatch operator to close S/G PORV isolation valve.
__ d. Verify condenser dump valves - CLOSED.	d. **IF** steam dumps required to be closed, **THEN** perform the following: 1) Select "OFF RESET" on the following switches: __ • "STEAM DUMP INTLK BYP TRN A" __ • "STEAM DUMP INTLK BYP TRN B". __ 2) **IF** valve will not close, **THEN** dispatch operator to close affected condenser dump valve isolation valve. (RNO continued on next page)

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 8 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	3) **WHEN** leaking condenser dump valve isolated **OR** repaired, **THEN** perform the following: a) **IF** steam dumps in pressure mode, **THEN** perform the following: __ (1) Place "STM DUMP CTRL" in manual. __ (2) Adjust "STM DUMP CTRL" to 0% demand. b) Return the following switches to "ON": __ • "STEAM DUMP INTLK BYP TRN A" __ • "STEAM DUMP INTLK BYP TRN B". c) **IF** steam dumps in pressure mode, **THEN** perform the following: __ (1) Adjust "STM DUMP CTRL" to control steam header pressure at value required by controlling procedure in effect. __ (2) **WHEN** desired, **THEN** place "STM DUMP CTRL" in auto.

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 9 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
__ e. Verify atmospheric dump valves - CLOSED.	e. **IF** steam dumps required to be closed, **THEN** perform the following:
	1) Select "OFF RESET" on the following switches:
	__ • "STEAM DUMP INTLK BYP TRN A"
	__ • "STEAM DUMP INTLK BYP TRN B".
	__ 2) **IF** valve will not close, **THEN** CLOSE affected atmospheric dump valve isolation valve.
	__ 3) **IF** isolation valve will not close, **THEN** dispatch operator to fail air to affected atmospheric dump valve.
	4) **WHEN** leaking atmospheric dump valve isolated **OR** repaired, **THEN** perform the following:
	a) **IF** steam dumps in pressure mode, **THEN** perform the following:
	__ (1) Place "STM DUMP CTRL" in manual.
	__ (2) Adjust "STM DUMP CTRL" to 0% demand.
	b) Return the following switches to "ON":
	__ • "STEAM DUMP INTLK BYP TRN A"
	__ • "STEAM DUMP INTLK BYP TRN B".
	(RNO continued on next page)

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 10 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	c) **IF** steam dumps in pressure mode, **THEN** perform the following: __ (1) Adjust "STM DUMP CTRL" to control steam header pressure at value required by controlling procedure in effect. __ (2) **WHEN** desired, **THEN** place "STM DUMP CTRL" in auto.
__ f. Verify CA PMP #1 - OFF.	__ f. **IF** operation of CA PMP #1 causing uncontrolled cooldown **AND** flow from CA PMP #1 not required, **THEN** stop CA PMP #1.
g. **IF** leak suspected to be in a doghouse, **THEN** CLOSE the following valves for affected doghouse: • Outside DH: __ • 1SM-77A (S/G 1A Otlt Hdr Bldwn C/V) __ • 1SM-74B (S/G 1D Otlt Hdr Bldwn C/V). OR • Inside DH: __ • 1SM-76B (S/G 1B Otlt Hdr Bldwn C/V) __ • 1SM-75A (S/G 1C Otlt Hdr Bldwn C/V).	

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 11 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
__ h. **WHEN** location of steam leak known, **THEN** notify plant personnel by plant page to stay clear of the area, as required.	
10. **Determine required notifications:**	
__ • **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO)	
__ • **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation).	
11. **Ensure compliance with appropriate Tech Specs and Selected Licensee Commitments Manual:**	
__ • 3.6.3 (Containment Isolation Valves)	
__ • 3.7.4 (Steam Generator Power Operated Relief Valves (SG PORVs)).	
__ 12. **Notify RP of leak.**	
__ 13. **Verify - LEAK ISOLATED.**	__ **GO TO Step 15.**
__ 14. **Determine long term plant status. RETURN TO procedure and step in effect.**	
__ 15. **Verify UST level - STABLE OR TRENDING UP.**	**Perform the following:** __ a. Initiate makeup to UST. __ b. Notify Secondary Chemistry of increased makeup.

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 12 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16. **Evaluate Unit 1 shutdown as follows:**	
__ a. Verify Unit 1 status - IN MODE 1 **OR** 2.	__ a. **GO TO** Step 19.
b. Determine if Unit 1 shutdown or load reduction warranted based on the following criteria:	
__ • Size of leak	
__ • Location of leak	
__ • Rate of depletion of secondary inventory	
__ • Steam leak cannot be isolated or repaired at power	
__ • SM judgment	
__ • **IF** steam leaking from secondary heater relief **OR** MSR relief valve, **THEN** reducing turbine load may reduce pressure enough to close relief valve.	
__ • **IF** turbine trip will isolate steam leak, **THEN** it may be desirable to perform an orderly shutdown of the turbine and maintain reactor power in Mode 1.	
__ c. Verify Unit 1 shutdown or load reduction - REQUIRED.	c. Perform the following: __ 1) Maintain present plant conditions until leak can be isolated or repaired. __ 2) **RETURN TO** procedure and step in effect.
__ d. Verify immediate isolation of steam leak - REQUIRED.	__ d. **GO TO** Step 16.i.

Note to Evaluator:

Once determination of Unit shutdown being required, and at the discretion of the lead evaluator, the scenario may continue by having the booth operator insert Trigger 7 (Steam Line Break Inside Containment / MSIVs fail to close on MSI signal / Auxiliary Building Unfiltered Exhaust Fans Fail to secure on S/I).

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK	PAGE NO. 13 of 43 Revision 12

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16. (Continued)	
__ e. Verify steam leak - KNOWN TO BE ISOLABLE BY TURBINE TRIP.	e. Perform the following: __ 1) Trip Unit 1 reactor. 2) CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves. __ 3) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ f. Verify reactor power - GREATER THAN 69%.	f. Perform the following: __ 1) Trip turbine. __ 2) **GO TO** AP/1/A/5500/002 (Turbine Generator Trip).
__ g. Trip Unit 1 reactor.	
__ h. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).	
i. Initiate Unit 1 shutdown. **REFER TO** one of the following: __ • OP/1/A/6100/003 (Controlling Procedure For Unit Operation) OR __ • OP/1/A/6100/002 (Controlling Procedure For Unit Shutdown) OR __ • AP/1/A/5500/009 (Rapid Downpower).	

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	73	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

Control Room Indications
1FO-1, D/5 "HI CONT PRESS S/I RX TRIP" – LIT
Unit 1 Reactor Trip and Safety Injection

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	74	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 4 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 6 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Verify Reactor Trip:** __ • All rod bottom lights - LIT __ • All reactor trip and bypass breakers - OPEN __ • I/R power - TRENDING DOWN.	**Perform the following:** __ a. Trip reactor. b. **IF** reactor will not trip, **THEN** concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • **GO TO** EP/1/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS).
3. **Verify Turbine Trip:** __ • All turbine stop valves - CLOSED.	**Perform the following:** __ a. Trip turbine. b. **IF** turbine will not trip, **THEN** perform the following: __ 1) Depress "MANUAL" pushbutton on turbine control panel. __ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons. 3) **IF** control valves will not close, **THEN** CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves.

Op Test No.: 301 Scenario # 1 Event # 7,8,9	Page 75 of 153
Event Description:	Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 5 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 4. **Verify 1ETA and 1ETB - ENERGIZED.**	**Perform the following:** __ a. **IF** 1ETA **AND** 1ETB de-energized, **THEN GO TO** EP/1/A/5000/ECA-0.0 (Loss of All AC Power). __ b. **WHEN** time allows, **THEN** attempt to restore power to de-energized switchgear while continuing with this procedure. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power).

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 6 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify S/I actuated:**	
__ a. "SAFETY INJECTION ACTUATED" status light (1SI-13) - LIT.	a. Perform the following: 1) Verify conditions requiring S/I: __ • Pzr pressure - LESS THAN 1845 PSIG OR __ • Containment pressure - GREATER THAN 1.2 PSIG. __ 2) <u>IF</u> S/I required, <u>**THEN**</u> initiate S/I. 3) <u>IF</u> S/I not required, <u>**THEN**</u> concurrently perform the following: • <u>IF</u> 1ETA <u>**OR**</u> 1ETB de-energized, <u>**THEN**</u> ensure the following pumps running on energized bus: __ • NV pump __ • KC pumps __ • RN pump. __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees). __ • <u>**GO TO**</u> EP/1/A/5000/ES-0.1 (Reactor Trip Response).
__ b. Both E/S load sequencer actuated status lights (1SI-14) - LIT.	__ b. Initiate S/I.
__ 6. **Announce "Unit 1 Safety Injection".**	

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 7 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Determine required notifications:**	
__ • **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO)	
__ • **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation).	
__ 8. **Verify all Feedwater Isolation status lights (1SI-5) - LIT.**	**Perform the following:** __ a. Initiate Feedwater Isolation. __ b. **IF** proper status light indication not obtained, **THEN** CLOSE valves.
9. **Verify Phase A Containment Isolation status as follows:**	
__ a. Phase A "RESET" lights - DARK.	__ a. Initiate Phase A Isolation.
__ b. Monitor Light Panel Group 5 St lights on energized train(s) - LIT.	__ b. Align valves as necessary to ensure each penetration isolated by at least one isolation valve.

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	78	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 8 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

10. **Verify Phase B actuation as follows:**

__ a. Verify containment pressure - HAS REMAINED LESS THAN 3 PSIG.

a. Perform the following:

1) Verify Phase B Isolation actuated as follows:

__ a) Phase B Isolation "RESET" lights - DARK.

N/A__ b) **IF** Phase B Isolation "RESET" lights lit, **THEN** initiate Phase B Isolation.

c) Verify following monitor light panel lights on energized train(s) - LIT:

__ • Group 1 Sp lights

__ • Group 5 Sp lights

__ • Group 5 St light L/11.

N/A__ d) **IF** monitor light panel not in correct alignment, **THEN** ensure correct alignment.

__ 2) Stop all NC pumps.

__ 3) Maintain seal injection flow.

__ 4) Energize H_2 igniters.

(RNO continued on next page)

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	79	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 9 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
	5) Dispatch operator to perform the following:
	__ a) Secure all ice condenser air handling units. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 11 (Securing All Ice Condenser Units).
	__ b) Place containment H_2 analyzers in service. **REFER TO** OP/1/A/6450/010 (Containment Hydrogen Control Systems).
Note to Evaluator: **Enclosure 5 can be found as Attachment 7 in the back of this document.**	__ 6) **WHEN** 9 minutes elapsed, **THEN** verify proper VX System operation. **REFER TO** Enclosure 5 (VX System Operation).
	__ 7) **GO TO** Step 11.
__ b. **IF AT ANY TIME** containment pressure exceeds 3 PSIG while in this procedure, **THEN** perform Step 10.a.	

Op Test No.: 301	Scenario # 1	Event # 7,8,9	Page 80 of 153				

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 10 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **Verify proper CA pump status as follows:**	
__ a. Motor driven CA pumps - ON.	a. Perform the following for affected train(s): __ 1) Reset ECCS. __ 2) Reset D/G load sequencer(s). __ 3) Start affected pump(s). __ 4) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.
__ b. 3 S/G N/R levels - GREATER THAN 11%.	b. Perform the following: __ 1) Place CA PMP #1 control switch to ON. __ 2) Ensure CA pump #1 - RUNNING.
12. **Verify all of the following S/I pumps - ON:** __ • NV pumps __ • ND pumps __ • NI pumps.	**Perform the following for affected train(s):** __ a. Reset ECCS. __ b. Reset D/G load sequencer(s). __ c. Start affected pump(s). __ d. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.

Op Test No.: 301 Scenario # 1 Event # 7,8,9	Page 81 of 153
Event Description:	Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 11 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 13. **Verify all KC pumps - ON.**	**Perform the following for affected train(s):** __ a. Reset ECCS. __ b. Reset D/G load sequencer(s). __ c. Start affected pump(s). __ d. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on. __ e. **IF** KC flow cannot be established to NC pumps, **THEN** stop all NC pumps.
__ 14. **Verify all Unit 1 and Unit 2 RN pumps - ON.**	**Perform the following:** __ a. **IF** any Unit 2 RN pump off, **THEN** start affected pump(s). b. **IF** any Unit 1 RN pump off, **THEN** perform the following for affected train(s): __ 1) Reset ECCS. __ 2) Reset D/G load sequencer(s). __ 3) Start affected pump(s). __ 4) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.
15. **Verify proper ventilation systems operation as follows:** __ • **REFER TO** Enclosure 2 (Ventilation System Verification). __ • Notify Unit 2 operator to perform Enclosure 3 (Opposite Unit Ventilation Verification).	**Note to Evaluator:** **Enclosure 2 can be found as Attachment 8 in the back of this document. BOP will use this Enclosure to secure the Auxiliary Building Unfiltered Exhaust Fans that failed to automatically secure following the S/I.**

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	82 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 12 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 16. **Verify all S/G pressures - GREATER THAN 775 PSIG.**	**Perform the following:**
	a. Verify Main Steam Isolation as follows: __ • All MSIVs - CLOSED __ • All MSIV bypass valves - CLOSED __ • All S/G PORVs - CLOSED.
	b. **IF** any valve open, **THEN** perform the following:
	__ 1) Initiate Main Steam Isolation.
CRITICAL TASK #2	__ 2) **IF** any valve still open, **THEN** CLOSE valve.
17. **Verify proper S/I flow as follows:**	
__ a. "NV S/I FLOW" - INDICATING FLOW.	__ a. Start NV pump(s) and align valves.

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	83 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 13 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ b. NC pressure - LESS THAN 1620 PSIG.	b. Perform the following: __ 1) Ensure ND pump miniflow valve on operating ND pump(s) - OPEN. 2) **IF** ND pump miniflow valve(s) cannot be opened, **THEN** perform the following for affected train(s): __ a) Reset ECCS. __ b) Reset D/G load sequencer. __ c) Stop ND pump. __ d) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on. __ e) **IF AT ANY TIME** NC pressure goes down to less than 285 PSIG in uncontrolled manner, **THEN** restart ND pump. __ 3) **GO TO** Step 18.
__ c. NI pumps - INDICATING FLOW.	__ c. Start NI pump(s) and align valves.

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 14 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ d. NC pressure - LESS THAN 285 PSIG.	d. Perform the following:
	__ 1) Ensure ND pump miniflow valve on operating ND pump(s) - OPEN.
	N/A 2) **IF** ND pump miniflow valve(s) cannot be opened, **THEN** perform the following for affected train(s):
	__ a) Reset ECCS.
	__ b) Reset D/G load sequencer.
	__ c) Stop ND pump.
	__ d) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.
	__ e) **IF AT ANY TIME** NC pressure goes down to less than 285 PSIG in uncontrolled manner, **THEN** restart ND pump.
	__ 3) **GO TO** Step 18.
__ e. ND pumps - INDICATING FLOW TO C-LEGS.	__ e. Start ND pump(s) and align valves.
__ 18. **WHEN time and manpower permit (within two hours of event), THEN monitor Spent Fuel Pool level and temperature. REFER TO EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 1 (Unit 1 Spent Fuel Pool Monitoring).**	

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	85	of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 15 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. **Control S/G levels as follows:**	
__ a. Verify total CA flow - GREATER THAN 450 GPM.	a. Perform the following: 1) **IF** N/R level in all S/Gs less than 11% (29% ACC), **THEN** perform the following: __ • Start CA pumps __ • Ensure correct valve alignment. 2) **IF** N/R level in all S/Gs less than 11% (29% ACC) **AND** feed flow greater than 450 GPM cannot be established, **THEN** concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • **GO TO** EP/1/A/5000/FR-H.1 (Response to Loss of Secondary Heat Sink).
__ b. **WHEN** each S/G N/R level greater than 11% (29% ACC), **THEN** control feed flow to maintain that S/G N/R level between 11% (29% ACC) and 50%.	
__ 20. **Verify all CA isolation valves on intact S/Gs - OPEN.**	__ **OPEN valve(s).**
__ 21. **Verify S/I equipment status based on monitor light panel(s) - IN PROPER ALIGNMENT.**	__ **Align equipment.**

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	86 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 16 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE Enclosure 4 (NC Temperature Control) shall remain in effect until subsequent procedures provide alternative NC temperature control guidance.

Note to Evaluator:

Enclosure 4 can be found as Attachment 9 in the back of this document.

__ 22. **Control NC temperature. REFER TO Enclosure 4 (NC Temperature Control).**

23. **Verify Pzr PORV and Pzr Spray Valve status as follows:**

__ a. All Pzr PORVs - CLOSED.

RNO:

a. IF Pzr pressure less than 2315 PSIG, THEN perform the following:

__ 1) CLOSE Pzr PORV(s).

__ 2) IF any Pzr PORV cannot be closed, THEN CLOSE its isolation valve.

3) IF 1NC-32B OR 1NC-34A cannot be closed OR isolated, THEN perform the following:

a) Align N_2 to PORVs by opening the following valves:

__ • 1NI-438A (Emer N2 From CLA A To 1NC-34A)

__ • 1NI-439B (Emer N2 From CLA B To 1NC-32B).

__ b) CLOSE affected Pzr PORV.

(RNO continued on next page)

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	87 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 17 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

23. (Continued)

4) **IF** any Pzr PORV cannot be closed **OR** isolated, **THEN** perform the following:

__ a) Energize H_2 igniters.

b) Dispatch operator to perform the following:

__ (1) Secure all ice condenser air handling units. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 11 (Securing All Ice Condenser Units).

__ (2) Place containment H_2 analyzers in service. **REFER TO** OP/1/A/6450/010 (Containment Hydrogen Control Systems).

c) **IF AT ANY TIME** both the following conditions exist:

__ • Containment pressure - HAS REMAINED LESS THAN 3 PSIG

__ • Containment pressure - BETWEEN 1 PSIG **AND** 3 PSIG,

__ **THEN** start one VX fan and secure normal containment ventilation. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 18 (VX and Containment Ventilation Control).

(RNO continued on next page)

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	88 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 18 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23. (Continued)	
	d) Concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • **GO TO** EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant).
__ b. Normal Pzr spray valves - CLOSED.	b. **IF** Pzr pressure less than 2150 PSIG, **THEN** perform the following: __ 1) CLOSE spray valve(s). 2) **IF** spray valve(s) cannot be closed, **THEN** perform the following: __ a) Stop NC pumps 1A and 1B. __ b) **IF** both 1C **AND** 1D NC pumps on, **THEN** stop one additional pump.
__ c. At least one Pzr PORV isolation valve - OPEN.	__ c. **IF** power available, **THEN** OPEN one Pzr PORV isolation valve unless it was closed to isolate an open Pzr PORV.
__ 24. **Verify NC subcooling based on core exit T/Cs - GREATER THAN 0°F.**	**IF any NV OR NI pump on, THEN perform the following:** __ a. Ensure all NC pumps - OFF. __ b. Maintain seal injection flow.

Op Test No.: 301 Scenario # 1 Event # 7,8,9 Page 89 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 19 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
25. **Verify main steamlines intact:**	**IF any S/G faulted, THEN perform the following:**
__ • All S/G pressures - STABLE OR TRENDING UP	__ a. Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees).
__ • ALL S/Gs - PRESSURIZED.	__ b. **GO TO** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation).
26. **Verify S/G tubes intact as follows:**	**IF any EMF trip 1 light lit OR any S/G level trending up in uncontrolled manner, THEN concurrently:**
__ • All S/G levels - STABLE OR TRENDING UP IN A CONTROLLED MANNER.	__ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees).
• Verify the following EMF trip 1 lights - DARK:	__ • **GO TO** EP/1/A/5000/E-3 (Steam Generator Tube Rupture).
__ • 1EMF-33 (Condenser Air Ejector Exhaust)	
__ • 1EMF-26 (Steamline 1A)	
__ • 1EMF-27 (Steamline 1B)	
__ • 1EMF-28 (Steamline 1C)	
__ • 1EMF-29 (Steamline 1D).	

Op Test No.: 301	Scenario # 1	Event # 7,8,9	Page 90 of 153				
Event Description:	Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I						

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 2 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 10 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
__ 2. **Maintain any faulted S/G or secondary break isolated during subsequent recovery actions unless needed for NC System cooldown.**	
3. **Verify the following valves - CLOSED:** __ • All MSIVs __ • All MSIV bypass valves.	__ **CLOSE valve(s).**
4. **WHEN TSC staffed, THEN notify TSC of the following:** __ • IF feedline OR steamline break has occurred inside doghouse, THEN ensure affected doghouse curtains opened within 24 hours of rupture inside of the doghouse.	
__ 5. **Verify any S/G pressure - STABLE OR TRENDING UP.**	__ **IF all S/Gs faulted, THEN GO TO EP/1/A/5000/ECA-2.1 (Uncontrolled Depressurization of All Steam Generators).**
6. **Identify faulted S/G(s) as follows:** __ • Verify any S/G pressure - TRENDING DOWN IN UNCONTROLLED MANNER OR __ • Verify any S/G - DEPRESSURIZED.	**Perform the following:** a. Dispatch operators to search for initiating break at the following locations: __ • Main steamlines __ • Main feedlines __ • Other secondary piping. __ b. GO TO Step 11.

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	91 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 3 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 7. **Maintain at least one S/G available for NC System cooldown in subsequent steps.**	
__ 8. **Verify faulted S/G(s) PORV - CLOSED.**	**Perform the following:** __ a. CLOSE faulted S/G(s) PORV. __ b. **IF** S/G PORV cannot be closed, **THEN** CLOSE S/G PORV isolation valve. __ c. **IF** S/G PORV isolation valve cannot be closed, **THEN** dispatch operator to close valve.
__ 9. **Ensure CA System valve control - RESET.**	
10. **Isolate all faulted S/G(s) as follows:**	
• S/G 1A:	
__ a. Verify S/G 1A Feedwater Isolation status light (1SI-5) - LIT.	a. Perform the following: 1) Ensure the following valves - CLOSED: __ • 1CF-28 (S/G 1A CF Ctrl) __ • 1CF-30 (S/G 1A CF Byp Ctrl) __ • 1CF-33 (S/G 1A CF Cont Isol) __ • 1CF-90 (S/G 1A CF Cont Isol Byp) __ • 1CA-149 (S/G 1A CF Byp To CA Nozzle) __ • 1CA-185 (S/G 1A CA Nozz Tempering Isol). (RNO continued on next page)

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	92	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 4 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

10. (Continued)

2) **IF** 1CA-185 (S/G 1A CA Nozz Tempering Isol) cannot be closed, **THEN** perform the following:

   a) CLOSE the following valves:

   __ • 1CF-100 (S/G CA Nozz Tempering Ctrl)

   __ • 1CF-156 (Byp Valve For 1CF-100).

   b) **IF** 1CF-100 **OR** 1CF-156 cannot be closed, **THEN** dispatch operator to close affected valve(s):

   __ • 1CF-100 (S/G CA Nozz Tempering Ctrl) (TB1-580, 1H-33) (Ladder needed)

   __ • 1CF-156 (Byp Valve For 1CF-100) (TB1-577, 1H-33) (Ladder needed).

__ 3) **IF** more than one Feedwater Isolation valve above open **AND** CM still aligned to feed faulted S/G, **THEN** evaluate alternate means to stop CM flow to faulted S/G.

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	93	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 5 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
b. CLOSE the following valves:	
__ 1) 1SM-77A (S/G 1A Otlt Hdr Bldwn C/V).	__ 1) Dispatch operator to close 1SM-77A (S/G 1A Otlt Hdr Bldwn C/V) (DH-583, FF-GG, 43-44, Rm 591).
__ 2) 1CA-62A (CA Pmp A Disch To S/G 1A Isol).	2) Perform the following: __ a) CLOSE 1CA-60 (CA Pump 1A Flow To S/G 1A). __ b) Dispatch operator to close 1CA-62A (CA Pmp A Disch To S/G 1A Isol) (DH-587, DD-EE, 44-45, Rm 591). __ c) **IF** exterior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-59 (CA Pump 1A Disch To S/G 1A Ctrl Inlet Isol) (AB-551, BB,49-50, Rm 250) (Ladder needed) (Key #633).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	94 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 6 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 3) 1CA-66B (CA Pmp 1 Disch To S/G 1A Isol).	3) Perform the following: __ a) CLOSE 1CA-64 (CA Pump #1 Flow To S/G 1A). __ b) Dispatch operator to close 1CA-66B (CA Pmp 1 Disch To S/G 1A Isol) (DH-584, DD-EE, 44-45, Rm 591). __ c) **IF** exterior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-63 (CA Pump No 1 Disch To S/G 1A Ctrl Inlet Isol) (AB-556, BB-50, Rm 250) (Ladder needed) (Key #633).
c. Verify the following blowdown isolation valves - CLOSED:	
__ 1) 1BB-56A (S/G 1A Bldwn Cont Isol Insd).	__ 1) CLOSE valve.

Op Test No.: 301 Scenario # 1 Event # 7,8,9	Page 95 of 153
Event Description:	Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 7 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 2) 1BB-148B (S/G 1A Bldwn Cont Isol Byp).	2) Perform the following: __ a) CLOSE valve. b) **<u>IF</u>** valve will not close **<u>AND</u>** 1BB-56A open, **<u>THEN</u>** perform the following: __ (1) Ensure "S/G A BLDWN FLOW CTRL" - CLOSED. (2) Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-148B (S/G 1A Bldwn Cont Isol Byp) (DH-580, EE-FF, 44-45, Rm 591) __ • 1BB-81 (1A S/G Blowdown Penetration Valve Test Isol) (DH-583, EE-FF, 44, Rm 591).

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 8 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 3) 1BB-57B (S/G 1A Bldwn Cont Isol Otsd).	3) Perform the following: __ a) CLOSE valve. b) **IF** valve will not close **AND** 1BB-56A open, **THEN** perform the following: __ (1) Ensure "S/G A BLDWN FLOW CTRL" - CLOSED. (2) Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-57B (S/G 1A Bldwn Cont Isol Otsd) (DH-580, EE-FF, 44-45, Rm 591) __ • 1BB-81 (1A S/G Blowdown Penetration Valve Test Isol) (DH-583, EE-FF, 44, Rm 591).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	97 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 9 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
• S/G 1B:	
__ a. Verify S/G 1B Feedwater Isolation status light (1SI-5) - LIT.	a. Perform the following:
	1) Ensure the following valves - CLOSED:
	__ • 1CF-37 (S/G 1B CF Ctrl)
	__ • 1CF-39 (S/G 1B CF Byp Ctrl)
	__ • 1CF-42 (S/G 1B CF Cont Isol)
	__ • 1CF-89 (S/G 1B CF Cont Isol Byp)
	__ • 1CA-150 (S/G 1B CF Byp To CA Nozzle)
	__ • 1CA-186 (S/G 1B CA Nozz Tempering Isol).
	2) **IF** 1CA-186 (S/G 1B CA Nozz Tempering Isol) cannot be closed, **THEN** perform the following:
	a) CLOSE the following valves:
	__ • 1CF-100 (S/G CA Nozz Tempering Ctrl)
	__ • 1CF-156 (Byp Valve For 1CF-100).
	(RNO continued on next page)

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 10 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
	b) **IF** 1CF-100 **OR** 1CF-156 cannot be closed, **THEN** dispatch operator to close affected valve(s): __ • 1CF-100 (S/G CA Nozz Tempering Ctrl) (TB1-580, 1H-33) (Ladder needed) __ • 1CF-156 (Byp Valve For 1CF-100) (TB1-577, 1H-33) (Ladder needed).
	__ 3) **IF** more than one Feedwater Isolation valve above open **AND** CM still aligned to feed faulted S/G, **THEN** evaluate alternate means to stop CM flow to faulted S/G.
b. CLOSE the following valves:	
__ 1) 1SM-76B (S/G 1B Otlt Hdr Bldwn C/V).	__ 1) Dispatch operator to close 1SM-76B (S/G 1B Otlt Hdr Bldwn C/V) (DH-583, FF-53, Rm 572).

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 11 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 2) 1CA-58A (CA Pmp A Disch To S/G 1B Isol).	2) Perform the following: __ a) CLOSE 1CA-56 (CA Pump 1A Flow To S/G 1B). __ b) Dispatch operator to close 1CA-58A (CA Pmp A Disch To S/G 1B Isol) (DH-586, DD-EE, 52-53, Rm 572). __ c) **IF** interior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-55 (CA Pump 1A Disch To S/G 1B Inlet Isol) (AB-550, DD-52, Rm 250) (Key #633).
__ c. Verify CA Pump 1A or 1B - AVAILABLE.	c. **IF** CA Pump #1 only source of feedwater, **THEN** perform the following: __ 1) Maintain at least one S/G available to supply steam to CA Pump #1. __ 2) Ensure feed flow maintained available to S/G used to supply steam to CA Pump #1. __ 3) **IF** desired to isolate CA Pump #1 from 1B S/G, **THEN GO TO** Step 10.d. __ 4) **GO TO** Step 10.f.

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 12 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ d. CLOSE 1CA-54B (CA Pmp 1 Disch To S/G 1B Isol).	d. Perform the following: __ 1) CLOSE 1CA-52 (CA Pump #1 Flow To S/G 1B). __ 2) Dispatch operator to close 1CA-54B (CA Pmp 1 Disch To S/G 1B Isol) (DH-584, DD-EE, 52-53, Rm 572). __ 3) **IF** interior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-51 (CA Pump No 1 Disch To S/G 1B Inlet Isol) (AB-552, DD-52, Rm 250) (Key #633).
__ e. Dispatch operator to unlock and close 1SA-1 (1B S/G Main Steam to CAPT Maintenance Isol) (DH-624, FF-53, Rm 572) (Breakaway lock).	__ e. Dispatch operator to unlock and close 1SA-3 (1B S/G Main Steam to CAPT Stop Check) (AB-551, DD-53, Rm 217) (Breakaway lock).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	101 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 13 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
f. Verify the following blowdown isolation valves - CLOSED:	
__ 1) 1BB-19A (S/G 1B Bldwn Cont Isol Insd).	__ 1) CLOSE valve.
__ 2) 1BB-150B (S/G 1B Bldwn Cont Isol Byp).	2) Perform the following: __ a) CLOSE valve. b) **IF** valve will not close **AND** 1BB-19A open, **THEN** perform the following: __ (1) Ensure "S/G B BLDWN FLOW CTRL" - CLOSED. (2) Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-150B (S/G 1B Bldwn Cont Isol Byp) (DH-580, FF, 52-53, Rm 572) __ • 1BB-83 (1B S/G Blowdown Penetration Valve Test Isol) (DH-580, FF-53, Rm 572).

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 14 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 3) 1BB-21B (S/G 1B Bldwn Cont Isol Otsd).	3) Perform the following: __ a) CLOSE valve. b) **IF** valve will not close **AND** 1BB-19A open, **THEN** perform the following: __ (1) Ensure "S/G B BLDWN FLOW CTRL" - CLOSED. (2) Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-21B (S/G 1B Bldwn Cont Isol Otsd) (DH-580, FF, 52-53, Rm 572) __ • 1BB-83 (1B S/G Blowdown Penetration Valve Test Isol) (DH-580, FF-53, Rm 572).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	103	of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 15 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued) • S/G 1C: __ a. Verify S/G 1C Feedwater Isolation status light (1SI-5) - LIT.	a. Perform the following: 1) Ensure the following valves - CLOSED: __ • 1CF-46 (S/G 1C CF Ctrl) __ • 1CF-48 (S/G 1C CF Byp Ctrl) __ • 1CF-51 (S/G 1C CF Cont Isol) __ • 1CF-88 (S/G 1C CF Cont Isol Byp) __ • 1CA-151 (S/G 1C CF Byp To CA Nozzle) __ • 1CA-187 (S/G 1C CA Nozz Tempering Isol). 2) **IF** 1CA-187 (S/G 1C CA Nozz Tempering Isol) cannot be closed, **THEN** perform the following: a) CLOSE the following valves: __ • 1CF-100 (S/G CA Nozz Tempering Ctrl) __ • 1CF-156 (Byp Valve For 1CF-100). (RNO continued on next page)

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 16 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
	b) **IF** 1CF-100 **OR** 1CF-156 cannot be closed, **THEN** dispatch operator to close affected valve(s): __ • 1CF-100 (S/G CA Nozz Tempering Ctrl) (TB1-580, 1H-33) (Ladder needed) __ • 1CF-156 (Byp Valve For 1CF-100) (TB1-577, 1H-33) (Ladder needed).
	__ 3) **IF** more than one Feedwater Isolation valve above open **AND** CM still aligned to feed faulted S/G, **THEN** evaluate alternate means to stop CM flow to faulted S/G.
b. CLOSE the following valves:	
__ 1) 1SM-75A (S/G 1C Otlt Hdr Bldwn C/V).	__ 1) Dispatch operator to close 1SM-75A (S/G 1C Otlt Hdr Bldwn C/V) (DH-580, GG-52/53, Rm 572).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	105	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 17 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 2) 1CA-46B (CA Pmp B Disch To S/G 1C Isol).	2) Perform the following: __ a) CLOSE 1CA-44 (CA Pump 1B Flow To S/G 1C). __ b) Dispatch operator to close 1CA-46B (CA Pmp B Disch To S/G 1C Isol) (DH-586, DD, 53-54, Rm 572). __ c) **IF** interior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-43 (CA Pump 1B Disch To S/G 1C Ctrl Inlet Isol) (AB-552, DD-52, Rm 250) (Key #633).
__ c. Verify CA Pump 1A or 1B - AVAILABLE.	c. **IF** CA Pump #1 only source of feedwater, **THEN** perform the following: __ 1) Maintain at least one S/G available to supply steam to CA Pump #1. __ 2) Ensure feed flow maintained available to S/G used to supply steam to CA Pump #1. __ 3) **IF** desired to isolate CA Pump #1 from 1C S/G, **THEN GO TO** Step 10.d. __ 4) **GO TO** Step 10.f.

Op Test No.: 301	Scenario # 1	Event # 7,8,9	Page 106 of 153				

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 18 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ d. CLOSE 1CA-50A (CA Pmp 1 Disch To S/G 1C Isol).	d. Perform the following: __ 1) CLOSE 1CA-48 (CA Pump #1 Flow To S/G 1C). __ 2) Dispatch operator to close 1CA-50A (CA Pmp 1 Disch To S/G 1C Isol) (DH-584, EE-53, Rm 572). __ 3) **IF** interior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-47 (CA Pump No 1 Disch To S/G 1C Ctrl Inlet Isol) (AB-552, DD-53, Rm 250) (Key #633).
__ e. Dispatch operator to unlock and close 1SA-4 (1C S/G Main Steam to CAPT Maintenance Isol) (DH-624, FF-53, Rm 572) (Breakaway lock installed).	__ e. Dispatch operator to unlock and close 1SA-6 (1C S/G Main Steam to CAPT Stop Check) (AB-551, DD-53, Rm 217) (Breakaway lock) (Ladder needed).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	107	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 19 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
f. Verify the following blowdown isolation valves - CLOSED:	
__ 1) 1BB-60A (S/G 1C Bldwn Cont Isol Insd).	__ 1) CLOSE valve.
__ 2) 1BB-149B (S/G 1C Bldwn Cont Isol Byp).	2) Perform the following: __ a) CLOSE valve. b) **IF** valve will not close **AND** 1BB-60A open, **THEN** perform the following: __ (1) Ensure "S/G C BLDWN FLOW CTRL" - CLOSED. (2) Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-149B (S/G 1C Bldwn Cont Isol Byp) (DH-578, FF-GG, 52, Rm 572) __ • 1BB-82 (1C S/G Blowdown Penetration Valve Test Isol) (DH-583, FF-53, Rm 572).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	108	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 20 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 3) 1BB-61B (S/G 1C Bldwn Cont Isol Otsd).	3) Perform the following: __ a) CLOSE valve. b) **IF** valve will not close **AND** 1BB-60A open, **THEN** perform the following: __ (1) Ensure "S/G C BLDWN FLOW CTRL" - CLOSED. (2) Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-61B (S/G 1C Bldwn Cont Isol Otsd) (DH-578, FF-GG, 52, Rm 572) __ • 1BB-82 (1C S/G Blowdown Penetration Valve Test Isol) (DH-583, FF-53, Rm 572).

Op Test No.: 301	Scenario # 1	Event # 7,8,9	Page 109 of 153
Event Description:	Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I		

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 21 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
• <u>S/G 1D</u>:	
__ a. Verify S/G 1D Feedwater Isolation status light (1SI-5) - LIT.	a. Perform the following: 1) Ensure the following valves - CLOSED: __ • 1CF-55 (S/G 1D CF Ctrl) __ • 1CF-57 (S/G 1D CF Byp Ctrl) __ • 1CF-60 (S/G 1D CF Cont Isol) __ • 1CF-87 (S/G 1D CF Cont Isol Byp) __ • 1CA-152 (S/G 1D CF Byp To CA Nozzle) __ • 1CA-188 (S/G 1D CA Nozz Tempering Isol). 2) <u>IF</u> 1CA-188 (S/G 1D CA Nozz Tempering Isol) cannot be closed, <u>**THEN**</u> perform the following: a) CLOSE the following valves: __ • 1CF-100 (S/G CA Nozz Tempering Ctrl) __ • 1CF-156 (Byp Valve For 1CF-100). (RNO continued on next page)

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 22 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
	b) **IF** 1CF-100 **OR** 1CF-156 cannot be closed, **THEN** dispatch operator to close affected valve(s):
	__ • 1CF-100 (S/G CA Nozz Tempering Ctrl) (TB1-580, 1H-33) (Ladder needed)
	__ • 1CF-156 (Byp Valve For 1CF-100) (TB1-577, 1H-33) (Ladder needed).
	__ 3) **IF** more than one Feedwater Isolation valve above open **AND** CM still aligned to feed faulted S/G, **THEN** evaluate alternate means to stop CM flow to faulted S/G.
b. CLOSE the following valves:	
__ 1) 1SM-74B (S/G 1D Otlt Hdr Bldwn C/V).	__ 1) Dispatch operator to close 1SM-74B (S/G 1D Otlt Hdr Bldwn C/V) (DH-583, FF-GG, 44-45, Rm 591).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	111 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 23 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 2) 1CA-42B (CA Pmp B Disch To S/G 1D Isol).	2) Perform the following: __ a) CLOSE 1CA-40 (CA Pump 1B Flow To S/G 1D). __ b) Dispatch operator to close 1CA-42B (CA Pmp B Disch To S/G 1D Isol) (DH-586, DD-EE, 43-44, Rm 591). __ c) **IF** exterior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-39 (CA Pump 1B Disch To S/G 1D Ctrl Inlet Isol) (AB-551, BB, 49-50, Rm 250) (Ladder needed) (Key #633).
__ 3) 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol).	3) Perform the following: __ a) CLOSE 1CA-36 (CA Pump #1 Flow To S/G 1D). __ b) Dispatch operator to close 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol) (DH-584, DD-EE, 43-44, Rm 591). __ c) **IF** exterior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-35 (CA Pump No 1 Disch To S/G 1D Ctrl Inlet Isol) (AB-555, BB-50, Rm 250) (Ladder needed) (Key #633).

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 24 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
c. Verify the following blowdown isolation valves - CLOSED:	
__ 1) 1BB-8A (S/G 1D Bldwn Cont Isol Insd).	__ 1) CLOSE valve.
__ 2) 1BB-147B (S/G 1D Bldwn Cont Isol Byp).	2) Perform the following: __ a) CLOSE valve. b) **IF** valve will not close **AND** 1BB-8A open, **THEN** perform the following: __ (1) Ensure "S/G D BLDWN FLOW CTRL" - CLOSED. (2) Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-147B (S/G 1D Bldwn Cont Isol Byp) (DH-582, EE-FF, 44, Rm 591) __ • 1BB-80 (1D S/G Blowdown Penetration Valve Test Isol) (DH-583,EE-FF, 44, Rm 591).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	113 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 25 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
__ 3) 1BB-10B (S/G 1D Bldwn Cont Isol Otsd).	3) Perform the following: __ a) CLOSE valve. b) **IF** valve will not close **AND** 1BB-8A open, **THEN** perform the following: __ (1) Ensure "S/G D BLDWN FLOW CTRL" - CLOSED. (2) Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-10B (S/G 1D Bldwn Cont Isol Otsd) (DH-582, EE-FF, 44, Rm 591) __ • 1BB-80 (1D S/G Blowdown Penetration Valve Test Isol) (DH-583,EE-FF, 44, Rm 591).
__ 11. **WHEN NC T-Hots start to trend up, THEN dump steam from intact S/G PORVs to stabilize NC T-Hots.**	

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	114	of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 26 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. **Verify secondary radiation normal as follows:**	
a. Ensure the following signals - RESET:	
__ 1) Phase A containment isolations.	
__ 2) CA System valve control.	
__ 3) KC NC NI NM St signals.	
__ b. Align all S/Gs for Chemistry sampling.	
c. Perform at least one of the following:	
__ • Notify Chemistry to sample all S/Gs for activity	
OR	
__ • Notify RP to frisk all cation columns for activity.	
d. Verify the following EMF trip 1 lights - DARK:	__ d. **GO TO** EP/1/A/5000/E-3 (Steam Generator Tube Rupture).
__ • 1EMF-33 (Condenser Air Ejector Exhaust)	
__ • 1EMF-26 (Steamline 1A)	
__ • 1EMF-27 (Steamline 1B)	
__ • 1EMF-28 (Steamline 1C)	
__ • 1EMF-29 (Steamline 1D).	
__ e. Verify S/G(s) fault - INSIDE CONTAINMENT.	e. Request RP to perform the following: __ 1) Monitor area of steam fault for radiation. __ 2) Notify Control Room of any abnormal radiation conditions.

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	115	of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION	PAGE NO. 27 of 28 Revision 16

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. (Continued)	
__ f. **WHEN** activity results reported, **THEN** notify Station Management to evaluate S/G(s) activity results.	
13. **Verify S/I termination criteria:**	
__ a. NC subcooling based on core exit T/Cs - GREATER THAN 0°F.	__ a. **GO TO** Step 14.
b. Verify secondary heat sink as follows:	__ b. **GO TO** Step 14.
__ • Any intact S/G N/R level - GREATER THAN 11% (29% ACC)	
OR	
__ • Total feed flow to intact S/Gs - GREATER THAN 450 GPM.	
__ c. NC pressure - STABLE OR TRENDING UP.	__ c. **GO TO** Step 14.
__ d. Pzr level - GREATER THAN 11% (30% ACC).	__ d. **GO TO** Step 14.
__ e. **GO TO** EP/1/A/5000/ES-1.1 (Safety Injection Termination).	
__ 14. **GO TO EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant).**	

END

Op Test No.: 301 Scenario # 1 Event # 7,8,9	Page 116 of 153
Event Description:	Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/ES-1.1	SAFETY INJECTION TERMINATION	PAGE NO. 2 of 58 Revision 35

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 11 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Reset the following:**	
__ a. ECCS.	__ a. Locally reset ECCS. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 4 (ECCS Master Reset).
__ b. D/G load sequencers.	b. Dispatch operator to open affected sequencer(s) control power breaker: __ • 1EDE-F01F (Diesel Generator Load Sequencer Panel 1DGLSA) (AB-577, BB-46, Rm 496) __ • 1EDF-F01F (Diesel Generator Load Sequencer Panel 1DGLSB) (AB-560, BB-46, Rm 372).
__ c. Phase A.	
__ d. Phase B.	
__ e. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.	

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	117 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/ES-1.1	SAFETY INJECTION TERMINATION	PAGE NO. 3 of 58 Revision 35

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **Establish VI to Containment as follows:** __ • Ensure 1VI-77B (VI Cont Isol) - OPEN __ • Verify VI pressure - GREATER THAN 85 PSIG.	**Perform the following:** a. Align N_2 to Pzr PORVs by opening the following valves: __ • 1NI-438A (Emer N2 From CLA A To 1NC-34A) __ • 1NI-439B (Emer N2 From CLA B To 1NC-32B). b. **IF** VI pressure less than 85 PSIG, **THEN** perform the following: __ 1) Dispatch operator to ensure proper VI compressor operation. __ 2) Restore VI while continuing with this procedure. **REFER TO** AP/0/A/5500/022 (Loss of Instrument Air).
__ 4. **Ensure only one NV pump - ON.**	
__ 5. **Verify NC pressure - STABLE OR TRENDING UP.**	**Perform the following:** __ a. Ensure Pzr spray valves - CLOSED. __ b. **IF** NC pressure continues to trend down, **THEN GO TO** EP/1/A/5000/ES-1.2 (POST LOCA Cooldown and Depressurization).
__ 6. **Verify VI pressure - GREATER THAN 50 PSIG.**	**In subsequent steps, Control Room control is lost for the following valves and local operation will be required:** __ • 1NV-294 (NV Pmps A&B Disch Flow Ctrl) __ • 1NV-309 (Seal Water Injection Flow).

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	118 of 153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/ES-1.1	SAFETY INJECTION TERMINATION	PAGE NO. 4 of 58 Revision 35

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Isolate NV S/I flowpath as follows:**	
a. Verify the following valves - OPEN: __ • 1NV-252A (NV Pumps Suct From FWST) __ • 1NV-253B (NV Pumps Suct From FWST).	a. **IF** NV pump suction aligned to discharge of ND pumps in Cold Leg Recirc, **THEN** perform the following: __ 1) Align charging with NV miniflow isolated. **GO TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 13 (Aligning Charging With NV Miniflow Valves Closed). __ 2) **GO TO** Step 9.
b. Verify the following valves - OPEN: __ • 1NV-203A (NV Pumps A&B Recirc Isol) __ • 1NV-202B (NV Pmps A&B Recirc Isol).	b. Perform the following: __ 1) OPEN affected valve(s). __ 2) **IF** 1NV-203A **AND** 1NV-202B open, **THEN GO TO** Step 7.c. 3) Dispatch operator to open affected valve(s): __ • 1NV-203A (NV Pumps A&B Recirc Isol) (AB-554, HH-JJ, 54-55, Rm 231) (Ladder needed) __ • 1NV-202B (NV Pmps A&B Recirc Isol) (AB-554, HH-JJ, 54-55, Rm 231) (Ladder needed). __ 4) Align charging with NV miniflow isolated. **GO TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 13 (Aligning Charging With NV Miniflow Valves Closed). __ 5) **WHEN** 1NV-203A **AND** 1NV-202B open, **THEN** charging flow may be reduced below 80 GPM. __ 6) **GO TO** Step 9.

Op Test No.:	301	Scenario #	1	Event #	7,8,9	Page	119	of	153

Event Description: Steam Line Break Inside Containment / MSIVs fail to close on MSI / Aux. Bldg. UFXFs fail to secure on S/I

CNS EP/1/A/5000/ES-1.1	SAFETY INJECTION TERMINATION	PAGE NO. 5 of 58 Revision 35

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	
c. CLOSE the following valves: __ • 1NI-9A (NV Pmp C/L Inj Isol) __ • 1NI-10B (NV Pmp C/L Inj Isol).	c. Perform the following: 1) Dispatch operator to close affected valve(s). **REFER TO** the following enclosure(s) for affected valve(s): __ • EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 5 (Locally Close 1NI-9A) __ • EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 7 (Locally Close 1NI-10B). __ 2) Do not continue until both 1NI-9A and 1NI-10B - CLOSED.
Note to Evaluator: **This is the end of the scenario. At the lead evaluators discretion, the scenario may be terminated by having the booth operator place the simulator in FREEZE.**	
8. **Establish charging as follows:**	
a. Verify all of the following valves - OPEN: __ • 1NV-44A (NC Pmp A Seal Supply Cont Isol) __ • 1NV-55A (NC Pmp B Seal Supply Cont Isol) __ • 1NV-66A (NC Pmp C Seal Supply Cont Isol) __ • 1NV-77A (NC Pmp D Seal Supply Cont Isol).	a. **IF** all valves closed, **THEN** perform the following: __ 1) OPEN 1NV-309 (Seal Water Injection Flow). __ 2) **IF** control of 1NV-309 lost from Control Room, **THEN** dispatch operator with radio to open 1NV-311 (Seal Wtr Inj Flow Ctrl Byp) (AB-555, JJ-54, Rm 233). 3) OPEN the following valves: __ • 1NV-312A (Chrg Line Cont Isol) __ • 1NV-314B (Chrg Line Cont Isol). (RNO continued on next page)

Attachment List

Scenario 1

ATTACHMENT 1 -	Crew Critical Task Summary
ATTACHMENT 2 -	Shift Turnover Information
ATTACHMENT 3 -	AP/1/A/5500/021 Enclosure 1 (Foldout Page)
ATTACHMENT 4 -	AP/1/A/5500/003 Enclosure 3 (Rod Insertion Limit Boration)
ATTACHMENT 5 -	AP/1/A/5500/028 Enclosure 1 (Foldout Page)
ATTACHMENT 6 -	EP/1/A/5000/E-0 Enclosure 1 (Foldout Page)
ATTACHMENT 7 -	EP/1/A/5000/E-0 Enclosure 5 (VX System Operation)
ATTACHMENT 8 -	EP/1/A/5000/E-0 Enclosure 2 (Ventilation System Verification)
ATTACHMENT 9 -	EP/1/A/5000/E-0 Enclosure 4 (NC Temperature Control)
ATTACHMENT 10	EP/1/A/5000/E-2 Enclosure 1 (Foldout Page)
ATTACHMENT 11	EP/1/A/5000/ES-1.1 Enclosure 1 (Foldout Page)
ATTACHMENT 12 -	Scenario Specific Technical Specifications

ATTACHMENT 1

CREW CRITICAL TASK SUMMARY			
SAT	**UNSAT**	**CT #**	**CRITICAL TASK**
		1	Runback the Main Turbine prior to Main Turbine Trip (must be below amps for 332 MW @ 0.9 Power Factor within 3.5 minutes).
		2	Close MSIVs prior to a severe challenge (Orange or Red Path) to the NC Integrity CSF status tree.

Comments:

ATTACHMENT 2

SHIFT TURNOVER INFORMATION			
Unit 1 Status			
Power Level	**Power History**	**NCS Boron**	**Xenon**
50 **%**	MOL	1117 PPM	per OAC
Controlling Procedure			
• OP/1/A/6100/003 (Controlling Procedure for Unit Operation), Enclosure 4.1 (Power Increase). The steps up to step 3.52 are complete.			
Other Information Needed to Assume the Shift			
• Unit 1 is at 50% power at the MOL. Unit 2 is at 100% power. Direction for the crew is to raise reactor power to 100%. A reactivity plan has been provided by Reactor Engineering for raising reactor power.			
AOs Available			
Eight AOs are available as listed on the status board			
METEOROLOGICAL CONDITIONS			
• Upper wind direction = 125 degrees, speed = 3 mph • Lower wind direction = 127 degrees, speed = 4.5 mph • Forecast calls for clear skies over the next 24 hours.			

ATTACHMENT 3

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING Enclosure 1 - Page 1 of 2 **Foldout Page**	PAGE NO. 22 of 59 Revision 48

1. **SSF Manning Criteria:**

CAUTION **Failure to restore NC pump seal cooling via thermal barrier cooling or NV seal injection within ten minutes will cause damage to the NC pump seals resulting in NC System inventory loss.**

IF AT ANY TIME KC **AND** NV seal cooling for any NC pump lost, **THEN** perform the following:

__ a. Dispatch operator to SSF to establish NC pump seal injection. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 19 (Establishing NC Makeup/Seal Injection From The SSF).

b. **IF** 1EMXS de-energized, **THEN** perform the following:

__ 1) Dispatch operator to 1ETA switchgear room to align alternate power supply to 1EMXS. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 20 (Align Alternate Power Supply To 1EMXS).

__ 2) Notify operator at SSF (Ext. 5251 or 5212) operator has been dispatched to align alternate power supply to 1EMXS.

2. **NC Pump Trip Criteria:**

- **IF** any of the following NC pump trip criteria met:

__ • KC flow unavailable to NC pumps - GREATER THAN 10 MINUTES

OR

__ • #1 Seal outlet temperature - GREATER THAN 235°F

OR

__ • Lower bearing temperature - GREATER THAN 225°F

OR

__ • Motor bearing temperature - GREATER THAN 195°F,

__ **THEN GO TO** Enclosure 6 (Rx Trip Sequence).

ATTACHMENT 3

CNS AP/1/A/5500/021	LOSS OF COMPONENT COOLING Enclosure 1 - Page 2 of 2 **Foldout Page**	PAGE NO. 23 of 59 Revision 48

NOTE The following step prevents damage to the 1B2 KC pump as a result of deadheading. (NCR #01406467)

3. **IF AT ANY TIME the following conditions met:**

__ • Train B KC non-essential header isolation valves - CLOSED

AND

__ • 1KC-81B (KC To ND Hx 1B Sup Isol) - CLOSED,

__ **THEN** ensure less than 2 Train B KC pumps - IN SERVICE.

NOTE Monitoring of the following steps must continue while KC malfunction exists even if a transition is made to the emergency procedures.

__ 4. **IF AT ANY TIME both trains of KC lost, THEN RETURN TO Section C. (Operator Actions), Step 2.**

5. **IF operators dispatched to align alternate cooling to NV pump 1A, THEN perform the following:**

__ a. **WHEN** alternate cooling aligned, **THEN** perform Enclosure 5 (Maximize NV Pump Run Time), Step 7.

6. **IF AT ANY TIME KC cooling to operating KF pump(s) lost, THEN perform the following:**

__ • **IF** annunciator 1AD-13, D/6 "KF PUMP A MTR CLR HI TEMP" lit, **THEN** secure 1A KF pump and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).

__ • **IF** annunciator 1AD-13, D/7 "KF PUMP B MTR CLR HI TEMP" lit, **THEN** secure 1B KF pump and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).

ATTACHMENT 4

CNS AP/1/A/5500/003	LOAD REJECTION Enclosure 3 - Page 1 of 2 **Rod Insertion Limit Boration**	PAGE NO. 53 of 55 Revision 47

CAUTION **Failure to initiate boration within one hour of exceeding rod insertion limits may violate Tech Spec 3.1.6.**

NOTE OAC point C1L4409 (Ctrl Bank Tech Spec Insertion Lmt Reached) and R.O.D Book (Section 2.2) provide rod insertion limit indication.

1. **IF control rods cannot be maintained above rod insertion limits, THEN perform the following:**

   a. Stop any dilutions in progress.

   b. Ensure control rods restored above insertion limits within 2 hours of exceeding limits.

   c. Ensure compliance with Tech Spec 3.1.6 (Control Bank Insertion Limits).

2. **Perform one of the following to restore control rods above insertion limits:**

   a. **IF** initial reactor power was 100%, **THEN** borate NC System as required to restore control rods above insertion limits. **REFER TO** Unit 1 R.O.D. book, section 4.8 (Reactivity Data Sheet).

CNS AP/1/A/5500/003	LOAD REJECTION Enclosure 3 - Page 2 of 2 **Rod Insertion Limit Boration**	PAGE NO. 54 of 55 Revision 47

2. (Continued)

b. **<u>IF</u>** initial reactor power was less than 100% **<u>OR</u>** Unit 1 R.O.D. book, section 4.8 (Reactivity Data Sheet) is **<u>NOT</u>** available, **<u>THEN</u>** perform the following as required to restore control rods above insertion limits:

<u>NOTE</u> OAC point C1P1448 (Low Bank Insertion Limit Margin) and R.O.D Book (Section 2.2) provide rod insertion limit indication.

1) Determine control rod insertion limit. _________.

2) Calculate "A" (reactivity difference between required rod position and current rod position). **<u>REFER TO</u>** Unit 1 R.O.D. book section 5.6.3.

R = Required rod position IRW _________ PCM

P = Current rod position IRW _________ PCM

(R - P = A __________ PCM).

3) Determine "B" (differential boron worth). **<u>REFER TO</u>** Unit 1 R.O.D. book section 5.5 ______________ PCM/PPM.

4) Calculate "C" (difference in reactivity) as follows:

A / B = C ______________ PPM.

5) Calculate "D" (required boron concentration) as follows:

E = Current Boron concentration _________ PPM.

E + C = D__________ PPM.

6) Determine required boric acid needed to raise NC System boron concentration to value "D" calculated in Step 2.b.5. **<u>REFER TO</u>** Unit 1 R.O.D. book table 4.1 or REACT Boration/Dilution module. ___________.

<u>NOTE</u>

- The boric acid added to the NC System should be added in several increments within the first hour of the runback.
- Due to the post transient Xenon build-in rate, the total boric acid value calculated in Step 2.b.6, may not need to be added to restore control rods above insertion limits.

7) Borate NC System as required to restore control rods above insertion limits.

ATTACHMENT 5

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK Enclosure 1 - Page 1 of 2 **Foldout Page**	PAGE NO. 26 of 43 Revision 12

1. **Reactor trip criteria:**

IF AT ANY TIME containment pressure reaches 1.2 PSIG, **THEN** perform the following:

__ a. Ensure Unit 1 reactor tripped.

__ b. Ensure S/I initiated.

c. CLOSE the following valves:

__ • All MSIVs

__ • All MSIV bypass valves.

__ d. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

IF any of the following conditions exist:

__ • Steam leak endangering personnel or jeopardizing plant equipment

__ • S/G levels - TRENDING DOWN IN UNCONTROLLED MANNER

__ • Tavg 5°F less than T-Ref **AND** trending down in uncontrolled manner

__ • Reactor power - TRENDING UP IN UNCONTROLLED MANNER

__ • Secondary condensate inventory - TRENDING DOWN IN UNCONTROLLED MANNER,

THEN perform the following:

__ a. Trip Unit 1 reactor.

b. CLOSE the following valves:

__ • All MSIVs

__ • All MSIV bypass valves.

__ c. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

CNS AP/1/A/5500/028	SECONDARY STEAM LEAK Enclosure 1 - Page 2 of 2 **Foldout Page**	PAGE NO. 27 of 43 Revision 12

2. **Uncontrolled cooldown:**

IF Tavg less than 554°F **AND** trending down, **THEN** perform one of the following:

- **IF** reactor power less than 1%, **THEN** perform the following:
  - a. CLOSE the following valves:
    - __ • All MSIVs
    - __ • All MSIV bypass valves.
  - b. **IF** cooldown continues, **THEN** perform the following:
    - __ 1) Trip Unit 1 reactor.
    - __ 2) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
- **IF** reactor power less than 69%, **THEN** perform the following:
  - __ a. Trip Unit 1 turbine.
  - b. **IF** cooldown continues, **THEN** perform the following:
    - __ 1) Trip Unit 1 reactor.
    - 2) CLOSE the following valves:
      - __ • All MSIVs
      - __ • All MSIV bypass valves.
    - __ 3) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
  - __ c. **REFER TO** AP/1/A/5500/002 (Turbine Generator Trip).

__ 3. **IF steam leak size goes up, THEN RETURN TO C. (Operator Actions), Step 2.**

ATTACHMENT 6

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 1 of 3 **Foldout Page**	PAGE NO. 34 of 49 Revision 46

1. **NC Pump Trip Criteria:**

- **IF** the following conditions satisfied, **THEN** trip all NC pumps while maintaining seal injection flow:
  - Any NV or NI pump - DELIVERING S/I FLOW TO NC SYSTEM
  - NC subcooling based on core exit T/Cs - LESS THAN OR EQUAL TO 0°F
  - Reactor power - LESS THAN 5%.

2. **Open Phase Criteria:**

- **IF** operating NV **AND** KC pumps automatically trip, **THEN** perform the following:
  a. Start the following pumps on opposite train:
     - NV pump
     - KC pumps
     - RN pump.

  b. **IF** pumps do not start, **OR** trip after starting, **THEN** restart pumps on previously operating train.

  c. **IF** all KC pumps off, **THEN** ensure all NC pumps - OFF.

  d. **IF** Unit 2 4160V bus energized by Unit 1 busline, **THEN** immediately notify Unit 2 to perform same actions on Unit 2.

3. **CA Suction Source Switchover Criterion:**

- **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

4. **Position Criteria for 1NV-202B and 1NV-203A (NV Pumps A&B Recirc Isol):**

- **IF** NC pressure less than 1500 PSIG **AND** NV S/I flowpath aligned, **THEN** CLOSE 1NV-202B and 1NV-203A.
- **IF** NC pressure greater than 2000 PSIG, **THEN** OPEN 1NV-202B and 1NV-203A.

ATTACHMENT 6

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 2 of 3 **Foldout Page**	PAGE NO. 35 of 49 Revision 46

NOTE CA flow control valves fail open on CA auto start. Isolating flow with the motor operated isolation valve will not require re-isolation on subsequent CA auto starts.

5. **Ruptured S/G CA Isolation Criteria:**

- **IF** both the following conditions met, **THEN** stop CA flow to affected S/G(s):
  - Level going up in uncontrolled manner or radiation level in that S/G abnormal
  - N/R level - GREATER THAN 11% (29% ACC).

NOTE CA flow control valves fail open on CA auto start. Isolating flow with the motor operated isolation valve will not require re-isolation on subsequent CA auto starts.

6. **Faulted S/G CA isolation Criteria:**

- **IF** all the following conditions met, **THEN** stop CA flow to affected S/G:
  - S/G pressure trends down in uncontrolled manner or completely depressurized
  - Only one S/G diagnosed as faulted
  - Secondary heat sink criteria met:
    - Total CA flow - GREATER THAN 450 GPM

      OR
    - ANY S/G(s) N/R level - GREATER THAN 11% (29% ACC).

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 3 of 3 **Foldout Page**	PAGE NO. 36 of 49 Revision 46

7. **NS Pump Trip Criterion:**

- **IF** NS pump in recirc and S/I occurs, **THEN** perform one of the following:
  - **IF** train affected ECCS and D/G load sequencer - RESET, **THEN** stop NS pump

    OR
  - **WHEN** sequencer loading complete, **THEN** perform the following for affected train:

    a. Notify Control Room Supervisor.

    b. Reset ECCS.

    c. Reset D/G load sequencer.

    d. Secure NS pump.

    e. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.

8. **IF AT ANY TIME KC cooling to operating KF pump(s) lost, THEN perform the following:**

- **IF** annunciator 1AD-13, D/6 "KF PUMP A MTR CLR HI TEMP" lit, **THEN** secure 1A KF pump and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).
- **IF** annunciator 1AD-13, D/7 "KF PUMP B MTR CLR HI TEMP" lit, **THEN** secure 1B KF pump and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).

ATTACHMENT 7

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 5 - Page 1 of 1 **VX System Operation**	PAGE NO. 49 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **Verify the following containment air return fan dampers - OPEN:** __ • ARF-D-2 (ARF-1A Ret Fan Damper) (1MD-4, I/5) __ • ARF-D-4 (ARF-1B Ret Fan Damper) (1MD-4, I/8).	__ **<u>IF</u> equipment not in proper alignment, <u>THEN</u> align equipment.**
2. **Verify the following equipment alignment:** __ • 1VX-1A (HSF-1A Inlt Isol) (1MD-4, I/6) - OPEN __ • 1VX-2B (HSF-1B Inlt Isol) (1MD-4, I/7) - OPEN __ • ARF-1A (Cont Air Return Fan) (1MD-4, I/3) - ON __ • ARF-1B (Cont Air Return Fan) (1MD-4, I/10) - ON __ • HSF-1A (H_2 Skimmer Fan) (1MD-4, I/4) - ON __ • HSF-1B (H_2 Skimmer Fan) (1MD-4, I/9) - ON.	**<u>IF</u> equipment not in proper alignment, <u>THEN</u> perform the following:** a. <u>**IF**</u> containment pressure less than 0.3 PSIG, <u>**THEN**</u> verify the following Monitor Light Panel Group 1 Sp lights - DARK: __ • I/3 __ • I/10. __ b. Align or start affected component(s). __ c. <u>**IF**</u> any VX System equipment cannot be started, <u>**THEN REFER TO**</u> OP/1/A/6450/010 (Containment Hydrogen Control Systems), for further actions.
3. **Verify containment air return fans operate as containment pressure changes as follows:** __ • <u>**IF AT ANY TIME**</u> containment pressure greater than 0.9 PSIG, <u>**THEN**</u> ensure containment air return fans - ON. __ • <u>**IF AT ANY TIME**</u> containment pressure less than 0.3 PSIG, <u>**THEN**</u> ensure containment air return fans - OFF.	

ATTACHMENT 8

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 2 - Page 1 of 6 **Ventilation System Verification**	PAGE NO. 37 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **Verify proper VC/YC operation as follows:**	
a. Verify one train of the following equipment in operation: __ • YC chiller __ • CR AHU-1 __ • CRA AHU-1 __ • CRA PFT-1.	a. Perform the following: __ 1) Shift operating VC/YC trains. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 17 (Shifting Operating VC/YC Trains). 2) **IF** no train can be properly aligned, **THEN** dispatch operator and IAE/Maintenance to restore at least one train of VC/YC. **REFER TO** the following: __ • OP/0/A/6450/011 (Control Room Area Ventilation/Chilled Water System) __ • EM/0/A/5200/001 (Troubleshooting Cause For Improper Operation of VC/YC System).
b. Verify the following alarms - DARK: __ • 1AD-18, A/8 "UNIT 1 INTAKE HI CHLORINE 1A" __ • 1AD-18, B/8 "UNIT 1 INTAKE HI CHLORINE 1B" __ • 1AD-18, D/8 "UNIT 2 INTAKE HI CHLORINE 2A" __ • 1AD-18, E/8 "UNIT 2 INTAKE HI CHLORINE 2B".	b. **IF** chlorine odor detected in Control Room, **THEN** perform the following based on the status of given alarms: 1) **IF** detectors on both unit intakes in alarm, **THEN** perform the following: a) Ensure the following VC intake dampers - CLOSED: __ • 1VC-5B (CRA Filt Inlet) __ • 1VC-6A (CRA Filt Inlet) __ • 2VC-5B (CRA Filt Inlet) __ • 2VC-6A (CRA Filt Inlet). __ b) **GO TO** Step 1.d. (RNO continued on next page)

ATTACHMENT 8

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 2 - Page 2 of 6 **Ventilation System Verification**	PAGE NO. 38 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (Continued)	
	2) **IF** Unit 1 intake HI chlorine detector(s) in alarm, **THEN** perform the following: a) Ensure the following VC dampers - CLOSED: __ • 1VC-5B (CRA Filt Inlet) __ • 1VC-6A (CRA Filt Inlet). b) Ensure the following dampers - OPEN: __ • 2VC-5B (CRA Filt Inlet) __ • 2VC-6A (CRA Filt Inlet). __ c) **GO TO** Step 1.d.
	3) **IF** Unit 2 intake Hi chlorine detector(s) in alarm, **THEN** perform the following: a) Ensure the following VC dampers - CLOSED: __ • 2VC-5B (CRA Filt Inlet) __ • 2VC-6A (CRA Filt Inlet). b) Ensure the following dampers - OPEN: __ • 1VC-5B (CRA Filt Inlet) __ • 1VC-6A (CRA Filt Inlet). __ c) **GO TO** Step 1.d.
c. Ensure the following VC dampers - OPEN: __ • 1VC-5B (CRA Filt Inlet) __ • 1VC-6A (CRA Filt Inlet) __ • 2VC-5B (CRA Filt Inlet) __ • 2VC-6A (CRA Filt Inlet).	

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 2 - Page 3 of 6 **Ventilation System Verification**	PAGE NO. 39 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. (Continued)

d. Repeat Step 1 of this enclosure until notified by station management as follows:

__ • At least once every 8 hours

OR

__ • Any time VC/YC related annunciators on 1AD-18 actuate.

2. **Ensure proper VA System operation as follows:**

• Ensure the following fans - OFF:

__ • ABUXF 1A
__ • ABUXF 1B.

• Ensure VA system filter in service as follows:

__ • 1ABF-D-12 & 19 (VA Filter A Bypass Dampers) - CLOSED

__ • 1ABF-D-5 & 20 (VA Filter B Bypass Dampers) - CLOSED.

• Ensure the following fans - ON:

__ • ABFXF-1A
__ • ABFXF 1B.

ATTACHMENT 8

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **Verify proper VE System operation as follows:**	
__ a. VE fans - ON.	__ a. Start fan(s).
__ b. Annulus pressure - BETWEEN -1.4 IN. WC AND -1.8 IN. WC.	b. Perform the following: 1) **<u>IF</u>** annulus pressure more positive than -1.4 in. WC, **<u>THEN</u>** perform the following: a) Verify flow indicated on the following indications: __ • 1VEP5180 (VE 1A Flow To Stack) __ • 1VEP5200 (VE 1B Flow To Stack). b) **<u>IF</u>** flow not indicated, **<u>THEN</u>** dispatch operator to verify status of the following dampers based on their local indication or their operating piston rods being extended 4" to 6": __ • 1AVS-D-2 (VE A Trn Recirc Damp) (AB-603, JJ-51, Rm 500) - CLOSED __ • 1AVS-D-7 (VE B Trn Recirc Damp) (AB-603, HH-52, Rm 500) - CLOSED __ • 1AVS-D-3 (VE A Trn Exh Damp) (AB-603, JJ-52, Rm 500) - OPEN __ • 1AVS-D-8 (VE B Trn Exh Damp) (AB-603, HH-52, Rm 500) - OPEN. (RNO continued on next page)

ATTACHMENT 8

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. (Continued)	
	__ c) Consult plant engineering staff and notify IAE/Maintenance to troubleshoot and repair. **REFER TO** EM/1/A/5200/002 (Troubleshooting Cause For VE System Hi/Lo Pressure).
	__ d) **GO TO** Step 3.c.
	2) **IF** annulus pressure more negative than -1.8 in. WC, **THEN** perform the following:
	__ a) Determine which VE train indicates highest discharge flow to stack.
	__ b) Within 2 hours, ensure VE train that indicates highest discharge flow to stack secured.
	__ c) Consult plant engineering staff and notify IAE/Maintenance to troubleshoot and repair. **REFER TO** EM/1/A/5200/002 (Troubleshooting Cause For VE System Hi/Lo Pressure).
__ c. Repeat Step 3.b every 30 minutes until notified by Station Management.	

ATTACHMENT 8

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 2 - Page 6 of 6 **Ventilation System Verification**	PAGE NO. 42 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

__ 4. **Record time ventilation systems verified on following table:**

TIME	SYSTEM (VC, VE)	INITIALS

ATTACHMENT 9

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 4 - Page 1 of 5 **NC Temperature Control**	PAGE NO. 44 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **Verify any NC pump - ON.**	**Perform the following:** a. Use NC T-Colds to determine NC temperature as required in subsequent steps. b. **GO TO** Step 4.
2. **Use NC T-Avg to determine NC temperature as required in subsequent steps.**	
3. **IF AT ANY TIME all NC pumps tripped, THEN use NC T-Colds to determine NC temperature as required in subsequent steps.**	
4. **Verify one of the following:** • NC temperature - STABLE AT LESS THAN OR EQUAL TO 557°F OR • NC temperature - TRENDING TO 557°F.	**GO TO Step 8.**
5. **Continue to monitor NC temperature.**	
6. **Notify Control Room Supervisor of NC temperature control status.**	

ATTACHMENT 9

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Do not continue in this enclosure until one of the following occurs:** • NC temperature - GREATER THAN 557°F AND TRENDING UP IN AN UNCONTROLLED MANNER OR • NC temperature - GREATER THAN 557°F AND STABLE OR • NC temperature - LESS THAN 557°F AND TRENDING DOWN IN UNCONTROLLED MANNER.	
8. **Verify NC temperature - LESS THAN 557°F AND TRENDING DOWN.**	**Perform the following:** a. **IF** NC temperature greater than 557°F **AND** trending up, **THEN** stabilize NC temperature at 557°F as follows: 1) **IF** steam dumps available, **THEN** use steam dumps. 2) **IF** steam dumps not available, **THEN** use S/G PORVs. b. **IF** the following conditions exist: • NC temperature greater than 557°F and stable • Time and manpower available, **THEN** stabilize NC temperature at 557°F as follows: 1) **IF** steam dumps available, **THEN** use steam dumps. 2) **IF** steam dumps not available, **THEN** use S/G PORVs. c. **GO TO** Step 10.

ATTACHMENT 9

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 4 - Page 3 of 5 **NC Temperature Control**	PAGE NO. 46 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Attempt to stop NC cooldown as follows:**	
a. Ensure all steam dumps - CLOSED.	
b. Ensure all S/G PORVs - CLOSED.	b. **IF** any S/G PORV cannot be closed, **THEN** CLOSE its isolation valve.
c. Ensure S/G blowdown isolated.	
d. CLOSE the following valves: • 1SM-77A (S/G 1A Otlt Hdr Bldwn C/V) • 1SM-76B (S/G 1B Otlt Hdr Bldwn C/V) • 1SM-75A (S/G 1C Otlt Hdr Bldwn C/V) • 1SM-74B (S/G 1D Otlt Hdr Bldwn C/V).	
e. Verify MSR Second Stage steam supply valves - CLOSED • 1HM-1 (MSRH 1A&1B SSRH Stm Source) • 1HM-2 (MSRH 1C&1D SSRH Stm Source).	e. Perform the following: 1) CLOSE MSR Second Stage steam supply valve(s). 2) **IF** steam flowpath cannot be isolated from Control Room, **THEN** CLOSE the following valves: • All MSIVs • All MSIV bypass valves.

ATTACHMENT 9

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 4 - Page 4 of 5 **NC Temperature Control**	PAGE NO. 47 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
f. Depress and hold "S/V BEFORE SEAT DRN" "CLOSE" pushbutton (1MC-3) to close the following valves: • 1SM-41 (Stop Vlv #1 Before Seat Drn) • 1SM-44 (Stop Vlv #2 Before Seat Drn) • 1SM-43 (Stop Vlv #3 Before Seat Drn) • 1SM-42 (Stop Vlv #4 Before Seat Drn).	
g. Verify NC cooldown - STOPPED.	g. **IF** cooldown continues, **THEN** THROTTLE feed flow as follows: 1) **IF** S/G N/R level less than 11% (29% ACC) in all S/G's, **THEN** THROTTLE feed flow to achieve the following: • Minimize cooldown • Maintain total feed flow greater than 450 GPM. 2) **WHEN** N/R level greater than 11% (29% ACC) in any S/G, **THEN** THROTTLE feed flow further to achieve the following: • Minimize cooldown • Maintain at least one S/G N/R level greater than 11% (29% ACC). 3) **IF** cooldown continues, **THEN** CLOSE the following valves: • All MSIVs • All MSIV bypass valves.

ATTACHMENT 9

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 4 - Page 5 of 5 **NC Temperature Control**	PAGE NO. 48 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Continue to perform actions of this enclosure as required to ensure one of the following:** • NC temperature - STABLE AT LESS THAN OR EQUAL TO 557°F OR • NC temperature - TRENDING TO 557°F.	
11. **Notify Control Room Supervisor of NC temperature control status.**	

ATTACHMENT 10

CNS EP/1/A/5000/E-2	FAULTED STEAM GENERATOR ISOLATION Enclosure 1 - Page 1 of 1 **Foldout Page**	PAGE NO. 28 of 28 Revision 16

1. **Cold Leg Recirc Switchover Criterion:**
   - **IF** FWST level lowers to 20% 1AD-9, D/8 "FWST 2/4 LO LEVEL" lit), **THEN GO TO** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation).

2. **CA Suction Source Switchover Criterion:**
   - **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

3. **Position Criteria for 1NV-202B and 1NV-203A (NV Pumps A&B Recirc Isol):**
   - **IF** NC pressure less than 1500 PSIG **AND** NV S/I flowpath aligned, **THEN** CLOSE 1NV-202B and 1NV-203A.
   - **IF** NC pressure greater than 2000 PSIG, **THEN** OPEN 1NV-202B and 1NV-203A.

4. **N_2 to Pzr PORV Criteria:**
   - **IF** Containment pressure greater than or equal to 3 psig and VI isolated to Containment, **THEN** align N_2 to Pzr PORVs by opening the following valves:
     - 1NI-438A (Emer N2 From CLA A To 1NC-34A)
     - 1NI-439B (Emer N2 From CLA B To 1NC-32B).

ATTACHMENT 11

CNS EP/1/A/5000/ES-1.1	SAFETY INJECTION TERMINATION Enclosure 1 - Page 1 of 1 **Foldout Page**	PAGE NO. 38 of 58 Revision 35

1. **S/I Reinitiation Criteria:**

   - **IF** NC subcooling based on core exit T/Cs less than 0°F **OR** Pzr level cannot be maintained greater than 11% (30% ACC), **THEN** perform the following:

     a. Start one or more S/I pumps.

     b. Realign NV S/I flow path. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 14 (NV Alignment To S/I Mode).

     c. **IF** Step 11 has been completed, **THEN GO TO** EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant).

2. **Secondary Integrity Criteria:**

   - **IF** any unisolated S/G pressure trending down in uncontrolled manner **OR** completely depressurized, **THEN GO TO** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation).

3. **Cold Leg Recirc Switchover Criterion:**

   - **IF** FWST level lowers to 20% (1AD-9, D/8 "FWST 2/4 LO LEVEL"), **THEN GO TO** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation).

4. **CA Suction Source Switchover Criterion:**

   - **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

ATTACHMENT 12

Event #3
KC Miniflow Valve 1KC-C37A Failed Open

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCW train inoperable.	A.1 -------------NOTE------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for residual heat removal loops made inoperable by CCW. ---------------------------------- Restore CCW train to OPERABLE status.	 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

ATTACHMENT 12

Event #3

KC Miniflow Valve 1KC-C37A Failed Open

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----NOTE----- Isolation of CCW flow to individual components does not render the CCW System inoperable. -----	
	Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	Verify each CCW automatic valve in the flow path servicing safety related equipment that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

ATTACHMENT 12

Event #3
KC Miniflow Valve 1KC-C37A Failed Open

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

-----NOTE-----
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable when entering MODE 1.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable. OR -----NOTE----- Only applicable if Mode 2 has not been entered following refueling. --- One turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO

(continued)

ATTACHMENT 12

Event #3
KC Miniflow Valve 1KC-C37A Failed Open

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met. OR Two AFW trains inoperable in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours
D. Three AFW trains inoperable in MODE 1, 2, or 3.	D.1 ------------NOTE------------ LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. -------------------------------- Initiate action to restore one AFW train to OPERABLE status.	Immediately
E. Required AFW train inoperable in MODE 4.	E.1 Initiate action to restore AFW train to OPERABLE status.	Immediately

ATTACHMENT 12

Event #3

KC Miniflow Valve 1KC-C37A Failed Open

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	-----NOTE----- Not applicable to automatic valves when THERMAL POWER is ≤ 10% RTP. ----- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	-----NOTE----- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. ----- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	-----NOTE----- Not applicable in MODE 4 when steam generator is relied upon for heat removal. ----- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

ATTACHMENT 12

Event #3

KC Miniflow Valve 1KC-C37A Failed Open

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	-------------------------------NOTES------------------------------ 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 600 psig in the steam generator. 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. --- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.5	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage system to each steam generator.	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days

ATTACHMENT 12

Event #5
1A NS Pump Loss of Power

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray System

LCO 3.6.6 Two containment spray trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	-----NOTE----- Not required to be met for system vent flow paths opened under administrative control. ----- Verify each containment spray manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

ATTACHMENT 12

Event #5

1A NS Pump Loss of Power

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.6.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.6.3	Deleted.	
SR 3.6.6.4	Deleted.	
SR 3.6.6.5	Verify that each spray pump is de-energized and prevented from starting upon receipt of a terminate signal and is allowed to manually start upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.6	Verify that each spray pump discharge valve closes or is prevented from opening upon receipt of a terminate signal and is allowed to manually open upon receipt of a start permissive from the Containment Pressure Control System (CPCS).	In accordance with the Surveillance Frequency Control Program
SR 3.6.6.7	Verify each spray nozzle is unobstructed.	Following activities which could result in nozzle blockage
SR 3.6.6.8	Verify containment spray locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can effect reactivity. (R.M.)

1.2 Low load operation limits:

1.2.1 The unit can be operated continuously at low loads when exhaust hood temperature is < 175°F. The load shall, however, be increased slowly until the temperature decreases below 125°F before increasing load at normal rate (Multipoint Recorder on 1MC3).

1.2.2 Excessive use of the exhaust hood sprays shall be avoided to prevent accelerated blade erosion.

1.3 Do **NOT** exceed the load, hydrogen pressure, and power factor limits per the Unit One Revised Data Book Figure 43.

1.4 If the limits of the Unit One Revised Data Book Figure 43 (Generator Capability Curves) are exceeded, the Turbine Generator shall be tripped.

1.5 Under certain environmental conditions, indicated condenser vacuum less than 24.3 inches Hg may be reached at full load. Exhaust hood temperatures are a more accurate indicator of true vacuum. It is recommended the turbine **NOT** be operated under the following conditions at full load:

- Exhaust Hood 1A temperature ≥ 136°F
- Exhaust Hood 1B temperature ≥ 129°F
- Exhaust Hood 1C temperature ≥ 124.5 °F

1.6 The maximum differential pressure between adjacent LP shell pressures shall **NOT** exceed 2.0 inches Hg. (main condenser vacuum gauges on 1MC13, OAC points C1P1669 (D/P between A & B Condensers) and C1P1670 (D/P between B & C Condensers) or Main Condenser graphic (CMCOND)).

1.7 A sudden downward trend on an LP turbine's lower extraction temperature shall be investigated as a possible indication of water induction into the turbine. This is indicated on the recorder on the rear of 1MC8 labeled "TURBINE WATER DETECTION", using any of the LP 8th stage lower temperatures.

1.8 A "LOAD RATE" > "6.2 MW/MIN" shall **NOT** be used during normal load changes.

1.9 Differential temperature between adjacent exhaust hoods shall **NOT** exceed 30°F unless evaluated and approved by the responsible engineer (Turbine Generator System Expert). (OAC points C1P1667 (A & B Exhaust Hoods Metal Delta Temp) and C1P1668 (B & C Exhaust Hoods Metal Delta Temp) or Main Condenser graphic (CMCOND)).

1.10 The Main Turbine OIU Work Station has the capability to perform control functions for the Main Turbine, including tripping and resetting of the turbine. If a control function window is inadvertently selected while manipulating the Main Turbine OIU Work Station, the window shall be closed to prevent actuation of the control function.

1.11 To reduce potential for Turbine rubs at low power levels (< 30% Turbine Load) observe the following:

- Steam Seal Header Pressure between 3 and 5 psig.
- Gland Steam Condenser Header vacuum of 10 - 12" H_2O vacuum.
- Condenser Vacuum < 28.0" Hg
- Minimize time that Turbine is at speed no load.
- Minimize time between Turbine Shell Warming and rolling the Turbine.

1.12 Exhaust hood water sprays are used to cool the last-stage buckets and to minimize temporary distortion of the low-pressure hood and shell structures. These sprays have a significant potential for quenching the LP turbine structure, and if they are applied manually should be undertaken very gradually. Large and rapid changes in the temperature of the exhaust hood can also have an impact on bearing alignment and may cause a rub to develop. Excessive use of the sprays may cause unnecessary erosion of the long last-stage buckets during low flow conditions.

2. Initial Conditions

AA ____ Verify Turbine Generator is On Line per Enclosure 4.1 (Turbine Generator Startup).

3. Procedure

CAUTION:
1. The load, hydrogen pressure and power factor limits per the Unit One Revised Data Book Figure 43 shall **NOT** be exceeded.
2. Rate of change of First-Stage Bowl Inner Surface Temperature shall **NOT** exceed 150°F/hr (OAC point C1P1283 (First Stage Metal Temp Rate)).
3. Control valve casing difference, OAC point C1A0961 (Turb Valve Chest Inner Surface Metal Temp) minus C1A0967 (Turb Valve Chest Outer Surface Metal Temp), shall **NOT** exceed curve "Allowable Temp Difference on Turbine Valve Chest" in the Unit 1 OAC Databook.
4. OAC point C1A1140 (Turbine Lower Inner Shell Temp) vs. Percent Steam Flow (OAC point C1P1588 (Design Total Main Steam Flow, Measured (%))) shall be maintained above and to the left of the curve in the Unit One OAC Databook "Load-Changing Recommendations".

NOTE:
- Several of the parameters required for this procedure can be found on OAC graphics, and a list of all OAC points are found on Enclosure 4.8 (Turbine Generator Roll Computer Points).
- Step 3.1 and Step 3.2 may be performed in any order.

_____ 3.1 **IF** increasing turbine generator load, perform the following:

_____ 3.1.1 Ensure the proper reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 3.1.2 **WHILE** increasing Turbine Generator load, perform the following:

_____ 3.1.2.1 **IF AT ANY TIME** Turbine load is < 30%, operate the RC system pumps and fans as required per OP/1/B/6400/001 A (Condenser Circulating Water System) to maintain vacuum in Condenser C < 28" Hg.

_____ 3.1.2.2 **IF** applicable, verify Groups B and C valves on Enclosure 4.6 (Valve Checklist) close at 15% of full load (184 MWe, 107 psig Turbine Impulse Pressure).

_____ 3.1.2.3 **IF** applicable, verify the following valves close at 15% of full load (184 MWe, 107 psig Turbine Impulse Pressure):

☐ 1SM-21 (Ctrl Vlv #2 Stm Lead Drn)

☐ 1SM-29 (Ctrl Vlv #1 Stm Lead Drn)

_____ 3.1.2.4 **IF** applicable, **WHEN** CV3 comes off of its fully closed seat (65% of full load, 796 MWe), verify 1SM-25 (Ctrl Vlv #3 Stm Lead Drn) closes.

_____ 3.1.2.5 **IF** applicable, **WHEN** CV4 comes off of its fully closed seat (92% of full load, 1127 MWe), verify 1SM-33 (Ctrl Vlv #4 Stm Lead Drn) closes.

CAUTION:

1. Until it is recognized that the first stage shell metal temperature change rate stays below the allowable limit (150°F/hr), the following loading rate shall **NOT** be exceeded:
   - 1/2%/min - First Stage Inner Shell Temperature (1MC3 or OAC point C1A1140 (Turbine Lower Inner Shell Temp)) $\leq$ 350°F
   - 1%/min - First Stage Inner Shell Temperature (1MC3 or OAC point C1A1140 (Turbine Lower Inner Shell Temp)) > 350°F
2. Normal steady-state load changes shall be made without exceeding the limits shown on Enclosure 4.7 (Generator Operating Limits) and in the Unit 1 OAC Databook "Recommended Startup and Loading Curves".
3. Unit One Reactor Operating Data, Section 2.4 shall be referred to for allowable ramp rates. A "LOAD RATE" > 6.2 MW/MIN shall **NOT** be used during normal load changes.

3.1.3 Increase turbine generator load by performing the following:

_____ 3.1.3.1 Select "LOAD RATE" and verify it illuminates.

_____ 3.1.3.2 Input the desired load rate.

_____ 3.1.3.3 Select "ENTER" and verify "LOAD RATE" goes dark.

_____ 3.1.3.4 Select "TARGET" and verify it illuminates.

_____ 3.1.3.5 Input the desired load target.

_____ 3.1.3.6 Select "ENTER" and verify "TARGET" light goes dark.

_____ 3.1.3.7 Verify new load target appears on Target Display.

_____ 3.1.3.8 Select "GO" and verify it illuminates to start load increase.

_____ 3.1.3.9 Coordinate with Secondary Chemistry to adjust S/G blowdown flowrates to obtain maximum blowdown for the appropriate load.

CAUTION:

1. The load, hydrogen pressure and power factor limits per the Unit One Revised Data Book Figure 43 shall **NOT** be exceeded.
2. Rate of change of First-Stage Bowl Inner Surface Temperature shall **NOT** exceed 150°F/hr (OAC point C1P1283 (First Stage Metal Temp Rate)).
3. OAC point C1A1140 (Turbine Lower Inner Shell Temp) vs. Percent Steam Flow (OAC point C1P1588 (Design Total Main Steam Flow, Measured (%)) shall be maintained above and to the left of curve in the Unit One OAC Databook "Load-Changing Recommendations".
4. Control valve casing difference, OAC point C1A0961 (Turb Valve Chest Inner Surface Metal Temp) minus C1A0967 (Turb Valve Chest Outer Surface Metal Temp), shall **NOT** exceed curve "Allowable Temp Difference on Turbine Valve Chest" in the Unit 1 OAC Databook.

N/A 3.2 **IF** decreasing turbine generator load, perform the following:

_____ 3.2.1 Ensure the proper reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 3.2.2 **WHILE** decreasing turbine generator load, perform the following:

_____ 3.2.2.1 **IF AT ANY TIME** Turbine load is < 30%, operate the RC system pumps and fans as required per OP/1/B/6400/001 A (Condenser Circulating Water System) to maintain vacuum in Condenser C < 28" Hg.

_____ 3.2.2.2 **IF** CV4 fully closes (92% of full load, 1127 MWe), verify 1SM-33 (Ctrl Vlv #4 Stm Lead Drn) opens.

_____ 3.2.2.3 **IF** CV3 fully closes (65% of full load, 796 MWe), verify 1SM-25 (Ctrl Vlv #3 Stm Lead Drn) opens.

CAUTION: 1. Normal steady-state load change shall be made without exceeding limits shown on Enclosure 4.7 (Generator Operating Limits) and in the Unit 1 OAC Databook "Recommended Starting and Loading Curves".

2. Unit One Reactor Operating Data, Section 2.4 shall be referred to for allowable ramp rates.

3.2.3 Decrease turbine generator load by performing the following:

_____ 3.2.3.1 Select "LOAD RATE" and verify it illuminates.

_____ 3.2.3.2 Input the desired load rate.

_____ 3.2.3.3 Select "ENTER" and verify "LOAD RATE" goes dark

_____ 3.2.3.4 Select "TARGET" and verify it illuminates.

_____ 3.2.3.5 Input the desired load target.

_____ 3.2.3.6 Select "ENTER" and verify "TARGET" goes dark.

_____ 3.2.3.7 Verify new load target appears on Target Display.

_____ 3.2.3.8 Select "GO" and verify it illuminates to start load decrease.

_____ 3.2.3.9 Coordinate with Secondary Chemistry to adjust S/G blowdown flowrates to obtain maximum blowdown for the appropriate load.

3.3 Do **NOT** file a copy of this enclosure in the designated storage cabinet.

_____ 3.48.7 Input the desired load target.

_____ 3.48.8 Select "ENTER" and verify "TARGET" light goes dark.

_____ 3.48.9 Verify new load target appears on Target Display.

_____ 3.48.10 Select "GO" and verify it illuminates to start load increase. (R.M.)

_____ 3.48.11 Verify "HOLD" light is dark.

NOTE:	IST partial stroke test requirement for the following check valves is satisfied based on the CF flow requirement to the lower S/G nozzle to reach 50% power.

AA 3.49 **WHEN** reactor power > 50%, perform PT/1/A/4200/005 E (Various Check Valve Forward Flow Test) for following:

- ☑ 1CF-31 (1A S/G Feedwater Containment Isol Inlet Check)
- ☑ 1CF-40 (1B S/G Feedwater Containment Isol Inlet Check)
- ☑ 1CF-49 (1C S/G Feedwater Containment Isol Inlet Check)
- ☑ 1CF-58 (1D S/G Feedwater Containment Isol Inlet Check)

N/A 3.50 **IF** required, notify IAE to adjust nuclear instrumentation per Model W/O #00874628.
Person notified ____________________________

AA 3.51 Notify Secondary Chemistry to determine required S/G blowdown flow rate and adjust as needed.
Person notified **Tony Story**

AA 3.52 Prior to exceeding P-9 (69% Reactor Power), ensure PT/1/A/4250/002 B (Monthly Main Turbine Valve Movement Test) has been completed within the previous 31 days.

_____ 3.53 At 69% Reactor power, verify "P9 - REACTOR TRIP ON TURBINE TRIP BLOCKED" light on 1SI-18 goes dark.

NOTE:	Procedure may continue while performing Step 3.54.

_____ 3.54 At approximately 65% turbine load (796 MWe), place Second Stage Reheater high load valves in service per power escalation enclosure of OP/1/B/6250/013 (Moisture Separator Reheater Operation).

_____ 3.55 **<u>IF</u>** returning to this enclosure after resolving an issue with Main Turbine Generator that required a power reduction to approximately 66% perform following:

_____ 3.55.1 Ensure Second Stage Reheater high load valves aligned as required per power escalation enclosure of OP/1/B/6250/013 (Moisture Separator Reheater Operation).

_____ 3.55.2 Continue with normal power escalation.

_____ 3.55.3 At 69% reactor power, verify "P9 - REACTOR TRIP ON TURBINE TRIP BLOCKED" light on 1SI-18 goes dark.

NOTE:
1. Placing megawatt feedback loop in and out of service is a bumpless transfer. When MW IN is selected the TARGET and setpoint will become ACTUAL megawatts. When MW OUT is selected the TARGET becomes present valve reference converted to MWs. In either condition, load changes in progress will be seen as complete with TARGET change and GO light extinguished.
2. Previously inputted load rates are **<u>NOT</u>** changed when changing MW feedback status.

_____ 3.56 **<u>IF</u>** MW IN in service on the Main turbine, prior to placing "C" Heater Drain pumps in service perform following to take Main Turbine control back to MW OUT: (R.M.)

_____ 3.56.1 Ensure Main Turbine is in HOLD.

_____ 3.56.2 Depress "MW IN/MW OUT" pushbutton.

_____ 3.56.3 Verify green "MW OUT" light is illuminated.

NOTE:
1. "C" Heater Drain pumps shall **<u>NOT</u>** be placed in service prior to reaching 70% power due to restrictions associated with the isolation of the flowpath through 1CM-127. {PIP 98-1726}
2. Procedure may continue while performing Step 3.57.

_____ 3.57 At 70% turbine load (857 MWe), place available "C" Heater Drain pumps in service per OP/1/B/6250/004 (Feedwater Heaters, Vents, Drains and Bleed System).

_____ 3.58 **<u>WHILE</u>** available "C" Heater Drain pumps are being placed in service, continue turbine load increase per OP/1/B/6300/001 (Turbine Generator) enclosure for load changing.

_____ 3.59 **<u>IF</u>** required due to Generator/Automatic Voltage Regulator (AVR) testing at approximately 75% turbine load (~919 MWe), perform following:

_____ 3.59.1 **<u>IF</u>** performing Generator/Automatic Voltage Regulator (AVR) testing, **<u>HOLD</u>** until Generator/AVR personnel are ready for Operations to continue with Unit 1 power increase.

_____ 3.59.2 Once notified by AVR personnel that AVR testing is complete, at this power, begin power increase.

Person making notification____________________

CAUTION:	Alternate indications of reactor power shall be monitored to verify reactor power level and help prevent NI miscalibration.

_____ 3.60 At 75% reactor power , compare OAC heat balance point C1P1385 (Reactor Thermal Power, Best) to nuclear instrumentation.

_____ 3.61 **<u>IF</u>** required, notify IAE to adjust nuclear instrumentation per Model W/O #00874628.
Person notified ________________________________

NOTE:

1. Placing megawatt feedback loop in and out of service is a bumpless transfer. When MW IN is selected the TARGET and setpoint will become ACTUAL megawatts. When MW OUT is selected the TARGET becomes present valve reference converted to MWs. In either condition, load changes in progress will be seen as complete with TARGET change and GO light extinguished.

2. Previously inputted load rates are **<u>NOT</u>** changed when changing MW feedback status.

_____ 3.62 **<u>IF</u>** Step 3.56 was performed **<u>AND</u>** it is desired to place MW IN in service on Main turbine, perform following: (R.M.)

_____ 3.62.1 Ensure Main Turbine is in HOLD.

_____ 3.62.2 Verify Secondary feedwater and steam systems stable.

_____ 3.62.3 Depress "MW IN/MW OUT" pushbutton. (R.M.)

_____ 3.62.4 Verify red "MW IN" light is illuminated.

_____ 3.62.5 Ensure desired load rate.

_____ 3.62.6 Select "TARGET" and verify it illuminates.

_____ 3.62.7 Input desired load target.

_____ 3.62.8 Select "ENTER" and verify "TARGET" light goes dark.

_____ 3.62.9 Verify new load target appears on Target Display.

_____ 3.62.10 Select "GO" and verify it illuminates to start load increase. (R.M.)

_____ 3.62.11 Verify "HOLD" light is dark.

NOTE:	DCS alarms associated with OTΔT will alarm if power is below approximately 85% due to inputs over ranging. Performing the below will reinstate the associated alarms. These alarms will clear once power increases and inputs are no longer over ranged.

3.63 Perform following to enable DCS alarms associated with OTΔT on DCS graphic 6068:

_____ • Set alarm checking to "ON" for NCAA5422.

_____ • Set alarm checking to "ON" for NCAA5462.

_____ • Set alarm checking to "ON" for NCAA5502.

_____ • Set alarm checking to "ON" for NCAA5542.

_____ • Set alarm checking to "ON" for OTDELTAT-FAIL.

CAUTION:	Failure to perform following step as written may result in lifting the AS Header relief valve.

NOTE:	OAC Graphics CF Pump Details, CFPMP1A and CFPMP1B, shall be referred to while swapping steam supplies for CFPTs.

3.64 At approximately 85% turbine load (1041 MWe), perform following while ensuring CF pump speed is maintained:

_____ 3.64.1 Slowly open 1SP-3 (SC To CFPT 1A & 1B) (TB-640, 1G-24).

_____ 3.64.2 Adjust 1AS-2 (Main Stm To Aux Stm) setpoint to maintain 125 psig.

_____ 3.64.3 **IF** 1AS-12 (AS To CFPT Isol) can be closed from the control room,
THEN slowly close 1AS-12 (AS To CFPT Isol).

_____ 3.64.4 **IF** 1AS-12 (AS To CFPT Isol) will **NOT** close from the control room, **THEN** perform the following:

_____ 3.64.4.1 Place breaker 1MXH-R02A (AUX STEAM TO CF PUMP TURBINE ISOL MOTOR (1AS12)) in the "OFF" position.

_____ 3.64.4.2 Manually close 1AS-12 (AS TO CFPT ISOL) (TB1 /E 605/1M 27) to move valve off it's open seat.

_____ 3.64.4.3 Place breaker 1MXH-R02A (AUX STEAM TO CF PUMP TURBINE ISOL MOTOR (1AS12)) in the "ON" position

_____ 3.64.4.4 Slowly close 1AS-12 (AS To CFPT Isol).

_____ 3.64.5 **IF** Unit 1 and Unit 2 AS headers are cross-tied, adjust setpoint of 2AS-2 (Main Stm To Aux Stm) to maintain desired AS header pressure.

_____ 3.65 **IF** a temporary PC was installed at the local MSR Panel for placing the MSR's in service, remove the PC.

_____ 3.66 **IF** desired to increase turbine load to greater than 85%, go to Enclosure 4.3 (Unit Operation Between 85% and 100% Power).

3.67 File a copy of this enclosure in the designated cabinet.

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can affect core reactivity by changing control rod position. (R.M.)

1.2 The following Limits and Precautions are Reactivity Management related: (R.M.)

1.2.1 When rods are being moved, observe "RODS IN/RODS OUT" light for proper direction.

1.2.2 When rods are being moved, observe the demand position and actual (digital) position to verify proper operation of the Rod Control System.

1.2.3 Adjusting T-Avg ± 1°F of T-Ref before transferring rod control to "AUTO" will prevent undesired rod movement.

1.2.4 Monitor startup rate continuously during any rod motion to ensure < 0.5 DPM stable startup rate.

1.3 Automatic rod control shall **NOT** be used when less than 15% (184 MW_e) turbine power.

1.4 Individual control bank positions on "CRD BANK SELECT" switch shall not be used to position rods manually. (The automatic overlap feature is disabled.)

1.5 After releasing Rod Motion lever, waiting 2 seconds before attempting to move rods again will allow all signals to clear the firing cards.

1.6 A rod motion demand below zero steps may result in the movable grippers **NOT** properly engaging the drive shaft.

2. Initial Conditions

AA 2.1 Ensure Reactivity Management controls established per AD-OP-ALL-0203 (Reactivity Management. (RM)

AA 2.2 Verify Unit 1 is **NOT** in an EP or AP.

AA 2.3 Verify one of the following exist:

- ☑ Control Bank movement required to increase/decrease Reactor Power
- ☐ Control Bank movement required to increase/decrease Tavg
- ☐ Control Bank movement required to maintain AFD
- ☐ Control Bank manual control required to support testing/maintenance activity

3. Procedure

NOTE:	Steps 3.1 through 3.6 may be signed off as time allows ensuring operator maintains proper focus on reactivity management.

AA 3.1 Monitor the following:

- Tavg/Tref
- Demand Counter positions
- DRPI rod positions
- ROD MOTION RODS-IN/RODS-OUT Light
- ROD MOTION DEMAND SIGNALS - TEMP ERROR/POWER MISMATCH
- Power Range instruments
- IR SUR (Startup Rate)

AA 3.2 **IF** MANUAL ROD movement is desired, perform the following:

- Verify the "ALM" LED on circuit card A206 in the left side of 1ERCC0006 (Rod Control Logic Cabinet) is **NOT** illuminated.
- Verify one GRP select light is illuminated on each power cabinet.

AA 3.3 **IF** plant conditions require, place the "CRD BANK SELECT" switch in "MAN".

_____ 3.4 **IF** withdrawing Control Banks, pull and hold the "ROD MOTION" lever "OUT" as required until control rods are in the desired position. (R.M.)

_____ 3.5 **IF** inserting Control Banks, push and hold the "ROD MOTION" lever "IN" as required until control rods are in the desired position. (R.M.)

_____ 3.6 **IF** automatic rod control is desired, perform the following:

_____ 3.6.1 Verify Unit 1 Reactor Power is ≥ 15% RTP.

_____ 3.6.2 **WHEN** Tavg is within 1°F of Tref, place "CRD BANK SELECT" in "AUTO".

3.7 Do **NOT** file this enclosure.

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can affect core reactivity by changing boron concentration. (R.M.)

1.2 The following Limits and Precautions are Reactivity Management related: (R.M.)

1.2.1 When performing dilutions at or near 100% power, batch additions to the VCT (instead of continuous dilution at low flow rates) are the preferred method. {PIP C99-0587}

1.2.2 If the unit has operated continuously for several months, significant Boron 10 depletion may have occurred. The effective boron concentration of the NC System may be lower than indicated by Chemistry samples.

1.3 Continuous dilution operations will affect the NC System H_2 concentration.

1.4 With the "NC MAKEUP MODE SELECT" switch in the "DILUTE" position, the makeup flow rate is limited by letdown flow, the VCT spray nozzle, and VCT pressure. The maximum expected makeup flow rate is approximately 95 gpm.

1.5 With a reactor makeup water pump running following an auto start, repositioning its control switch to "ON" can cause the pump motor breaker to trip. Therefore, the control switches for these pumps shall only be repositioned with the associated pump off.

1.6 Maintaining VCT pressure as low as practical during large makeups will minimize gas absorption. VCT pressure can be reduced by diverting letdown or by VCT purge.

1.7 Due to Electromagnetic Interference within the Unit 1 Reactor Makeup Control System, the Unit 1 Boric Acid Counter may sporadically count up during dilution activities. OFF indications for the Boric Acid Xfer Pumps and Closed indication for valve 1NV-238A can be used by the Reactor Operators to validate that sporadic counts are indication only. (NCR 02081372).

2. Initial Conditions

AA 2.1 Ensure R2 reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

AA 2.2 Verify Unit 1 is in Mode 1 or 2.

AA 2.3 Verify the NV System is in operation per OP/1/A/6200/001 (Chemical and Volume Control System).

AA 2.4 Verify sufficient RHT volume is available to receive the reactor coolant displaced during the planned dilution operation.

AA 2.5 Verify the NB System is in operation per OP/1/A/6200/012 (Reactor Makeup Water).

N/A 2.6 **IF** NC System boron concentration will be changed by ≥ 50 ppm, initiate PZR spray to equalize the boron concentration throughout the system by operating backup heaters per OP/0/A/6200/055 (Miscellaneous Component Operation).

3. Procedure

NOTE: This enclosure will affect reactivity of the core and is therefore designated important to Reactivity Management per the guidelines of AD-OP-ALL-0203 (Reactivity Management). (R.M.)

AA 3.1 Ensure valves are aligned per Enclosure 4.8 (Valve Checklist).

_____ 3.2 **IF** the blender is set for automatic makeup per Enclosure 4.1 (Automatic Makeup), record the setpoint on 1NV-242A (RMWST To B/A Blender Ctrl): ___________gpm

NOTE: The purpose of the following step is to minimize the pressure fluctuation caused by manually diverting to the RHT and effects on Reactor Water Makeup flow.

_____ 3.3 **IF** desired, adjust the setpoint for 1NV-172A (VCT-LEVEL CTRL) (1NVSS5760) to 55%.

3.4 Ensure the following valve control switches in "AUTO":

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ 3.5 Ensure 1NV-242A (RMWST To B/A Blender Ctrl) controller in auto.

_____ 3.6 Ensure at least one reactor makeup water pump is in "AUTO" or "ON".

☐ 3.7 Record the desired volume of reactor makeup water to be added. __________ gallons

_____ 3.8 Adjust the total makeup counter to the desired volume of reactor makeup water to be added. (R.M.)

_____ 3.9 Place the "NC MAKEUP MODE SELECT" switch to the "DILUTE" position.

NOTE: High letdown flow rates result in increased backpressure on the letdown line. If letdown flow is ≥ 90 gpm, it may be desirable to reduce the dilution flow rate to 80 gpm to avoid the Rx Make-up Flow Deviation alarm and associated automatic actions.

_____ 3.10 **IF** required, adjust the setpoint for 1NV-242A (RMWST To B/A Blender Ctrl) to the desired flow.

_____ 3.11 **IF AT ANY TIME** it is desired to divert letdown to the RHT manually operate 1NV-172A (3-Way Divert To VCT-RHT) as follows:

_____ 3.11.1 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) to the "RHT" position.

_____ 3.11.2 Ensure VCT level is monitored continuously while diverting to the RHT.

NOTE: Procedure may continue while performing the following step.

3.11.3 **WHEN** desired VCT level is reached return 1NV-172A (3-Way Divert To VCT-RHT) to auto as follows:

_____ 3.11.3.1 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) in the "VCT" position.

_____ 3.11.3.2 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) in the "AUTO" position.

_____ 3.12 **IF AT ANY TIME** during the makeup it becomes necessary to change the makeup flow rate, adjust the setpoint for 1NV-242A (RMWST To B/A Blender Ctrl) as necessary to achieve the desired flow.

_____ 3.13 **IF AT ANY TIME** while dilution is in progress it becomes necessary **OR** it is desired to stop the dilution (for example: loss of all NC Pumps, unexpected results observed or a large makeup being made in multiple batches), perform the following:

_____ 3.13.1 Place the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.13.2 Ensure the following valves close:

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ 3.13.3 **IF** in "AUTO", verify the reactor makeup water pump stops.

☐ 3.13.4 Record reactor makeup water volume added as indicated on the total makeup counter. _____________ gallons

3.13.5 **WHEN** conditions allow resuming the dilution, perform the following:

☐ 3.13.5.1 Determine remaining volume to be added by subtracting the amount previously added (Step 3.13.4) from the desired volume to be added (Step 3.7).

_____________ - ___________ = ____________ gallons
(Step 3.7) (Step 3.13.4)

_____ 3.13.5.2 Adjust total makeup counter to the volume of reactor makeup water determined in Step 3.13.5.1. (R.M.)

_____ 3.13.5.3 Place the "NC MAKEUP CONTROL" switch in the "START" position. (R.M.)

_____ 3.13.5.4 Verify the following:

☐ 1NV-242A (RMWST To B/A Blender Ctrl) modulates to establish desired flow
☐ 1NV-181A (B/A Blender Otlt To VCT) opens

_____ 3.13.5.5 **IF** in "AUTO", verify the reactor makeup water pump starts.

_____ 3.14 **WHILE** makeup is in progress, monitor the following for expected results:

☐ Control rod motion
☐ NC System Tavg
☐ Reactor Power

NOTE:	If a small makeup is being performed, placekeeping for Steps 3.15 through 3.17 may be performed after Step 3.18 is performed.

_____ 3.15 Place the "NC MAKEUP CONTROL" switch in the "START" position. (R.M.)

_____ 3.16 Verify the following:

- ☐ 1NV-242A (RMWST To B/A Blender Ctrl) modulates to establish desired flow
- ☐ 1NV-181A (B/A Blender Otlt To VCT) opens

_____ 3.17 **IF** in "AUTO", verify the reactor makeup water pump starts.

NOTE:	The total makeup counter may count up 1 - 5 gallons after termination.

3.18 **WHEN** the desired volume of reactor makeup water is reached on the total makeup counter, ensure the following valves close. (R.M.)

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ 3.19 **IF** Step 3.3 was performed, return the setpoint for 1NV-172A (VCT LEVEL CRL) (1NVSS5760) to 75.0 %.

_____ 3.20 **IF** automatic makeup is desired, perform one of the following:

_____ 3.20.1 **IF** it is desired to change the blender outlet boron concentration, refer to Enclosure 4.1 (Automatic Makeup).

OR

_____ 3.20.2 **IF** makeup at the previous concentration is acceptable **AND** the system was previously aligned per Enclosure 4.1 (Automatic Makeup), perform the following:

_____ 3.20.2.1 Ensure the controller for 1NV-242A (RMWST To B/A Blender Ctrl) is set to the value recorded in Step 3.2. (R.M.)

_____ 3.20.2.2 Place the "NC MAKEUP MODE SELECT" switch in "AUTO".

_____ 3.20.2.3 Place the "NC MAKEUP CONTROL" switch to the "START" position. (R.M.)

3.21 Do **NOT** file this enclosure.

Elapsed Time (Hours)	Power (%FP)	Rod Bank Position (SWD)	Chemical & Volume Control System Changes	Comments
0.0	50.0	CD @ 139	459 Gallons of WATER	
0.5	55.0	CD @ 149	498 Gallons of WATER	
1.0	60.0	CD @ 156	454 Gallons of WATER	
1.5	65.0	CD @ 163	485 Gallons of WATER	
2.0	70.0	CD @ 169	531 Gallons of WATER	
2.5	75.0	CD @ 174	468 Gallons of WATER	
3.0	80.0	CD @ 181	508 Gallons of WATER	
3.5	85.0	CD @ 188	566 Gallons of WATER	
4.0	90.0	CD @ 195	196 Gallons of WATER	
4.5	92.7	CD @ 200	--	
5.0	95.3	CD @ 215	498 Gallons of WATER	
5.5	98.0	CD @ 215	153 Gallons of WATER	
6.0	99.0	CD @ 215	91 Gallons of WATER	
6.5	99.5	CD @ 215	63 Gallons of WATER	
7.0	100.0	CD @ 215	--	

2021 INITIAL LICENSE NRC EXAM
SCENARIO # 2

Facility:	Catawba NRC Exam 2021	Scenario No.: 2	Op Test No.: 2021301
Examiners:		Operators:	SRO
			RO
			BOP

Initial Conditions: Unit 1 is at 75% power at the MOL. Unit 2 is at 100% power.

Turnover: Unit 1 is at 75% power at the MOL following maintenance of the 1A CFPT LF pumps. 1A CFPT has been placed back in service and current power level is being maintained while maintenance continues to monitor LF system performance. Unit 2 is at 100% power. Direction for the crew is to swap LCVUs by securing 1A LCVU and starting 1C LCVU using OP/1/A/6450/001 Encl. 4.13.

Event No.	Malf. No.	Event Type*	Event Description
1		N – BOP N – SRO	Swap operating LCVUs
2		C – BOP C – SRO TS - SRO	1B RN Pump trip
3		C – RO C – SRO	Main Generator Voltage Regulator Failure
4		C – BOP C – SRO TS – SRO	1C SGTL
5		R – RO R – SRO N – BOP	AP/09 Rapid Downpower
6		C – RO C – SRO	Rods fail to insert in automatic during AP/09 Rapid Shutdown
7		M – ALL	1C Steam Generator Tube Rupture
8		C – RO C – SRO	1C CA Flow Control Valve will not close
9		C – BOP C – SRO	1NI-9A and 1NI-10B fail to open automatically on S/I
10		C – RO C – SRO	MSIVs fail to close manually

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Scenario 2 – Summary

Initial Condition
Unit 1 is at 75% power at the MOL. Unit 2 is at 100% power.

Turnover:
Unit 1 is at 75% power at the MOL following maintenance of the 1A CFPT LF pumps. 1A CFPT has been placed back in service and current power level is being maintained while maintenance continues to monitor LF system performance. Unit 2 is at 100% power. Direction for the crew is to swap LCVUs by securing 1A LCVU and starting 1C LCVU using OP/1/A/6450/001 Encl. 4.13.

Event 1
BOP will swap operating LCVUs per OP/1/A/6450/001 Encl. 4.13.

Event History: This event has previously been performed as a JPM (2019), but has never been used as a normal evolution in a scenario.

Event 2
1B RN pump will trip. Crew will enter AP/0/A/5500/020 (Loss of Nuclear Service Water) Case 1 (Loss of RN train) and start an RN pump. SRO will address Tech Specs.

Verifiable Action – BOP will manually start an RN pump as directed.

Event History: This failure last used 13 (3).

Event 3
The Unit 1 Main Generator voltage regulator will fail low. Crew will enter AP/1/A/5500/037 (Generator Voltage and Electric Grid Disturbances) Case 1 (Abnormal Generator or Grid Voltage). Crew will manually adjust Unit 1 Generator Voltage to return voltage to normal per the voltage schedule.

Verifiable Action – The RO will manually adjust Generator voltage to restore to normal per the voltage schedule.

Event History: Voltage Regulator failure last used 17 (3). However, this is the first time that voltage has been failed low.

Event 4
A S/G Tube Leak will develop on the 1C S/G. Crew will enter AP/1/A/5500/010 (Reactor Coolant Leak) Case 1 (Steam Generator Tube Leak). Crew will adjust charging and letdown flow to stabilize Pressurizer level, will quantify the leak size, and initiate a rapid downpower. SRO will address Tech Specs.

Verifiable Action – BOP will throttle charging and letdown flow to stabilize Pressurizer level and quantify the size of the tube leak.

Event History: The last S/G Tube leak was on 1B S/G on 17(3). A 1C S/G tube leak has not been used before.

Event 5
Crew will enter AP/1/A/5500/009 (Rapid Downpower) when directed by AP/10.

Verifiable Action – RO will enter desired turbine target load and load rate and place the main turbine in 'GO' to begin power decrease. BOP will perform boration.

Event History: Rapid downpower last used in 17 (3).

Event 6
Control rods will fail to insert automatically during the rapid downpower. The RO will place control rods in MANUAL to control NC system Tavg.

Verifiable Action – RO will manually insert control rods to mimic automatic control.

Event History: The last automatic control rod failure was in 19 (3), but was during a spurious turbine trip. This is the first time that this failure has been used during a rapid downpower.

Event 7
The S/G tube leak size will increase requiring the crew to manually trip the reactor and initiate safety injection due to not being able to maintain Pressurizer level. Crew will enter EP/1/A/5000/E-0 (Reactor Trip or Safety Injection) to ensure safeguards equipment is in service as required. Crew will eventually transition to EP/1/A/5000/E-3 (Steam Generator Tube Rupture).

Verifiable Action – RO will manually trip Unit 1 reactor. BOP will initiate Safety Injection.

Event History: The last S/G tube rupture was in 17 (3) on 1B S/G. A 1C S/G tube rupture has not been used before.

Event 8
The CA flow control valve will fail to close when attempted by the RO per E-0 foldout page guidance.

Verifiable Action – RO will isolate CA flow to 1C S/G using the motor operated isolation valve.

Event History: This failure is new.

Event 9
NV pumps to Cold Leg isolation valves 1NI-9A and 1NI-10B will fail to automatically open on the Safety Injection signal.

Verifiable Action – BOP will manually open 1NI-9A and 1NI-10B.

Event History: Similar failure used on 19 (4) except 1NI-9A would not open automatically or manually.

Event 10
During implementation of E-3, the MSIVs on all S/Gs will fail to close manually. Since the ruptured S/G cannot be isolated from the intact S/Gs, a transition to EP/1/A/5000/ECA-3.1 (SGTR With Loss of Reactor Coolant – Subcooled Recovery Desired) will be made. Crew will initiate a less than 100°F/hour cooldown.

Verifiable Action – RO will initiate a less than 100°F/hour cooldown.

Event History: This failure is new.

Manual Control of Automatic Functions		
Event	**Position**	**Description**
3	RO	Main Generator Voltage Regulator Failure
6	RO	Manual Insertion of Control Rods during Rapid Downpower

***Critical Task 1* – Restore RN flow prior to any NC pump motor bearing reaching 195°F.**

***Critical Task 2* – Establish high head ECCS flow prior to transition from E-0.**

***Critical Task 3* – Initiate Cooldown at less than 100°F/hour.**

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Total malfunctions (5–8)	8
2. Malfunctions after EOP entry (1–2)	3
3. Abnormal events (2–4)	4
4. Major transients (1–2)	1
5. EOPs entered/requiring substantive actions (1–2)	2
6. EOP contingencies requiring substantive actions (0–2)	1
7. Critical tasks (2–3)	3

EXERCISE GUIDE WORKSHEET

1. INITIAL CONDITIONS:

   1.1 Reset to IC # 169 (password = gallus21) and load schedule file for NRC Scenario 2

START TIME: __________

✓	✓	Trigger	Instructor Action	Final	Delay	Ramp	Delete In	Event
		2	LOA-RN003 (RACKOUT RN PUMP 1B)	RACK-OUT				2
		3	OVR-EGB004D (VOLTAGE ADJUST LOWER PB)	ON			:10	3
		4	MAL-SG001C (S/G C TUBE LEAK)	65		:10		4
			MAL-IRX009 (RODS FAIL TO MOVE)	AUTO				6
		7	MAL-SG001C (S/G C TUBE LEAK)	400				7
			MAL-CA009C (MD CA CTRL VLV CA44 FAILURE)	100				8
			VLV-NI001A (NI9A B.I.T. DISCHARGE ISOL VLV FAIL AUTO ACTIONS)	ACTIVE				9
			VLV-NI002A (NI10B B.I.T. DISCHARGE ISOL VLV FAIL AUTO ACTIONS)	ACTIVE				9
			MAL-SM006E (ALL MSI (MSIV) VLV FAIL)	ACTIVE				10
		13	LOA-VC039 (MNL RST OF LATCH FOR CHILLER B HI COND PRESS TRP DUE TO LOSS RN)	RESET	5:00			2

2. SIMULATOR BRIEFING

2.1 Control Room Assignments:

Position	Name
CRS	
RO	
BOP	

2.2 Give a copy of Attachment 2 (Shift Turnover Information) to the CRS.

3. EXERCISE PRESENTATION

3.1 Familiarization Period

A. Allow examinees time to familiarize themselves with the Control Board alignments.

3.2 **Scenario EVENT 2**, 1B RN Pump Trip

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 2** to cause 1B RN Pump to trip.

✓	BOOTH INSTRUCTOR ACTION
	IF dispatched as an AO to perform a post start check of 1A/2A/2B RN pump, **REPEAT** the information.
	After 5 minutes, contact the control room crew and **REPORT**: • "Post start check of 1A/2A/2B RN pump is complete. The pump looks good for continued operation."

✓	BOOTH INSTRUCTOR ACTION
	WHEN contacted as an AO to locally reset YC chiller, **REPEAT** the information. **INSERT SIMULATOR Trigger 13**.
	After 5 minutes **REPORT**: • "YC Chiller 'B' has been reset and has re-started."

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for 1B RN Pump, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as Environmental chemistry to report RN pump shifts, **REPEAT** the information.

3.3 **Scenario EVENT 3**, Main Generator Voltage Regulator Failure

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 3** to cause Main Generator Voltage to fail low.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as DEC TOP to monitor RTCA indications, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for the Main Generator Voltage Regulator, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as DEC BA to determine CNS Generator Voltage requirements, **REPORT:** • "Adjust voltage per the operating schedule."

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as DEC TOP or BA to determine if switchyard (grid) voltage adequate and reliable, **REPORT**: • "Yes, switchyard and grid voltage are adequate and reliable."

3.4 **Scenario EVENT 4**, 1C S/G Tube Leak

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 4** to cause a 1C S/G Tube Leak.

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for 1C S/G tube leakage, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as RP to frisk all S/G cation columns, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as Secondary Chemistry to sample all S/Gs for activity, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as RP to notify of the size of the S/G tube leak, **REPEAT** the information.

3.5 **Scenario EVENTS 7/8/9/10**, 1C SGTR / 1C CA Flow Ctrl Vlv failed open / 1NI-9A & 1NI-10B fail to auto open / MSIVs fail to close manually

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 7** to cause a 1C S/G Tube Rupture.

✓	BOOTH INSTRUCTOR ACTION
	IF dispatched as an AO to unlock and close 1SA-4, **REPEAT** the information.

Enclosure 4.13

Shifting Operating Lower Containment Ventilation Units And Pipe Tunnel Booster Fans

OP/1/A/6450/001
Page 1 of 7

1. Limits and Precautions

1.1 Observe the upper and lower containment temperature limits of Tech Spec 3.6.5.

1.2 All operating lower containment ventilation units, and the operating pipe tunnel booster fan, are normally operated at the same speed.

1.3 When CLAs are pressurized above 175 psig, lower containment temperature shall be maintained greater than 60°F to maintain accumulator temperature greater than 60°F due to brittle fracture concerns of the accumulator vessel.

2. Initial Conditions

None

3. Procedure

N/A 3.1 **IF** two LCVUs are operating **AND** it is desired, shift the operating units as follows:

_____ 3.1.1 **IF** the LCVUs are operating in "LOW" speed, perform the following:

NOTE:	If two LCVUs are to remain in service, it is preferable to run a vent unit in each fan room (A/D, B/C) in order to maximize air distribution in the lower containment. Due to the temperature characteristics in lower containment and the Digital Rod Position Indication (DRPI) Panels area, use of the 1D LCVU is preferred. Failure to operate with at least one vent unit in each fan room during Modes 1 - 3 can result in high pressurizer and/or steam generator cavity air temperatures.

3.1.1.1 Start an idle LCVU by placing its control switch in the "LOW" position:

_____ • "VV LCVU 1A"
_____ • "VV LCVU 1B"
_____ • "VV LCVU 1C"
_____ • "VV LCVU 1D"

_____ 3.1.1.2 Verify the red indicating light illuminates for the LCVU placed in service.

_____ 3.1.1.3 Verify the red "OPEN" indicating light illuminates for the LCVU dampers associated with the LCVU placed in service.

_____ 3.1.1.4 Verify the red "MAX" indicating light illuminates for the LCVU placed in service.

Enclosure 4.13

Shifting Operating Lower Containment Ventilation Units And Pipe Tunnel Booster Fans

OP/1/A/6450/001
Page 2 of 7

3.1.1.5 Stop the LCVU to be removed from service by placing its control switch in the "OFF" position:

_____ • "VV LCVU 1A"
_____ • "VV LCVU 1B"
_____ • "VV LCVU 1C"
_____ • "VV LCVU 1D"

NOTE: The procedure may continue up to and including Step 3.1.1.9 before completing the following step.

_____ 3.1.1.6 Verify the green indicating light illuminates for the LCVU stopped.

_____ 3.1.1.7 Verify the red "MAX" indicating light extinguishes for the LCVU stopped.

_____ 3.1.1.8 Verify the green "CLOSED" indicating light illuminates for the LCVU damper associated with the LCVU stopped.

_____ 3.1.2 IF the LCVUs are operating in "HIGH" speed, perform the following:

CAUTION: Operating the lower containment vent units in "HIGH" speed for more than 24 hours will cause bearing problems in the fans.

NOTE:
- It is preferable to run a vent unit in each fan room (A/D, B/C) in order to maximize air distribution in the lower containment. Due to the temperature characteristics in lower containment and the Digital Rod Position Indication (DRPI) Panels area, use of the 1D LCVU is preferred. Failure to operate with at least one vent unit in each fan room during Modes 1 - 3 can result in high pressurizer and/or steam generator cavity air temperatures.
- Operating the lower containment vent units in "HIGH" speed will fail the bypass chilled water valves open.

3.1.2.1 Start an idle LCVU by placing its control switch in the "HIGH" position:

_____ • "VV LCVU 1A"
_____ • "VV LCVU 1B"
_____ • "VV LCVU 1C"
_____ • "VV LCVU 1D"

_____ 3.1.2.2 Verify the red indicating light illuminates for the LCVU placed in service.

Enclosure 4.13
Shifting Operating Lower Containment Ventilation Units And Pipe Tunnel Booster Fans

OP/1/A/6450/001
Page 3 of 7

_____ 3.1.2.3 Verify the red "OPEN" indicating light illuminates for the LCVU damper associated with the LCVU started.

_____ 3.1.2.4 Verify the red "OPEN" light illuminates for the valve corresponding to the LCVU started (rear of 1MC3):

- ☐ "1RN-473 LWR CONT VENT UNT 1A FULL FLOW"
- ☐ "1RN-455 LWR CONT VENT UNT 1B FULL FLOW"
- ☐ "1RN-447 LWR CONT VENT UNT 1C FULL FLOW"
- ☐ "1RN-481 LWR CONT VENT UNT 1D FULL FLOW"

3.1.2.5 Stop the LCVU to be removed from service by placing its control switch in the "OFF" position:

- _____ "VV LCVU 1A"
- _____ "VV LCVU 1B"
- _____ "VV LCVU 1C"
- _____ "VV LCVU 1D"

NOTE: The procedure may continue up to and including Step 3.1.2.8 before completing the following step.

_____ 3.1.2.6 Verify the green indicating light illuminates for the LCVU stopped.

_____ 3.1.2.7 Verify the green "CLOSED" indicating light illuminates for the LCVU damper associated with the LCVU stopped.

_____ 3.1.2.8 Verify the green "CLOSED" light illuminates for the valve corresponding to the LCVU stopped (rear of 1MC3):

- ☐ "1RN-473 LWR CONT VENT UNT 1A FULL FLOW"
- ☐ "1RN-455 LWR CONT VENT UNT 1B FULL FLOW"
- ☐ "1RN-447 LWR CONT VENT UNT 1C FULL FLOW"
- ☐ "1RN-481 LWR CONT VENT UNT 1D FULL FLOW"

_____ 3.1.2.9 Inform Engineering that lower containment vent units have been placed in "HIGH" speed.

Engineer notified ________________________________

Op Test No.:	301	Scenario #	2	Event #	1	Page	14 of 155
Event Description:	Swap Operating Lower Containment Ventilation Units						

Enclosure 4.13
Shifting Operating Lower Containment Ventilation Units And Pipe Tunnel Booster Fans

OP/1/A/6450/001
Page 4 of 7

_____ 3.2 **IF** three LCVUs are operating **AND** it is desired, shift the operating units as follows:

_____ 3.2.1 **IF** the LCVUs are operating in "LOW" speed, perform the following:

3.2.1.1 Stop the LCVU to be removed from service by placing its control switch in the "OFF" position:

- _____ "VV LCVU 1A"
- _____ "VV LCVU 1B"
- _____ "VV LCVU 1C"
- _____ "VV LCVU 1D"

NOTE:	The procedure may continue up to and including Step 3.2.1.9 before completing the following step.

_____ 3.2.1.2 Verify the green indicating light illuminates for the LCVU stopped.

_____ 3.2.1.3 Verify the red "MAX" indicating light extinguishes for the LCVU stopped.

_____ 3.2.1.4 Verify the green "CLOSED" indicating light illuminates for the LCVU damper associated with the LCVU stopped.

NOTE:	• If Unit 1 is in Mode 1 and LCVU 1C or 1D is the idle unit that is being placed in service, a delay of approximately 15 to 30 minutes may be needed before starting LCVU 1C or 1D to allow lower containment air temperature to trend up. This will prevent exceeding the Tech Spec low limit for air temperature. {PIP 00-0763, PIP 05-3785} • Adequate margin of VQ pressure may be required to allow containment temperature to trend up if waiting 15 to 30 minutes to start LCVU 1C or 1D.

3.2.1.5 Start the idle LCVU by placing its control switch in the "LOW" position:

- _____ "VV LCVU 1A"
- _____ "VV LCVU 1B"
- _____ "VV LCVU 1C"
- _____ "VV LCVU 1D"

_____ 3.2.1.6 Verify the red indicating light illuminates for the LCVU placed in service.

_____ 3.2.1.7 Verify the red "OPEN" indicating light illuminates for the LCVU damper associated with the LCVU started.

Enclosure 4.13

Shifting Operating Lower Containment Ventilation Units And Pipe Tunnel Booster Fans

OP/1/A/6450/001
Page 5 of 7

_____ 3.2.1.8 Verify the red "MAX" indicating light illuminates for the LCVU placed in service.

N/A 3.2.2 IF the LCVUs are operating in "HIGH" speed, perform the following:

3.2.2.1 Stop the lower containment vent unit to be removed from service by placing its control switch in the "OFF" position:

_____ • "VV LCVU 1A"
_____ • "VV LCVU 1B"
_____ • "VV LCVU 1C"
_____ • "VV LCVU 1D"

NOTE: The procedure may continue up to and including Step 3.2.2.8 before completing the following step.

_____ 3.2.2.2 Verify the green indicating light illuminates for the LCVU stopped.

_____ 3.2.2.3 Verify the green "CLOSED" indicating light illuminates for the LCVU damper associated with the LCVU stopped.

_____ 3.2.2.4 Verify the green "CLOSED" light illuminates for the valve corresponding to the LCVU stopped (rear of 1MC3):

☐ "1RN-473 LWR CONT VENT UNT 1A FULL FLOW"
☐ "1RN-455 LWR CONT VENT UNT 1B FULL FLOW"
☐ "1RN-447 LWR CONT VENT UNT 1C FULL FLOW"
☐ "1RN-481 LWR CONT VENT UNT 1D FULL FLOW"

CAUTION: Operating the lower containment vent units in "HIGH" speed for more than 24 hours will cause bearing problems in the fans.

NOTE: Operating the lower containment vent units in "HIGH" speed will fail the bypass chilled water valves open.

3.2.2.5 Start the idle lower containment vent unit by placing its control switch in the "HIGH" position:

_____ • "VV LCVU 1A"
_____ • "VV LCVU 1B"
_____ • "VV LCVU 1C"
_____ • "VV LCVU 1D"

Enclosure 4.13

Shifting Operating Lower Containment Ventilation Units And Pipe Tunnel Booster Fans

OP/1/A/6450/001
Page 6 of 7

_____ 3.2.2.6 Verify the red indicating light illuminates for the LCVU placed in service.

_____ 3.2.2.7 Verify the red "OPEN" indicating light illuminates for the LCVU damper associated with the LCVU started.

_____ 3.2.2.8 Verify the red "OPEN" light illuminates for the valve corresponding to the LCVU started (rear of 1MC3):

- ☐ "1RN-473 LWR CONT VENT UNT 1A FULL FLOW"
- ☐ "1RN-455 LWR CONT VENT UNT 1B FULL FLOW"
- ☐ "1RN-447 LWR CONT VENT UNT 1C FULL FLOW"
- ☐ "1RN-481 LWR CONT VENT UNT 1D FULL FLOW"

_____ 3.2.2.9 Inform Engineering that lower containment vent units have been placed in "HIGH" speed.

Engineer notified ________________________________

N/A 3.3 **IF** shifting the operating pipe tunnel booster fan, perform the following:

3.3.1 Stop the operating fan by placing its control switch in the "OFF" position:

- _____ • "PIPE TUNNEL BSTR FAN 1A"
- _____ • "PIPE TUNNEL BSTR FAN 1B"

_____ 3.3.2 Verify the green indicating light illuminates for the pipe tunnel booster fan stopped.

_____ 3.3.3 **IF** the operating lower containment ventilation units are running in "LOW" speed, start the pipe tunnel booster fan to be placed in service by placing its control switch in the "LOW" speed position:

- _____ • "PIPE TUNNEL BSTR FAN 1A"
- _____ • "PIPE TUNNEL BSTR FAN 1B"

_____ 3.3.4 **IF** the operating lower containment ventilation units are running in "HIGH" speed, start the pipe tunnel booster fan to be placed in service by placing its control switch in the "HIGH" speed position:

- _____ • "PIPE TUNNEL BSTR FAN 1A"
- _____ • "PIPE TUNNEL BSTR FAN 1B"

_____ 3.3.5 Verify that the red indicating light illuminates for the pipe tunnel booster fan placed in service.

Op Test No.:	301	Scenario #	2	Event #	1	Page	17	of	155
Event Description:	Swap Operating Lower Containment Ventilation Units								

Enclosure 4.13

Shifting Operating Lower Containment Ventilation Units And Pipe Tunnel Booster Fans

OP/1/A/6450/001
Page 7 of 7

_____ 3.4 Indicate below the operating Pipe Tunnel Booster Fan:

- ☐ "PIPE TUNNEL BSTR FAN 1A"
- ☐ "PIPE TUNNEL BSTR FAN 1B"

_____ 3.5 Indicate below the operating LCVUs:

- ☐ "VV LCVU 1A"
- ☐ "VV LCVU 1B"
- ☐ "VV LCVU 1C"
- ☐ "VV LCVU 1D"

3.6 File a copy of this enclosure in the designated storage cabinet.

Note to Evaluator:

At the discretion of the Lead Evaluator, the scenario may continue to the next event by instructing the booth operator to INSERT Trigger 2 (1B RN pump trip).

Op Test No.:	301	Scenario #	2	Event #	2	Page	18	of	155
Event Description:	1B RN Pump Trip								

Control Room Indications
1AD-12, A/2 “RN ESSENTIAL HDR A PRESSURE – LO” – LIT
1AD-12, A/5 “RN ESSENTIAL HDR B PRESSURE – LO” – LIT
1AD-11, D/1 “4KV ESS PWR TRAIN B TROUBLE” – LIT
Indicating lights for 1B RN pump – DARK
OAC alarm for RN Header A & B pressure – LOW

Op Test No.:	301	Scenario #	2	Event #	2	Page	19 of 155
Event Description:	1B RN Pump Trip						

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Case I Loss of RN Train	PAGE NO. 6 of 137 Revision 49

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

Critical Task #1

__ 1. **Start idle RN pump(s) as required.**

Note to Evaluator:

Enclosure 1 can be found as Attachment 3 in the back of this document.

__ 2. **Ensure Unit 1 and Unit 2 OATC monitors Enclosure 1 (Foldout Page).**

3. **Verify RN System in normal alignment as follows:**
   - __ Both RN Supply headers - ALIGNED

   AND

   - __ Both RN Discharge headers - ALIGNED.

Perform the following:

__ a. **IF** RN aligned for single supply header operation with "A" train supply header isolated, **THEN GO TO** Case III (Loss of RN With "A" Supply Header Isolated).

__ b. **IF** RN aligned for single supply header operation with "B" train supply header isolated, **THEN GO TO** Case IV (Loss of RN With "B" Supply Header Isolated).

__ c. **IF** RN aligned for single discharge header operation with "A" train discharge header isolated, **THEN GO TO** Case V (Loss of RN With "A" Discharge Header Isolated).

__ d. **IF** RN aligned for single discharge header operation with "B" train discharge header isolated, **THEN GO TO** Case VI (Loss of RN With "B" Discharge Header Isolated).

__ e. **IF** RN aligned for single SNSWP discharge header operation with "A" train discharge header isolated, **THEN GO TO** Case VII (Loss of RN With "A" Pond Return Header Isolated).

__ f. **IF** RN aligned for single SNSWP discharge header operation with "B" train discharge header isolated, **THEN GO TO** Case VIII (Loss of RN With "B" Pond Return Header Isolated).

Op Test No.:	301	Scenario #	2	Event #	2	Page	20 of 155
Event Description:	1B RN Pump Trip						

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Case I Loss of RN Train	PAGE NO. 7 of 137 Revision 49

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 4. **Verify each operating RN pump discharge flow - GREATER THAN 8,600 GPM.**	**Perform the following:** __ a. Stop any RN pump(s) not required to support system operation. b. Ensure the following suction valves to Lake - OPEN: __ • 1RN-1A (RN P/H Pit A Isol From Lake) __ • 1RN-2B (RN P/H Pit A Isol From Lake) __ • 1RN-5A (RN P/H Pit B Isol From Lake) __ • 1RN-6B (RN P/H Pit B Isol From Lake). c. Ensure the following essential header isolation valves for required trains - OPEN: __ • 1RN-67A (RN Hdr 1A Supply Isol) __ • 1RN-69B (RN Hdr 1B Supply Isol) __ • 2RN-67A (RN Hdr 2A Supply Isol) __ • 2RN-69B (RN Hdr 2B Supply Isol). d. Ensure the following RN to RL discharge valves - OPEN: __ • 1RN-57A (Station RN Disch To RL Sys) __ • 1RN-843B (Station RN Disch To RL Sys). (RNO continued on next page)

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Case I Loss of RN Train	PAGE NO. 8 of 137 Revision 49

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
	e. Ensure one of the following RL discharge valves - OPEN: __ • 1RL-54 (RN System Disch To A RL Header) OR __ • 1RL-62 (RN System Disch To B RL Header). f. Ensure the following station RN discharge header crossover valves - OPEN: __ • 1RN-54A (Station RN Disch Hdr X-Over) __ • 1RN-53B (Station RN Disch Hdr X-Over). g. **IF** either of the following conditions met: __ • RN cannot be aligned to Lake **OR** __ • No flow indicated on operating RN pump(s), (RNO continued on next page)

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Case I Loss of RN Train	PAGE NO. 9 of 137 Revision 49

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	**THEN** align RN to SNSWP as follows: __ 1) Align valves for RN swap to SNSWP. **REFER TO** Enclosure 2 (RN Valve Alignment for RN Swap to SNSWP). __ 2) **IF** WL discharge in progress, **THEN** coordinate with Radwaste Chemistry to secure all controlled WL discharges. __ 3) **IF** any RN chemical addition in progress, **THEN** notify Chemistry to secure it. __ 4) **WHEN** corrective action has been taken, **THEN** restore RN to normal alignment. **REFER TO** Enclosure 3 (Returning RN alignment To Normal After Transfer To SNSWP). h. Verify the following alarms - DARK: __ • 1AD-12, C/2 "RN PMP A STRAINER HI D/P" __ • 1AD-12, C/5 "RN PMP B STRAINER HI D/P" __ • 2AD-12, C/2 "RN PMP A STRAINER HI D/P" __ • 2AD-12, C/5 "RN PMP B STRAINER HI D/P". __ i. **IF** any of the previous alarms lit, **THEN** backflush affected strainer. **REFER TO** OP/0/A/6400/006C (Nuclear Service Water System).

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Case I Loss of RN Train	PAGE NO. 10 of 137 Revision 49

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 5. **Verify each operating RN pump discharge flow - LESS THAN 23,000 GPM.**	**Perform the following:**

CAUTION **The following steps may result in loss of an essential header until an opposite train pump is started.**

a. Ensure the following RN isolation valves - CLOSED:

- __ 1RN-47A (RN Supply X-Over Isol)
- __ 1RN-48B (RN Supply X-Over Isol)
- __ 2RN-47A (RN Supply X-Over Isol)
- __ 2RN-48B (RN Supply X-Over Isol).
- __ 1RN-51A (Non-Ess Ret Hdr Isol)
- __ 1RN-52B (Non-Ess Ret Hdr Isol)
- __ 2RN-51A (Non-Ess Ret Hdr Isol)
- __ 2RN-52B (Non-Ess Ret Hdr Isol).

__ b. Ensure 1RN-58B (RN Hdr B Ret To SNSWP) - OPEN.

c. **WHEN** 1RN-58B open, **THEN** CLOSE the following valves:

- __ 1RN-54A (Station RN Disch Hdr X-Over)
- __ 1RN-53B (Station RN Disch Hdr X-Over).

__ d. **IF** flow returning to normal, **THEN GO TO** Step 6.

(RNO continued on next page)

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Case I Loss of RN Train	PAGE NO. 11 of 137 Revision 49

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. (Continued)	e. **IF** flow still excessive, **THEN** perform the following: __ 1) Ensure both RN pump(s) on affected train - OFF. __ 2) Dispatch operators to locate any piping leaks. **REFER TO** AP/0/A/5500/030 (Plant Flooding).
__ 6. **Ensure RN pumps - IN OPERATION AS NEEDED.**	
7. **Ensure proper alignment of RN to KC Hx(s) as follows:**	
__ a. Verify RN - ALIGNED TO IN SERVICE KC HX(s).	a. Shift KC train in service as needed. **REFER TO** the following procedures: __ • OP/1/A/6400/005 (Component Cooling Water System) __ • OP/2/A/6400/005 (Component Cooling Water System).
__ b. Ensure KC Hx Otlt Mode switches - PROPERLY ALIGNED.	
__ 8. **Verify each operating RN pump discharge flow - GREATER THAN 8,600 GPM.**	**Perform the following:** __ a. Do not exceed 4650 GPM through an NS Hx. __ b. Align RN flow through NS Hx(s) as needed to raise each operating RN pump's discharge flow to greater than 8,600 GPM. **REFER TO** OP/0/A/6400/006C (Nuclear Service Water System).

Op Test No.:	301	Scenario #	2	Event #	2	Page	25 of 155
Event Description:	1B RN Pump Trip						

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Case I Loss of RN Train	PAGE NO. 12 of 137 Revision 49

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 9. **Verify RN - AVAILABLE TO ALL UNIT 1 AND UNIT 2 D/G(s).**	**Dispatch operator to remove any D/G(s) without cooling water supply from standby readiness. <u>REFER TO</u> the following procedures:** __ • OP/1/A/6350/002 (Diesel Generator Operation) __ • OP/2/A/6350/002 (Diesel Generator Operation).
10. **Determine VC/YC status as follows:** __ • Verify VC/YC - ALIGNED TO OPERATING RN TRAIN __ • Verify YC Chiller - RUNNING.	**Perform the following:** __ • **<u>IF</u>** YC Chiller tripped due to loss of RN flow and RN flow restored, **<u>THEN</u>** dispatch operator to reset YC Chiller trip. __ • Align VC/YC to operating RN train as required. **<u>REFER TO</u>** OP/0/A/6450/011 (Control Room Area Ventilation/Chilled Water System).
__ 11. **Determine and correct cause of loss of RN train.**	

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Case I Loss of RN Train	PAGE NO. 13 of 137 Revision 49

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

12. **Ensure compliance with appropriate Tech Specs and Selected Licensee Commitments Manual:**

- __ SLC 16.7-6 (RN Discharge Instrumentation)
- __ 3.6.5 (Containment Air Temperature)
- __ 3.6.6 (Containment Spray System)
- __ 3.6.17 (Containment Valve Injection Water System (CVIWS))
- __ 3.7.5 (Auxiliary Feedwater (AFW) System)
- __ 3.7.7 (Component Cooling Water (CCW) System)
- __ 3.7.8 (Nuclear Service Water System (NSWS))
- __ 3.7.10 (Control Room Area Ventilation System (CRAVS))
- __ 3.7.11 (Control Room Area Chilled Water System (CRACWS))
- __ 3.8.1 (A.C. Sources - Operating)
- __ 3.8.2 (A.C. Sources - Shutdown).

TECH SPEC EVALUATION

See Attachment 11 for applicable Tech Specs.

Both Units are in:

T.S. 3.7.8

Condition A: Restore Nuclear Service Water System Train to OPERABLE in 72 hours.

Unit 1 is temporarily in T.S. 3.7.11 Condition A (Restore CRACWS train to OPERABLE in 30 days) until the B YC Chiller is restarted.

13. **Determine required notifications:**

- __ **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO)
- __ **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation).

Note to Evaluator:

At the discretion of the Lead Evaluator, the scenario may continue to the next event by instructing the booth operator to INSERT Trigger 3 (Main Generator Voltage Regulator Failure).

__ 14. **Notify Environmental Chemistry of any RN pump shifts.**

__ 15. **Determine long term plant status. RETURN TO procedure in effect.**

END

Op Test No.:	301	Scenario #	2	Event #	3	Page	27 of 155

Event Description: Main Generator Voltage Regulator Failure

Control Room Indications
OAC alarm for Generator Voltage – LOW
1AD-1, D/6 "EHC SYSTEM FAULT" – LIT

Op Test No.: 301	Scenario # 2	Event # 3	Page 28 of 155				
Event Description:	Main Generator Voltage Regulator Failure						

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 4 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. **Operator Actions**	
__ 1. **Verify Generator - TIED TO GRID.**	**Perform the following:** a. **IF** notified by DEC TOP (Transmission Operations) degraded switchyard (grid) voltage conditions exist, **THEN** ensure compliance with the following Tech Specs: __ • 3.8.1 (AC Sources - Operating) __ • 3.8.2 (AC Sources - Shutdown). __ b. **RETURN TO** procedure and step in effect.
2. **Verify Generator MVARs - EXCEED GENERATOR CAPABILITY CURVE limits. REFER TO one of the following:** __ • GENCAP (OAC Graphic Display) __ • Enclosure 1 (Unit 1 Generator Capability Curves).	**Perform the following:** __ a. **IF AT ANY TIME** GENERATOR CAPABILITY CURVE limits exceeded, **THEN GO TO** Step 3. __ b. Observe Note prior to Step 5 and **GO TO** Step 5.

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 5 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **Perform the following to adjust Generator MVARs:**	
a. Adjust Generator MVARs to operate within Generator Capability Curve limits as follows: __ • Depress "LOWER" on VOLTAGE ADJUST to reduce lagging MVARs **OR** __ • Depress "RAISE" on VOLTAGE ADJUST to reduce leading MVARs.	a. Perform the following: 1) **IF** voltage regulator in "AUTO", **THEN** perform the following: __ a) Place voltage regulator in "MANUAL". __ b) Adjust MVARs to within GENERATOR CAPABILITY CURVE limits. 2) **IF** unable to maintain MVARs within Generator Capability Curve limits, **THEN** remove generator from service as follows: a) **IF** reactor power greater than or equal to 69%, **THEN** perform the following: __ (1) Trip reactor. __ (2) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). b) **IF** reactor power less than 69%, **THEN** perform the following: __ (1) Trip turbine. __ (2) **GO TO** AP/1/A/5500/002 (Turbine Generator Trip).
__ b. **IF** Generator voltage operated outside Generator Voltage Schedule for more than 30 continuous minutes, **THEN** notify DEC TOP (Transmission Operations)/DEC BA (Balancing Authority).	

Event Description: Main Generator Voltage Regulator Failure

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 6 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. (Continued) c. **IF** voltage regulator in "MANUAL", **THEN** perform the following: __ 1) Notify DEC TOP (Transmission Operations)/DEC BA (Balancing Authority) voltage regulator in manual. __ 2) **WHEN** voltage regulator returned to auto, **THEN** notify DEC TOP (Transmission Operations)/DEC BA (Balancing Authority).	
__ 4. **Notify Engineering to evaluate generator abnormal operating conditions.**	

NOTE The DEC TOP "Real Time Contingency Analysis" (RTCA) program determines if the Unit will have adequate switchyard voltage available for ECCS loads following a Unit Trip with Safety Injection actuation.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 5. **Verify DEC TOP (Transmission Operations) reported "Real Time Contingency Analysis" (RTCA) indicates CNS switchyard voltage would NOT be adequate if the unit should trip.**	**Perform the following:** __ a. **IF** DEC TOP (Transmission Operations) has not reported "RTCA" indications, **THEN** notify DEC TOP to monitor "RTCA" and report results to Control Room Supervisor. __ b. **IF AT ANY TIME** DEC TOP (Transmission Operations) reports "RTCA" indicates CNS switchyard voltage would **NOT** be adequate if the unit should trip, **THEN GO TO** Step 6. __ c. Observe Note prior to Step 17 and **GO TO** Step 17.

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 7 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

__ 6. **Record time DEC TOP (Transmission Operations) "RTCA" indicated CNS switchyard (grid) voltage NOT adequate:**

____

__ 7. **Start 2 hour timer from time DEC TOP (Transmission Operations) "RTCA" indicated CNS switchyard (grid) voltage NOT adequate.**

NOTE When DEC TOP "RTCA" indicates switchyard voltage would **NOT** be adequate if the unit trips, the following conditions exist:

- Both trains of offsite (normal) power are inoperable.
- Both ECCS trains are in an unanalyzed condition and inoperable.

8. **Perform Tech Spec assessment as follows:**

   a. Ensure compliance with the following Tech Specs due to both trains of the following systems inoperable:

   __ • 3.8.1 (AC Sources - Operating)

   __ • 3.8.2 (AC Sources - Shutdown)

   __ • 3.5.2 (ECCS - Operating).

   __ b. Ensure compliance with Tech Spec 3.0.3 due to both trains of ECCS inoperable.

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 8 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE The DEC TOP (Transmission Operations)/DEC BA (Balancing Authority) response to a degraded grid voltage condition may restore voltage to adequate status within one hour.

9. **Activate TSC as follows:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ a. Verify DEC TOP (Transmission Operations) expects switchyard (grid) voltage restoration to adequate status in less than one hour.	a. Perform the following: __ 1) Activate TSC. **<u>REFER TO</u>** RP/0/B/5000/027 (Augmentation of Shift Utilizing the Emergency Response Organization without Emergency Declaration) __ 2) Observe Note prior to Step 10 and **<u>GO TO</u>** Step 10.
__ b. Evaluate activating TSC. **<u>REFER TO</u>** RP/0/B/5000/027 (Augmentation of Shift Utilizing the Emergency Response Organization without Emergency Declaration)	

NOTE The degraded switchyard (grid) voltage condition places ECCS in an unanalyzed condition reportable per 10CFR50.72(b)(3)(ii).

10. **Determine required notifications:**
    - __ **<u>REFER TO</u>** AD-EP-ALL-0111 (Control Room Activation of the ERO)
    - __ **<u>REFER TO</u>** AD-LS-ALL-0006 (Notification/Reportability Evaluation).

11. **Evaluate the following:**
    - __ Stopping in progress surveillance testing.
    - __ Stopping in progress maintenance activities.
    - __ Returning systems to normal/functional status.

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 9 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

12. **IF AT ANY TIME DEC TOP (Transmission Operations) requests actions to restore grid reliability, THEN perform the following:**

CAUTION **The actions implemented to restore grid reliability should not exceed any regulatory or equipment operating limits.**

__ a. Notify Shift Manager to evaluate DEC TOP (Transmission Operations) requested actions to restore grid reliability.

__ b. Notify DEC TOP (Transmission Operations) of actions that cannot be performed.

__ c. Perform Shift Manager approved actions to restore grid reliability.

__ d. Document all DEC TOP (Transmission Operations) requested action(s) and resolution in logbook.

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 10 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE The DEC BA (Balancing Authority)/DEC TOP (Transmission Operations) is allowed 2 hours to restore switchyard (grid) voltage to adequate. The 2 hour time limit may be adjusted based on Shift Manager assessment of plant or grid conditions.

13. **Do not continue in this procedure until one of the following conditions exist:**

__ • 2 hours elapsed since DEC TOP (Transmission Operations) "RTCA" indicated switchyard voltage **NOT** adequate

OR

__ • Notification from DEC TOP (Transmission Operations) that "RTCA" indicates switchyard voltage would be adequate should the unit trip

OR

__ • Shift Manager adjust 2 hour time limit based on assessment of plant conditions.

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 11 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 14. **Verify DEC TOP (Transmission Operations) reported that "RTCA" indicates switchyard voltage would be adequate should the unit trip.**	**Perform the following:** **NOTE** • The jumpers bypass the 4160 degraded voltage 5 second time delay relay. • If an S/I occurs with the jumpers installed, the 4160V incoming breakers trip open when LOCA Sequencer actuates. The D/Gs will supply power. __ a. Notify SPOC to install jumpers per AM/1/A/5100/008 (4Kv Essential Power (EPC) System Degraded Voltage Logic. **NOTE** When jumpers are installed offsite power and shared systems remain inoperable until "RTCA" indicates switchyard voltage is adequate should the unit trip. __ b. **WHEN** jumpers installed per AM/1/A/5100/008 (4Kv Essential Power (EPC) System Degraded Voltage Logic, **THEN** Unit 1 exits L.C.O. 3.0.3 due to both trains of ECCS inoperable Tech Spec 3.5.2.
__ 15. **Do not continue in this procedure until DEC TOP (Transmission Operations) reports "RTCA" indicates switchyard voltage would be adequate should the unit trip.**	

Op Test No.:	301	Scenario #	2	Event #	3	Page	36 of 155

Event Description: Main Generator Voltage Regulator Failure

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 12 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

16. **__WHEN__ DEC TOP (Transmission Operations) reports "RTCA" indicates switchyard voltage returned to adequate voltage should the unit trip, __THEN__ perform the following:**

__ a. **__IF__** jumpers installed, **__THEN__** notify SPOC to remove jumpers per, AM/1/A/5100/008 (4Kv Essential Power (EPC) System Degraded Voltage Logic.

b. Evaluate exiting the following Tech Spec LCO actions:

__ • 3.8.1 (AC Sources - Operating)

__ • 3.8.2 (AC Sources - Shutdown)

__ • 3.5.2 (ECCS - Operating)

__ • Tech Spec LCO 3.0.3.

__ c. Notify Regulatory Compliance of Tech Spec LCO action status.

__NOTE__ Do not exceed any generator limits when adjusting generator voltage.

17. **Coordinate with DEC BA (Balancing Authority) and perform generator voltage adjustments as follows:**

__ • Adjust Unit 1 Generator Bus Voltage per, Unit 1 Revised Data Book Figure 23 (CNS Generator Voltage Operating Schedule)

OR

__ • Adjust Unit 1 Generator Bus Voltage per DEC BA (Balancing Authority) request.

CNS AP/1/A/5500/037	GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES Case I Abnormal Generator or Grid Voltage	PAGE NO. 13 of 21 Revision 08

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 18. **WHEN DEC TOP (Transmission Operations) or DEC BA (Balancing Authority) verifies switchyard (grid) voltage adequate and reliable, THEN RETURN TO procedure step in effect.**	

END

Note to Evaluator:

At the discretion of the Lead Evaluator, the scenario may continue to the next event by instructing the booth operator to INSERT Trigger 4 (1C S/G Tube Leak).

Event Description: 1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto

Control Room Indications
1RAD-1, B/1 "1EMF-33 CSAE EXHAUST HI RAD" – LIT
1RAD-1, B/5 "1EMF-72 S/G B LEAKAGE HI RAD" – LIT
1RAD-1, C/1 "1EMF-73 S/G C LEAKAGE HI RAD" - LIT
Count rate on 1EMF-28 – RISING
Pressurizer Level – LOWERING

Op Test No.:	301	Scenario #	2	Event #	4,5,6	Page	39	of 155

Event Description: 1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 3 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

Note to Evaluator:

Enclosure 1 can be found as Attachment 4 in the back of this document.

__ 1. **Monitor Enclosure 1 (Case I Steam Generator Tube Leak Foldout Page).**

__ 2. **Verify Pzr level - STABLE OR TRENDING UP.**

Perform the following:

__ a. Maintain charging flow less than 180 GPM.

__ b. THROTTLE 1NV-294 (NV Pmps A&B Disch Flow Ctrl) to stabilize Pzr level.

__ c. **IF** Pzr level stable **OR** trending up, **THEN GO TO** Step 3.

d. **IF** Pzr level continues to trend down, **THEN** reduce or isolate letdown as follows:

1) **IF** desired to reduce letdown flow, **THEN** perform the following:

a) **IF** 1NV-10A (Letdn Orif 1B Otlt Cont Isol) open, **THEN** perform the following:

__ (1) Control 1NV-148 (Letdn Press Control) to establish letdown pressure between 375 - 400 PSIG.

__ (2) THROTTLE 1NV-849 (Letdn Flow Var Orif Ctrl) for 45 GPM letdown flow.

__ (3) **WHEN** 45 GPM letdown flow established, **THEN** adjust 1NV-148 (Letdn Press Control) to maintain letdown pressure at 350 PSIG.

__ (4) **WHEN** letdown pressure stable at 350 PSIG, **THEN** place 1NV-148 (Letdn Press Control) in auto.

(RNO continued on next page)

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 4 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. (Continued)

N/A b) **IF** 1NV-13A (Letdn Orif 1A Otlt Cont Isol) open, **THEN** perform the following:

__ (1) Control 1NV-148 (Letdn Press Control) to establish letdown pressure between 150 - 200 PSIG.

__ (2) OPEN 1NV-11A (Letdn Orif 1C Otlt Cont Isol).

__ (3) Adjust 1NV-148 (Letdn Press Control) to establish letdown pressure between 375 - 400 PSIG.

__ (4) CLOSE 1NV-13A (Letdn Orif 1A Otlt Cont Isol).

__ (5) Adjust 1NV-148 (Letdn Press Control) to maintain letdown pressure at 350 PSIG.

__ (6) **WHEN** letdown pressure stable at 350 PSIG, **THEN** place 1NV-148 (Letdn Press Control) in auto.

N/A 2) **IF** letdown isolation required, **THEN** ensure the following valves - CLOSED:

__ • 1NV-10A (Letdn Orif 1B Otlt Cont Isol)

__ • 1NV-11A (Letdn Orif 1C Otlt Cont Isol)

__ • 1NV-13A (Letdn Orif 1A Otlt Cont Isol).

__ 3) **IF** Pzr level stable **OR** trending up, **THEN GO TO** Step 3.

(RNO continued on next page)

Op Test No.:	301	Scenario #	2	Event #	4,5,6	Page	41 of 155

Event Description: 1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 5 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	4) **<u>IF</u>** Pzr level continues to trend down **<u>OR</u>** Pzr level cannot be maintained greater than 4%, **<u>THEN</u>** perform the following: __ a) Trip Unit 1 reactor. __ b) **<u>WHEN</u>** reactor tripped verified, **<u>THEN</u>** initiate S/I. __ c) **<u>GO TO</u>** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ 3. **<u>IF AT ANY TIME</u> Pzr level trends down in uncontrolled manner or cannot be maintained greater than 4%, <u>THEN</u> perform Step 2.**	

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 6 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE In subsequent steps the term "affected S/G" is a S/G with primary to secondary leakage.

4. **Identify affected S/G(s) as follows:**

__ • Notify RP to frisk all cation columns

OR

__ • Any S/G N/R level - TRENDING UP IN AN UNCONTROLLED MANNER

OR

NOTE S/G Leakage EMFs are highly sensitive which may cause EMFs located on adjacent steamline to be trending up or in alarm.

• Verify any of the following S/G leakage EMF indication(s) - TRENDING UP OR IN ALARM:

__ • 1EMF-71 (S/G A Leakage)
__ • 1EMF-72 (S/G B Leakage)
__ • 1EMF-73 (S/G C Leakage)
__ • 1EMF-74 (S/G D Leakage).

OR

• Verify any of the following S/G steamline EMF indication(s) - TRENDING UP OR IN ALARM:

__ • 1EMF-26 (Steamline 1A)
__ • 1EMF-27 (Steamline 1B)
__ • 1EMF-28 (Steamline 1C)
__ • 1EMF-29 (Steamline 1D).

OR

__ • Verify CF flow - LOWER TO ANY S/G AS COMPARED TO OTHERS

OR

__ • Notify Secondary Chemistry to determine affected S/G by sampling.

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 7 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE Consideration should be given to evacuating non-essential personnel from the Unit 1 Turbine Building and affected Doghouse based on severity of the leak.	
__ 5. **Announce event on plant page.**	
6. **Determine required notifications:** __ • **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO) __ • **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation).	
NOTE If reactor trip required, the E-0 foldout page item "Ruptured S/G CA Isolation Criteria" does not apply unless S/I is initiated.	
7. **Verify VCT level able to be maintained by normal makeup as follows:**	
a. One of the following conditions exists: __ • S/G tube leak less than 90 gpm **OR** __ • Automatic makeup stabilizes or raises VCT level **OR** __ • Manual makeup stabilizes or raises VCT level.	a. Perform the following: 1) Align NV pumps to FWST as follows: __ a) OPEN 1NV-252A (NV Pumps Suct From FWST). __ b) OPEN 1NV-253B (NV Pumps Suct From FWST). __ c) CLOSE 1NV-188A (VCT Otlt Isol). __ d) CLOSE 1NV-189B (VCT Otlt Isol). (RNO continued on next page)

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 8 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	2) **IF** reactor trip breakers closed, **THEN** perform the following: __ a) Continue concurrent use of this procedure for S/G tube leak at Step 8. __ b) Trip Unit 1 reactor. __ c) **IF** Unit 1 was in Mode 3 below 1955 PSIG, **THEN GO TO** AP/1/A/5500/005 (Reactor Trip or Inadvertent S/I Below P-11). __ d) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). __ 3) **IF** reactor trip breakers open, **THEN GO TO** Step 8.
b. **IF AT ANY TIME** the following conditions exist: __ • 1AD-7, I/1 "VCT LO LVL" alarm - LIT **AND** • Reactor trip breakers closed, __ **THEN** perform Step 7.a RNO.	
__ 8. **IF AT ANY TIME S/G tube leak greater than 25 GPM, THEN notify Emergency Coordinator.**	

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 9 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- OAC calculated leak rate may be invalid during mode changes and/or transient conditions unless there is current, coordinated sampling data available.
- Tritium sampling may be the most effective method for determining leak rate for small tube leaks when in Mode 2 or 3.
- OAC Primary to Secondary Leakage points may be accessed by turn on code "GD EROSLEAK".

9. **Determine S/G leak rate by any of the following methods:**
   - Monitor the following computer points:
     - __ EROSLEAK (Primary To Secondary Leakage)
     - __ C1P0187 (Estimated Total Pri To Sec Leakrate)
     - __ C1P0189 (Pri To Sec Leakrate 15 Min Running Avg).

   OR

   NOTE
   - 36,000 GPD is equivalent to 25 GPM.
   - S/G Leakage EMFs are highly sensitive which may cause EMFs located on the adjacent steamline to be trending up or in alarm.

   - S/G leakage EMF indication(s):
     - __ 1EMF-71 (S/G A Leakage)
     - __ 1EMF-72 (S/G B Leakage)
     - __ 1EMF-73 (S/G C Leakage)
     - __ 1EMF-74 (S/G D Leakage).

   OR

   - __ Compare charging flow and letdown flow

   OR

   - __ Monitor OAC NV Graphic

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 10 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9. (Continued)

OR

__ • Initiate OAC Program "RCSLEAK"

OR

__ • Monitor OAC point C1P0976 (U1 Gross NC System Leak Rate, Ten Min Run Avg)

OR

__ • Secondary Chemistry performance of PT/1/B/4600/028 (Determination of Steam Generator Tube Leak For Unit 1).

10. **Minimize Secondary contamination as follows:**

a. Remove CM polishing demineralizers from service as follows:

__ 1) Ensure "POLSH DEMIN BYP CTRL" - IN MANUAL.

__ 2) Ensure "POLSH DEMIN BYP CTRL" - OPEN.

__ 3) Notify Secondary Chemistry CM polishing demineralizers bypassed.

__ b. Align auxiliary systems to minimize secondary side contamination. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 2 (Minimizing Secondary Side Contamination).

__ c. Stop any transfer of water between both Unit's CSTs.

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 11 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **Ensure compliance with appropriate Tech Specs and Selected Licensee Commitments Manual:** __ • 3.4.13 (RCS Operational Leakage) __ • 3.4.14 (RCS Pressure Isolation Valve (PIV) Leakage) __ • 3.4.18 (Steam Generator (SG) Tube Integrity __ • 3.5.5 (Seal Injection Flow) __ • 3.7.17 (Secondary Specific Activity) __ • SLC 16.7-9 (Standby Shutdown System).	**TECH SPEC EVALUATION** ***See Attachment 11 for applicable Tech Specs.*** **T.S. 3.4.13 & 3.4.18** **Condition B: Be in Mode 3 in 6 hours AND be in Mode 5 in 36 hours.** **SLC 16.7-9** **Condition B: Declare the standby makeup pump non-functional and enter Condition A Immediately.**
__ 12. **Verify Unit 1 in Mode 1.**	**Perform the following:** a. IF any of the following exist: __ • Leak rate greater than or equal to 75 gpd in any one S/G __ • Leak rate trending up and approaching 75 gpd in any one S/G, __ THEN stop any startup activities in progress. __ b. Notify Secondary Chemistry to validate leakage by performing PT/1/B/4600/028 (Determination of Steam Generator Tube Leak For Unit 1). __ c. IF Unit 1 in Mode 2, THEN GO TO Step 13. __ d. GO TO Step 19.

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 12 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- The following leak rates are based on leakage in one S/G.
- In the event of an oscillating leak, leak rate shall be determined based on peak value of the spike.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 13. **Verify leak rate - GREATER THAN OR EQUAL TO 5 GPD.**	**Perform the following:** __ a. Notify Secondary Chemistry to perform PT/1/B/4600/028 (Determination of Steam Generator Tube Leak For Unit 1). __ b. **RETURN TO** procedure and step in effect.
__ 14. **Verify leak rate - GREATER THAN OR EQUAL TO 30 GPD.**	**Perform the following:** a. Initiate increased monitoring as follows: 1) Notify RP of the following: __ • Current value of leakage __ • Perform HP/0/B/1009/003 (Radiation Protection Response Following A Primary To Secondary Leak) • Reset setpoints of the following EMFs per HP/0/B/1000/010 (Determination of Radiation Monitor Setpoints): __ • 1EMF-33 (Condenser Air Ejector Exhaust) __ • 1EMF-71 (S/G A Leakage) __ • 1EMF-72 (S/G B Leakage) __ • 1EMF-73 (S/G C Leakage) __ • 1EMF-74 (S/G D Leakage). (RNO continued on next page)

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 13 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. (Continued)	2) Notify Secondary Chemistry of the following: __ • Current value of leakage __ • Perform PT/1/B/4600/028 (Determination of Steam Generator Tube Leak For Unit 1) immediately **AND** increase frequency to daily. 3) **IF AT ANY TIME** the following conditions met: __ • Any main steam line N-16 radiation monitor - INOPERABLE **AND** __ • C1P0187 (Estimated Total Pri To Sec Leakrate) - INVALID, __ **THEN** notify Secondary Chemistry determine frequency to perform PT/1/B/4600/028 (Determination of Steam Generator Tube Leak For Unit 1). 4) Monitor the following EMFs: __ • 1EMF-33 (Condenser Air Ejector Exhaust) __ • 1EMF-71 (S/G A Leakage) __ • 1EMF-72 (S/G B Leakage) __ • 1EMF-73 (S/G C Leakage) __ • 1EMF-74 (S/G D Leakage). (RNO continued on next page)

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 14 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. (Continued)	5) **REFER TO** the following Tech Specs: __ • 3.4.13 (RCS Operational Leakage) __ • 3.7.17 (Secondary Specific Activity). __ 6) Notify Station Management of trends. __ b. **RETURN TO** procedure and step in effect while continuing to monitor leakage for requirements of this procedure.
15. **Verify at least one of the following:** __ • Leak rate greater than or equal to 100 gpd OR __ • Leak rate greater than or equal to 75 gpd has been sustained for at least 1 hour OR • All of the following: __ • Leak rate greater than or equal to 75 gpd __ • C1P0187 (Estimated Total Pri To Sec Leakrate) - INVALID __ • Any main steam line N-16 radiation monitor - INOPERABLE.	**Perform the following:** a. Initiate increased monitoring as follows: 1) Notify RP of the following: __ a) Current value of leakage. __ b) Perform HP/0/B/1009/003 (Radiation Protection Response Following A Primary To Secondary Leak). c) **IF** 1RAD-1, B/1 "1EMF 33 CSAE EXHAUST HI RAD - LIT, **THEN** ensure blowdown flow controllers set to 0 gpm: __ • S/G A BLDWN FLOW CTRL __ • S/G B BLDWN FLOW CTRL __ • S/G C BLDWN FLOW CTRL __ • S/G D BLDWN FLOW CTRL. (RNO continued on next page)

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 15 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

15. (Continued)

d) Reset setpoints of the following EMFs per HP/0/B/1000/010 (Determination of Radiation Monitor Setpoints):

__ • 1EMF-33 (Condenser Air Ejector Exhaust)
__ • 1EMF-71 (S/G A Leakage)
__ • 1EMF-72 (S/G B Leakage)
__ • 1EMF-73 (S/G C Leakage)
__ • 1EMF-74 (S/G D Leakage).

2) Notify Secondary Chemistry of the following:

__ • Current value of leakage

__ • Perform PT/1/B/4600/028 (Determination of Steam Generator Tube Leak For Unit 1) immediately **AND** increase frequency to daily.

3) **IF AT ANY TIME** the following conditions met:

__ • Any main steam line N-16 radiation monitor - INOPERABLE

AND

__ • C1P0187 (Estimated Total Pri To Sec Leakrate) - INVALID,

__ **THEN** notify Secondary Chemistry to determine frequency to perform PT/1/B/4600/028 (Determination of Steam Generator Tube Leak For Unit 1).

(RNO continued on next page)

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 16 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15. (Continued)	

NOTE EMFs should be monitored every 15 minutes until leak rates have stabilized.

4) Monitor the following EMFs:

- __ 1EMF-33 (Condenser Air Ejector Exhaust)
- __ 1EMF-71 (S/G A Leakage)
- __ 1EMF-72 (S/G B Leakage)
- __ 1EMF-73 (S/G C Leakage)
- __ 1EMF-74 (S/G D Leakage).

__ 5) Initiate review of applicable procedures to be utilized by Operations, Chemistry and RP in the event leak rate goes up.

__ 6) Evaluate secondary contamination potential. Review EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 3 (Turbine Building Sump Isolation).

7) **REFER TO** the following Tech Specs:

- __ 3.4.13 (RCS Operational Leakage)
- __ 3.7.17 (Secondary Specific Activity).

__ 8) Notify Station Management of trends.

__ b. **RETURN TO** procedure and step in effect while continuing to monitor leakage for the requirements of this procedure.

Event Description: 1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 17 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16. **Perform the following:**	
NOTE Leakage indications are validated by EMFs trending in the same direction. Precise duplication of leakage values is not required.	
a. Verify at least one of the following EMFs validate leakage indication: __ • 1EMF-71 (S/G A Leakage) __ • 1EMF-72 (S/G B Leakage) __ • 1EMF-73 (S/G C Leakage) __ • 1EMF-74 (S/G D Leakage).	__ a. Notify Secondary Chemistry to perform PT/1/B/4600/028 (Determination of Steam Generator Tube Leak For Unit 1).
NOTE Unit shutdown should not be postponed while waiting for chemistry calculation results.	
b. **IF AT ANY TIME** chemistry calculations contradict leakage indications, **THEN** perform the following: __ • Unit 1 shutdown may be suspended __ • **RETURN TO** Step 11.	
NOTE EMFs should be monitored every 15 minutes.	
c. Monitor the following EMFs: __ • 1EMF-33 (Condenser Air Ejector Exhaust) __ • 1EMF-71 (S/G A Leakage) __ • 1EMF-72 (S/G B Leakage) __ • 1EMF-73 (S/G C Leakage) __ • 1EMF-74 (S/G D Leakage).	

Op Test No.: 301	Scenario # 2	Event # 4,5,6	Page 54	of 155			
Event Description:	1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto						

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 18 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

16. (Continued)

d. Notify RP of the following:

__ 1) Current value of leakage.

__ 2) Perform HP/0/B/1009/003 (Radiation Protection Response Following A Primary To Secondary Leak).

__ e. Evaluate secondary contamination potential. Review EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 3 (Turbine Building Sump Isolation).

Event Description: 1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 19 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

17. **Determine Unit 1 shutdown requirements as follows:**

a. **IF AT ANY TIME** leak rate greater than or equal to 100 gpd, **THEN** perform the following:

__ 1) Ensure reactor power less than 50% within 1 hr.

__ 2) Ensure Unit 1 in Mode 3 within the following 2 hrs.

__ 3) **IF** leak rate greater than or equal to 150 gpd, **THEN** ensure Unit 1 in Mode 5 within 36 hours.

__ 4) Observe Note prior to Step 18 and **GO TO** Step 18.

b. **IF** leak rate greater than or equal to 75 gpd and less than 100 gpd, **THEN** perform the following:

- **IF AT ANY TIME** the following conditions met:

  __ • Any main steam line N-16 radiation monitor - INOPERABLE

  AND

  __ • C1P0187 (Estimated Total Pri To Sec Leakrate) - INVALID,

  THEN perform the following:

  __ 1) Ensure reactor power less than 50% within 1 hr.

  __ 2) Ensure Unit 1 in Mode 3 within the following 2 hrs.

  __ 3) Observe Note prior to Step 18 and **GO TO** Step 18.

__ c. **IF** leak rate greater than or equal to 75 gpd and less than 100 gpd sustained for one hour, **THEN** ensure Unit 1 in Mode 3 within 24 hrs.

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Case I Steam Generator Tube Leak	PAGE NO. 20 of 164 Revision 62

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE EMF indications may trend down during unit shutdown. Unit shutdown should not be suspended based solely on lower radiation monitor indications.

18. **Shutdown Unit 1 as follows:**

__ a. Notify Engineering of occurrence.

__ b. Verify reactor power - GREATER THAN 15%.

b. Perform the following:

__ 1) Initiate Unit 1 shutdown. **REFER TO** OP/1/A/6100/002 (Controlling Procedure For Unit Shutdown).

__ 2) Ensure adequate shutdown margin maintained. **REFER TO** Unit 1 ROD Book, Section 5.11.

__ 3) **GO TO** Step 19.

NOTE A more rapid shutdown is prudent for larger leaks to minimize secondary contamination or offsite dose.

c. Initiate Unit 1 shutdown. **REFER TO** the following procedures:

__ • OP/1/A/6100/003 (Controlling Procedure For Unit Operation)

OR

__ • AP/1/A/5500/009 (Rapid Downpower).

__ d. Ensure adequate shutdown margin maintained. **REFER TO** Unit 1 ROD Book, Section 5.11.

__ 19. **IF AT ANY TIME 1RAD-1, B/1 "1EMF-33 CSAE EXHAUST HI RAD" - LIT, THEN ensure "UNIT 1 CSAE EXH" aligned as required by annunciator response.**

Op Test No.:	301	Scenario #	2	Event #	4,5,6	Page	57 of 155
Event Description:	1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto						

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 2 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

Note to Evaluator:

Enclosure 1 can be found as Attachment 5 in the back of this document.

__ 1. **Monitor Enclosure 1 (Foldout Page).**

2. **Determine required notifications:**

- __ **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO)
- __ **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation).

__ 3. **IF AT ANY TIME prompt separation from grid required, THEN GO TO Step 26.**

N/A 4. **IF load reduction due to grid instability, THEN perform the following:**

NOTE In manual mode, the control valves are capable of full travel within 3 minutes.

__ a. Select "MANUAL" and "CONTROL VALVE LOWER" to reduce turbine load as required.

__ b. **GO TO** Step 12.

__ 5. **Verify Turbine Control - IN AUTO.**

RESPONSE NOT OBTAINED:

Perform the following:

__ a. **IF** manual turbine control desired, **THEN GO TO** Step 9.

b. **IF** auto turbine control desired, **THEN** perform the following:

__ 1) Verify "AUTO" - FLASHING.

__ 2) Depress "AUTO" pushbutton.

Op Test No.:	301	Scenario #	2	Event #	4,5,6	Page	58 of 155

Event Description: 1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 3 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. **Verify the following load reduction criteria - KNOWN:** __ • Time required to reduce load __ • Target load power level.	**Perform the following:** __ a. **WHEN** required target power level and available time known, **THEN** perform Steps 5 through 10. __ b. **GO TO** Step 10.
__ 7. **Verify time required to reduce load - GREATER THAN OR EQUAL TO 15 MINUTES.**	__ **GO TO Step 10.**

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 4 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE The following tables are estimates only and can be used for rapid entry into the turbine control panel.

__ 8. **Determine required power reduction rate (MW/Min) from table below:**

Time to Reduce Load (Min)	Total Power Change Required (%)									
	10	20	30	40	50	60	70	80	90	100
15	8	16	24	32	40	48	56	64	72	80
20	6	12	18	24	30	36	42	48	54	60
30	4	8	12	16	20	24	28	32	36	40
45	2.7	5.3	8	10.7	13.3	16	18.7	21.3	24	26.7
60		4	6	8	10	12	14	16	18	20
75		3.2	4.8	6.4	8	9.6	11.2	12.8	14.4	16
90		2.7	4	5.3	6.7	8	9.3	10.7	12	13.3
120			3	4	5	6	7	8	9	10
150				3.2	4	4.8	5.6	6.4	7.2	8
180					3.3	4	4.7	5.3	6	6.7
210					2.9	3.4	4	4.6	5.1	5.7
240						3	3.5	4	4.5	5
270							3.1	3.6	4	4.4
300								3.2	3.6	4
330									3.3	3.6
360									3	3.3

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 5 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

__ 9. **Determine target load from table below:**

Reactor Power (%)	Target (MW)
0	40
10	120
20	240
30	360
40	480
48	576
50	600
60	720
69	828
70	840
80	960
90	1080

10. **Initiate turbine load reduction as follows:**

NOTE
- Any load reduction rate of greater than 25 MW/Min must be performed in the manual mode.
- Unloading rates greater than 60 MW/Min (5%/minute) will meet C-7A interlock and may result in steam dump actuation.

__ a. Verify automatic turbine load reduction - DESIRED.

a. Reduce turbine load as follows:

NOTE In manual mode, the control valves are capable of full travel within 3 minutes.

__ 1) Select "MANUAL" and "CONTROL VALVE LOWER" to reduce turbine load as required.

(RNO continued on next page)

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 6 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
	__ 2) **IF AT ANY TIME** auto turbine control available **AND** desired, **THEN** perform Steps 5 through 11.
	__ 3) **GO TO** Step 12.
__ b. Enter desired "LOAD RATE" on turbine control panel.	
__ c. Enter desired "TARGET" on turbine control panel.	
__ d. Depress "GO" pushbutton on turbine control panel.	
__ e. Verify turbine load - TRENDING DOWN AS REQUIRED.	e. Perform the following:
	NOTE In manual mode, the control valves are capable of full travel within 3 minutes.
	__ 1) Select "MANUAL" and "CONTROL VALVE LOWER" to reduce turbine load as required.
	__ 2) **GO TO** Step 12.
__ f. **IF AT ANY TIME** turbine controls fail to respond properly, **THEN** perform Step 10.e.	
__ 11. **IF AT ANY TIME turbine load reduction rate OR target load must be changed, THEN RETURN TO Step 5.**	
__ 12. **Adjust power factor as necessary. REFER TO Unit 1 Revised Data Book Figure 43.**	

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 7 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. **Attempt to control T-Avg as follows:**	
__ a. Verify T-Ref instrumentation - AVAILABLE.	__ a. **IF** T-Avg Coastdown in progress, **THEN** determine T-Ref from table. **REFER TO** Enclosure 5 (T-Ref Value Following Runback/Power Reduction).
__ b. Verify control rods - IN AUTO AND STEPPING IN.	__ b. **IF** T-Avg greater than 1.5°F higher than T-Ref, **THEN** insert control rods as required to maintain T-Avg within 1°F of T-Ref.
__ c. Maintain T-Avg greater than or equal to 551°F.	
__ d. Verify Reactor Engineering Power Maneuvering Guidance provided by Duty Reactor Engineer.	d. Perform the following: **NOTE** • The boric acid added to the NC System should be added in several increments. • The boric acid added to the NC System should be added only during the first hour of the downpower event. __ 1) Borate NC System as required. **REFER TO** Unit 1 R.O.D. book (section 4.8). __ 2) **GO TO** Step 13.f.
__ e. Borate NC System as required. **REFER TO** Reactor Engineering Power Maneuvering Guidance.	
__ f. Ensure operator monitors Enclosure 2 (Rod Insertion Limit Boration).	

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 8 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. **Verify Pzr PORV and Pzr spray valve status as follows:**	
__ a. All Pzr PORVs - CLOSED.	a. **IF** Pzr pressure less than 2315 PSIG, **THEN** perform the following: __ 1) CLOSE Pzr PORV(s). __ 2) **IF** any Pzr PORV cannot be closed, **THEN** CLOSE its isolation valve. 3) **IF** Pzr PORV isolation valve cannot be closed, **THEN** perform the following: __ a) Trip reactor. __ b) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ b. Normal Pzr spray valves - CLOSED.	b. **IF** Pzr pressure trending down in uncontrolled manner, **THEN** perform the following: __ 1) CLOSE spray valve. __ 2) **IF** affected spray valve cannot be closed, **THEN REFER TO** AP/1/A/5500/011 (Pressurizer Pressure Anomolies).
__ 15. **Operate RC pumps and fans as necessary to maintain RC temperature greater than 60°F. REFER TO OP/1/B/6400/001A (Condenser Circulating Water System).**	

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 9 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 16. **Verify reactor power - LESS THAN 85%.**	**Perform the following:** a. **IF** target load less than 85%, **THEN** perform the following: __ 1) **WHEN** time and manpower permit, **THEN** perform applicable steps of OP/1/A/6100/003 (Controlling Procedure For Unit Operation). __ 2) Do not continue in this procedure until reactor power less than 85%. __ 3) **WHEN** reactor power less than 85%, **THEN GO TO** Step 17. b. **WHEN** target load reached, **THEN** perform the following: __ 1) Stabilize unit at current power level. __ 2) Maintain control rods above insertion limits. 3) Adjust the following as required to maintain T-Avg within 1°F of T-Ref: __ • Turbine load __ • Control rods __ • Boron concentration. __ c. **GO TO** Step 41.
__ 17. **Dispatch operator to secure both C-Htr drain pumps. REFER TO OP/1/B/6250/004 (Feedwater Heater Vents, Drains and Bleed System).**	

Op Test No.:	301	Scenario #	2	Event #	4,5,6	Page	65 of 155
Event Description:	1C Steam Generator Tube Leak / Rapid Downpower / Rods Fail to Insert in Auto						

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 10 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18. **Align AS supply to CF pumps as follows:**	
__ a. Adjust 1AS-2 (Main Stm To Aux Steam) as necessary to maintain AS header pressure 165 PSIG.	
__ b. Ensure 1AS-12 (AS To CFPT Isol) - OPEN.	
__ c. Dispatch operator to close 1SP-3 (SC To CFPT 1A & 1B) (TB1-640, 1G-24).	
__ 19. **Adjust 1TL-4 (Stm Seal Reg Byp) as necessary to maintain steam seal pressure between 3 PSIG - 5 PSIG.**	

CNS AP/1/A/5500/009	RAPID DOWNPOWER	PAGE NO. 11 of 40 Revision 34

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 20. **Verify reactor power - LESS THAN 69%.**	**Perform the following:** a. **IF** target load less than 69%, **THEN** perform the following: __ 1) **WHEN** time and manpower permit, **THEN** perform applicable steps of OP/1/A/6100/003 (Controlling Procedure For Unit Operation). __ 2) Do not continue in this procedure until reactor power less than 69%. __ 3) **WHEN** reactor power less than 69%, **THEN GO TO** Step 21. b. **WHEN** target load reached, **THEN** perform the following: __ 1) Stabilize unit at current power level. __ 2) Maintain control rods above insertion limits. 3) Adjust the following as required to maintain T-Avg within 1°F of T-Ref: __ • Turbine load __ • Control rods __ • Boron concentration. __ c. **GO TO** Step 41.
Note to Evaluator: **At the discretion of the Lead Evaluator, the scenario may continue to the next event by instructing the booth operator to INSERT Trigger 7 (1C S/G Tube Rupture).**	
__ 21. **Verify "P9-REACTOR TRIP ON TURBINE TRIP BLOCKED" status light (1SI-18) - LIT.**	

Op Test No.: 301	Scenario # 2	Event # 7,8,9,10	Page 67	of 155			

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

Control Room Indications
Pressurizer Level – LOWERING UNCONTROLLED
1RAD-3, E/5 "1EMF26, 27, 28, 29, S/G A, B, C, D STEAMLINE" – LIT
1C S/G N/R Level – RISING UNCONTROLLED

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	68	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 4 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 6 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Verify Reactor Trip:** __ • All rod bottom lights - LIT __ • All reactor trip and bypass breakers - OPEN __ • I/R power - TRENDING DOWN.	**Perform the following:** __ a. Trip reactor. b. **IF** reactor will not trip, **THEN** concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • **GO TO** EP/1/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS).
3. **Verify Turbine Trip:** __ • All turbine stop valves - CLOSED.	**Perform the following:** __ a. Trip turbine. b. **IF** turbine will not trip, **THEN** perform the following: __ 1) Depress "MANUAL" pushbutton on turbine control panel. __ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons. 3) **IF** control valves will not close, **THEN** CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves.

Op Test No.: 301	Scenario # 2	Event # 7,8,9,10	Page 69 of 155				

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 5 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 4. **Verify 1ETA and 1ETB - ENERGIZED.**	**Perform the following:** __ a. **IF** 1ETA **AND** 1ETB de-energized, **THEN GO TO** EP/1/A/5000/ECA-0.0 (Loss of All AC Power). __ b. **WHEN** time allows, **THEN** attempt to restore power to de-energized switchgear while continuing with this procedure. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power).

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 6 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify S/I actuated:**	
__ a. "SAFETY INJECTION ACTUATED" status light (1SI-13) - LIT.	a. Perform the following: 1) Verify conditions requiring S/I: __ • Pzr pressure - LESS THAN 1845 PSIG OR __ • Containment pressure - GREATER THAN 1.2 PSIG. __ 2) **IF** S/I required, **THEN** initiate S/I. 3) **IF** S/I not required, **THEN** concurrently perform the following: • **IF** 1ETA **OR** 1ETB de-energized, **THEN** ensure the following pumps running on energized bus: __ • NV pump __ • KC pumps __ • RN pump. __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees). __ • **GO TO** EP/1/A/5000/ES-0.1 (Reactor Trip Response).
__ b. Both E/S load sequencer actuated status lights (1SI-14) - LIT.	__ b. Initiate S/I.
__ 6. **Announce "Unit 1 Safety Injection".**	**Note to Evaluator:** **Once immediate actions are complete, the crew should use foldout page guidance #5 to isolate CA flow to the 1C S/G. The normal method to do this is using the CA flow control valve. This valve is failed open, so the operator will have to use the CA motor operated isolation valve to complete this task.**

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	71 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 7 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Determine required notifications:**	
__ • **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO)	
__ • **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation).	
__ 8. **Verify all Feedwater Isolation status lights (1SI-5) - LIT.**	**Perform the following:** __ a. Initiate Feedwater Isolation. __ b. **IF** proper status light indication not obtained, **THEN** CLOSE valves.
9. **Verify Phase A Containment Isolation status as follows:**	
__ a. Phase A "RESET" lights - DARK.	__ a. Initiate Phase A Isolation.
__ b. Monitor Light Panel Group 5 St lights on energized train(s) - LIT.	__ b. Align valves as necessary to ensure each penetration isolated by at least one isolation valve.

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	72	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 8 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Verify Phase B actuation as follows:**	
__ a. Verify containment pressure - HAS REMAINED LESS THAN 3 PSIG.	a. Perform the following: 1) Verify Phase B Isolation actuated as follows: __ a) Phase B Isolation "RESET" lights - DARK. __ b) **IF** Phase B Isolation "RESET" lights lit, **THEN** initiate Phase B Isolation. c) Verify following monitor light panel lights on energized train(s) - LIT: __ • Group 1 Sp lights __ • Group 5 Sp lights __ • Group 5 St light L/11. __ d) **IF** monitor light panel not in correct alignment, **THEN** ensure correct alignment. __ 2) Stop all NC pumps. __ 3) Maintain seal injection flow. __ 4) Energize H_2 igniters. (RNO continued on next page)

Op Test No.: 301	Scenario # 2	Event # 7,8,9,10	Page 73 of 155				

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 9 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	5) Dispatch operator to perform the following:
	__ a) Secure all ice condenser air handling units. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 11 (Securing All Ice Condenser Units).
	__ b) Place containment H_2 analyzers in service. **REFER TO** OP/1/A/6450/010 (Containment Hydrogen Control Systems).
	__ 6) **WHEN** 9 minutes elapsed, **THEN** verify proper VX System operation. **REFER TO** Enclosure 5 (VX System Operation).
	__ 7) **GO TO** Step 11.
__ b. **IF AT ANY TIME** containment pressure exceeds 3 PSIG while in this procedure, **THEN** perform Step 10.a.	

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	74 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 10 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **Verify proper CA pump status as follows:**	
__ a. Motor driven CA pumps - ON.	a. Perform the following for affected train(s): __ 1) Reset ECCS. __ 2) Reset D/G load sequencer(s). __ 3) Start affected pump(s). __ 4) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.
__ b. 3 S/G N/R levels - GREATER THAN 11%.	b. Perform the following: __ 1) Place CA PMP #1 control switch to ON. __ 2) Ensure CA pump #1 - RUNNING.
12. **Verify all of the following S/I pumps - ON:** __ • NV pumps __ • ND pumps __ • NI pumps.	**Perform the following for affected train(s):** __ a. Reset ECCS. __ b. Reset D/G load sequencer(s). __ c. Start affected pump(s). __ d. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 11 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 13. **Verify all KC pumps - ON.**	**Perform the following for affected train(s):** __ a. Reset ECCS. __ b. Reset D/G load sequencer(s). __ c. Start affected pump(s). __ d. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on. __ e. **IF** KC flow cannot be established to NC pumps, **THEN** stop all NC pumps.
__ 14. **Verify all Unit 1 and Unit 2 RN pumps - ON.** **Note to Evaluator:** **Crew may make the decision to transition to the RNO at this step due to 1B RN pump tripping earlier in the scenario. However resetting ECCS and D/G load sequencer will not allow starting of this pump, so the crew may determine that all available RN pumps are in service.**	**Perform the following:** __ a. **IF** any Unit 2 RN pump off, **THEN** start affected pump(s). b. **IF** any Unit 1 RN pump off, **THEN** perform the following for affected train(s): __ 1) Reset ECCS. __ 2) Reset D/G load sequencer(s). __ 3) Start affected pump(s). __ 4) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.
15. **Verify proper ventilation systems operation as follows:** __ • **REFER TO** Enclosure 2 (Ventilation System Verification) __ • Notify Unit 2 operator to perform Enclosure 3 (Opposite Unit Ventilation Verification).	**Note to Evaluator:** **Enclosure 2 can be found as Attachment 7 in the back of this document.**

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	76 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 12 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 16. **Verify all S/G pressures - GREATER THAN 775 PSIG.**	**Perform the following:** a. Verify Main Steam Isolation as follows: __ • All MSIVs - CLOSED __ • All MSIV bypass valves - CLOSED __ • All S/G PORVs - CLOSED. b. <u>IF</u> any valve open, <u>THEN</u> perform the following: __ 1) Initiate Main Steam Isolation. __ 2) <u>IF</u> any valve still open, <u>THEN</u> CLOSE valve.
17. **Verify proper S/I flow as follows:**	**Critical Task #2**
__ a. "NV S/I FLOW" - INDICATING FLOW.	__ a. Start NV pump(s) and align valves.

Note to Evaluator:

BOP will have to manually open 1NI-9A and/or 1NI-10B to establish high head ECCS flow.

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 12 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 16. **Verify all S/G pressures - GREATER THAN 775 PSIG.**	**Perform the following:** a. Verify Main Steam Isolation as follows: __ • All MSIVs - CLOSED __ • All MSIV bypass valves - CLOSED __ • All S/G PORVs - CLOSED. b. IF any valve open, **THEN** perform the following: __ 1) Initiate Main Steam Isolation. __ 2) IF any valve still open, **THEN** CLOSE valve.
17. **Verify proper S/I flow as follows:**	
__ a. "NV S/I FLOW" - INDICATING FLOW.	__ a. Start NV pump(s) and align valves.

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	78 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 13 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ b. NC pressure - LESS THAN 1620 PSIG.	b. Perform the following: __ 1) Ensure ND pump miniflow valve on operating ND pump(s) - OPEN. N/A 2) **IF** ND pump miniflow valve(s) cannot be opened, **THEN** perform the following for affected train(s): __ a) Reset ECCS. __ b) Reset D/G load sequencer. __ c) Stop ND pump. __ d) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on. __ e) **IF AT ANY TIME** NC pressure goes down to less than 285 PSIG in uncontrolled manner, **THEN** restart ND pump. __ 3) **GO TO** Step 18.
__ c. NI pumps - INDICATING FLOW.	__ c. Start NI pump(s) and align valves.

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	79 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 14 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ d. NC pressure - LESS THAN 285 PSIG.	d. Perform the following: __ 1) Ensure ND pump miniflow valve on operating ND pump(s) - OPEN. 2) **IF** ND pump miniflow valve(s) cannot be opened, **THEN** perform the following for affected train(s): __ a) Reset ECCS. __ b) Reset D/G load sequencer. __ c) Stop ND pump. __ d) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on. __ e) **IF AT ANY TIME** NC pressure goes down to less than 285 PSIG in uncontrolled manner, **THEN** restart ND pump. __ 3) **GO TO** Step 18.
__ e. ND pumps - INDICATING FLOW TO C-LEGS.	__ e. Start ND pump(s) and align valves.
__ 18. **WHEN time and manpower permit (within two hours of event), THEN monitor Spent Fuel Pool level and temperature. REFER TO EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 1 (Unit 1 Spent Fuel Pool Monitoring).**	

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	80 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 15 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19. **Control S/G levels as follows:**	
__ a. Verify total CA flow - GREATER THAN 450 GPM.	a. Perform the following: 1) **IF** N/R level in all S/Gs less than 11% (29% ACC), **THEN** perform the following: __ • Start CA pumps __ • Ensure correct valve alignment. 2) **IF** N/R level in all S/Gs less than 11% (29% ACC) **AND** feed flow greater than 450 GPM cannot be established, **THEN** concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • **GO TO** EP/1/A/5000/FR-H.1 (Response to Loss of Secondary Heat Sink).
__ b. **WHEN** each S/G N/R level greater than 11% (29% ACC), **THEN** control feed flow to maintain that S/G N/R level between 11% (29% ACC) and 50%.	
__ 20. **Verify all CA isolation valves on intact S/Gs - OPEN.**	__ **OPEN valve(s).**
__ 21. **Verify S/I equipment status based on monitor light panel(s) - IN PROPER ALIGNMENT.**	__ **Align equipment.** Critical Task #2

Note to Evaluator:

BOP will have to manually open 1NI-9A and/or 1NI-10B to establish high head ECCS flow.

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 16 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE Enclosure 4 (NC Temperature Control) shall remain in effect until subsequent procedures provide alternative NC temperature control guidance.

Note to Evaluator:

Enclosure 4 can be found as Attachment 8 in the back of this document.

__ 22. **Control NC temperature. REFER TO Enclosure 4 (NC Temperature Control).**

23. **Verify Pzr PORV and Pzr Spray Valve status as follows:**

__ a. All Pzr PORVs - CLOSED.

a. **IF** Pzr pressure less than 2315 PSIG, **THEN** perform the following:

__ 1) CLOSE Pzr PORV(s).

__ 2) **IF** any Pzr PORV cannot be closed, **THEN** CLOSE its isolation valve.

3) **IF** 1NC-32B **OR** 1NC-34A cannot be closed **OR** isolated, **THEN** perform the following:

a) Align N_2 to PORVs by opening the following valves:

__ • 1NI-438A (Emer N2 From CLA A To 1NC-34A)

__ • 1NI-439B (Emer N2 From CLA B To 1NC-32B).

__ b) CLOSE affected Pzr PORV.

(RNO continued on next page)

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	82	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 17 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

23. (Continued)

4) **IF** any Pzr PORV cannot be closed **OR** isolated, **THEN** perform the following:

__ a) Energize H_2 igniters.

b) Dispatch operator to perform the following:

__ (1) Secure all ice condenser air handling units. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 11 (Securing All Ice Condenser Units).

__ (2) Place containment H_2 analyzers in service. **REFER TO** OP/1/A/6450/010 (Containment Hydrogen Control Systems).

c) **IF AT ANY TIME** both the following conditions exist:

__ • Containment pressure - HAS REMAINED LESS THAN 3 PSIG

__ • Containment pressure - BETWEEN 1 PSIG **AND** 3 PSIG,

__ **THEN** start one VX fan and secure normal containment ventilation. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 18 (VX and Containment Ventilation Control).

(RNO continued on next page)

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	83	of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 18 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23. (Continued)	
	d) Concurrently perform the following: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • **<u>GO TO</u>** EP/1/A/5000/E-1 (Loss of Reactor or Secondary Coolant).
__ b. Normal Pzr spray valves - CLOSED.	b. **<u>IF</u>** Pzr pressure less than 2150 PSIG, **<u>THEN</u>** perform the following: __ 1) CLOSE spray valve(s). 2) **<u>IF</u>** spray valve(s) cannot be closed, **<u>THEN</u>** perform the following: __ a) Stop NC pumps 1A and 1B. __ b) **<u>IF</u>** both 1C **<u>AND</u>** 1D NC pumps on, **<u>THEN</u>** stop one additional pump.
__ c. At least one Pzr PORV isolation valve - OPEN.	__ c. **<u>IF</u>** power available, **<u>THEN</u>** OPEN one Pzr PORV isolation valve unless it was closed to isolate an open Pzr PORV.
__ 24. **Verify NC subcooling based on core exit T/Cs - GREATER THAN 0°F.**	**<u>IF</u> any NV <u>OR</u> NI pump on, <u>THEN</u> perform the following:** __ a. Ensure all NC pumps - OFF. __ b. Maintain seal injection flow.

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 19 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
25. **Verify main steamlines intact:**	**IF any S/G faulted, THEN perform the following:**
__ • All S/G pressures - STABLE OR TRENDING UP	__ a. Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees).
__ • ALL S/Gs - PRESSURIZED.	__ b. **GO TO** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation).
26. **Verify S/G tubes intact as follows:**	**IF any EMF trip 1 light lit OR any S/G level trending up in uncontrolled manner, THEN concurrently:**
__ • All S/G levels - STABLE OR TRENDING UP IN A CONTROLLED MANNER.	__ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees).
• Verify the following EMF trip 1 lights - DARK:	__ • **GO TO** EP/1/A/5000/E-3 (Steam Generator Tube Rupture).
__ • 1EMF-33 (Condenser Air Ejector Exhaust)	
__ • 1EMF-26 (Steamline 1A)	
__ • 1EMF-27 (Steamline 1B)	
__ • 1EMF-28 (Steamline 1C)	
__ • 1EMF-29 (Steamline 1D).	

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	85	of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 2 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 9 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Identify ruptured S/G(s) as follows:**	**Perform the following:**
__ • Any S/G level - TRENDING UP IN UNCONTROLLED MANNER	__ a. Continue to monitor S/G N/R levels and steamline EMFs.
OR	b. Notify RP to perform the following:
__ • Chemistry or RP determined ruptured S/G	__ • Frisk all Unit 1 S/G cation columns for activity
OR	__ • Notify Control Room of results.
• Any of the following EMF trip 1 lights - LIT: __ • 1EMF-26 (Steamline 1A) __ • 1EMF-27 (Steamline 1B) __ • 1EMF-28 (Steamline 1C) __ • 1EMF-29 (Steamline 1D).	c. <u>IF</u> S/G sampling required to identify ruptured S/G(s), <u>**THEN**</u> perform the following: 1) Ensure the following signals - RESET: __ a) Phase A Containment Isolations. __ b) CA System valve control. __ c) KC NC NI NM St signals. __ 2) Align all S/Gs for Chemistry sampling. 3) Notify Chemistry to perform the following: __ • Sample all S/Gs for activity __ • Notify Control Room of results.
	__ d. <u>**WHEN**</u> ruptured S/G(s) identified, <u>**THEN**</u> immediately <u>**RETURN TO**</u> Step 3.
	__ e. <u>**GO TO**</u> Step 11.

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 3 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 3. **Verify at least one intact S/G - AVAILABLE FOR NC SYSTEM COOLDOWN.**	__ **Maintain at least one S/G available for NC System cooldown in subsequent steps.**
4. **Isolate flow from ruptured S/G(s) as follows:**	
__ a. Verify all ruptured S/G(s) PORV - CLOSED.	a. **WHEN** ruptured S/G(s) pressure less than 1090 PSIG, **THEN** perform the following:
	__ 1) Ensure ruptured S/G(s) PORV - CLOSED.
	__ 2) **IF** ruptured S/G(s) PORV will not close, **THEN** CLOSE ruptured S/G(s) PORV isolation valve.
	__ 3) **IF** ruptured S/G(s) PORV isolation valve will not close, **THEN** dispatch operator to close ruptured S/G(s) PORV isolation valve.

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 4 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
__ b. Verify S/G(s) 1B and 1C - INTACT.	b. Perform the following:
	__ 1) **IF** both motor driven CA pumps available, **THEN** CLOSE "CAPT TRIP T/V CTRL".
	N/A 2) **IF** CA Pump #1 only source of feedwater, **THEN** maintain steam flow to CA Pump #1 from at least one S/G.
	N/A 3) **IF** S/G 1B ruptured, **THEN** perform the following:
	__ a) Dispatch two operators to unlock and close 1SA-1 (1B S/G Main Steam to CAPT Maintenance Isol) (DH-624, FF-53, Rm 572) (Breakaway lock).
	__ b) **IF** 1SA-1 cannot be closed, **THEN** dispatch two operators to unlock and close 1SA-3 (1B S/G Main Steam to CAPT Stop Check) (AB-551, DD-53, Rm 217) (Breakaway lock).
	4) **IF** S/G 1C ruptured, **THEN** perform the following:
	__ a) Dispatch two operators to unlock and close 1SA-4 (1C S/G Main Steam to CAPT Maintenance Isol) (DH-624, FF-53, Rm 572) (Breakaway lock installed).
	__ b) **IF** 1SA-4 cannot be closed, **THEN** dispatch two operators to unlock and close 1SA-6 (1C S/G Main Steam to CAPT Stop Check) (AB-551, DD-53, Rm 217) (Breakaway lock) (Ladder needed).
	(RNO continued on next page)

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	88	of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 5 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
	__ 5) **WHEN** ruptured S/G steam supply to CA Pump #1 isolated, **THEN** OPEN "CAPT TRIP T/V CTRL".
c. Isolate blowdown and steam drain on all ruptured S/G(s) as follows:	
• S/G 1A:	
__ 1) CLOSE 1SM-77A (S/G 1A Otlt Hdr Bldwn C/V).	__ 1) Dispatch operator to close 1SM-77A (S/G 1A Otlt Hdr Bldwn C/V) (DH-583, FF-GG, 43-44, Rm 591).
2) Verify the following blowdown isolation valves - CLOSED:	
__ a) 1BB-56A (S/G 1A Bldwn Cont Isol Insd).	__ a) CLOSE valve.

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	89 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 6 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
__ b) 1BB-148B (S/G 1A Bldwn Cont Isol Byp).	b) Perform the following: __ (1) CLOSE valve. (2) **IF** valve will not close **AND** 1BB-56A open, **THEN** perform the following: __ 1. Ensure "S/G A BLDWN FLOW CTRL" - CLOSED. 2. Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-148B (S/G 1A Bldwn Cont Isol Byp) (DH-580, EE-FF, 44-45, Rm 591) __ • 1BB-81 (1A S/G Blowdown Penetration Valve Test Isol) (DH-583, EE-FF, 44, Rm 591).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	90 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 7 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
__ c) 1BB-57B (S/G 1A Bldwn Cont Isol Otsd).	c) Perform the following: __ (1) CLOSE valve. (2) **IF** valve will not close **AND** 1BB-56A open, **THEN** perform the following: __ 1. Ensure "S/G A BLDWN FLOW CTRL" - CLOSED. 2. Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-57B (S/G 1A Bldwn Cont Isol Otsd) (DH-580, EE-FF, 44-45, Rm 591) __ • 1BB-81 (1A S/G Blowdown Penetration Valve Test Isol) (DH-583, EE-FF, 44, Rm 591).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	91	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 8 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
• S/G 1B:	
__ 1) CLOSE 1SM-76B (S/G 1B Otlt Hdr Bldwn C/V).	__ 1) Dispatch operator to close 1SM-76B (S/G 1B Otlt Hdr Bldwn C/V) (DH-583, FF-53, Rm 572).
2) Verify the following blowdown isolation valves - CLOSED:	
__ a) 1BB-19A (S/G 1B Bldwn Cont Isol Insd).	__ a) CLOSE valve.
__ b) 1BB-150B (S/G 1B Bldwn Cont Isol Byp).	b) Perform the following: __ (1) CLOSE valve. (2) **IF** valve will not close **AND** 1BB-19A open, **THEN** perform the following: __ 1. Ensure "S/G B BLDWN FLOW CTRL" - CLOSED. 2. Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-150B (S/G 1B Bldwn Cont Isol Byp) (DH-580, FF, 52-53, Rm 572) __ • 1BB-83 (1B S/G Blowdown Penetration Valve Test Isol) (DH-580, FF-53, Rm 572).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	92	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 9 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
__ c) 1BB-21B (S/G 1B Bldwn Cont Isol Otsd).	c) Perform the following: __ (1) CLOSE valve. (2) **IF** valve will not close **AND** 1BB-19A open, **THEN** perform the following: __ 1. Ensure "S/G B BLDWN FLOW CTRL" - CLOSED. 2. Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-21B (S/G 1B Bldwn Cont Isol Otsd) (DH-580, FF, 52-53, Rm 572) __ • 1BB-83 (1B S/G Blowdown Penetration Valve Test Isol) (DH-580, FF-53, Rm 572).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	93	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 10 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
• S/G 1C:	
__ 1) CLOSE 1SM-75A (S/G 1C Otlt Hdr Bldwn C/V).	__ 1) Dispatch operator to close 1SM-75A (S/G 1C Otlt Hdr Bldwn C/V) (DH-580, GG-52/53, Rm 572).
2) Verify the following blowdown isolation valves - CLOSED:	
__ a) 1BB-60A (S/G 1C Bldwn Cont Isol Insd).	__ a) CLOSE valve.
__ b) 1BB-149B (S/G 1C Bldwn Cont Isol Byp).	b) Perform the following: __ (1) CLOSE valve. (2) **IF** valve will not close **AND** 1BB-60A open, **THEN** perform the following: __ 1. Ensure "S/G C BLDWN FLOW CTRL" - CLOSED. 2. Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-149B (S/G 1C Bldwn Cont Isol Byp) (DH-578, FF-GG, 52, Rm 572) __ • 1BB-82 (1C S/G Blowdown Penetration Valve Test Isol) (DH-583, FF-53, Rm 572).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	94 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 11 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
__ c) 1BB-61B (S/G 1C Bldwn Cont Isol Otsd).	c) Perform the following: __ (1) CLOSE valve. (2) **IF** valve will not close **AND** 1BB-60A open, **THEN** perform the following: __ 1. Ensure "S/G C BLDWN FLOW CTRL" - CLOSED. 2. Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-61B (S/G 1C Bldwn Cont Isol Otsd) (DH-578, FF-GG, 52, Rm 572) __ • 1BB-82 (1C S/G Blowdown Penetration Valve Test Isol) (DH-583, FF-53, Rm 572).

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 12 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
• <u>S/G 1D</u>:	
__ 1) CLOSE 1SM-74B (S/G 1D Otlt Hdr Bldwn C/V).	__ 1) Dispatch operator to close 1SM-74B (S/G 1D Otlt Hdr Bldwn C/V) (DH-583, FF-GG, 44-45, Rm 591).
2) Verify the following blowdown isolation valves - CLOSED:	
__ a) 1BB-8A (S/G 1D Bldwn Cont Isol Insd).	__ a) CLOSE valve.
__ b) 1BB-147B (S/G 1D Bldwn Cont Isol Byp).	b) Perform the following: __ (1) CLOSE valve. (2) <u>**IF**</u> valve will not close <u>**AND**</u> 1BB-8A open, <u>**THEN**</u> perform the following: __ 1. Ensure "S/G D BLDWN FLOW CTRL" - CLOSED. 2. Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-147B (S/G 1D Bldwn Cont Isol Byp) (DH-582, EE-FF, 44, Rm 591) __ • 1BB-80 (1D S/G Blowdown Penetration Valve Test Isol) (DH-583,EE-FF, 44, Rm 591).

Op Test No.: 301 Scenario # 2 Event # 7,8,9,10	Page 96 of 155
Event Description:	1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 13 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. (Continued)	
__ c) 1BB-10B (S/G 1D Bldwn Cont Isol Otsd).	c) Perform the following: __ (1) CLOSE valve. (2) **<u>IF</u>** valve will not close **<u>AND</u>** 1BB-8A open, **<u>THEN</u>** perform the following: __ 1. Ensure "S/G D BLDWN FLOW CTRL" - CLOSED. 2. Dispatch operators to ensure the following valves - CLOSED: __ • 1BB-10B (S/G 1D Bldwn Cont Isol Otsd) (DH-582, EE-FF, 44, Rm 591) __ • 1BB-80 (1D S/G Blowdown Penetration Valve Test Isol) (DH-583,EE-FF, 44, Rm 591).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	97	of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE	PAGE NO. 14 of 95 Revision 44

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **CLOSE the following valves on all ruptured S/G(s):** __ • MSIV __ • MSIV bypass valve. **Note to Evaluator:** **Operators will not be able to close any of the MSIVs from the control room. Therefore the ruptured S/G can not be isolated from the intact S/Gs requiring a transition to EP/1/A/5000/ECA-3.1 (SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired).**	**Perform the following:** a. CLOSE the following valves on remaining S/Gs: __ • MSIV __ • MSIV bypass valve. __ b. Place "STM DUMP CTRL" in manual. __ c. Adjust "STM DUMP CTRL" output to 0%. __ d. Place "STEAM DUMP SELECT" switch in pressure mode. N/A e. **IF** any intact S/G MSIV and associated bypass valve closed, **THEN GO TO** Step g. in this RNO. __ f. **GO TO** EP/1/A/5000/ECA-3.1 (SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired). __ g. Place steam dump interlock bypass switches in "OFF RESET". h. Transfer turbine steam seal supply to AS as follows: __ 1) OPEN 1TL-8 (Aux Stm To Stm Seal Reg). __ 2) CLOSE 1TL-2 (Main Stm To Stm Seal Reg). (RNO continued on next page)

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 3 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 10 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Reset the following:**	
__ a. ECCS.	__ a. Locally reset ECCS. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 4 (ECCS Master Reset).
__ b. D/G load sequencers.	b. Dispatch operator to open affected sequencer(s) control power breaker: __ • 1EDE-F01F (Diesel Generator Load Sequencer Panel 1DGLSA) (AB-577, BB-46, Rm 496) __ • 1EDF-F01F (Diesel Generator Load Sequencer Panel 1DGLSB) (AB-560, BB-46, Rm 372).
__ c. Phase A.	
__ d. Phase B.	
__ e. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.	

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	99 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 4 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **Establish VI to containment as follows:** __ • Ensure 1VI-77B (VI Cont Isol) - OPEN __ • Verify VI pressure - GREATER THAN 85 PSIG.	**Perform the following:** a. Align N_2 to Pzr PORVs by opening the following valves: __ • 1NI-438A (Emer N2 From CLA A To 1NC-34A) __ • 1NI-439B (Emer N2 From CLA B To 1NC-32B). b. **IF** VI pressure less than 85 PSIG, **THEN** perform the following: __ 1) Dispatch operator to ensure proper VI compressor operation. __ 2) Restore VI while continuing with this procedure. **REFER TO** AP/0/A/5500/022 (Loss of Instrument Air).
4. **Verify all AC busses energized by offsite power as follows:** • A Train: __ • "FTA B/O NORM FDR FRM ATC" - CLOSED __ • "D/G 1A BKR TO ETA" - OPEN __ • 1ETA - ENERGIZED. • B Train: __ • "FTB B/O NORM FDR FRM ATD" - CLOSED __ • "D/G 1B BKR TO ETB" - OPEN __ • 1ETB - ENERGIZED.	**Perform the following:** __ a. Restore offsite power while continuing with this procedure. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power). b. Start the following equipment: __ • Start all available CRD vent fans __ • Dispatch operator to start available VI compressors.

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	100	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 5 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
CAUTION **Pzr heaters shall remain off until directed by subsequent steps.**	
__ 5. **Place all Pzr heaters in manual and off.**	
6. **Control ruptured S/G(s) level as follows:**	
__ a. Verify ruptured S/G(s) N/R level - GREATER THAN 11% (29% ACC).	a. Perform the following: __ 1) **IF** any ruptured S/G also faulted, **THEN** do not establish feed flow to ruptured S/G unless needed for NC System cooldown. 2) **IF** any ruptured S/G(s) not faulted **OR** required for cooldown, **THEN** perform the following: __ a) Establish and maintain feed flow to affected S/G(s). __ b) **WHEN** affected S/G(s) N/R level greater than 11% (29% ACC), **THEN** perform Steps 6.b and 6.c. __ 3) **GO TO** Step 7.

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 6 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
b. Isolate feed flow to all ruptured S/G(s) as follows:	
• For S/G 1A:	
__ 1) CLOSE 1CA-62A (CA Pmp A Disch To S/G 1A Isol).	1) Perform the following: __ a) CLOSE 1CA-60 (CA Pump 1A Flow To S/G 1A). __ b) Dispatch operator to close 1CA-62A (CA Pmp A Disch To S/G 1A Isol) (DH-587, DD-EE, 44-45, Rm 591). __ c) **IF** exterior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-59 (CA Pump 1A Disch To S/G 1A Ctrl Inlet Isol) (AB-551, BB,49-50, Rm 250) (Ladder needed) (Key #633).
__ 2) CLOSE 1CA-66B (CA Pmp 1 Disch To S/G 1A Isol).	2) Perform the following: __ a) CLOSE 1CA-64 (CA Pump #1 Flow To S/G 1A). __ b) Dispatch operator to close 1CA-66B (CA Pmp 1 Disch To S/G 1A Isol) (DH-584, DD-EE, 44-45, Rm 591). __ c) **IF** exterior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-63 (CA Pump No 1 Disch To S/G 1A Ctrl Inlet Isol) (AB-556, BB-50, Rm 250) (Ladder needed) (Key #633).

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 7 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
• For S/G 1B:	
__ 1) CLOSE 1CA-58A (CA Pmp A Disch To S/G 1B Isol).	1) Perform the following: __ a) CLOSE 1CA-56 (CA Pump 1A Flow To S/G 1B). __ b) Dispatch operator to close 1CA-58A (CA Pmp A Disch To S/G 1B Isol) (DH-586, DD-EE, 52-53, Rm 572). __ c) **IF** interior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-55 (CA Pump 1A Disch To S/G 1B Inlet Isol) (AB-550, DD-52, Rm 250) (Key #633).
__ 2) CLOSE 1CA-54B (CA Pmp 1 Disch To S/G 1B Isol).	2) Perform the following: __ a) CLOSE 1CA-52 (CA Pump #1 Flow To S/G 1B). __ b) Dispatch operator to close 1CA-54B (CA Pmp 1 Disch To S/G 1B Isol) (DH-584, DD-EE, 52-53, Rm 572). __ c) **IF** interior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-51 (CA Pump No 1 Disch To S/G 1B Inlet Isol) (AB-552, DD-52, Rm 250) (Key #633).

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 8 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
• For S/G 1C:	
__ 1) CLOSE 1CA-46B (CA Pmp B Disch To S/G 1C Isol).	1) Perform the following: __ a) CLOSE 1CA-44 (CA Pump 1B Flow To S/G 1C). __ b) Dispatch operator to close 1CA-46B (CA Pmp B Disch To S/G 1C Isol) (DH-586, DD, 53-54, Rm 572). __ c) **IF** interior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-43 (CA Pump 1B Disch To S/G 1C Ctrl Inlet Isol) (AB-552, DD-52, Rm 250) (Key #633).
__ 2) CLOSE 1CA-50A (CA Pmp 1 Disch To S/G 1C Isol).	2) Perform the following: __ a) CLOSE 1CA-48 (CA Pump #1 Flow To S/G 1C). __ b) Dispatch operator to close 1CA-50A (CA Pmp 1 Disch To S/G 1C Isol) (DH-584, EE-53, Rm 572). __ c) **IF** interior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-47 (CA Pump No 1 Disch To S/G 1C Ctrl Inlet Isol) (AB-552, DD-53, Rm 250) (Key #633).

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 9 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
• For S/G 1D:	
__ 1) CLOSE 1CA-42B (CA Pmp B Disch To S/G 1D Isol).	1) Perform the following: __ a) CLOSE 1CA-40 (CA Pump 1B Flow To S/G 1D). __ b) Dispatch operator to close 1CA-42B (CA Pmp B Disch To S/G 1D Isol) (DH-586, DD-EE, 43-44, Rm 591). __ c) **IF** exterior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-39 (CA Pump 1B Disch To S/G 1D Ctrl Inlet Isol) (AB-551, BB, 49-50, Rm 250) (Ladder needed) (Key #633).
__ 2) CLOSE 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol).	2) Perform the following: __ a) CLOSE 1CA-36 (CA Pump #1 Flow To S/G 1D). __ b) Dispatch operator to close 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol) (DH-584, DD-EE, 43-44, Rm 591). __ c) **IF** exterior doghouse not accessible **OR** CA cannot be isolated, **THEN** dispatch operator to unlock and close 1CA-35 (CA Pump No 1 Disch To S/G 1D Ctrl Inlet Isol) (AB-555, BB-50, Rm 250) (Ladder needed) (Key #633).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	105	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 10 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
__ c. **IF AT ANY TIME** ruptured S/G(s) N/R level less than 11% (29% ACC), **THEN** perform Step 6.	
7. **Verify criteria to stop operating ND pumps as follows:**	
__ a. Any ND pump - ON.	__ a. **GO TO** Step 7.f.
__ b. Running ND pump suction - ALIGNED TO FWST.	__ b. **GO TO** Step 8.
__ c. NC pressure - GREATER THAN 285 PSIG.	__ c. **GO TO** Step 8.
__ d. NC pressure - STABLE OR TRENDING UP.	__ d. **GO TO** Step 8.
__ e. Stop ND pumps.	
__ f. **IF AT ANY TIME** NC pressure trends down to less than 285 PSIG in uncontrolled manner, **THEN** restart ND pumps.	

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	106 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 11 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. **Initiate evaluation of plant status as follows:**	
a. Verify auxiliary building radiation normal as follows:	a. Perform the following:
__ • EMF-41 (Aux Bldg Ventilation) trip 1 light - DARK	__ 1) Monitor OAC EMF alarms, OAC VA Graphic, and area monitor EMFs to determine location of activity.
__ • All area monitor EMF trip 1 lights - DARK.	__ 2) Dispatch operator to locate and isolate potential Unit 1 leak.
b. Ensure the following signals - RESET:	
__ 1) Phase A Containment Isolations.	
__ 2) KC NC NI NM St signals.	
__ c. **<u>WHEN</u>** TSC activated **<u>AND</u>** staffed, **<u>THEN</u>** notify TSC to perform Enclosure 2 (TSC Actions).	
9. **Verify main steamlines intact as follows:**	**<u>IF</u> any S/G pressure trending down in uncontrolled manner <u>OR</u> depressurized, <u>THEN</u> perform the following:**
__ • All S/G pressures - STABLE OR TRENDING UP	__ a. **<u>IF</u>** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation) has been implemented for all faulted S/G(s), **<u>THEN</u> <u>GO TO</u>** Step 10.
__ • All S/Gs - PRESSURIZED.	__ b. **<u>IF</u>** faulted S/G(s) needed for NC System cooldown, **<u>THEN</u> <u>GO TO</u>** Step 10.
	__ c. **<u>IF</u>** any S/G(s) faulted, **<u>THEN</u> <u>GO TO</u>** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	107 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 12 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Control intact S/G levels as follows:**	
__ a. Verify N/R level in any intact S/Gs - GREATER THAN 11% (29% ACC).	a. Perform the following: __ 1) Maintain total feed flow greater than 450 GPM to intact S/Gs until at least one intact S/G N/R level greater than 11% (29% ACC). 2) __IF__ total feed flow greater than 450 GPM cannot be established, **__THEN__** contact Station Management for guidance to establish feed flow from the following alternate sources: __ • CF __ • CM __ • Alternate low pressure water source.
__ b. THROTTLE feed flow to maintain all intact S/G N/R levels between 16% (29% ACC) and 50%.	b. __IF__ N/R level in any intact S/G continues to trend up in uncontrolled manner, **__THEN__** perform the following: __ 1) Stop NC System cooldown. __ 2) **__GO TO__** EP/1/A/5000/E-3 (Steam Generator Tube Rupture).

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	108 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 13 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- TSC will monitor shutdown margin during cooldown. Shutdown margin results are not required prior to initiating cooldown.
- After low steamline pressure main steam isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation.
- OAC graphic SMRATES to monitor S/G pressure rates can be accessed via a hot button in the center of the SM graphic.

11. **Initiate NC System cooldown to Cold Shutdown as follows:**

a. **WHEN** "P-11 PZR S/I BLOCK PERMISSIVE" status light (1SI-18) lit, **THEN** perform the following:

__ 1) Depress ECCS steam pressure "BLOCK" pushbuttons.

__ 2) Verify main steam isolation blocked status lights (1SI-13) - LIT.

N/A b. **IF** ND System in RHR mode, **THEN** initiate cooldown with ND System while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour.

__ c. Verify "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT.

__ c. **GO TO** Step 11.i.

Op Test No.: 301	Scenario # 2	Event # 7,8,9,10	Page 109 of 155				

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 14 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
__ d. Verify MSIVs on all intact S/Gs - OPEN.	d. Perform the following:
	__ 1) **IF** any S/G faulted, **THEN GO TO** Step 11.i.
	__ 2) **IF** any ruptured S/G MSIV not fully closed, **THEN GO TO** Step 11.i.
	__ 3) **IF** intact S/G MSIVs required closed to isolate leak, **THEN GO TO** Step 11.i.
	4) Reset Main Steam Isolation signal as follows:
	__ a) Ensure manual loaders for all MSIV bypass valves - ADJUSTED TO 0%.
	__ b) Reset SM Isolation.
	__ c) Reset S/G PORVs.
	5) Place steam dumps in pressure mode as follows:
	__ a) Place "STM DUMP CTRL" in manual.
	__ b) Adjust "STM DUMP CTRL" to 0% demand.
	__ c) Place steam dumps in pressure mode.
	(RNO continued on next page)

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 15 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	6) Perform the following to equalize pressure across MSIVs on intact S/Gs: __ a) OPEN MSIV bypass valve on intact S/Gs. __ b) **IF AT ANY TIME** pressure does not equalize as required, **THEN** isolate steam loads off main steam header. **REFER TO** Enclosure 3 (Equalizing Across MSIVs). 7) **WHEN** all intact S/Gs pressure within 50 psig of steam header pressure, **THEN** perform the following: __ a) OPEN all MSIVs on intact S/Gs. __ b) CLOSE all MSIV bypass valves. __ c) Restore any alignments made using Enclosure 3 (Equalizing Across MSIVs). __ d) **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK". __ e) Dump steam to condenser while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour. __ f) **WHEN** condenser dumps established, **THEN** S/G PORVs may be closed. (RNO continued on next page)

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 16 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
	__ 8) **GO TO** Step 11.i to dump steam using S/G PORVs while pressure equalizing across MSIVs.
__ e. Verify steam dumps - IN PRESSURE MODE.	e. Place steam dumps in pressure mode as follows: __ 1) Place "STM DUMP CTRL" slim station in manual. __ 2) Place steam dumps in pressure mode.
__ f. **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK".	
__ g. Dump steam to condenser while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour. **Critical Task #3**	__ g. **IF** steam cannot be dumped to condenser, **THEN GO TO** Step 11.i.
__ h. **GO TO** Step 11.j.	

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	112	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 17 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
__ i. Dump steam from intact S/G(s) PORV while maintaining cooldown rate based on NC T-Colds less than 100°F in an hour.	i. Perform the following: 1) **IF** any intact S/G PORV cannot be operated from Control Room, **THEN** perform the following: __ a) Dispatch operator(s) to operate affected S/G(s) PORV. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 10 (Local Operation of S/G PORVs). __ b) Obtain sound powered phone from storage box on rear wall of Control Room. __ c) Connect sound powered phone to jack on 1MC-11. __ d) Monitor sound powered phone for communication from Doghouse(s). __ e) **IF** 1B **OR** 1C S/G intact and associated S/G PORV unavailable, **THEN** evaluate starting CA Pump #1. 2) **IF** no intact S/G available **AND** ND not in RHR mode, **THEN** contact Station Management to determine which of the following to perform: __ • Use faulted S/G OR __ • Use ruptured S/G.

Op Test No.: 301 Scenario # 2 Event # 7,8,9,10 Page 113 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 18 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
__ j. Verify main steam isolation blocked status lights (1SI-13) - LIT.	j. Perform the following: 1) Depressurize NC System to less than 1955 PSIG using one of the following: __ • Pzr spray OR __ • Pzr PORV. 2) **WHEN** "P-11 PZR S/I BLOCK PERMISSIVE" status light (1SI-18) lit, **THEN** perform the following: __ a) Depress ECCS steam pressure "BLOCK" pushbuttons. __ b) Verify main steam isolation blocked status lights (1SI-13) - LIT. __ 3) Maintain NC pressure less than 1955 PSIG.

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	114	of	155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 19 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. **Determine whether subcooled recovery appropriate as follows:**	
__ a. Verify FWST level - GREATER THAN 62%.	a. Perform the following: __ 1) Determine required Containment Sump level based on existing FWST level. **REFER TO** Enclosure 4 (Adequate Water Inventory). __ 2) **IF AT ANY TIME** Containment Sump level less than required level **OR** transfer to Cold Leg Recirc not available, **THEN GO TO** EP/1/A/5000/ECA-3.2 (SGTR With Loss of Reactor Coolant - Saturated Recovery Desired). __ 3) **IF AT ANY TIME** while in this procedure ruptured S/G(s) N/R level greater than 92% (80% ACC), **THEN** perform Step 12.b RNO. __ 4) **GO TO** Step 13.
__ b. Verify all ruptured S/G(s) N/R level - LESS THAN 92% (80% ACC).	b. Perform the following: __ 1) Consult with Station Management to determine if recovery should be completed using EP/1/A/5000/ECA-3.2 (SGTR With Loss of Reactor Coolant - Saturated Recovery Desired). __ 2) **IF AT ANY TIME** while in this procedure FWST level less than 62%, **THEN** perform Step 12.a RNO. __ 3) **GO TO** Step 13.

Op Test No.:	301	Scenario #	2	Event #	7,8,9,10	Page	115 of 155

Event Description: 1C Steam Generator Tube Rupture / 1C CA Flow Control Valve Failed Open / 1NI-9A & 1NI-10B Fail to Auto Open / MSIVs Fail to Close Manually

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED	PAGE NO. 20 of 102 Revision 40

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. (Continued)	
__ c. **<u>IF AT ANY TIME</u>** while in this procedure FWST level less than 62% **<u>OR</u>** ruptured S/G(s) level greater than 92% (80% ACC), **<u>THEN</u>** perform Step 12.	
__ 13. **Verify NC subcooling based on core exit T/Cs - GREATER THAN 0°F.**	__ **<u>GO TO</u> Step 30.**
14. **Verify S/I in service as follows:**	__ **<u>GO TO</u> Step 24.**
__ • Any NI pump - ON OR __ • 1NI-9A (NV Pmp C/L Inj Isol) - OPEN OR __ • 1NI-10B (NV Pmp C/L Inj Isol) - OPEN OR __ • Any ND pump - ON WITH SUCTION ALIGNED TO FWST **<u>OR</u>** CONTAINMENT SUMP.	

Note to Evaluator:

At this point all Critical Tasks are complete. At the lead evaluators discretion, the scenario may be terminated by having the booth operator place the simulator in FREEZE.

Attachment List

Scenario 1

ATTACHMENT 1 -	Crew Critical Task Summary
ATTACHMENT 2 -	Shift Turnover Information
ATTACHMENT 3 -	AP/0/A/5500/020 Enclosure 1 (Foldout Page)
ATTACHMENT 4 -	AP/1/A/5500/010 Enclosure 1 (Foldout Page)
ATTACHMENT 5 -	AP/1/A/5500/009 Enclosure 1 (Foldout Page)
ATTACHMENT 6 -	EP/1/A/5000/E-0 Enclosure 1 (Foldout Page)
ATTACHMENT 7 -	EP/1/A/5000/E-0 Enclosure 2 (Ventilation System Verification)
ATTACHMENT 8 -	EP/1/A/5000/E-0 Enclosure 4 (NC Temperature Control)
ATTACHMENT 9	EP/1/A/5000/E-3 Enclosure 1 (Foldout Page)
ATTACHMENT 10	EP/1/A/5000/ECA-3.1 Enclosure 1 (Foldout Page)
ATTACHMENT 11 -	Scenario Specific Technical Specifications

ATTACHMENT 1

CREW CRITICAL TASK SUMMARY			
SAT	**UNSAT**	**CT #**	**CRITICAL TASK**
		1	Restore RN flow prior to any NC pump motor bearing temperature reaching 195°F.
		2	Establish high head ECCS flow prior to transition from E-0.
		3	Initiate cooldown at less than 100°F/hour.

Comments:

ATTACHMENT 2

SHIFT TURNOVER INFORMATION			
Unit 1 Status			
Power Level	**Power History**	**NCS Boron**	**Xenon**
75 **%**	MOL	998 PPM	per OAC
Controlling Procedure			
• OP/1/A/6100/003 (Controlling Procedure for Unit Operation), Enclosure 4.1 (Power Increase). The steps up to step 3.49 are complete.			
Other Information Needed to Assume the Shift			
• Unit 1 is at 75% power at the MOL following maintenance of the 1A CFPT LF pumps. 1A CFPT has been placed back in service and current power level is being maintained while maintenance continues to monitor LF system performance. Unit 2 is at 100% power. Direction for the crew is to swap LCVUs by securing 1A LCVU and starting 1C LCVU using OP/1/A/6450/001 Encl. 4.13.			
AOs Available			
Eight AOs are available as listed on the status board			
METEOROLOGICAL CONDITIONS			
• Upper wind direction = 125 degrees, speed = 3 mph • Lower wind direction = 127 degrees, speed = 4.5 mph • Forecast calls for clear skies over the next 24 hours.			

ATTACHMENT 3

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Enclosure 1 - Page 1 of 2 **Foldout Page**	PAGE NO. 71 of 137 Revision 49

1. **SSF Manning Criteria:**

- **IF AT ANY TIME** 1EMXS **OR** 2EMXS de-energized, **THEN** perform the following:

  a. Dispatch operator to align alternate power supply(s). **REFER TO** Enclosure 4 (Align Alternate Power Supply To 1EMXS OR 2EMXS).

  b. Notify operator at SSF (Ext. 5251 or 5212) operator has been dispatched to align alternate power supply to 1EMXS (2EMXS).

CAUTION **Higher KC System temperature due to loss of RN could result in a loss of KC and NV pumps supplying NC pump seal cooling. Failure to restore NC pump seal cooling via thermal barrier cooling or NV seal injection within ten minutes will cause damage to NC pump seals resulting in NC System inventory loss.**

- **IF AT ANY TIME** RN cooling to operating KC Hx lost, **THEN** dispatch operator to SSF to standby at SSF to establish seal injection.

- **IF AT ANY TIME** KC **AND** NV seal cooling for any NC pump lost, **THEN** ensure operator dispatched to SSF to establish NC pump seal injection. **REFER TO** Enclosure 5 (Establishing NC Makeup/Seal Injection From The SSF) for affected Unit(s).

2. **Alternate Cooling to NV Pump Criteria:**

   a. **IF** S/I actuated on either Unit, **THEN** discontinue monitoring this criterion.

   b. **IF** RN **NOT** available to KC, **THEN** perform the following:

   CAUTION **YD can only supply one Unit's NV pump at a time.**

   1) Determine which Unit will receive alternate NV pump cooling from YD.

   2) **IF** Unit 1 selected, **THEN** align alternate YD cooling to 1A NV pump. **REFER TO** Enclosure 8 (Maximize NV Pump 1A Run Time).

   3) **IF** Unit 2 selected, **THEN** align alternate YD cooling to 2A NV pump. **REFER TO** Enclosure 9 (Maximize NV Pump 2A Run Time).

ATTACHMENT 3

CNS AP/0/A/5500/020	LOSS OF NUCLEAR SERVICE WATER Enclosure 1 - Page 2 of 2 **Foldout Page**	PAGE NO. 72 of 137 Revision 49

3. **Spent Fuel Pool Criteria:**

- **IF AT ANY TIME** RN cooling to operating KC Hx lost, **THEN** perform the following:

  a. **IF** either of the following Unit 1 annunciators lit, **THEN** secure KF pump(s) and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level):

  - 1AD-13, D/6 "KF PUMP A MTR CLR HI TEMP"

    OR

  - 1AD-13, D/7 "KF PUMP B MTR CLR HI TEMP".

  b. **IF** either of the following Unit 2 annunciators lit, **THEN** secure KF pump and **REFER TO** AP/2/A/5500/041 (Loss of Spent Fuel Cooling or Level):

  - 2AD-13, D/6 "KF PUMP A MTR CLR HI TEMP"

    OR

  - 2AD-13, D/7 "KF PUMP B MTR CLR HI TEMP".

ATTACHMENT 4

CNS AP/1/A/5500/010	REACTOR COOLANT LEAK Enclosure 1 - Page 1 of 1 **Case I Steam Generator Tube Leak Foldout Page**	PAGE NO. 84 of 164 Revision 62

Case I (Steam Generator Tube Leak) Safety Injection Initiation Criteria:

a. **IF** Pzr level cannot be maintained greater than 4% **OR** Pzr pressure trending down in uncontrolled manner, **THEN** perform the following:

1) Trip Unit 1 reactor.

2) **WHEN** reactor trip verified, **THEN** initiate S/I.

3) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

ATTACHMENT 5

CNS AP/1/A/5500/009	RAPID DOWNPOWER Enclosure 1 - Page 1 of 2 **Foldout Page**	PAGE NO. 30 of 40 Revision 34

1. **Turbine Trip Criteria:**

   - **IF** T-Avg less than T-Ref **AND** trending down in uncontrolled manner, **THEN** perform the following:

     a. **IF** reactor power greater than or equal to 69%, **THEN** trip reactor.

     b. Ensure turbine - TRIPPED.

     c. **IF** reactor tripped, **THEN** **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

     d. **GO TO** AP/1/A/5500/002 (Turbine Generator Trip).

   - **IF** any turbine trip parameter reached **OR** automatic turbine trip occurs, **THEN** perform the following:

     a. **IF** reactor power greater than or equal to 69%, **THEN** trip reactor.

     b. Ensure turbine - TRIPPED.

     c. **IF** reactor tripped, **THEN** **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

     d. **GO TO** AP/1/A/5500/002 (Turbine Generator Trip).

2. **Reactor Trip Criteria:**

   IF any of the following conditions exist:

   - NC T-Avg cannot be maintained greater than 551°F

     OR

   - Both of the following:

     - NC T-Avg greater than 587°F
     - NC T-Avg stable or trending up.

     OR

   - Any reactor trip setpoint exceeded,

   THEN trip reactor and **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

3. **Manual Rod Control Operation Criterion:**

   - **IF** "C-5 LO TURB IMPULSE PRESS ROD BLOCK" status light (1SI-18) lit, **THEN** ensure rod control in manual.

ATTACHMENT 5

CNS AP/1/A/5500/009	RAPID DOWNPOWER Enclosure 1 - Page 2 of 2 **Foldout Page**	PAGE NO. 31 of 40 Revision 34

4. **IF AT ANY TIME prompt separation from grid required AND it is desired to carry in-house loads, THEN GO TO Section C. (Operator Actions), Step 26.**

ATTACHMENT 6

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 1 of 3 **Foldout Page**	PAGE NO. 34 of 49 Revision 46

1. **NC Pump Trip Criteria:**

   - **IF** the following conditions satisfied, **THEN** trip all NC pumps while maintaining seal injection flow:
     - Any NV or NI pump - DELIVERING S/I FLOW TO NC SYSTEM
     - NC subcooling based on core exit T/Cs - LESS THAN OR EQUAL TO 0°F
     - Reactor power - LESS THAN 5%.

2. **Open Phase Criteria:**

   - **IF** operating NV **AND** KC pumps automatically trip, **THEN** perform the following:

     a. Start the following pumps on opposite train:
        - NV pump
        - KC pumps
        - RN pump.

     b. **IF** pumps do not start, **OR** trip after starting, **THEN** restart pumps on previously operating train.

     c. **IF** all KC pumps off, **THEN** ensure all NC pumps - OFF.

     d. **IF** Unit 2 4160V bus energized by Unit 1 busline, **THEN** immediately notify Unit 2 to perform same actions on Unit 2.

3. **CA Suction Source Switchover Criterion:**

   - **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

4. **Position Criteria for 1NV-202B and 1NV-203A (NV Pumps A&B Recirc Isol):**

   - **IF** NC pressure less than 1500 PSIG **AND** NV S/I flowpath aligned, **THEN** CLOSE 1NV-202B and 1NV-203A.
   - **IF** NC pressure greater than 2000 PSIG, **THEN** OPEN 1NV-202B and 1NV-203A.

ATTACHMENT 6

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 2 of 3 **Foldout Page**	PAGE NO. 35 of 49 Revision 46

NOTE CA flow control valves fail open on CA auto start. Isolating flow with the motor operated isolation valve will not require re-isolation on subsequent CA auto starts.

5. **Ruptured S/G CA Isolation Criteria:**

- **IF** both the following conditions met, **THEN** stop CA flow to affected S/G(s):
  - Level going up in uncontrolled manner or radiation level in that S/G abnormal
  - N/R level - GREATER THAN 11% (29% ACC).

NOTE CA flow control valves fail open on CA auto start. Isolating flow with the motor operated isolation valve will not require re-isolation on subsequent CA auto starts.

6. **Faulted S/G CA isolation Criteria:**

- **IF** all the following conditions met, **THEN** stop CA flow to affected S/G:
  - S/G pressure trends down in uncontrolled manner or completely depressurized
  - Only one S/G diagnosed as faulted
  - Secondary heat sink criteria met:
    - Total CA flow - GREATER THAN 450 GPM

      OR
    - ANY S/G(s) N/R level - GREATER THAN 11% (29% ACC).

ATTACHMENT 6

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 3 of 3 **Foldout Page**	PAGE NO. 36 of 49 Revision 46

7. **NS Pump Trip Criterion:**

- **IF** NS pump in recirc and S/I occurs, **THEN** perform one of the following:
  - **IF** train affected ECCS and D/G load sequencer - RESET, **THEN** stop NS pump

    OR
  - **WHEN** sequencer loading complete, **THEN** perform the following for affected train:

    a. Notify Control Room Supervisor.

    b. Reset ECCS.

    c. Reset D/G load sequencer.

    d. Secure NS pump.

    e. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.

8. **IF AT ANY TIME KC cooling to operating KF pump(s) lost, THEN perform the following:**

- **IF** annunciator 1AD-13, D/6 "KF PUMP A MTR CLR HI TEMP" lit, **THEN** secure 1A KF pump and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).
- **IF** annunciator 1AD-13, D/7 "KF PUMP B MTR CLR HI TEMP" lit, **THEN** secure 1B KF pump and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).

ATTACHMENT 7

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 2 - Page 1 of 6 **Ventilation System Verification**	PAGE NO. 37 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **Verify proper VC/YC operation as follows:**	
a. Verify one train of the following equipment in operation: __ • YC chiller __ • CR AHU-1 __ • CRA AHU-1 __ • CRA PFT-1.	a. Perform the following: __ 1) Shift operating VC/YC trains. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 17 (Shifting Operating VC/YC Trains). 2) **IF** no train can be properly aligned, **THEN** dispatch operator and IAE/Maintenance to restore at least one train of VC/YC. **REFER TO** the following: __ • OP/0/A/6450/011 (Control Room Area Ventilation/Chilled Water System) __ • EM/0/A/5200/001 (Troubleshooting Cause For Improper Operation of VC/YC System).
b. Verify the following alarms - DARK: __ • 1AD-18, A/8 "UNIT 1 INTAKE HI CHLORINE 1A" __ • 1AD-18, B/8 "UNIT 1 INTAKE HI CHLORINE 1B" __ • 1AD-18, D/8 "UNIT 2 INTAKE HI CHLORINE 2A" __ • 1AD-18, E/8 "UNIT 2 INTAKE HI CHLORINE 2B".	b. **IF** chlorine odor detected in Control Room, **THEN** perform the following based on the status of given alarms: 1) **IF** detectors on both unit intakes in alarm, **THEN** perform the following: a) Ensure the following VC intake dampers - CLOSED: __ • 1VC-5B (CRA Filt Inlet) __ • 1VC-6A (CRA Filt Inlet) __ • 2VC-5B (CRA Filt Inlet) __ • 2VC-6A (CRA Filt Inlet). __ b) **GO TO** Step 1.d. (RNO continued on next page)

ATTACHMENT 7

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 2 - Page 2 of 6 **Ventilation System Verification**	PAGE NO. 38 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (Continued)	
	2) **IF** Unit 1 intake HI chlorine detector(s) in alarm, **THEN** perform the following: a) Ensure the following VC dampers - CLOSED: __ • 1VC-5B (CRA Filt Inlet) __ • 1VC-6A (CRA Filt Inlet). b) Ensure the following dampers - OPEN: __ • 2VC-5B (CRA Filt Inlet) __ • 2VC-6A (CRA Filt Inlet). __ c) **GO TO** Step 1.d.
	3) **IF** Unit 2 intake Hi chlorine detector(s) in alarm, **THEN** perform the following: a) Ensure the following VC dampers - CLOSED: __ • 2VC-5B (CRA Filt Inlet) __ • 2VC-6A (CRA Filt Inlet). b) Ensure the following dampers - OPEN: __ • 1VC-5B (CRA Filt Inlet) __ • 1VC-6A (CRA Filt Inlet). __ c) **GO TO** Step 1.d.
c. Ensure the following VC dampers - OPEN: __ • 1VC-5B (CRA Filt Inlet) __ • 1VC-6A (CRA Filt Inlet) __ • 2VC-5B (CRA Filt Inlet) __ • 2VC-6A (CRA Filt Inlet).	

ATTACHMENT 7

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

1. (Continued)

   d. Repeat Step 1 of this enclosure until notified by station management as follows:

   __ • At least once every 8 hours

   OR

   __ • Any time VC/YC related annunciators on 1AD-18 actuate.

2. **Ensure proper VA System operation as follows:**

   • Ensure the following fans - OFF:

   __ • ABUXF 1A
   __ • ABUXF 1B.

   • Ensure VA system filter in service as follows:

   __ • 1ABF-D-12 & 19 (VA Filter A Bypass Dampers) - CLOSED

   __ • 1ABF-D-5 & 20 (VA Filter B Bypass Dampers) - CLOSED.

   • Ensure the following fans - ON:

   __ • ABFXF-1A
   __ • ABFXF 1B.

ATTACHMENT 7

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **Verify proper VE System operation as follows:**	
__ a. VE fans - ON.	__ a. Start fan(s).
__ b. Annulus pressure - BETWEEN -1.4 IN. WC AND -1.8 IN. WC.	b. Perform the following: 1) **IF** annulus pressure more positive than -1.4 in. WC, **THEN** perform the following: a) Verify flow indicated on the following indications: __ • 1VEP5180 (VE 1A Flow To Stack) __ • 1VEP5200 (VE 1B Flow To Stack). b) **IF** flow not indicated, **THEN** dispatch operator to verify status of the following dampers based on their local indication or their operating piston rods being extended 4" to 6": __ • 1AVS-D-2 (VE A Trn Recirc Damp) (AB-603, JJ-51, Rm 500) - CLOSED __ • 1AVS-D-7 (VE B Trn Recirc Damp) (AB-603, HH-52, Rm 500) - CLOSED __ • 1AVS-D-3 (VE A Trn Exh Damp) (AB-603, JJ-52, Rm 500) - OPEN __ • 1AVS-D-8 (VE B Trn Exh Damp) (AB-603, HH-52, Rm 500) - OPEN. (RNO continued on next page)

ATTACHMENT 7

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 2 - Page 5 of 6 **Ventilation System Verification**	PAGE NO. 41 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. (Continued)	
	__ c) Consult plant engineering staff and notify IAE/Maintenance to troubleshoot and repair. **REFER TO** EM/1/A/5200/002 (Troubleshooting Cause For VE System Hi/Lo Pressure).
	__ d) **GO TO** Step 3.c.
	2) **IF** annulus pressure more negative than -1.8 in. WC, **THEN** perform the following:
	__ a) Determine which VE train indicates highest discharge flow to stack.
	__ b) Within 2 hours, ensure VE train that indicates highest discharge flow to stack secured.
	__ c) Consult plant engineering staff and notify IAE/Maintenance to troubleshoot and repair. **REFER TO** EM/1/A/5200/002 (Troubleshooting Cause For VE System Hi/Lo Pressure).
__ c. Repeat Step 3.b every 30 minutes until notified by Station Management.	

ATTACHMENT 7

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 2 - Page 6 of 6 **Ventilation System Verification**	PAGE NO. 42 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

__ 4. **Record time ventilation systems verified on following table:**

TIME	SYSTEM (VC, VE)	INITIALS

ATTACHMENT 8

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 4 - Page 1 of 5 **NC Temperature Control**	PAGE NO. 44 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **Verify any NC pump - ON.**	**Perform the following:** a. Use NC T-Colds to determine NC temperature as required in subsequent steps. b. **GO TO** Step 4.
2. **Use NC T-Avg to determine NC temperature as required in subsequent steps.**	
3. **IF AT ANY TIME all NC pumps tripped, THEN use NC T-Colds to determine NC temperature as required in subsequent steps.**	
4. **Verify one of the following:** • NC temperature - STABLE AT LESS THAN OR EQUAL TO 557°F OR • NC temperature - TRENDING TO 557°F.	**GO TO Step 8.**
5. **Continue to monitor NC temperature.**	
6. **Notify Control Room Supervisor of NC temperature control status.**	

ATTACHMENT 8

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 4 - Page 2 of 5 **NC Temperature Control**	PAGE NO. 45 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Do not continue in this enclosure until one of the following occurs:** • NC temperature - GREATER THAN 557°F AND TRENDING UP IN AN UNCONTROLLED MANNER OR • NC temperature - GREATER THAN 557°F AND STABLE OR • NC temperature - LESS THAN 557°F AND TRENDING DOWN IN UNCONTROLLED MANNER.	
8. **Verify NC temperature - LESS THAN 557°F AND TRENDING DOWN.**	**Perform the following:** a. **IF** NC temperature greater than 557°F **AND** trending up, **THEN** stabilize NC temperature at 557°F as follows: 1) **IF** steam dumps available, **THEN** use steam dumps. 2) **IF** steam dumps not available, **THEN** use S/G PORVs. b. **IF** the following conditions exist: • NC temperature greater than 557°F and stable • Time and manpower available, **THEN** stabilize NC temperature at 557°F as follows: 1) **IF** steam dumps available, **THEN** use steam dumps. 2) **IF** steam dumps not available, **THEN** use S/G PORVs. c. **GO TO** Step 10.

ATTACHMENT 8

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 4 - Page 3 of 5 **NC Temperature Control**	PAGE NO. 46 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Attempt to stop NC cooldown as follows:**	
a. Ensure all steam dumps - CLOSED.	
b. Ensure all S/G PORVs - CLOSED.	b. **IF** any S/G PORV cannot be closed, **THEN** CLOSE its isolation valve.
c. Ensure S/G blowdown isolated.	
d. CLOSE the following valves: • 1SM-77A (S/G 1A Otlt Hdr Bldwn C/V) • 1SM-76B (S/G 1B Otlt Hdr Bldwn C/V) • 1SM-75A (S/G 1C Otlt Hdr Bldwn C/V) • 1SM-74B (S/G 1D Otlt Hdr Bldwn C/V).	
e. Verify MSR Second Stage steam supply valves - CLOSED • 1HM-1 (MSRH 1A&1B SSRH Stm Source) • 1HM-2 (MSRH 1C&1D SSRH Stm Source).	e. Perform the following: 1) CLOSE MSR Second Stage steam supply valve(s). 2) **IF** steam flowpath cannot be isolated from Control Room, **THEN** CLOSE the following valves: • All MSIVs • All MSIV bypass valves.

ATTACHMENT 8

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
f. Depress and hold "S/V BEFORE SEAT DRN" "CLOSE" pushbutton (1MC-3) to close the following valves: • 1SM-41 (Stop Vlv #1 Before Seat Drn) • 1SM-44 (Stop Vlv #2 Before Seat Drn) • 1SM-43 (Stop Vlv #3 Before Seat Drn) • 1SM-42 (Stop Vlv #4 Before Seat Drn).	
g. Verify NC cooldown - STOPPED.	g. **IF** cooldown continues, **THEN** THROTTLE feed flow as follows: 1) **IF** S/G N/R level less than 11% (29% ACC) in all S/G's, **THEN** THROTTLE feed flow to achieve the following: • Minimize cooldown • Maintain total feed flow greater than 450 GPM. 2) **WHEN** N/R level greater than 11% (29% ACC) in any S/G, **THEN** THROTTLE feed flow further to achieve the following: • Minimize cooldown • Maintain at least one S/G N/R level greater than 11% (29% ACC). 3) **IF** cooldown continues, **THEN** CLOSE the following valves: • All MSIVs • All MSIV bypass valves.

ATTACHMENT 8

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 4 - Page 5 of 5 **NC Temperature Control**	PAGE NO. 48 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Continue to perform actions of this enclosure as required to ensure one of the following:** • NC temperature - STABLE AT LESS THAN OR EQUAL TO 557°F OR • NC temperature - TRENDING TO 557°F.	
11. **Notify Control Room Supervisor of NC temperature control status.**	

ATTACHMENT 9

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE Enclosure 1 - Page 1 of 2 **Foldout Page**	PAGE NO. 71 of 95 Revision 44

1. **NC Pump Trip Criteria:**

   - **IF** the following conditions satisfied, **THEN** trip all NC pumps while maintaining seal injection flow:
     - Any NV or NI pump - ON
     - NC subcooling based on core exit T/Cs - LESS THAN OR EQUAL TO 0°F.

2. **Position Criteria for 1NV-202B and 1NV-203A (NV Pumps A&B Recirc Isol):**

   - **IF** NC pressure less than 1500 PSIG **AND** NV S/I flowpath aligned, **THEN** CLOSE 1NV-202B and 1NV-203A.
   - **IF** NC pressure greater than 2000 PSIG, **THEN** OPEN 1NV-202B and 1NV-203A.

3. **S/I Reinitiation Criteria:**

   - **IF** NC subcooling based on core exit T/Cs less than 0°F **OR** Pzr level cannot be maintained greater than 11% (30% ACC), **THEN** perform the following:

     a. Perform the following as necessary to maintain subcooling greater than 0°F and Pzr level greater than 11% (30% ACC):

        1) Start one or more S/I pumps.

        2) Align NV S/I flow path. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 14 (NV Alignment To S/I Mode).

     b. **IF** S/I reinitiation occurs after Section C. (Operator Actions), Step 24, **THEN GO TO** EP/1/A/5000/ECA-3.1 (SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired).

4. **Secondary Integrity Criteria:**

   - **IF** any unisolated S/G pressure trending down in uncontrolled manner **OR** has completely depressurized, **THEN GO TO** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation) unless needed for NC System cooldown.

5. **Cold Leg Recirc Switchover Criterion:**

   - **IF** FWST level lowers to 20% (1AD-9, D/8 "FWST 2/4 LO LEVEL"), **THEN GO TO** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation).

ATTACHMENT 9

CNS EP/1/A/5000/E-3	STEAM GENERATOR TUBE RUPTURE Enclosure 1 - Page 2 of 2 **Foldout Page**	PAGE NO. 72 of 95 Revision 44

6. **CA Suction Source Switchover Criterion:**

   - **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

7. **Multiple Tube Rupture Criteria:**

   - **IF** level in any intact S/G trends up in uncontrolled manner **OR** any intact S/G indicates abnormal radiation, **THEN** perform the following:

     a. Stop any operator controlled cooldown and depressurization in progress.

     b. **RETURN TO** EP/1/A/5000/E-3 (Steam Generator Tube Rupture) Step 1.

ATTACHMENT 10

CNS EP/1/A/5000/ECA-3.1	SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED Enclosure 1 - Page 1 of 1 **Foldout Page**	PAGE NO. 67 of 102 Revision 40

1. **S/I Reinitiation Criteria:**

   - **IF** NC subcooling based on core exit T/Cs less than 0°F **OR** Pzr level cannot be maintained greater than 11% (30% ACC), **THEN** perform the following as necessary to restore subcooling and Pzr level:
     - Start one or more S/I pumps
     - Align NV S/I flow path. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 14 (NV Alignment To S/I Mode).

2. **Secondary Integrity Criteria:**

   - **IF** any unisolated S/G pressure trending down in uncontrolled manner **OR** has completely depressurized, **THEN GO TO** EP/1/A/5000/E-2 (Faulted Steam Generator Isolation) unless needed for NC System cooldown.

3. **Cold Leg Recirc Switchover Criterion:**

   - **IF** FWST level lowers to 20% (1AD-9, D/8 "FWST 2/4 LO LEVEL"), **THEN GO TO** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation).

4. **CA Suction Source Switchover Criterion:**

   - **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

5. **Position Criteria for 1NV-202B and 1NV-203A (NV Pumps A&B Recirc Isol):**

   - **IF** NC pressure less than 1500 PSIG **AND** NV S/I flowpath aligned, **THEN** CLOSE 1NV-202B and 1NV-203A.
   - **IF** NC pressure greater than 2000 PSIG, **THEN** OPEN 1NV-202B and 1NV-203A.

ATTACHMENT 11

Event #2

1B RN Pump Trip

3.7 PLANT SYSTEMS

3.7.8 Nuclear Service Water System (NSWS)

LCO 3.7.8 Two NSWS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------ Not applicable while in Condition C of this LCO unless entry is directed by Note 2 of Condition C. ------------------------------- One NSWS train inoperable.	A.1 ------------NOTES------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for emergency diesel generator made inoperable by NSWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for residual heat removal loops made inoperable by NSWS. -------------------------------- Restore NSWS train to OPERABLE status.	72 hours

(continued)

ATTACHMENT 11

Event #2

1B RN Pump Trip

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. -----------NOTES----------- 1. Entry into this Condition shall only be allowed for pre-planned activities as described in the Bases of this Specification. 2. Immediately enter Condition A of this LCO if one or more NSWS components become inoperable while in this Condition and one NSWS train remains OPERABLE. 3. Immediately enter LCO 3.0.3 if one or more NSWS components become inoperable while in this Condition and no NSWS train remains OPERABLE. ------------------------------- One NSWS supply header inoperable due to NSWS being aligned for single supply header operation.	B.1 Restore NSWS supply header to OPERABLE status.	30 days

(continued)

ATTACHMENT 11

Event #2

1B RN Pump Trip

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----------NOTES----------- 1. Entry into this Condition shall only be allowed for Unit 1 and for pre-planned activities as described in the Bases of this Specification. Entry into this Condition shall not be allowed while Unit 2 is in MODE 1, 2, 3, or 4. 2. Immediately enter Condition A of this LCO if one or more Unit 1 required NSWS components become inoperable while in this Condition and one NSWS train remains OPERABLE. 3. Immediately enter LCO 3.0.3 if one or more Unit 1 required NSWS components become inoperable while in this Condition and no NSWS train remains OPERABLE. ------------------------------- One NSWS train inoperable due to NSWS being aligned for single Auxiliary Building discharge header operation.	C.1 Restore NSWS train to OPERABLE status.	14 days

(continued)

ATTACHMENT 11

Event #2
1B RN Pump Trip

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. -----------NOTES----------- 1. Entry into this Condition shall only be allowed for pre-planned activities. 2. Immediately enter Condition A of this LCO if one or more NSWS components become inoperable while in this Condition and one NSWS train remains OPERABLE. 3. Immediately enter LCO 3.0.3 if one or more NSWS components become inoperable while in this Condition and no NSWS train remains OPERABLE. 4. Entry into this Condition shall only be allowed for 60 days per 12-month period. -------------------------------- One NSWS Pond return header inoperable due to NSWS being aligned for single Pond return header operation.	D.1 Restore NSWS Pond return header to OPERABLE status.	30 days

(continued)

ATTACHMENT 11

Event #2

1B RN Pump Trip

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	------NOTE------ Isolation of NSWS flow to individual components does not render the NSWS inoperable. ------------ Verify each NSWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	------NOTE------ Not required to be met for valves that are maintained in position to support NSWS single supply header operation, single Auxiliary Building discharge header operation, or single Pond return header operation. ------------ Verify each NSWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each NSWS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

ATTACHMENT 11

Event #4
1C S/G Tube Leak

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day (Unit 1) and 45 gallons per day (Unit 2) primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

ATTACHMENT 11

Event #4
1C S/G Tube Leak

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	----------NOTES---------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. -------------------- Verify RCS Operational LEAKAGE within limits by performance of RCS water inventory balance.	------NOTE------ Only required to be performed during steady state operation -------------------- In accordance with the Surveillance Frequency Control Program
SR 3.4.13.2	----------NOTE---------- Not required to be performed until 12 hours after establishment of steady state operation. -------------------- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day (Unit 1) and $\leq$ 45 gallons per day (Unit 2) through any one SG.	--------NOTE-------- Only required to be performed during steady state operation -------------------- In accordance with the Surveillance Frequency Control Program

ATTACHMENT 11

Event #4
1C S/G Tube Leak

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next inspection. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection

(continued)

Event #4
1C S/G Tube Leak

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.18.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

ATTACHMENT 11

Event #4
1C S/G Tube Leak

16.7 INSTRUMENTATION

16.7-9 Standby Shutdown System (SSS)

COMMITMENT The SSS shall be FUNCTIONAL.

APPLICABILITY: MODES 1, 2, and 3.

REMEDIAL ACTIONS

--NOTE--
SLC 16.2.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SSS non-functional.	A.1 Restore SSS to FUNCTIONAL status.	7 days
B. Total accumulative LEAKAGE from unidentified LEAKAGE, identified LEAKAGE, and reactor coolant pump seal LEAKAGE > 20 gpm.	B.1 Declare the standby makeup pump non-functional and enter Condition A.	Immediately
C. A required cell in a 24-Volt battery bank is < 1.36 volts on float charge.	C.1 Enter Condition A.	Immediately
D. Required Action and associated Completion Time of Condition A not met.	D.1 Prepare and submit a Special Report to the NRC outlining the extent of repairs required, schedule for completing repairs, and basis for continued operation.	14 days

ATTACHMENT 11

Event #4

1C S/G Tube Leak

TESTING REQUIREMENTS

TEST		FREQUENCY
TR 16.7-9-1	Verify that the electrolyte level of each SSS diesel starting 24-Volt battery is $\geq$ the low mark and $\leq$ the high mark.	7 days
TR 16.7-9-2	Verify that the overall SSS diesel starting 24-Volt battery voltage is $\geq$ 24 volts on float charge.	7 days
TR 16.7-9-3	Verify that the requirements of SLC 16.9-21 are met and the boron concentration in the storage pool is $\geq$ the minimum specified in the COLR.	7 days
TR 16.7-9-4	Verify the fuel level in the SSS diesel generator fuel storage tank is $\geq$ 67 inches.	31 days
TR 16.7-9-5	Verify the SSS diesel generator starts from ambient conditions and operates for $\geq$ 30 minutes at $\geq$ 700 kW.	31 days
TR 16.7-9-6	Verify that the electrolyte level of each SSS 250/125-Volt battery is above the plates.	31 days
TR 16.7-9-7	Verify the total SSS 250/125-Volt battery terminal voltage is $\geq$ 258/129 volts on float charge.	31 days
TR 16.7-9-8	Perform CHANNEL CHECK of each SSS instrumentation device.	31 days
TR 16.7-9-9	Verify the fuel oil properties of new and stored fuel oil for the SSS diesel generator are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
TR 16.7-9-10	Verify that the individual battery cell voltage of the required cells in the SSS diesel starting 24-Volt battery is $\geq$ 1.36 volts on float charge.	92 days

(continued)

ATTACHMENT 11

Event #4

1C S/G Tube Leak

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.7-9-11	Verify that the Standby Makeup Pump's developed head at the test flow point is $\geq$ the required developed head, in accordance with the Inservice Testing Program.	92 days
TR 16.7-9-12	Verify that the specific gravity of the SSS 250/125-Volt battery is appropriate for continued service of the battery.	92 days
TR 16.7-9-13	Subject the SSS diesel generator to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.	18 months
TR 16.7-9-14	Verify that the SSS diesel starting 24-Volt batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration.	18 months
TR 16.7-9-15	Verify that the SSS diesel starting 24-Volt battery-to-battery and terminal connections are clean, tight, and free of corrosion.	18 months
TR 16.7-9-16	Verify that the SSS 250/125-Volt batteries, cell plates, and battery racks show no visual indications of physical damage or abnormal deterioration.	18 months
TR 16.7-9-17	Verify that the SSS 250/125-Volt battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.	18 months
TR 16.7-9-18	Verify that the steam turbine driven auxiliary feedwater pump and controls from the Standby Shutdown Facility function as designed from the SSS.	18 months
TR 16.7-9-19	Perform CHANNEL CALIBRATION of each SSS instrumentation device.	18 months

(continued)

ATTACHMENT 11

Event #4
1C S/G Tube Leak

TESTING REQUIREMENTS (continued)

TEST		FREQUENCY
TR 16.7-9-20	Verify proper installation of pressurizer insulation.	18 months
TR 16.7-9-21	Verify pressurizer heaters powered from the SSS have a capacity of $\geq$ 65 kW measured at motor control center SMXG.	18 months
TR 16.7-9-22	Verify flowpath from the reactor vessel head through the valves powered from the SSS is unobstructed.	18 months

BASES

The SSS is designed to mitigate the consequences of certain postulated fire, security, and station blackout incidents by providing capability to maintain MODE 3 conditions and by controlling and monitoring vital systems from locations external to the main control room. This capability is consistent with the requirements of 10 CFR Part 50.48(c).

When the SSS is under Condition A and it is anticipated that Condition D will be utilized, establish the bases for continued operation (including any supporting actions) prior to entering Condition D. Risks associated with the continued operation under Condition D are evaluated and managed through existing processes and procedures. These risk contributors, risk insights, risk-informed information, and/or risk mitigation actions assessed and managed during periods when Condition D is applied, are to be included in the 14-day special report.

The TESTING REQUIREMENTS ensure that the SSS systems and components are capable of performing their intended functions. The required level in the SSS diesel generator fuel storage tank ensures sufficient fuel for 72 hours uninterrupted operation. It is assumed that, within 72 hours, either offsite power can be restored or additional fuel can be added to the storage tank.

Although the standby makeup pump is not nuclear safety related and was not designed according to ASME Code requirements, it is tested quarterly to ensure its FUNCTIONALITY. The TESTING REQUIREMENT concerning the standby makeup pump water supply ensures that an adequate water volume is available to supply the pump continuously for 72 hours.

Total accumulative LEAKAGE is calculated in the NC System Leakage Calculation procedure as identified + unidentified + seal leakoff (References 2 and 3). The REMEDIAL ACTION limit of 20 gpm total accumulative LEAKAGE provides additional margin to allow for

BASES (continued)

instrument inaccuracy, and for the predicted increase in seal leakoff rate due to heatup of the reactor coolant pump seal injection water supply temperature following the SSS event (due to spent fuel pool heatup). Following the increase in seal injection temperature, the standby makeup pump flow of 26 gpm is sufficient to provide in excess of this total accumulative LEAKAGE, thereby assuring that reactor coolant system inventory is maintained at MODE 3 conditions. The supporting evaluation is provided in CNC-1223.04-00-0072 (Ref. 4).

A visual inspection of the diesel starting 24-volt batteries, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. Since the battery cell jars are not transparent, a direct visual inspection of the cell plates cannot be performed. Instead, the cell plates are inspected for physical damage and abnormal deterioration by: 1) visually inspecting the jar sides of each cell for excessive bowing and/or deformation, and 2) visually inspecting the electrolyte of each cell for abnormal appearance.

Verifying individual cell voltage while on float charge for the SSS diesel starting 24-Volt batteries ensures that each cell is capable of supporting its intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or battery cell) in a fully charged state. The battery cell voltage limit of 1.36 volts is consistent with the nominal design voltage of the battery and is based on the manufacturer's recommended minimum float charge voltage for a fully charged cell with adequate capacity. The 24-Volt starting battery is designed with two battery banks, each battery bank contains 20 individual battery cells. The 24-Volt starting battery has sufficient capacity margin to maintain SSS diesel starting functionality with one cell in each battery bank to be fully degraded with a voltage < 1.36 volts. The 24-Volt starting battery is required to have 19 individual battery cells per battery bank to maintain SSS diesel starting functionality with sufficient capacity margin. The battery sizing calculation accounts for one degraded cell in each battery bank by assuming the degraded cells undergo a worst case polarity reversal during SSS diesel starting. The supporting evaluation is provided in CNC-1381.06-00-0056 (Ref.12).

Verification of proper installation of pressurizer insulation ensures that pressurizer heat losses during an SSS event do not exceed the capacity of the pressurizer heaters powered from the SSS.

Testing of the pressurizer heater capacity ensures the full capacity of the heaters is available to maintain a steam bubble in the pressurizer during an SSS event. The acceptance criterion includes an allowance for the voltage drop in the power cables between the SSS and the pressurizer.

ATTACHMENT 11

Event #4

1C S/G Tube Leak

BASES (continued)

Testing of the flowpath from the reactor vessel head to the pressurizer relief tank ensures sufficient flow capacity for reactor coolant inventory control during an SSS event.

REFERENCES

1. Letter from NRC to Gary R. Peterson, Duke, Issuance of Improved Technical Specifications Amendments for Catawba, September 30, 1998.
2. PT/1(2)/A/4150/001D, NC System Leakage Calculation.
3. PT/1(2)/A/4150/001I, Manual NC Leakage Calculation.
4. CNC-1223.04-00-0072, Reactor Coolant Pumps No. 1 Seal Leakoff Annunciator Alarm Setpoint for Unit 1 and Unit 2.
5. CNS-1560.SS-00-0001, Design Basis Specification for the Standby Shutdown Facility.
6. Catawba Technical Specification Amendments 206/200, July 10, 2003.
7. Catawba UFSAR, Section 18.2.4.
8. Catawba License Renewal Commitments, CNS-1274.00-00-0016, Section 4.5.
9. CNC-1223.03-00-0033, Determination of Pressurizer Heater Capacity Powered from the SSF Diesel.
10. Catawba Nuclear Station 10 CFR 50.48(c) Fire Protection Safety Evaluation (SE).
11. 10 CFR 50.48(c), Fire Protection.
12. CNC-1381.06-00-0056, SSF Diesel Generator Battery Sizing Calculation.

Shifting Operating Lower Containment Ventilation Units And Pipe Tunnel Booster Fans

1. Limits and Precautions

1.1 Observe the upper and lower containment temperature limits of Tech Spec 3.6.5.

1.2 All operating lower containment ventilation units, and the operating pipe tunnel booster fan, are normally operated at the same speed.

1.3 When CLAs are pressurized above 175 psig, lower containment temperature shall be maintained greater than 60°F to maintain accumulator temperature greater than 60°F due to brittle fracture concerns of the accumulator vessel.

2. Initial Conditions

None

3. Procedure

_____ 3.1 **IF** two LCVUs are operating **AND** it is desired, shift the operating units as follows:

_____ 3.1.1 **IF** the LCVUs are operating in "LOW" speed, perform the following:

NOTE:	If two LCVUs are to remain in service, it is preferable to run a vent unit in each fan room (A/D, B/C) in order to maximize air distribution in the lower containment. Due to the temperature characteristics in lower containment and the Digital Rod Position Indication (DRPI) Panels area, use of the 1D LCVU is preferred. Failure to operate with at least one vent unit in each fan room during Modes 1 - 3 can result in high pressurizer and/or steam generator cavity air temperatures.

3.1.1.1 Start an idle LCVU by placing its control switch in the "LOW" position:

_____ • "VV LCVU 1A"
_____ • "VV LCVU 1B"
_____ • "VV LCVU 1C"
_____ • "VV LCVU 1D"

_____ 3.1.1.2 Verify the red indicating light illuminates for the LCVU placed in service.

_____ 3.1.1.3 Verify the red "OPEN" indicating light illuminates for the LCVU dampers associated with the LCVU placed in service.

_____ 3.1.1.4 Verify the red "MAX" indicating light illuminates for the LCVU placed in service.

3.1.1.5 Stop the LCVU to be removed from service by placing its control switch in the "OFF" position:

- _____ • "VV LCVU 1A"
- _____ • "VV LCVU 1B"
- _____ • "VV LCVU 1C"
- _____ • "VV LCVU 1D"

NOTE: The procedure may continue up to and including Step 3.1.1.9 before completing the following step.

_____ 3.1.1.6 Verify the green indicating light illuminates for the LCVU stopped.

_____ 3.1.1.7 Verify the red "MAX" indicating light extinguishes for the LCVU stopped.

_____ 3.1.1.8 Verify the green "CLOSED" indicating light illuminates for the LCVU damper associated with the LCVU stopped.

_____ 3.1.2 **IF** the LCVUs are operating in "HIGH" speed, perform the following:

CAUTION: Operating the lower containment vent units in "HIGH" speed for more than 24 hours will cause bearing problems in the fans.

NOTE:
- It is preferable to run a vent unit in each fan room (A/D, B/C) in order to maximize air distribution in the lower containment. Due to the temperature characteristics in lower containment and the Digital Rod Position Indication (DRPI) Panels area, use of the 1D LCVU is preferred. Failure to operate with at least one vent unit in each fan room during Modes 1 - 3 can result in high pressurizer and/or steam generator cavity air temperatures.
- Operating the lower containment vent units in "HIGH" speed will fail the bypass chilled water valves open.

3.1.2.1 Start an idle LCVU by placing its control switch in the "HIGH" position:

- _____ • "VV LCVU 1A"
- _____ • "VV LCVU 1B"
- _____ • "VV LCVU 1C"
- _____ • "VV LCVU 1D"

_____ 3.1.2.2 Verify the red indicating light illuminates for the LCVU placed in service.

_____ 3.1.2.3 Verify the red "OPEN" indicating light illuminates for the LCVU damper associated with the LCVU started.

_____ 3.1.2.4 Verify the red "OPEN" light illuminates for the valve corresponding to the LCVU started (rear of 1MC3):

- ☐ "1RN-473 LWR CONT VENT UNT 1A FULL FLOW"
- ☐ "1RN-455 LWR CONT VENT UNT 1B FULL FLOW"
- ☐ "1RN-447 LWR CONT VENT UNT 1C FULL FLOW"
- ☐ "1RN-481 LWR CONT VENT UNT 1D FULL FLOW"

3.1.2.5 Stop the LCVU to be removed from service by placing its control switch in the "OFF" position:

- _____ "VV LCVU 1A"
- _____ "VV LCVU 1B"
- _____ "VV LCVU 1C"
- _____ "VV LCVU 1D"

NOTE: The procedure may continue up to and including Step 3.1.2.8 before completing the following step.

_____ 3.1.2.6 Verify the green indicating light illuminates for the LCVU stopped.

_____ 3.1.2.7 Verify the green "CLOSED" indicating light illuminates for the LCVU damper associated with the LCVU stopped.

_____ 3.1.2.8 Verify the green "CLOSED" light illuminates for the valve corresponding to the LCVU stopped (rear of 1MC3):

- ☐ "1RN-473 LWR CONT VENT UNT 1A FULL FLOW"
- ☐ "1RN-455 LWR CONT VENT UNT 1B FULL FLOW"
- ☐ "1RN-447 LWR CONT VENT UNT 1C FULL FLOW"
- ☐ "1RN-481 LWR CONT VENT UNT 1D FULL FLOW"

_____ 3.1.2.9 Inform Engineering that lower containment vent units have been placed in "HIGH" speed.

Engineer notified ______________________________________

_____ 3.2 **<u>IF</u>** three LCVUs are operating **<u>AND</u>** it is desired, shift the operating units as follows:

_____ 3.2.1 **<u>IF</u>** the LCVUs are operating in "LOW" speed, perform the following:

3.2.1.1 Stop the LCVU to be removed from service by placing its control switch in the "OFF" position:

- _____ "VV LCVU 1A"
- _____ "VV LCVU 1B"
- _____ "VV LCVU 1C"
- _____ "VV LCVU 1D"

NOTE:	The procedure may continue up to and including Step 3.2.1.9 before completing the following step.

_____ 3.2.1.2 Verify the green indicating light illuminates for the LCVU stopped.

_____ 3.2.1.3 Verify the red "MAX" indicating light extinguishes for the LCVU stopped.

_____ 3.2.1.4 Verify the green "CLOSED" indicating light illuminates for the LCVU damper associated with the LCVU stopped.

NOTE:

- If Unit 1 is in Mode 1 and LCVU 1C or 1D is the idle unit that is being placed in service, a delay of approximately 15 to 30 minutes may be needed before starting LCVU 1C or 1D to allow lower containment air temperature to trend up. This will prevent exceeding the Tech Spec low limit for air temperature. {PIP 00-0763, PIP 05-3785}
- Adequate margin of VQ pressure may be required to allow containment temperature to trend up if waiting 15 to 30 minutes to start LCVU 1C or 1D.

3.2.1.5 Start the idle LCVU by placing its control switch in the "LOW" position:

- _____ "VV LCVU 1A"
- _____ "VV LCVU 1B"
- _____ "VV LCVU 1C"
- _____ "VV LCVU 1D"

_____ 3.2.1.6 Verify the red indicating light illuminates for the LCVU placed in service.

_____ 3.2.1.7 Verify the red "OPEN" indicating light illuminates for the LCVU damper associated with the LCVU started.

_____ 3.2.1.8 Verify the red "MAX" indicating light illuminates for the LCVU placed in service.

_____ 3.2.2 **IF** the LCVUs are operating in "HIGH" speed, perform the following:

3.2.2.1 Stop the lower containment vent unit to be removed from service by placing its control switch in the "OFF" position:

- _____ "VV LCVU 1A"
- _____ "VV LCVU 1B"
- _____ "VV LCVU 1C"
- _____ "VV LCVU 1D"

NOTE: The procedure may continue up to and including Step 3.2.2.8 before completing the following step.

_____ 3.2.2.2 Verify the green indicating light illuminates for the LCVU stopped.

_____ 3.2.2.3 Verify the green "CLOSED" indicating light illuminates for the LCVU damper associated with the LCVU stopped.

_____ 3.2.2.4 Verify the green "CLOSED" light illuminates for the valve corresponding to the LCVU stopped (rear of 1MC3):

- ☐ "1RN-473 LWR CONT VENT UNT 1A FULL FLOW"
- ☐ "1RN-455 LWR CONT VENT UNT 1B FULL FLOW"
- ☐ "1RN-447 LWR CONT VENT UNT 1C FULL FLOW"
- ☐ "1RN-481 LWR CONT VENT UNT 1D FULL FLOW"

CAUTION: Operating the lower containment vent units in "HIGH" speed for more than 24 hours will cause bearing problems in the fans.

NOTE: Operating the lower containment vent units in "HIGH" speed will fail the bypass chilled water valves open.

3.2.2.5 Start the idle lower containment vent unit by placing its control switch in the "HIGH" position:

- _____ "VV LCVU 1A"
- _____ "VV LCVU 1B"
- _____ "VV LCVU 1C"
- _____ "VV LCVU 1D"

_____ 3.2.2.6 Verify the red indicating light illuminates for the LCVU placed in service.

_____ 3.2.2.7 Verify the red "OPEN" indicating light illuminates for the LCVU damper associated with the LCVU started.

_____ 3.2.2.8 Verify the red "OPEN" light illuminates for the valve corresponding to the LCVU started (rear of 1MC3):

- ☐ "1RN-473 LWR CONT VENT UNT 1A FULL FLOW"
- ☐ "1RN-455 LWR CONT VENT UNT 1B FULL FLOW"
- ☐ "1RN-447 LWR CONT VENT UNT 1C FULL FLOW"
- ☐ "1RN-481 LWR CONT VENT UNT 1D FULL FLOW"

_____ 3.2.2.9 Inform Engineering that lower containment vent units have been placed in "HIGH" speed.

Engineer notified ______________________________________

_____ 3.3 **IF** shifting the operating pipe tunnel booster fan, perform the following:

3.3.1 Stop the operating fan by placing its control switch in the "OFF" position:

_____ • "PIPE TUNNEL BSTR FAN 1A"
_____ • "PIPE TUNNEL BSTR FAN 1B"

_____ 3.3.2 Verify the green indicating light illuminates for the pipe tunnel booster fan stopped.

_____ 3.3.3 **IF** the operating lower containment ventilation units are running in "LOW" speed, start the pipe tunnel booster fan to be placed in service by placing its control switch in the "LOW" speed position:

_____ • "PIPE TUNNEL BSTR FAN 1A"
_____ • "PIPE TUNNEL BSTR FAN 1B"

_____ 3.3.4 **IF** the operating lower containment ventilation units are running in "HIGH" speed, start the pipe tunnel booster fan to be placed in service by placing its control switch in the "HIGH" speed position:

_____ • "PIPE TUNNEL BSTR FAN 1A"
_____ • "PIPE TUNNEL BSTR FAN 1B"

_____ 3.3.5 Verify that the red indicating light illuminates for the pipe tunnel booster fan placed in service.

_____ 3.4 Indicate below the operating Pipe Tunnel Booster Fan:

- ☐ "PIPE TUNNEL BSTR FAN 1A"
- ☐ "PIPE TUNNEL BSTR FAN 1B"

_____ 3.5 Indicate below the operating LCVUs:

- ☐ "VV LCVU 1A"
- ☐ "VV LCVU 1B"
- ☐ "VV LCVU 1C"
- ☐ "VV LCVU 1D"

3.6 File a copy of this enclosure in the designated storage cabinet.

2021 INITIAL LICENSE NRC EXAM
SCENARIO # 3

Facility: Catawba NRC Exam 2021	Scenario No.: 3	Op Test No.: 2021301	
Examiners:		Operators:	SRO
			RO
			BOP
Initial Conditions: Unit 1 is at 100% power at the BOL. Unit 2 is at 100% power.			
Turnover: Unit 1 is at 100% power at the BOL. Unit 2 is at 100% power. 1B CA Pump is removed from service for PMs. 1B CA Pump has been inoperable for 3 hours and is expected to be returned to service in 6 hours. Direction for the crew is to initiate a downpower to ~85% in preparation for performing the Turbine Control Valve Movement PT.			

Event No.	Malf. No.	Event Type*	Event Description
1		R – RO N – BOP N – SRO	Unit 1 Downpower to ~85%
2		C – BOP C – SRO	1NV-294 fails open
3		C – RO C – SRO TS – SRO	1NC-32B fails open, able to be manually closed
4		C – BOP C – SRO TS – SRO	1ETA Blackout (D/G does not start)
5		C – RO C – SRO	Continuous rod motion / 2 stuck rods on reactor trip
6		M – ALL	CAPT#1 Overspeed Trip / Loss of Heat Sink
7		C – RO C – SRO	Loss of CFPT Vacuum
8		C – BOP C – SRO	1NV-37A failed closed

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Scenario 3 – Summary

Initial Condition
Unit 1 is at 100% power at the BOL. Unit 2 is at 100% power.

Turnover:
Unit 1 is at 100% power at the BOL. Unit 2 is at 100% power. 1B CA Pump is removed from service for PMs. 1B CA Pump has been inoperable for 3 hours and is expected to be returned to service in 6 hours. Direction for the crew is to initiate a downpower to ~85% in preparation for performing the Turbine Control Valve Movement PT.

Event 1
Unit 1 downpower to ~85%. BOP will perform a boration per OP/1/A/6150/009 (Boron Concentration Control) Encl. 4.2 (Boration). RO will input a target load and load rate into the Main Turbine control panel and initiate the downpower per OP/1/B/6300/001 (Turbine Generator).

Event History: This downpower at BOL has not been performed before.

Event 2
1NV-294 (Charging flow control valve) will fail full open.

Verifiable Action – BOP will place 1NV-294 in manual and will manually control charging flow to stabilize Pressurizer Level.

Event History: This failure has not been used before on an NRC exam.

Event 3
Pressurizer PORV 1NC-32B fails open. RO will manually close 1NC-32B. Crew will enter AP/1/A/5500/011 (Pressurizer Pressure Anomalies) Case 1 (Pressurizer Pressure Decreasing). SRO will refer to Tech Specs.

Verifiable Action – RO will manually close 1NC-32B.

Event History: This Pressurizer PORV failure last used on 17(1), but was not able to be manually closed.

Event 4
A Blackout will occur on essential bus 1ETA. D/G 1A will fail to start resulting in no power on 1ETA. Crew will enter AP/1/A/5500/007 (Loss of Normal Power) Case 2 (Loss of All Power to an Essential Train) to address the failure. SRO will refer to Tech Specs.

Verifiable Action – The BOP will start an RN pump and KC pumps on 'B' train on Unit 1. RO will maintain reactor power.

Event History: This failure has not been used before on an NRC exam.

Event 5
Control rods will begin to insert continuously. RO will perform the immediate actions of AP/1/A/5500/015 (Rod Control Malfunctions) Case II (Continuous Rod Movement) and trip the Unit 1 reactor.

Verifiable Action – RO will perform the immediate actions of AP/1/A/5500/015 (Rod Control Malfunctions) Case 2 (Continuous Rod Movement), and manually trip Unit 1 Reactor. During the reactor trip, 2 control rods will fail to fully insert. This will require the BOP to initiate emergency boration in EP/1/A/5000/ES-0.1 (Reactor Trip Response) to maintain shutdown margin.

Event History: Continuous rod movement last used in 16 (3). Having 2 stuck rods on reactor trip last used in 19 (1).

Event 6

Following reactor trip, the crew will enter EP/1/A/5000/E-0 (Reactor Trip or Safety Injection) and subsequently transition to EP/1/A/5000/ES-0.1 (Reactor Trip Response). CAPT#1 will eventually trip on overspeed. Once S/G N/R levels lower to < 11% in all S/Gs, crew will transition to EP/1/A/5000/FR-H.1 (Loss of Secondary Heat Sink).

Verifiable Action – BOP will trip the NC pumps and RO will depressurize S/Gs to allow feed flow to be provided to the S/Gs via Hotwell and Condensate Booster Pumps.

Event History: Similar failure was last used in 16 (2).

Event 7

Following reactor trip, both CFPTs will lose vacuum.

Verifiable Action – RO will manually depressurize the Steam Generators in FR-H.1 to allow them to be fed by the Main Condensate pumps.

Event History: Loss of CFPT vacuum failure last used in 19 (1) but was prior to the reactor trip.

Event 8

When step is reached in FR-H.1 to initiate NC system depressurization, Auxiliary Spray Valve 1NV-37A will not open.

Verifiable Action – BOP will use a Pressurizer PORV to lower NC system pressure.

Event History: This failure was last used in 16 (2).

Manual Control of Automatic Functions

Event	Position	Description
2	BOP	1NV-294 Failing open requiring manual control of Pressurizer Level

<u>Critical Task 1</u> – Control charging line flow to prevent a reactor trip on Pressurizer Hi level (2/3 Pressurizer Levels <u>></u> 92%).

<u>Critical Task 2</u> – Close Pressurizer PORV prior to a reactor trip on Pressurizer low pressure.

<u>Critical Task 3</u> – Establish feedwater flow to at least one S/G before NC feed and bleed is required (<24% W/R level in 3 out of 4 S/G).

Target Quantitative Attributes (Per Scenario; See Section D.5.d)	Actual Attributes
1. Total malfunctions (5–8)	7
2. Malfunctions after EOP entry (1–2)	2
3. Abnormal events (2–4)	4
4. Major transients (1–2)	1
5. EOPs entered/requiring substantive actions (1–2)	2
6. EOP contingencies requiring substantive actions (0–2)	1
7. Critical tasks (2–3)	3

EXERCISE GUIDE WORKSHEET

1. INITIAL CONDITIONS:

   1.1 Reset to IC # 152 and load schedule file for NRC Scenario 3

START TIME: __________

✓	✓	Trigger	Instructor Action	Final	Delay	Ramp	Delete In	Event
			LOA-CA018 (RACKOUT CA PUMP 1B)	RACK-OUT				
		2	MAL-NV012E (NV VLV 294 FAIL OPEN)	ACTIVE				2
		9	MAL-NV012E (NV VLV 294 FAIL OPEN)	ACTIVE			:01	2
		4	MAL-EP008A (LOSS OF 4160V BUS ETA)	ACTIVE				4
		3	ANN-AD19-F04 (YN CRITICAL TROUBLE)	ON			:05	3
		11	VLV-NC005F (NC32B PZR PORV FAIL TO POSITION)	1		:05		3
		13	VLV-NC005F (NC32B PZR PORV FAIL TO POSITION)	1			:01	3
		5	OVR-IRX005B (SEL SW – CRD BANK SELECT MAN POS)	ON				5
		5	OVR-IRX006A (ROD MOTION SW IN POS)	ON				5
		14	MAL-IRX015K8 (PERMANENTLY STUCK ROD K8-38A)	130				5
		14	MAL-IRX015D4 (PERMANENTLY STUCK ROD D4)	220				5
		14	MAL-CA005 (CA PUMP OVERSPEED TRIP)	MECH-ANICAL	1:00			6
		14	MAL-EHC002 (TURBINE TRIP FAILURE)	AUTO			:06	6
		14	MAL-CF001A (LOSS OF CFPT 1A VACUUM)	100				7
		14	MAL-CF001B (LOSS OF CFPT 1B VACUUM)	100				
			VLV-NV014F (NV37A PRZR SPRAY LINE ISOL VLV FAIL TO POSITION)	0				8
		Ensure red cover placed on 1B CA Pump						
		Ensure EVENT 9 = ov_d9mod111slimbytes(7) (1NV-294 Manual PB)						
		Ensure EVENT 11 = x5ri017a (AD-19 Alarm Acknowledge PB)						
		Ensure EVENT 13 = x10i279c						
		Ensure EVENT 14 = jpplp4(1) │ jpplp4(2) (Reactor Trip Either Train)						
		Ensure 1B NV pump is in service						
		Ensure control rods are in MANUAL						

2. SIMULATOR BRIEFING

2.1 Control Room Assignments:

Position	Name
CRS	
RO	
BOP	

2.2 Give a copy of Attachment 2 (Shift Turnover Information) to the CRS.

3. EXERCISE PRESENTATION

3.1 Familiarization Period

A. Allow examinees time to familiarize themselves with the Control Board alignments.

3.2 **Scenario EVENT 2**, 1NV-294 Fails Open

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 2** to cause 1NV-294 to fail open.

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for 1NV-294, **REPEAT** the information.

3.3 **Scenario EVENT 3**, YN Critical Trouble Alarm / Pressurizer PORV 1NC-32B fails open

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 3** to cause a YN critical trouble alarm.

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for 1NC-32B, **REPEAT** the information.

3.4 **Scenario EVENT 4**, 1ETA blackout

✓	BOOTH INSTRUCTOR ACTION
	WHEN directed by the lead examiner, **THEN INSERT SIMULATOR Trigger 4** to cause a blackout of 1ETA.

✓	BOOTH INSTRUCTOR ACTION
	IF the SWM is contacted to initiate an NCR or W/R for the 1ETA B/O, **REPEAT** the information.

✓	BOOTH INSTRUCTOR ACTION
	IF contacted as an AO to perform a load shed of 1ETA per Enclosure 8, wait 5 minutes and then run schedule file to load shed 1ETA.
	After 5 minutes contact the Control Room crew and **REPORT**: • "Load Shed of 1ETA per Enclosure 8 is complete. There is an 86N relay picked up on 1ETA."

3.5 **Scenario EVENT 5**, Continuous Rod Motion / 2 Stuck Rods on Reactor Trip

✓	BOOTH INSTRUCTOR ACTION
	WHEN control rods begin to insert, **THEN ENSURE SIMULATOR Trigger 5 INSERTED** to cause control rods to continuously insert.

3.6 **Scenario EVENTs 6, 7, 8**, CAPT#1 Overspeed Trip / Loss of Secondary Heat Sink / Loss of CFPT vacuum / 1NV-37A failed closed

Op Test No.: 301	Scenario # 3	Event # 1			Page 10	of	117

Event Description: Unit 1 Power Decrease

<u>Note To Evaluator</u>:

The scenario begins with a power decrease from 100% by the crew. This will involve several procedures to accomplish. The following procedures are included in this guide:

- **OP/1/A/6150/009 Enclosure 4.2 (Boration)**
- **OP/1/A/6150/008 Enclosure 4.16 (Control Bank Manual Operation At Power)**
- **OP/0/B/6300/001 Enclosure 4.2 (Load Changing)**

These procedures may be performed in any order by the crew. Instructions for continuing to the next Event are included at the end of OP/0/B/6300/001 Enclosure 4.2.

Enclosure 4.2

Boration

OP/1/A/6150/009
Page 1 of 6

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can affect core reactivity by changing boron concentration. (R.M.)

1.2 The following Limits and Precautions are Reactivity Management related: (R.M.)

1.2.1 If the boron concentration is being increased in the NC System, at least one NC pump or one ND pump shall be in operation, recirculating the NC System.

1.2.2 If the unit has operated continuously for several months, significant Boron 10 depletion may have occurred. The effective boron concentration of the NC System may be lower than indicated by Chemistry samples.

1.3 Maintaining VCT pressure as low as practical during large makeups will minimize gas absorption. VCT pressure can be reduced by diverting letdown or by VCT purge.

1.4 Due to Electromagnetic Interference within the Unit 1 Reactor Makeup Control System, the Unit 1 Boric Acid Counter may sporadically count up during dilution activities. OFF indications for the Boric Acid Xfer Pumps and Closed indication for valve 1NV-238A can be used by the Reactor Operators to validate that sporadic counts are indication only. (NCR 02081372).

2. Initial Conditions

AA 2.1 **IF** in Mode 1, 2 or 3, ensure R2 reactivity management controls established AD-OP-ALL-0203 (Reactivity Management). (R.M.)

AA 2.2 Verify the NV System is in operation per OP/1/A/6200/001 (Chemical and Volume Control System).

AA 2.3 Verify sufficient RHT volume is available to receive the reactor coolant displaced during the planned boration operation.

AA 2.4 **IF** NC System boron concentration will be changed by ≥ 50 ppm, initiate PZR spray to equalize the boron concentration throughout the system by operating backup heaters per OP/0/A/6200/055 (Miscellaneous Component Operation).

3. Procedure

NOTE:	This enclosure will affect reactivity of the core and is therefore designated important to Reactivity Management per the guidelines of AD-OP-ALL-0203 (Reactivity Management). (R.M.)

AA 3.1 Ensure valves are aligned per Enclosure 4.8 (Valve Checklist).

Enclosure 4.2

Boration

OP/1/A/6150/009
Page 2 of 6

3.2 Ensure the following valve control switches in "AUTO":

____ • 1NV-238A (B/A To Blender Ctrl Vlv)

____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

____ 3.3 Ensure 1NV-238A (B/A Xfer Pmp To Blender Ctrl) controller in auto.

____ 3.4 Ensure at least one boric acid transfer pump is in "AUTO" or "ON".

☐ 3.5 Record the desired volume of boric acid to be added. __________ gallons

____ 3.6 Adjust the boric acid counter to the desired volume of boric acid to be added. (R.M.)

____ 3.7 IF the blender is set up for automatic makeup per Enclosure 4.1 (Automatic Makeup), record the setpoint of the controller for 1NV-238A (B/A Xfer Pmp To Blender Ctrl). ________gpm

____ 3.8 Place the "NC MAKEUP MODE SELECT" switch in "BORATE".

NOTE:	Boric Acid flow rates > 32 gpm may result in a boric acid flow deviation annunciator.

____ 3.9 IF required, adjust the controller for 1NV-238A (B/A Xfer Pmp To Blender Ctrl) to the desired flow.

____ 3.10 **IF AT ANY TIME** it is desired to divert letdown to the RHT manually operate 1NV-172A (3-Way Divert To VCT-RHT) as follows:

____ 3.10.1 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) to the "RHT" position.

____ 3.10.2 Ensure VCT level is monitored continuously while diverting to the RHT.

NOTE:	Procedure may continue while performing the following step.

3.10.3 **WHEN** desired VCT level is reached return 1NV-172A (3-Way Divert To VCT-RHT) to auto as follows:

____ 3.10.3.1 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) in the "VCT" position.

____ 3.10.3.2 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) in the "AUTO" position.

____ 3.11 **IF AT ANY TIME** during the makeup it becomes necessary to change the makeup flow rate, adjust the setpoint for 1NV-238A (B/A Xfer Pmp To Blender Ctrl) as necessary to achieve the desired flow.

Op Test No.:	301	Scenario #	3	Event #	1	Page	13	of 117
Event Description:	Unit 1 Power Decrease							

Enclosure 4.2
Boration

OP/1/A/6150/009
Page 3 of 6

_____ 3.12 **IF AT ANY TIME** while boration is in progress it becomes necessary **OR** it is desired to stop the boration, perform the following:

_____ 3.12.1 Place the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.12.2 Ensure the following valves close: (R.M.)

- _____ 1NV-238A (B/A To Blender Ctrl Vlv)
- _____ 1NV-186A (B/A Blender Otlt To VCT Otlt)

☐ 3.12.3 Record boric acid volume added as indicated on the Boric Acid counter.
____________ gallons

3.12.4 **WHEN** conditions allow resuming the boration, perform the following:

☐ 3.12.4.1 Determine remaining volume to be added by subtracting the amount previously added (Step 3.12.3) from the desired volume to be added (Step 3.5).

__________ - __________ = ___________ gallons
(Step 3.5) (Step 3.12.3)

_____ 3.12.4.2 Adjust boric acid counter to the volume of boric acid determined in Step 3.12.4.1. (R.M.)

_____ 3.12.4.3 Place the "NC MAKEUP CONTROL" switch in the "START" position. (R.M.)

_____ 3.12.4.4 Verify the following:

☐ 1NV-238A (B/A To Blender Ctrl Vlv) modulates to establish desired flow
☐ 1NV-186A (B/A Blender Otlt To VCT Otlt) opens

_____ 3.12.4.5 **IF** in "AUTO", verify the boric acid pump starts.

_____ 3.13 **WHILE** makeup is in progress, monitor the following for expected results:

☐ Control rod motion
☐ NC System Tavg
☐ Reactor Power

Enclosure 4.2

Boration

OP/1/A/6150/009
Page 4 of 6

NOTE:	If a small makeup is being performed, placekeeping for Steps 3.14 through 3.17 may be performed after Step 3.18 is performed.

_____ 3.14 Place the "NC MAKEUP CONTROL" switch to the "START" position. (R.M.)

_____ 3.15 Verify the following:

- ☐ 1NV-238A (B/A To Blender Ctrl Vlv) modulates to establish desired flow
- ☐ 1NV-186A (B/A Blender Otlt To VCT Otlt) opens

_____ 3.16 IF in "AUTO", verify the boric acid transfer pump starts.

☐ 3.17 Verify proper flow by observing the Boric Acid Counter. {PIP 96-0137}

NOTE:	The boric acid counter may count up 1 - 5 gallons after termination.

3.18 WHEN the desired volume of boric acid is reached on the boric acid counter, ensure the following valves close: (R.M.)

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

Enclosure 4.2

Boration

OP/1/A/6150/009
Page 5 of 6

NOTE:	If additional borations will be performed over the course of the shift, flushing the makeup line is **NOT** recommended.

N/A 3.19 **IF** desired, flush the makeup line as follows:

☐ 3.19.1 Record the setpoint on 1NV-242A (RMWST To B/A Blender Ctrl):
___________gpm

_____ 3.19.2 Place controller for 1NV-242A (RMWST To B/A Blender Ctrl) in manual.

_____ 3.19.3 Increase demand on controller for 1NV-242A (RMWST To B/A Blender Ctrl) to full open.

NOTE:	It is essential for the operator to read and understand the following steps before initiating a flush of the makeup line. If a reactor makeup water pump is currently on, the following step will initiate flow to the makeup line. Steps 3.19.4, 3.19.5, and 3.19.6 may be performed prior to signing off the steps.

3.19.4 Open the following valves:

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

_____ 3.19.5 Ensure one reactor makeup water pump is in "ON".

NOTE:	Valves in the following step shall be positioned as sequenced to preclude unanticipated additional reactor makeup water flow due to seat leak by on 1NV-186A.

3.19.6 **WHEN** ~ 20 gallons of makeup water have been flushed through the makeup line, close the following valves:

_____ 3.19.6.1 1NV-242A (RMWST To B/A Blender Ctrl)

_____ 3.19.6.2 1NV-186A (B/A Blender Otlt To VCT Otlt)

3.19.7 Place the following valve control switches in "AUTO":

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

_____ 3.19.8 Ensure controller for 1NV-242A (RMWST To B/A Blender Ctrl) is set to the value recorded in Step 3.19.1. (R.M.)

_____ 3.19.9 Place controller for 1NV-242A (RMWST To B/A Blender Ctrl) in auto.

Enclosure 4.2

Boration

OP/1/A/6150/009
Page 6 of 6

_____ 3.19.10 **IF NOT** required for current plant operation, place the reactor makeup water pump started in Step 3.19.5 in "AUTO".

_____ 3.20 **IF** automatic makeup is desired, perform one of the following:

AA 3.20.1 **IF** it is desired to change the blender outlet boron concentration, refer to Enclosure 4.1 (Automatic Makeup).

OR

_____ 3.20.2 **IF** makeup at the previous concentration is acceptable **AND** the system was previously aligned per Enclosure 4.1 (Automatic Makeup), perform the following:

_____ 3.20.2.1 Ensure the controller for 1NV-238A (B/A Xfer Pmp To Blender Ctrl) is set to the value recorded in Step 3.7. (R.M.)

_____ 3.20.2.2 Place the "NC MAKEUP MODE SELECT" switch in "AUTO".

_____ 3.20.2.3 Place the "NC MAKEUP CONTROL" switch to the "START" position. (R.M.)

3.21 Do **NOT** file this enclosure.

Enclosure 4.16

Control Bank Manual Operation At Power

OP/1/A/6150/008
Page 1 of 2

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can affect core reactivity by changing control rod position. (R.M.)

1.2 The following Limits and Precautions are Reactivity Management related: (R.M.)

1.2.1 When rods are being moved, observe "RODS IN/RODS OUT" light for proper direction.

1.2.2 When rods are being moved, observe the demand position and actual (digital) position to verify proper operation of the Rod Control System.

1.2.3 Adjusting T-Avg ± 1°F of T-Ref before transferring rod control to "AUTO" will prevent undesired rod movement.

1.2.4 Monitor startup rate continuously during any rod motion to ensure < 0.5 DPM stable startup rate.

1.3 Automatic rod control shall **NOT** be used when less than 15% (184 MW_e) turbine power.

1.4 Individual control bank positions on "CRD BANK SELECT" switch shall not be used to position rods manually. (The automatic overlap feature is disabled.)

1.5 After releasing Rod Motion lever, waiting 2 seconds before attempting to move rods again will allow all signals to clear the firing cards.

1.6 A rod motion demand below zero steps may result in the movable grippers **NOT** properly engaging the drive shaft.

2. Initial Conditions

AA 2.1 Ensure Reactivity Management controls established per AD-OP-ALL-0203 (Reactivity Management. (RM)

AA 2.2 Verify Unit 1 is **NOT** in an EP or AP.

AA 2.3 Verify one of the following exist:

- ☑ Control Bank movement required to increase/decrease Reactor Power
- ☐ Control Bank movement required to increase/decrease Tavg
- ☐ Control Bank movement required to maintain AFD
- ☐ Control Bank manual control required to support testing/maintenance activity

Enclosure 4.16
Control Bank Manual Operation At Power

OP/1/A/6150/008
Page 2 of 2

3. Procedure

NOTE:	Steps 3.1 through 3.6 may be signed off as time allows ensuring operator maintains proper focus on reactivity management.

AA 3.1 Monitor the following:

- Tavg/Tref
- Demand Counter positions
- DRPI rod positions
- ROD MOTION RODS-IN/RODS-OUT Light
- ROD MOTION DEMAND SIGNALS - TEMP ERROR/POWER MISMATCH
- Power Range instruments
- IR SUR (Startup Rate)

AA 3.2 IF MANUAL ROD movement is desired, perform the following:

- Verify the "ALM" LED on circuit card A206 in the left side of 1ERCC0006 (Rod Control Logic Cabinet) is NOT illuminated.
- Verify one GRP select light is illuminated on each power cabinet.

AA 3.3 IF plant conditions require, place the "CRD BANK SELECT" switch in "MAN".

N/A 3.4 IF withdrawing Control Banks, pull and hold the "ROD MOTION" lever "OUT" as required until control rods are in the desired position. (R.M.)

____ 3.5 IF inserting Control Banks, push and hold the "ROD MOTION" lever "IN" as required until control rods are in the desired position. (R.M.)

____ 3.6 IF automatic rod control is desired, perform the following:

____ 3.6.1 Verify Unit 1 Reactor Power is ≥ 15% RTP.

____ 3.6.2 WHEN Tavg is within 1°F of Tref, place "CRD BANK SELECT" in "AUTO".

3.7 Do NOT file this enclosure.

Enclosure 4.2
Load Changing

OP/1/B/6300/001
Page 1 of 6

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can effect reactivity. (R.M.)

1.2 Low load operation limits:

1.2.1 The unit can be operated continuously at low loads when exhaust hood temperature is < 175°F. The load shall, however, be increased slowly until the temperature decreases below 125°F before increasing load at normal rate (Multipoint Recorder on 1MC3).

1.2.2 Excessive use of the exhaust hood sprays shall be avoided to prevent accelerated blade erosion.

1.3 Do **NOT** exceed the load, hydrogen pressure, and power factor limits per the Unit One Revised Data Book Figure 43.

1.4 If the limits of the Unit One Revised Data Book Figure 43 (Generator Capability Curves) are exceeded, the Turbine Generator shall be tripped.

1.5 Under certain environmental conditions, indicated condenser vacuum less than 24.3 inches Hg may be reached at full load. Exhaust hood temperatures are a more accurate indicator of true vacuum. It is recommended the turbine **NOT** be operated under the following conditions at full load:

- Exhaust Hood 1A temperature ≥ 136°F
- Exhaust Hood 1B temperature ≥ 129°F
- Exhaust Hood 1C temperature ≥ 124.5 °F

1.6 The maximum differential pressure between adjacent LP shell pressures shall **NOT** exceed 2.0 inches Hg. (main condenser vacuum gauges on 1MC13, OAC points C1P1669 (D/P between A & B Condensers) and C1P1670 (D/P between B & C Condensers) or Main Condenser graphic (CMCOND)).

1.7 A sudden downward trend on an LP turbine's lower extraction temperature shall be investigated as a possible indication of water induction into the turbine. This is indicated on the recorder on the rear of 1MC8 labeled "TURBINE WATER DETECTION", using any of the LP 8th stage lower temperatures.

1.8 A "LOAD RATE" > "6.2 MW/MIN" shall **NOT** be used during normal load changes.

Enclosure 4.2 OP/1/B/6300/001

Load Changing Page 2 of 6

1.9 Differential temperature between adjacent exhaust hoods shall **NOT** exceed 30°F unless evaluated and approved by the responsible engineer (Turbine Generator System Expert). (OAC points C1P1667 (A & B Exhaust Hoods Metal Delta Temp) and C1P1668 (B & C Exhaust Hoods Metal Delta Temp) or Main Condenser graphic (CMCOND)).

1.10 The Main Turbine OIU Work Station has the capability to perform control functions for the Main Turbine, including tripping and resetting of the turbine. If a control function window is inadvertently selected while manipulating the Main Turbine OIU Work Station, the window shall be closed to prevent actuation of the control function.

1.11 To reduce potential for Turbine rubs at low power levels (< 30% Turbine Load) observe the following:

- Steam Seal Header Pressure between 3 and 5 psig.
- Gland Steam Condenser Header vacuum of 10 - 12" H_2O vacuum.
- Condenser Vacuum < 28.0" Hg
- Minimize time that Turbine is at speed no load.
- Minimize time between Turbine Shell Warming and rolling the Turbine.

1.12 Exhaust hood water sprays are used to cool the last-stage buckets and to minimize temporary distortion of the low-pressure hood and shell structures. These sprays have a significant potential for quenching the LP turbine structure, and if they are applied manually should be undertaken very gradually. Large and rapid changes in the temperature of the exhaust hood can also have an impact on bearing alignment and may cause a rub to develop. Excessive use of the sprays may cause unnecessary erosion of the long last-stage buckets during low flow conditions.

2. Initial Conditions

AA Verify Turbine Generator is On Line per Enclosure 4.1 (Turbine Generator Startup).

Op Test No.:	301	Scenario #	3	Event #	1	Page	21	of	117
Event Description:	Unit 1 Power Decrease								

Enclosure 4.2 OP/1/B/6300/001
Load Changing Page 3 of 6

3. Procedure

CAUTION:
1. The load, hydrogen pressure and power factor limits per the Unit One Revised Data Book Figure 43 shall **NOT** be exceeded.
2. Rate of change of First-Stage Bowl Inner Surface Temperature shall **NOT** exceed 150°F/hr (OAC point C1P1283 (First Stage Metal Temp Rate)).
3. Control valve casing difference, OAC point C1A0961 (Turb Valve Chest Inner Surface Metal Temp) minus C1A0967 (Turb Valve Chest Outer Surface Metal Temp), shall **NOT** exceed curve "Allowable Temp Difference on Turbine Valve Chest" in the Unit 1 OAC Databook.
4. OAC point C1A1140 (Turbine Lower Inner Shell Temp) vs. Percent Steam Flow (OAC point C1P1588 (Design Total Main Steam Flow, Measured (%))) shall be maintained above and to the left of the curve in the Unit One OAC Databook "Load-Changing Recommendations".

NOTE:
- Several of the parameters required for this procedure can be found on OAC graphics, and a list of all OAC points are found on Enclosure 4.8 (Turbine Generator Roll Computer Points).
- Step 3.1 and Step 3.2 may be performed in any order.

N/A 3.1 **IF** increasing turbine generator load, perform the following:

_____ 3.1.1 Ensure the proper reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 3.1.2 **WHILE** increasing Turbine Generator load, perform the following:

_____ 3.1.2.1 **IF AT ANY TIME** Turbine load is < 30%, operate the RC system pumps and fans as required per OP/1/B/6400/001 A (Condenser Circulating Water System) to maintain vacuum in Condenser C < 28" Hg.

_____ 3.1.2.2 **IF** applicable, verify Groups B and C valves on Enclosure 4.6 (Valve Checklist) close at 15% of full load (184 MWe, 107 psig Turbine Impulse Pressure).

_____ 3.1.2.3 **IF** applicable, verify the following valves close at 15% of full load (184 MWe, 107 psig Turbine Impulse Pressure):

☐ 1SM-21 (Ctrl Vlv #2 Stm Lead Drn)

☐ 1SM-29 (Ctrl Vlv #1 Stm Lead Drn)

Enclosure 4.2 OP/1/B/6300/001

Load Changing Page 4 of 6

_____ 3.1.2.4 __IF__ applicable, __WHEN__ CV3 comes off of its fully closed seat (65% of full load, 796 MWe), verify 1SM-25 (Ctrl Vlv #3 Stm Lead Drn) closes.

_____ 3.1.2.5 __IF__ applicable, __WHEN__ CV4 comes off of its fully closed seat (92% of full load, 1127 MWe), verify 1SM-33 (Ctrl Vlv #4 Stm Lead Drn) closes.

CAUTION:

1. Until it is recognized that the first stage shell metal temperature change rate stays below the allowable limit (150°F/hr), the following loading rate shall __NOT__ be exceeded:
   - 1/2%/min - First Stage Inner Shell Temperature (1MC3 or OAC point C1A1140 (Turbine Lower Inner Shell Temp)) ≤ 350°F
   - 1%/min - First Stage Inner Shell Temperature (1MC3 or OAC point C1A1140 (Turbine Lower Inner Shell Temp)) > 350°F
2. Normal steady-state load changes shall be made without exceeding the limits shown on Enclosure 4.7 (Generator Operating Limits) and in the Unit 1 OAC Databook "Recommended Startup and Loading Curves".
3. Unit One Reactor Operating Data, Section 2.4 shall be referred to for allowable ramp rates. A "LOAD RATE" > 6.2 MW/MIN shall __NOT__ be used during normal load changes.

3.1.3 Increase turbine generator load by performing the following:

_____ 3.1.3.1 Select "LOAD RATE" and verify it illuminates.

_____ 3.1.3.2 Input the desired load rate.

_____ 3.1.3.3 Select "ENTER" and verify "LOAD RATE" goes dark.

_____ 3.1.3.4 Select "TARGET" and verify it illuminates.

_____ 3.1.3.5 Input the desired load target.

_____ 3.1.3.6 Select "ENTER" and verify "TARGET" light goes dark.

_____ 3.1.3.7 Verify new load target appears on Target Display.

_____ 3.1.3.8 Select "GO" and verify it illuminates to start load increase.

_____ 3.1.3.9 Coordinate with Secondary Chemistry to adjust S/G blowdown flowrates to obtain maximum blowdown for the appropriate load.

Op Test No.: 301	Scenario # 3	Event # 1	Page 23 of 117
Event Description: Unit 1 Power Decrease			

Enclosure 4.2 OP/1/B/6300/001
Load Changing Page 5 of 6

CAUTION:
1. The load, hydrogen pressure and power factor limits per the Unit One Revised Data Book Figure 43 shall **NOT** be exceeded.
2. Rate of change of First-Stage Bowl Inner Surface Temperature shall **NOT** exceed 150°F/hr (OAC point C1P1283 (First Stage Metal Temp Rate)).
3. OAC point C1A1140 (Turbine Lower Inner Shell Temp) vs. Percent Steam Flow (OAC point C1P1588 (Design Total Main Steam Flow, Measured (%)) shall be maintained above and to the left of curve in the Unit One OAC Databook "Load-Changing Recommendations".
4. Control valve casing difference, OAC point C1A0961 (Turb Valve Chest Inner Surface Metal Temp) minus C1A0967 (Turb Valve Chest Outer Surface Metal Temp), shall **NOT** exceed curve "Allowable Temp Difference on Turbine Valve Chest" in the Unit 1 OAC Databook.

_____ 3.2 **IF** decreasing turbine generator load, perform the following:

AA 3.2.1 Ensure the proper reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 3.2.2 **WHILE** decreasing turbine generator load, perform the following:

_____ 3.2.2.1 **IF AT ANY TIME** Turbine load is < 30%, operate the RC system pumps and fans as required per OP/1/B/6400/001 A (Condenser Circulating Water System) to maintain vacuum in Condenser C < 28" Hg.

_____ 3.2.2.2 **IF** CV4 fully closes (92% of full load, 1127 MWe), verify 1SM-33 (Ctrl Vlv #4 Stm Lead Drn) opens.

_____ 3.2.2.3 **IF** CV3 fully closes (65% of full load, 796 MWe), verify 1SM-25 (Ctrl Vlv #3 Stm Lead Drn) opens.

Enclosure 4.2 OP/1/B/6300/001

Load Changing Page 6 of 6

CAUTION: 1. Normal steady-state load change shall be made without exceeding limits shown on Enclosure 4.7 (Generator Operating Limits) and in the Unit 1 OAC Databook "Recommended Starting and Loading Curves".

2. Unit One Reactor Operating Data, Section 2.4 shall be referred to for allowable ramp rates.

3.2.3 Decrease turbine generator load by performing the following:

_____ 3.2.3.1 Select "LOAD RATE" and verify it illuminates.

_____ 3.2.3.2 Input the desired load rate.

_____ 3.2.3.3 Select "ENTER" and verify "LOAD RATE" goes dark

_____ 3.2.3.4 Select "TARGET" and verify it illuminates.

_____ 3.2.3.5 Input the desired load target.

_____ 3.2.3.6 Select "ENTER" and verify "TARGET" goes dark.

_____ 3.2.3.7 Verify new load target appears on Target Display.

_____ 3.2.3.8 Select "GO" and verify it illuminates to start load decrease.

_____ 3.2.3.9 Coordinate with Secondary Chemistry to adjust S/G blowdown flowrates to obtain maximum blowdown for the appropriate load.

3.3 Do **NOT** file a copy of this enclosure in the designated storage cabinet.

Note to Evaluator:

At this time the power decrease is in progress. At the discretion of the Lead Evaluator, the scenario may continue to the next event by instructing the booth operator to INSERT Trigger 2 (1NV-294 Fails Open).

Op Test No.: 301	Scenario # 3	Event # 2	Page 25 of 117				
Event Description:	Charging Control Valve 1NV-294 Fails Open						

Control Room Indications
DCS alarm for 1NV-294 being full open
Pressurizer Level – RISING

Note To Evaluator:

The failing open of 1NV-294 will initially only cause a DCS alarm. The crew should notice the rise in charging line flow. The crew should place 1NV-294 in manual and set a control band for Pressurizer Level for the BOP to maintain. Preventing Pressurizer Level from raising above 92% is Critical Task #1. Once the BOP has control of 1NV-294, and Pressurizer Level is trending towards reference level, the scenario may continue at the discretion of the lead evaluator by having the booth operator INSERT Trigger 3 (YN Critical Trouble alarm). When this alarm is acknowledged on 1AD-19, Pressurizer PORV 1NC-32B will fail open.

Op Test No.:	301	Scenario #	3	Event #	3	Page	26 of 117

Event Description: YN Critical Trouble alarm / Pressurizer PORV 1NC-32B Fails Open

Control Room Indications
1AD-6, C/13 “PORV NC-32B ACTUATED” – LIT
1AD-6, F/8 “PZR LO PRESS CONTROL” – LIT
Pressurizer Pressure – LOWERING

Op Test No.:	301	Scenario #	3	Event #	3	Page	27 of 117
Event Description:	YN Critical Trouble alarm / Pressurizer PORV 1NC-32B Fails Open						

CNS AP/1/A/5500/011	PRESSURIZER PRESSURE ANOMALIES Case I Pressurizer Pressure Trending Down	PAGE NO. 2 of 10 Revision 25

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Verify all Pzr PORVs - CLOSED.**	**Perform the following:**
Critical Task #2	__ a. CLOSE Pzr PORV(s).
	N/A b. **IF** any Pzr PORV cannot be closed, **THEN**:
	__ 1) CLOSE affected PORV(s) isolation valve.
	2) **IF** Pzr PORV isolation valve cannot be closed, **THEN** perform the following:
	__ a) **IF** in Mode 3 with CLAs isolated **OR** in Mode 4, **THEN** **GO TO** AP/1/A/5500/027 (Shutdown LOCA).
	__ b) Trip Unit 1 reactor.
	__ c) **WHEN** reactor tripped **OR** S/I setpoint reached, **THEN** ensure S/I initiated.
	__ d) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Op Test No.:	301	Scenario #	3	Event #	3	Page	28 of 117
Event Description:	YN Critical Trouble alarm / Pressurizer PORV 1NC-32B Fails Open						

CNS AP/1/A/5500/011	PRESSURIZER PRESSURE ANOMALIES Case I Pressurizer Pressure Trending Down	PAGE NO. 3 of 10 Revision 25

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE Control rods may withdraw on NC pressure trending down.	
__ 2. **Verify Pzr spray valve(s) - CLOSED.**	**Perform the following:** __ a. CLOSE **malfunctioning** spray valve(s). b. **IF** affected spray valve(s) will not close, **THEN** perform the following: 1) Select "FAIL CLOSED" for affected spray valve(s) mode select switch: __ • "1 NC-27 PZR SPRAY VLV MODE SELECT" __ • "1 NC-29 PZR SPRAY VLV MODE SELECT".
N/A 3. **IF NC pressure continues to trend down due to malfunctioning spray valve, THEN perform the following:** a. **IF** in Modes 1 or 2, **THEN**: __ 1) Trip Unit 1 reactor. __ 2) **WHEN** reactor power less than 5%, **THEN** stop NC Pumps 1A and 1B. __ 3) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection). __ b. Stop NC Pumps 1A and 1B. __ c. **IF** 1C and 1D NCPs on, **THEN** stop one additional NCP. __ d. **REFER TO** AP/1/A/5500/004 (Loss of Reactor Coolant Pump).	

CNS AP/1/A/5500/011	PRESSURIZER PRESSURE ANOMALIES Case I Pressurizer Pressure Trending Down	PAGE NO. 4 of 10 Revision 25

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 4. **Verify all Pzr heaters - ENERGIZED.**	**Perform the following:** __ a. **IF** Pzr pressure less than 2220 PSIG **AND** stable or trending down, **THEN** ensure all Pzr heaters energized. __ b. **IF** Pzr pressure less than 2220 PSIG **AND** trending up, **THEN** operate Pzr heaters as required to stabilize Pzr pressure at 2235 PSIG. __ c. **WHEN** Pzr pressure returns to normal **AND** automatic Pzr pressure control desired, **THEN** place Pzr heaters in auto.
__ 5. **Ensure 1NV-37A (NV Supply To Pzr Aux Spray) - CLOSED.**	

NOTE Positive reactivity is inserted during a rise in NC pressure which may cause auto rod insertion.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 6. **Verify NC pressure - STABLE OR TRENDING UP.**	__ **IF pressure continues to trend down, THEN REFER TO AP/1/A/5500/010 (Reactor Coolant Leak).**
7. **WHEN NC pressure stable, THEN perform the following:** __ • Stabilize unit at appropriate power level. • Adjust the following as required to maintain T-Avg within 1°F of T-Ref: __ • Turbine load __ • Control rods __ • Boron concentration.	

Op Test No.: 301 Scenario # 3 Event # 3	Page 30 of 117
Event Description:	YN Critical Trouble alarm / Pressurizer PORV 1NC-32B Fails Open

CNS AP/1/A/5500/011	PRESSURIZER PRESSURE ANOMALIES Case I Pressurizer Pressure Trending Down	PAGE NO. 5 of 10 Revision 25

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
N/A 8. **<u>IF</u> Pzr pressure channel failed, <u>THEN</u> perform following:**	
__ a. Verify "P-11 PZR S/I BLOCK PERMISSIVE" status light (1SI-18) in required state for unit conditions.	__ a. Ensure compliance with Tech Spec 3.3.2 (Engineered Safety Features Actuation System (ESFAS) Instrumentation).
b. Notify IAE to fail following bistables for affected channel per Model W/O #00874531. Bistables shall be tripped within 72 hours: __ • Pzr low pressure S/I __ • OT Delta T __ • Pzr high pressure Reactor Trip __ • Pzr low pressure Reactor Trip.	

Op Test No.:	301	Scenario #	3	Event #	3	Page	31 of 117

Event Description: YN Critical Trouble alarm / Pressurizer PORV 1NC-32B Fails Open

CNS AP/1/A/5500/011	PRESSURIZER PRESSURE ANOMALIES Case I Pressurizer Pressure Trending Down	PAGE NO. 6 of 10 Revision 25

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Ensure compliance with appropriate Tech Specs:** __ • 3.3.1 (Reactor Trip System (RTS) Instrumentation) __ • 3.3.2 (Engineered Safety Features Actuation System (ESFAS) Instrumentation) __ • 3.3.3 (Post Accident Monitoring (PAM) Instrumentation) __ • 3.3.4 (Remote Shutdown System) __ • 3.4.1 (RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits) __ • 3.4.4 (RCS Loops - MODES 1 and 2) __ • 3.4.5 (RCS Loops - MODE 3) __ • 3.4.6 (RCS Loops - MODE 4) __ • 3.4.9 (Pressurizer) __ • 3.4.10 (Pressurizer Safety Valves) __ • 3.4.11 (Pressurizer Power Operated Relief Valves (PORVs)) __ • 3.4.13 (RCS Operational Leakage).	**TECH SPEC EVALUATION** ***See Attachment 10 for applicable Tech Specs.*** **T.S. 3.4.11** **Condition A: Close and maintain power to applicable block valve in 1 hour.**
__ 10. **Determine long term plant status. <u>RETURN TO</u> procedure in effect.**	

<u>END</u>

Note to Evaluator:

At the discretion of the Lead Evaluator, the scenario may continue to the next event by instructing the booth operator to INSERT Trigger 4 (Blackout of 1ETA).

Op Test No.: 301	Scenario # 3	Event # 4	Page 32 of 117
Event Description: 1ETA Blackout with failure of 1A D/G to start			

Control Room Indications
Multiple alarms for annunciator panel 1AD-11 – LIT
Alarms associated with RN and KC low flows – LIT
‘A’ train essential equipment indications – DARK

Op Test No.:	301	Scenario #	3	Event #	4	Page	33 of 117
Event Description:	1ETA Blackout with failure of 1A D/G to start						

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 24 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 3 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Verify essential loads powered from energized train as follows:**	
__ a. RN pump(s) - IN SERVICE AS NEEDED.	__ a. Ensure pump(s) placed in service as required.
__ b. KC pump(s) - IN SERVICE AS NEEDED.	b. Perform the following: __ 1) Ensure pump(s) placed in service as required. 2) **IF AT ANY TIME** all the following conditions exist: __ • Unit 1 in Mode 5, 6 or No Mode __ • Loss of KC pumps due to blackout **AND** energized train KC HX **NOT** available __ • KC cross train cooling alignment desired, **THEN** concurrently perform the following: __ • Continue in this procedure **AND** __ • Place KC in cross train cooling alignment. **REFER TO** Enclosure 39 (KC Cross Train Cooling).
__ c. At least one NV pump - ON.	__ c. **REFER TO** AP/1/A/5500/012 (Loss of Charging or Letdown).

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 25 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	
__ d. CA pump - ON.	__ d. **IF** CA pump required to maintain S/G levels, **THEN** ensure pump(s) placed in service as required.
__ e. Ensure KC Hx outlet mode switches - PROPERLY ALIGNED.	
__ f. VC/YC chiller - ON.	__ f. **REFER TO** OP/0/A/6450/011 (Control Room Area Ventilation/Chilled Water System).
__ 3. **Verify CA Pump #1 - ON.**	N/A __ **IF CA Pump #1 required to maintain S/G levels, THEN start CA Pump #1.**
4. **Maintain reactor power as follows:**	
__ • Maintain reactor power less than or equal to 100%	
__ • **IF** reactor power less than 100% prior to loss of power, **THEN** maintain reactor power stable at current power level.	
__ 5. **Verify Emergency D/G on affected bus - RUNNING.**	__ **GO TO Step 7.**

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 26 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 6. **Verify RN cooling to affected Emergency D/G.**	**Perform the following for affected D/G:**
	• D/G 1A:
	__ a. Depress and hold D/G "OFF" pushbutton.
	__ b. Dispatch operator to open 1EDE-F01F (Diesel Generator Load Sequencer Panel 1DGLSA) (AB-577, BB-46, Rm 496).
	__ c. **WHEN** 1EDE-F01F (Diesel Generator Load Sequencer Panel 1DGLSA) open, **THEN** release D/G "OFF" pushbutton.
	OR
	• D/G 1B:
	__ a. Depress and hold D/G "OFF" pushbutton.
	__ b. Dispatch operator to open 1EDF-F01F (Diesel Generator Load Sequencer Panel 1DGLSB) (AB-560, BB-46, Rm 372).
	__ c. **WHEN** 1EDF-F01F (Diesel Generator Load Sequencer Panel 1DGLSB) open, **THEN** release D/G "OFF" pushbutton.

Op Test No.: 301 Scenario # 3 Event # 4 Page 36 of 117

Event Description: 1ETA Blackout with failure of 1A D/G to start

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 27 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE The affected essential bus is the de-energized essential bus. If ESPS is energizing an essential bus then the opposite bus should be load shed.	
7. **Dispatch operator with screwdriver to load shed affected essential bus as follows:**	
__ • **REFER TO** Enclosure 8 (Local Load Shed Of 1ETA) OR __ • **REFER TO** Enclosure 9 (Local Load Shed Of 1ETB).	
__ 8. **Verify operating RN pump(s) flow - LESS THAN 23,000 GPM.**	__ **REFER TO AP/0/A/5500/020 (Loss of Nuclear Service Water).**
__ 9. **Stop any dilutions in progress.**	
10. **Verify S/I status as follows:**	
__ a. S/I - HAS ACTUATED.	__ a. **GO TO** Step 11.
__ b. **GO TO** Step 12.	

Op Test No.:	301	Scenario #	3	Event #	4	Page	37 of 117

Event Description: 1ETA Blackout with failure of 1A D/G to start

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 28 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **Verify ND System status as follows:**	
__ a. ND System - ALIGNED IN RESIDUAL HEAT REMOVAL MODE.	__ a. **GO TO** Step 12.
__ b. At least one ND pump - ON.	__ b. **REFER TO** AP/1/A/5500/019 (Loss of Residual Heat Removal System).

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 29 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
__ c. 1AD-11, K/3 "4KV B/O BUS FTA VOLTAGE LO" - DARK.	c. Perform the following: **NOTE** Both ND Hx Bypass valves fail closed on loss of 1LXI (1FTA). 1) **IF** ND pump 1A operating in Residual Heat Removal Mode, **THEN** perform the following: __ a) Place "PWR DISCON FOR 1NI173A" in "THROT". __ b) THROTTLE 1NI-173A (ND Hdr 1A To Cold Legs C&D) to stabilize NC temperature. __ c) **WHEN** 1AD-11, K/3 "4KV B/O BUS FTA VOLTAGE LO" dark, **THEN** return 1NI-173A to normal alignment. 2) **IF** ND pump 1B operating in Residual Heat Removal Mode, **THEN** perform the following: __ a) Place "PWR DISCON FOR 1NI178B" in "THROT". __ b) THROTTLE 1NI-178B (ND Hdr 1B To Cold Legs A&B) to stabilize NC temperature. __ c) **WHEN** 1AD-11, K/3 "4KV B/O BUS FTA VOLTAGE LO" dark, **THEN** return 1NI-178B to normal alignment.
__ 12. **Ensure CA System - RESET.**	

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 30 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. **Control S/G levels as follows:**	
__ a. Verify CF flow - MAINTAINING STABLE S/G LEVELS.	a. Perform the following: __ 1) **REFER TO** Enclosure 15 (S/G Level Control). __ 2) **GO TO** Step 14.
__ b. **IF AT ANY TIME** CF flow control to S/Gs lost, **THEN** perform Step 13.	

CAUTION **Battery depletion may occur as early as two hours. Battery depletion results in affected CA control valves failing full open. Failure to take local control of S/G level prior to battery depletion may result in S/G overfill.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ c. **IF AT ANY TIME** any vital or auxiliary control channel battery charger de-energized for greater than 1 hour, **THEN** dispatch operators to control affected CA flow path. **REFER TO** Enclosure 15 (S/G Level Control).	

Op Test No.:	301	Scenario #	3	Event #	4	Page	40 of 117
Event Description:	1ETA Blackout with failure of 1A D/G to start						

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 31 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 14. **Verify "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT.**	**To prevent overpressurizing condenser perform the following:** a. Dispatch operator to close the following valves: __ • 1SA-22 (Main Steam To CSAE) (TB1-594, 1M-32) __ • 1SA-27 (Aux Steam To CSAE) (TB-594, 1M-27). b. **WHEN** notified by dispatched operator SA supplies closed, **THEN** perform the following: __ 1) OPEN "COND A-B-C VAC BKR VLVS". 2) **IF** power not available to operate "COND A-B-C VAC BKR VLVS", **THEN** dispatch operator to open the following valves: __ • 1CM-368 (1A Main Cond Shell Vacuum Bkr) (TB1-600,1F-26) (Ladder needed) __ • 1CM-369 (1B Main Cond Shell Vacuum Bkr) (TB1-603, 1F-24) (Ladder needed) __ • 1CM-370 (1C Main Cond Shell Vacuum Bkr) (TB1-605, 1F-22) (Ladder needed). __ 3) **WHEN** time permits, **THEN** dispatch operator to complete breaking condenser vacuum. **REFER TO** OP/1/B/6300/006 (Main Vacuum). __ 4) Shutdown steam seals. **REFER TO** OP/1/B/6300/005 (Steam Seal System).

Op Test No.:	301	Scenario #	3	Event #	4	Page	41 of 117
Event Description:	1ETA Blackout with failure of 1A D/G to start						

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 32 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15. **Control charging as follows:**	
__ a. Maintain charging flow less than 180 GPM.	
__ b. Adjust charging flow as necessary to maintain Pzr level in program band.	
16. **Control letdown as follows:**	
__ a. Verify normal letdown - IN SERVICE.	a. Perform the following: __ 1) Attempt to restore letdown. **REFER TO** AP/1/A/5500/012 (Loss of Charging or Letdown). __ 2) **WHEN** normal letdown established, **THEN** place additional letdown orifice in service as necessary to control Pzr level. __ 3) **GO TO** Step 17.
__ b. Place additional letdown orifice in service as necessary to control Pzr level.	

NOTE Identifying tripped relays to the DEC TOP (Transmission Operations) will aid the power restoration.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 17. **Determine and correct cause of blackout.**	
__ 18. **Verify VI pressure - GREATER THAN 85 PSIG AND STABLE.**	__ **REFER TO AP/0/A/5500/022 (Loss of Instrument Air).**

Op Test No.:	301	Scenario #	3	Event #	4	Page	42 of 117
Event Description:	1ETA Blackout with failure of 1A D/G to start						

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 33 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

N/A 19. **IF Spent Fuel Pool instrumentation failed, THEN dispatch operator to monitor Unit 1 Spent Fuel Pool conditions. REFER TO EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 24 (Local Spent Fuel Pool Monitoring).**

CAUTION **Failure to disconnect battery prior to voltage less than 105 VDC will result in damage to battery.**

__ 20. **Verify all Vital and Aux Control Power DC busses - ALIGNED TO OPERATING CHARGER.**

RNO: __ **IF any Vital or Aux Control Power DC bus energized solely from its battery, THEN REFER TO AP/1/A/5500/029 (Loss of Vital or Aux Control Power).**

NOTE The following step is for pre-staging FLEX equipment. If both essential busses become de-energized, then FLEX equipment will be in place and ready to be aligned to plant equipment when required.

N/A 21. **IF only one Emergency D/G available and supplying an essential bus concurrent with loss of offsite power, THEN notify Shift Manager to initiate pre-staging FLEX equipment. REFER TO EP/1/A/5000/ECA-0.0 (Loss of All AC Power), Enclosure 45 (ELAP FSG Implementation) for implementation criteria, priority, and time limit considerations.**

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 34 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22. **Ensure compliance with appropriate Tech Specs:** __ • 3.4.9 (Pressurizer) __ • 3.8.1 (AC Sources - Operating) __ • 3.8.2 (AC Sources - Shutdown) __ • 3.8.4 (DC Sources - Operating) __ • 3.8.5 (DC Sources - Shutdown) __ • 3.8.7 (Inverters - Operating) __ • 3.8.8 (Inverters - Shutdown) __ • 3.8.9 (Distribution Systems - Operating) __ • 3.8.10 (Distribution Systems - Shutdown) __ • 3.7.7 (Component Cooling Water (CCW) System). __ • 3.7.8 (Nuclear Service Water System (NSWS)) __ • 3.7.10 (Control Room Area Ventilation System (CRAVS)) __ • 3.7.11 (Control Room Area Chilled Water System (CRACWS)) __ • 3.7.12 (Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)).	**TECH SPEC EVALUATION** ***See Attachment 10 for applicable Tech Specs.*** **T.S. 3.4.9** **Condition B: Restore required group of Pressurizer Heaters to OPERABLE in 72 hours.** **T.S. 3.8.9** **Condition A: Restore Electrical Power distribution subsystem to OPERABLE in 8 hours AND 16 hours from discovery of failure to meet the LCO.**
23. **Determine required notifications:** __ • <u>**REFER TO**</u> AD-EP-ALL-0111 (Control Room Activation of the ERO) __ • <u>**REFER TO**</u> AD-LS-ALL-0006 (Notification/Reportability Evaluation).	

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Case II Loss of All Power to an Essential Train	PAGE NO. 35 of 280 Revision 86

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 24. **Verify all 6.9 KV busses - ENERGIZED.**	**Perform the following:** **NOTE** SSF D/G is placed in service to maintain SSF batteries charged and Security equipment in service. __ a. **IF AT ANY TIME** offsite power not restored within 1 hour, **THEN** dispatch operator to place SSF D/G in service. **REFER TO** Enclosure 24 (Energize 1SLXG). __ b. **IF AT ANY TIME** IPB fans expected to be off greater than 48 hours, **THEN** notify TSC and Engineering to evaluate moisture accumulation in IPB ductwork prior to energizing 6.9 KV busses. __ c. Concurrently continue with Case II (Loss of All Power to an Essential Train) **AND** perform applicable steps of Case III. **REFER TO** Case III (Loss of All 6.9 KV Busses).
25. **Do not continue in this procedure until the following satisfied:** __ • Status of all lockout relays determined __ • Station Management approved power restoration to affected bus.	

Note to Evaluator:

At the discretion of the Lead Evaluator, the scenario may continue to the next event by instructing the booth operator to INSERT Trigger 5 (Continuous Rod Movement).

Control Room Indications
Control Rods – INSERTING with no demand
NC System Tavg – LOWERING

Op Test No.:	301	Scenario #	3	Event #	5	Page	46 of 117
Event Description:	Control Rods Insert Continuously / Two Rods Fail to Insert on Reactor Trip						

CNS AP/1/A/5500/015	ROD CONTROL MALFUNCTIONS Case II Continuous Rod Movement	PAGE NO. 8 of 10 Revision 17

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. **Operator Actions**	
__ 1. **Ensure "CRD BANK SELECT" switch - <u>NOT</u> IN AUTO.**	
__ 2. **Verify all rod motion - STOPS.**	**Perform the following:** __ a. Trip reactor. __ b. **<u>GO TO</u>** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

<u>NOTE</u> For T-Ref failures, T-Ref will need to be determined for current power level.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 3. **Adjust control rods as necessary to maintain T-Avg within 1°F of T-Ref.**	**Adjust the following as necessary to maintain T-Avg within 1°F of T-Ref:** __ • Turbine load __ • NC System boron concentration.

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	47 of 117
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed						

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 4 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 4 in the back of this document.**
__ 1. **Monitor Enclosure 1 (Foldout Page).**	
2. **Verify Reactor Trip:**	**Perform the following:**
__ • All rod bottom lights - LIT	__ a. Trip reactor.
__ • All reactor trip and bypass breakers - OPEN	b. IF reactor will not trip, THEN concurrently perform the following:
__ • I/R power - TRENDING DOWN.	__ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees) __ • GO TO EP/1/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS).
3. **Verify Turbine Trip:**	**Perform the following:**
__ • All turbine stop valves - CLOSED.	__ a. Trip turbine. b. IF turbine will not trip, THEN perform the following: __ 1) Depress "MANUAL" pushbutton on turbine control panel. __ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons. 3) IF control valves will not close, THEN CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves.

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 5 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 4. **Verify 1ETA and 1ETB - ENERGIZED.**	**Perform the following:** N/A a. **IF** 1ETA **AND** 1ETB de-energized, **THEN GO TO** EP/1/A/5000/ECA-0.0 (Loss of All AC Power). ___ b. **WHEN** time allows, **THEN** attempt to restore power to de-energized switchgear while continuing with this procedure. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power).

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION	PAGE NO. 6 of 49 Revision 46

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify S/I actuated:**	
__ a. "SAFETY INJECTION ACTUATED" status light (1SI-13) - LIT.	a. Perform the following: 1) Verify conditions requiring S/I: __ • Pzr pressure - LESS THAN 1845 PSIG OR __ • Containment pressure - GREATER THAN 1.2 PSIG. N/A __ 2) **IF** S/I required, **THEN** initiate S/I. 3) **IF** S/I not required, **THEN** concurrently perform the following: • **IF** 1ETA **OR** 1ETB de-energized, **THEN** ensure the following pumps running on energized bus: __ • NV pump __ • KC pumps __ • RN pump. __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees). __ • **GO TO** EP/1/A/5000/ES-0.1 (Reactor Trip Response).
__ b. Both E/S load sequencer actuated status lights (1SI-14) - LIT.	__ b. Initiate S/I.
__ 6. **Announce "Unit 1 Safety Injection".**	

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	50	of	117

Event Description: CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 2 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

Note to Evaluator:

Enclosure 1 can be found as Attachment 5 in the back of this document.

__ 1. **Monitor Enclosure 1 (Foldout Page).**

2. **Verify the following:**
   - __ All 6.9 KV busses - ENERGIZED
   - __ VI pressure - GREATER THAN 85 PSIG.

Perform the following:

a. **IF** loss of offsite power has occurred, **THEN** perform the following:
   - __ CLOSE all MSIVs
   - __ CLOSE all MSIV bypass valves
   - __ Ensure available RN pump(s) - IN SERVICE AS REQUIRED
   - __ **WHEN** time and manpower permit, **THEN** ensure available RL pump(s) in service as required. **REFER TO** OP/0/B/6400/003 (Low Pressure Service Water System).
   - __ **WHEN** time and manpower permit, **THEN** ensure available KR pump(s) in service as required. **REFER TO** OP/0/B/6400/004 (Recirculated Cooling Water System).

b. **IF AT ANY TIME** VI pressure less than or equal to 55 psig, **THEN** CLOSE the following valves:
   - __ All MSIVs
   - __ All MSIV bypass valves.

c. **IF** S/G N/R level less than 11% in all S/Gs, **THEN** THROTTLE feed flow to achieve the following:
   - __ Minimize cooldown
   - __ Maintain total feed flow greater than 450 GPM.

(RNO continued on next page)

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 3 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	
	d. **WHEN** N/R level greater than 11% in any S/G, **THEN** THROTTLE feed flow further to achieve the following: __ • Minimize cooldown __ • Maintain at least one S/G N/R level greater than 11%.
	e. **IF** VI pressure less than 85 PSIG, **THEN** perform the following: • Align N_2 to Pzr PORVs by opening the following valves: __ • 1NI-438A (Emer N2 From CLA A To 1NC-34A) __ • 1NI-439B (Emer N2 From CLA B To 1NC-32B). __ • Dispatch operator to ensure proper VI compressor operation. **REFER TO** AP/0/A/5500/022 (Loss of Instrument Air).
__ 3. **Announce "Unit 1 Reactor Trip, non-essential personnel stay out of Unit 1 Turbine bldg".**	

NOTE Enclosure 2 (NC Temperature Control) shall remain in effect until subsequent steps provide alternative NC temperature control guidance.

__ 4. **Control NC temperature. REFER TO Enclosure 2 (NC Temperature Control).**

Note to Evaluator:

Enclosure 2 can be found as Attachment 6 in the back of this document.

5. **Determine required notifications:**

__ • **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO)

__ • **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation).

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 4 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. **Verify Main Generator as follows:**	
__ a. Verify turbine generator megawatt output - LESS THAN **OR** EQUAL TO ZERO MW.	a. Perform the following: __ 1) Determine and correct cause of continued turbine generator output. __ 2) **WHEN** turbine generator megawatt output less than or equal to zero MW, **THEN** perform Step 6.b and Step 6.c. __ 3) **GO TO** Step 6.d.
b. Ensure the following breakers and MODs - OPEN: __ • MOD 1BG and 1BT __ • MOD 1AG and 1AT __ • Generator Breakers 1A and 1B.	
__ c. Ensure main generator "EXCITATION" - OFF.	
__ d. Verify "MAN/AUTO REG" select switch "MAN" mode light - LIT.	__ d. Transfer to manual mode.

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	53 of 117
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed						

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 5 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Verify feedwater status as follows:**	
__ a. T-Avg - LESS THAN 564°F.	a. Perform the following: __ 1) **WHEN** T-Avg less than 564°F, **THEN** perform Steps 7.b and 7.c. __ 2) **GO TO** Step 7.c.
__ b. All Feedwater Isolation status lights (1SI-5) - LIT.	b. Perform the following: __ 1) Initiate Feedwater Isolation. __ 2) **IF** any status light remains dark, **THEN** perform Enclosure 3 (Closure of Feedwater Isolation Valves).
__ c. Total feed flow to S/G(s) - GREATER THAN 450 GPM.	c. Establish feed flow to maintain at least one S/G N/R level greater than 11% **OR** total feed flow greater than 450 GPM using one of the following: __ • CA pumps OR __ • Main Feedwater System. **REFER TO** OP/1/A/6250/001 (Condensate and Feedwater System).

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 6 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. **Verify adequate shutdown margin as follows:**	
__ a. DRPI indication - AVAILABLE.	a. Verify adequate shutdown margin as follows: 1) Emergency borate 10,000 gallons of 7000 PPM boron solution as follows: __ a) OPEN 1NV-236B (Boric Acid To NV Pumps Suct). __ b) **IF** 1NV-236B will not open, **THEN** dispatch operator to open 1NV-236B (Boric Acid To NV Pumps Suct) (AB-550, HH-JJ, 53-54, Rm 234). c) **WHEN** 1NV-236B open, **THEN** perform the following: __ (1) Start boric acid transfer pumps. __ (2) Calculate required injection time based on boric acid flowrate. __ (3) **WHEN** required boric acid injected, **THEN** secure emergency boration. __ 2) Notify Reactor Group Duty Engineer to perform analysis to determine required shutdown margin. __ 3) **GO TO** Step 8.c.

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 7 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. (Continued)	
__ b. All control and shutdown rods - FULLY INSERTED.	b. **IF** two or more rods not fully inserted, **THEN** perform the following:
	1) Emergency borate 254 PPM for each rod not fully inserted as follows:
	__ a) OPEN 1NV-236B (Boric Acid To NV Pumps Suct).
	N/A b) **IF** 1NV-236B will not open, **THEN** dispatch operator to open 1NV-236B (Boric Acid To NV Pumps Suct) (AB-550, HH-JJ, 53-54, Rm 234).
Note to Evaluator: **Crew should calculate approximately 2850 gallons of boric acid need to be added to the NC system. At ~100 GPM, boric acid will need to be added for 28.5 minutes.**	c) **WHEN** 1NV-236B open, **THEN** perform the following:
	__ (1) Start boric acid transfer pumps.
	__ (2) Calculate required boron addition. **REFER TO** Unit 1 ROD Book, Section 4.1.
	__ (3) Calculate required injection time based on boric acid flowrate.
	__ (4) **WHEN** required boric acid injected, **THEN** secure emergency boration.
	__ 2) Notify Reactor Group Duty Engineer to perform analysis to determine required shutdown margin.
__ c. Stop any boron dilutions in progress.	

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	56 of 117
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed						

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 8 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. (Continued)	
__ d. All NC T-Colds - GREATER THAN 545°F.	d. Perform the following: __ 1) Determine lowest T-Cold. __ 2) Determine core burnup in effective full power days (EFPD) (OAC Point P1457 or from Reactor Operators logbook). __ 3) Verify lowest current T-Cold greater than or equal to allowable limit at present burnup. **REFER TO** Unit 1 ROD Book, Section 2.6. __ 4) **IF** lowest T-Cold less than allowable limit, **THEN** immediately add 40 gallons of 7000 PPM boron solution at greater than or equal to 30 GPM for each degree lowest T-Cold below limit of Unit 1 ROD Book, Section 2.6. __ 5) **GO TO** Step 9.
__ e. **IF AT ANY TIME** NC T-Colds trend down to less than or equal to 545°F, **THEN** perform Step 8.d.	

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 9 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Verify proper Pzr level control as follows:**	
__ a. Verify VI pressure - GREATER THAN 50 PSIG.	a. Perform the following: 1) **IF** Pzr Level less than 17%, **THEN** perform the following: __ a) Ensure normal letdown - ISOLATED. __ b) Ensure all Pzr heaters - OFF. __ c) Control charging to restore Pzr level to greater than 17% while maintaining flow less than 180 GPM. __ d) **WHEN** Pzr level greater than 17%, **THEN** Depress C Heater "ON" pushbutton. 2) **IF AT ANY TIME** NV controllers not maintaining stable charging flow, **THEN** perform the following: __ a) Maintain charging flow less than 180 GPM. b) Dispatch operator with radio to THROTTLE 1NV-295 (NV Pmps A & B Disch Ctrl Isol) (AB-551, JJ-55, Rm 231) to maintain the following: __ • Pzr level - GREATER THAN 17% __ • Pzr level - TRENDING TO "PZR REF LEVEL". (RNO continued on next page)

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 10 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	c) Dispatch operator with radio to perform the following: __ (1) THROTTLE 1NV-311 (Seal Wtr Inj Flow Ctrl Byp) (AB-555, JJ-54, Rm 233) as required to maintain 32 GPM total seal water flow in subsequent steps. __ (2) CLOSE 1NV-308 (Seal Wtr Inj Flow Ctrl Isol) (AB-554, JJ-54, Rm 233). __ d) **WHEN** dispatched operators throttled 1NV-295 and 1NV-311, **THEN** depress manual pushbutton and raise output to 100% demand position for 1NV-294 (NV Pmps A&B Disch Flow Ctrl). __ e) **WHEN** VI restored, **THEN** perform Step 9. __ f) **GO TO** Step 10.

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 11 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	
__ b. Verify Pzr level - GREATER THAN 17%.	b. Perform the following: __ 1) Ensure normal letdown - ISOLATED. __ 2) Ensure all Pzr heaters - OFF. __ 3) Control charging to restore Pzr level to greater than 17% while maintaining flow less than 180 GPM. 4) **WHEN** Pzr level greater than 17%, **THEN** perform the following: __ a) Establish normal letdown. **REFER TO** AP/1/A/5500/012 (Loss of Charging or Letdown). __ b) Depress C Heater "ON" pushbutton. __ 5) **GO TO** Step 9.d.
__ c. Verify charging and letdown - IN SERVICE.	__ c. Restore charging and letdown. **REFER TO** AP/1/A/5500/012 (Loss of Charging or Letdown).
__ d. Verify Pzr level - TRENDING TO "PZR REF LEVEL".	__ d. Control charging and letdown to maintain Pzr level at "PZR REF LEVEL".
__ 10. **Verify Pzr pressure - GREATER THAN 1845 PSIG.**	**Perform the following:** __ a. Ensure S/I - ACTUATED. __ b. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	60	of	117

Event Description: CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE	PAGE NO. 12 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 11. **Verify 1A and 1B NC pumps - ON.**	__ **Ensure Pzr spray valve for stopped NC pump - IN MANUAL AND CLOSED.**
__ 12. **Verify Pzr pressure - STABLE AT OR TRENDING TO 2235 PSIG.**	**Perform one of the following:**

Note to Evaluator:

Once all S/G N/R levels lower to < 11%, the crew will transition to EP/1/A/5000/FR-H.1 which begins on the next page of this guide.

a. **IF** Pzr pressure less than 2235 PSIG **AND** trending down, **THEN** perform the following:

__ 1) Ensure all Pzr PORVs - CLOSED.

__ 2) **IF** any Pzr PORV cannot be closed, **THEN** CLOSE its isolation valve.

3) **IF** 1NC-32B **OR** 1NC-34A cannot be closed **OR** isolated, **THEN** perform the following:

a) Align N_2 to PORVs by opening the following valves:

__ • 1NI-438A (Emer N2 From CLA A To 1NC-34A)

__ • 1NI-439B (Emer N2 From CLA B To 1NC-32B).

__ b) CLOSE affected Pzr PORV.

__ 4) Ensure Pzr spray valves - CLOSED.

5) **IF** spray valve(s) cannot be closed, **THEN** perform the following:

__ a) Stop NC pumps 1A and 1B.

__ b) **IF** both 1C **AND** 1D NC pumps on, **THEN** stop one additional NC pump.

__ 6) Ensure all Pzr heaters - ON.

__ 7) **GO TO** Step 13.

(RNO continued on next page)

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 2 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
N/A __ 1. **IF total feed flow less than 450 GPM due to operator action, THEN RETURN TO procedure and step in effect.**	
CAUTION **If a non-faulted S/G is available, then feed flow should only be established to non-faulted S/G(s) in subsequent steps.**	
2. **Verify secondary heat sink required as follows:**	
__ a. NC pressure - GREATER THAN ANY NON-FAULTED S/G PRESSURE.	__ a. **IF** LOCA in progress, **THEN RETURN TO** procedure and step in effect.
__ b. Any NC T-Hot - GREATER THAN 350°F.	b. Perform the following while continuing in this procedure: __ 1) Attempt to place ND in service. **REFER TO** OP/1/A/6200/004 (Residual Heat Removal System). __ 2) **WHEN** adequate ND cooling established, **THEN RETURN TO** procedure and step in effect.
__ 3. **Monitor Enclosure 1 (Foldout Page).**	**Note to Evaluator:** **Enclosure 1 can be found as Attachment 7 in the back of this document.**
__ 4. **Verify at least one NV pump - AVAILABLE.**	__ **GO TO Step 21.**

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	62 of 117
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed						

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 3 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify NC System feed and bleed required as follows:**	
__ a. W/R level in at least 3 S/Gs - LESS THAN 24% (36% ACC).	a. Perform the following: __ 1) Monitor feed and bleed initiation criteria. **<u>REFER TO</u>** Enclosure 1 (Foldout Page). __ 2) **<u>WHEN</u>** criteria satisfied, **<u>THEN GO TO</u>** Step 21. __ 3) **<u>GO TO</u>** Step 6.
__ b. **<u>GO TO</u>** Step 21.	
__ 6. **Ensure S/G BB and NM valves closed. <u>REFER TO</u> Enclosure 9 (S/G BB and NM Valve Checklist).**	**Note to Evaluator:** **Enclosure 9 can be found as Attachment 8 in the back of this document.**
7. **Attempt to establish CA flow to at least one S/G as follows:**	
__ a. Verify 1AD-8, B/1 "UST LO LEVEL" - DARK.	a. Perform the following: __ 1) **<u>REFER TO</u>** AP/1/A/5500/006 (Loss of S/G Feedwater). __ 2) **<u>GO TO</u>** Step 7.c.

Op Test No.: 301	Scenario # 3	Event # 6,7,8	Page 63 of 117				
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed						

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 4 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	
__ b. Verify 1CA-4 (CA Pmps Suct From UST) - OPEN.	b. Perform the following: __ 1) OPEN 1CA-4. 2) **IF** no suction source can be aligned, **THEN** OPEN the following valves: __ • 1RN-250A (RN Hdr A To CA Pmp Suct Isol) __ • 1CA-116A (CA Pump #1 Suct Frm RN Hdr A) __ • 1CA-15A (CA Pump 1A Suct Frm RN Isol) __ • 1CA-85B (CA Pump #1 Suct Frm RN Hdr B) __ • 1CA-18B (CA Pump 1B Suct Frm RN Isol) __ • 1RN-310B (RN Hdr B To CA Pmp Suct Isol).
c. Verify proper CA pump status as follows:	
__ 1) Power to both motor driven CA pumps - AVAILABLE.	1) Perform the following: __ • **IF** 1ETA **OR** 1ETB de-energized, **THEN** restore power to affected essential bus. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power). __ • **IF** essential bus energized, **THEN** dispatch operator to determine cause of breaker failure.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 5 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	
__ 2) 1AD-5, F/3 "CAPT MECH OS TRIP" - DARK.	2) Perform the following: __ a) Dispatch operator to reset CAPT trip and throttle valve. __ b) **IF AT ANY TIME** CAPT trip and throttle valve reset prior to reaching feed and bleed criteria, **THEN** perform Step 7. __ c) **GO TO** Step 7.d.
__ 3) "CAPT TRIP T/V CTRL" - OPEN.	3) Perform the following: __ a) OPEN valve. __ b) **IF** valve will not open, **THEN** dispatch operator to open CAPT trip and throttle valve.
4) Verify the following valves - OPEN: __ • 1SA-2 (S/G 1B SM To CAPT) __ • 1SA-5 (S/G 1C SM To CAPT).	__ 4) Place CA Pump #1 to "ON".
__ d. Ensure all CA isolation valves - OPEN.	
__ e. Verify all CA flow control valves - OPEN.	e. Perform the following: __ 1) **IF** valve(s) closed as required by Step 37, **THEN GO TO** Step 7.f. __ 2) OPEN affected valve(s).
__ f. Start all available CA pumps.	

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 6 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	
__ g. Verify total CA flow - GREATER THAN 450 GPM.	g. Perform the following:
	N/A 1) **IF** only one motor driven CA pump on, **AND** its discharge path cannot be aligned to associated S/Gs, **THEN** evaluate aligning flow to other S/Gs through motor driven CA train A/B cross-tie alignment. **REFER TO** Enclosure 3 (Motor Driven CA Pump Train A/B Cross-Tie Alignment).
	N/A 2) **IF** any CA pump on, **AND** Step 37 has been implemented, **THEN GO TO** Enclosure 7 (S/G CA Flow Restoration).
	N/A 3) **IF** any feed flow to at least one S/G verified, **THEN** perform the following:
	__ a) Maintain flow to restore N/R level in at least one S/G to greater than 11% (29% ACC).
	b) **IF AT ANY TIME** N/R level in at least one S/G trends up to greater than 11% (29% ACC) prior to reaching feed and bleed initiation criteria, **THEN** perform the following:
	(1) **IF** NC System **OR** S/G depressurization in progress to feed S/G(s) from CM, **THEN** stabilize the following:
	__ • NC pressure
	__ • S/G pressure.
	__ (2) **RETURN TO** procedure and step in effect.
	(RNO continued on next page)

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	66	of 117
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed							

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 7 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	

__ c) Dispatch operator to verify proper CA valve alignment. **REFER TO** Enclosure 2 (Local CA Flowpath Restoration).

__ d) **IF AT ANY TIME** CA flow restored greater than 450 GPM prior to meeting feed and bleed initiation criteria, **THEN** perform Step 7.

4) **IF** no CA flow indicated, **THEN** perform the following:

__ a) **IF** no CA pump can be started, **THEN** dispatch operator and maintenance to CA pumps to attempt to restore one CA pump to service. **REFER TO** EM/1/A/5200/007 (Troubleshooting Cause For CA Pump(s) Failing to Start).

__ b) Dispatch operator to verify proper CA valve alignment. **REFER TO** Enclosure 2 (Local CA Flowpath Restoration).

__ c) **IF AT ANY TIME** CA flow restored prior to meeting feed and bleed initiation criteria, **THEN** perform Step 7.

__ 5) **GO TO** Step 8.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 8 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	
__ h. Verify feed and bleed - PREVIOUSLY ESTABLISHED (Steps 22 through 26 completed).	h. Perform the following: 1) **IF** NC System **OR** S/G depressurization in progress to feed S/G(s) from CM, **THEN** stabilize the following: __ • NC pressure __ • S/G pressure. __ 2) **RETURN TO** procedure and step in effect.
__ i. **GO TO** Step 39.	
8. **Verify Condenser Steam Dump as follows:**	
a. Verify condenser available as follows: __ • "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT __ • Any MSIV - OPEN.	__ a. **GO TO** Step 9.
__ b. Verify steam dumps in - T-AVG MODE.	__ b. **GO TO** Step 9.
c. Place steam dumps in pressure mode as follows: __ 1) Place steam dumps in - PRESSURE MODE. __ 2) Ensure "STM DUMP CTRL" - SET AT 1090 PSIG STEAM HEADER PRESSURE.	
__ 9. **Stop all NC pumps.**	

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 9 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Verify CM System in service as follows:** __ • Hotwell pump(s) - ON __ • Condensate Booster pump(s) - ON.	**Perform the following:** __ a. Place CM System in service. **<u>REFER TO</u>** OP/1/A/6250/001 (Condensate and Feedwater System). __ b. **<u>IF</u>** CM System cannot be placed in service, **<u>THEN</u>** observe Note prior to Step 19 and **<u>GO TO</u>** Step 19.
11. **Reset Feedwater Isolation as follows:**	
a. Verify the following annunciators - DARK: __ • 1AD-8, D/7 "INNER DOGHOUSE TRAIN A LEVEL HI" __ • 1AD-8, E/7 "INNER DOGHOUSE TRAIN B LEVEL HI" __ • 1AD-8, D/8 "OUTER DOGHOUSE TRAIN A LEVEL HI" __ • 1AD-8, E/8 "OUTER DOGHOUSE TRAIN B LEVEL HI".	__ a. **<u>IF</u>** doghouse level greater than or equal to 11 inches, **<u>THEN</u>** notify IAE to bypass Feedwater Isolation due to Hi-Hi doghouse level. **<u>REFER TO</u>** EM/1/A/5200/008 (Bypassing Feedwater Isolation Due to Hi-Hi Doghouse Level).
__ b. Verify S/I - HAS PREVIOUSLY ACTUATED.	b. Perform the following: __ 1) Reset Feedwater Isolation. N/A 2) **<u>IF</u>** Feedwater Isolation will not reset, **<u>THEN</u>** perform the following: __ a) Notify IAE to bypass Feedwater Isolation. **<u>REFER TO</u>** EM/1/A/5200/009 (Bypassing Feedwater Isolation). __ b) **<u>WHEN</u>** IAE has bypassed Feedwater Isolation signal, **<u>THEN</u>** ensure Feedwater Isolation reset. __ 3) **<u>GO TO</u>** Step 11.f.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 10 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. (Continued)	
__ c. Notify IAE to bypass Feedwater Isolation. **REFER TO** EM/1/A/5200/009 (Bypassing Feedwater Isolation).	
__ d. **WHEN** IAE has bypassed Feedwater Isolation signal, **THEN** ensure Feedwater Isolation reset.	
e. Ensure S/I - RESET:	
__ 1) ECCS.	__ 1) Locally reset ECCS. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 4 (ECCS Master Reset).
__ 2) D/G load sequencers.	2) Dispatch operator to open affected sequencer(s) control power breaker: __ • 1EDE-F01F (Diesel Generator Load Sequencer Panel 1DGLSA) (AB-577, BB-46, Rm 496) __ • 1EDF-F01F (Diesel Generator Load Sequencer Panel 1DGLSB) (AB-560, BB-46, Rm 372).
__ 3) **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.	
__ f. **IF AT ANY TIME** subsequent Feedwater Isolation occurs, **THEN RETURN TO** Step 11.	

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	70 of 117

Event Description: CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 11 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. **Attempt to establish CF flow to at least one S/G as follows:**	
__ a. Verify CM System - IN SERVICE.	__ a. **IF** CM System cannot be placed in service, **THEN** observe Note prior to Step 19 and **GO TO** Step 19.
b. Place the following valves in manual and closed: __ • All CF control valves __ • All CF bypass control valves.	
c. Ensure at least one of the following valves - OPEN: __ • 1CF-10 (1A CF Pump Disch Isol) (TB1-579, 1E-21) __ • 1CF-17 (1B CF Pump Disch Isol) (TB1-579, 1E-20).	__ c. Observe Note prior to Step 19 and **GO TO** Step 19.
__ d. Verify at least one CF pump - AVAILABLE TO BE STARTED.	__ d. **IF** both CF pumps known to be incapable of starting, **THEN GO TO** Step 14.
e. Verify following feedwater pump recirc valves - FULLY OPEN: __ • 1CF-6 (CF Pump 1A Recirc Ctrl) __ • 1CF-13 (CF Pump 1B Recirc Ctrl).	__ e. OPEN valve(s).

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 12 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. (Continued)	
NOTE When CFPT reset, if speed demand raised above "0" the zero speed trip (≤ 2 RPM) arms. If speed has not raised above 2 RPM within 5 seconds CFPT will trip.	
__ f. Ensure CFPT to be started - RESET.	f. Perform the following: __ 1) Continue attempts to reset a CFPT. __ 2) **WHEN** CFPT reset **AND** feed from CF desired, **THEN RETURN TO** Step 10. __ 3) **GO TO** Step 14.
__ g. OPEN 1AS-12 (AS To CFPT Isol).	
__ h. Dispatch operator to close 1SP-3 (SC To CFPT 1A & 1B) (TB1-640, 1G-24).	
__ i. Verify CFPT selected to feed S/G(s) has remained running since loss of S/G feed event occurred.	i. OPEN the following valves for CFPT to be started: • CFPT 1A: __ • 1TE-3 (CFPT A LP S/V Above Seat Drn) __ • 1TE-7 (CFPT A HP S/V Above Seat Drn). • CFPT 1B: __ • 1TE-4 (CFPT B LP S/V Above Seat Drn) __ • 1TE-8 (CFPT B HP S/V Above Seat Drn).
__ j. Place "CFPT RUNBK ON RX TRIP" in - BYPASS.	

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 13 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

12. (Continued)

k. Issue work order to restore VI supply regulators to normal for the following SP valves:

__ • 1SP-29 (1A CFPT HP Drains) (TB-597, 1D-21) (W/O 02187004)

__ • 1SP-33 (1B CFPT HP Drains) (TB-597, 1D-19) (W/O 02187006)

__ • 1SP-40 (1A & 1B CFPT HP Drains) (TB-599, 1C-32) (W/O 02187005)

__ • 1SP-99 (1A CFPT HP Drains) (TB-598, 1C-23) (W/O 02187007).

l. Dispatch operator(s) to ensure VI isolation valve associated with the following SP valves - OPEN:

__ • 1VI-1298 (VI Supply to 1SPSV0290)(TB-597,1D-21)

__ • 1VI-2066 (VI Supply to 1SPSV0330)(TB-598,1C-19)

__ • 1VI-1101 (VI Supply to 1SPSV0400)(TB-600,1C-33)

__ • 1VI-1282 (VI Supply to 1SPSV0990)(TB-598,1C-23)

__ • 1VI-1291 (VI Supply to 1SPSV1230)(TB-606,1B-22).

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 14 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. (Continued)	
m. Ensure the following switches for CFPT to be started in - AUTO:	
• <u>CFPT 1A</u>:	
__ • "1SP-40, 29, 99 CFPT A HP DRNS"	
__ • "1SP-37, 19 CFPT A LP DRNS".	
• <u>CFPT 1B</u>:	
__ • "1SP-40, 33, 123 CFPT B HP DRNS"	
__ • "1SP-37, 23 CFPT B LP DRNS".	
__ n. Slowly raise speed of CFPT to be started to maintain CF pressure 200 PSIG greater than SM pressure for S/G(s) to be fed.	n. Perform the following: __ 1) Continue attempts to raise CFPT speed. __ 2) **<u>WHEN</u>** CFPT speed control available **<u>AND</u>** feed from CF desired, **<u>THEN</u> <u>RETURN</u> <u>TO</u>** Step 10. __ 3) **<u>GO</u> <u>TO</u>** Step 14.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 15 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. (Continued)	
o. CLOSE the following drain valves for CFPT started:	
• CFPT 1A:	
__ • 1TE-1 (CFPT 1A 1st Stage Drn)	
__ • 1TE-7 (CFPT A HP S/V Above Seat Drn)	
__ • 1TE-3 (CFPT A LP S/V Above Seat Drn)	
__ • 1TE-24 (CFPT A HP S/V Below Seat Drn)	
__ • 1TE-5 (CFPT A LP S/V Below Seat Drn).	
• CFPT 1B:	
__ • 1TE-2 (CFPT 1B 1st Stage Drn)	
__ • 1TE-8 (CFPT B HP S/V Above Seat Drn)	
__ • 1TE-4 (CFPT B LP S/V Above Seat Drn)	
__ • 1TE-28 (CFPT B HP S/V Below Seat Drn)	
__ • 1TE-6 (CFPT B LP S/V Below Seat Drn).	
__ p. Verify feed and bleed - PREVIOUSLY ESTABLISHED (Steps 22 through 26 completed).	__ p. **GO TO** Step 12.r.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 16 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. (Continued)	
<u>CAUTION</u> Restoring feed flow at very high flow rates may result in excessive NC System cooldown.	
__ q. Verify all S/G W/R levels - GREATER THAN 12% (21% ACC).	__ q. Establish feed flow to S/Gs. **<u>GO TO</u>** Enclosure 8 (S/G CM/CF Flow Restoration).
__ r. Verify CF Isolation signal - RESET OR BYPASSED.	r. Perform the following: __ 1) **<u>IF AT ANY TIME</u>** it is determined CF Isolation signal cannot be reset or bypassed, **<u>THEN</u>** observe Note prior to Step 19 and **<u>GO TO</u>** Step 19. __ 2) Do not continue until CF Isolation signal reset or bypassed.
s. OPEN at least one of the following valves: __ • 1CA-149 (S/G 1A CF Byp To CA Nozzle) __ • 1CA-150 (S/G 1B CF Byp To CA Nozzle) __ • 1CA-151 (S/G 1C CF Byp To CA Nozzle) __ • 1CA-152 (S/G 1D CF Byp To CA Nozzle).	s. Perform the following: 1) OPEN at least one of the following valves: __ • 1CF-33 (S/G 1A CF Cont Isol) __ • 1CF-42 (S/G 1B CF Cont Isol) __ • 1CF-51 (S/G 1C CF Cont Isol) __ • 1CF-60 (S/G 1D CF Cont Isol). __ 2) **<u>IF</u>** flow path cannot be established to at least one S/G, **<u>THEN</u>** observe Note prior to Step 19 and **<u>GO TO</u>** Step 19.
__ t. THROTTLE open CF control valve or CF bypass control valve for S/G(s) to be fed.	__ t. **<u>IF</u>** flow path cannot be established to at least one S/G, **<u>THEN</u>** observe Note prior to Step 19 and **<u>GO TO</u>** Step 19.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 17 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. **Verify S/G levels as follows:**	
__ a. Verify N/R level in at least one S/G - GREATER THAN 11% (29% ACC).	a. Perform the following: __ 1) **IF** feed flow indicated to at least one S/G, **THEN** maintain feed flow to restore S/G N/R level to greater than 11% (29% ACC). __ 2) **IF** feed flow not indicated, **THEN GO TO** Step 14.
__ b. Verify feed and bleed - PREVIOUSLY ESTABLISHED (Steps 22 through 26 completed).	__ b. **RETURN TO** procedure and step in effect.
__ c. **GO TO** Step 39.	
14. **Depressurize NC System to less than 1905 PSIG and perform blocks as follows:**	
__ a. Verify "P-11 PZR S/I BLOCK PERMISSIVE" status light (1SI-18) - DARK.	a. Perform the following: 1) Depress "BLOCK" pushbuttons for the following signals: __ • ECCS steam pressure __ • ECCS Pzr pressure. 2) Verify the following status lights (1SI-13) - LIT: __ • Main Steam Isol __ • Pzr low pressure S/I. __ 3) **GO TO** Step 15.
__ b. Ensure NC System pressure maintained greater than 1845 PSIG until ECCS Pzr pressure blocked.	

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	77	of 117

Event Description: CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 18 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

14. (Continued)

__ c. Ensure operator prepared to perform blocks in the following step as soon as Pzr pressure goes below P-11, to avoid inadvertent S/I.

d. **WHEN** "P-11 PZR S/I BLOCK PERMISSIVE" status light (1SI-18) lit, **THEN** perform the following:

1) Depress "BLOCK" pushbuttons for the following signals:

__ • ECCS steam pressure

__ • ECCS Pzr pressure.

2) Verify the following status lights (1SI-13) - LIT:

__ • Main Steam Isol

__ • Pzr low pressure S/I.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 19 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. (Continued)	
__ e. Verify letdown - IN SERVICE.	e. Perform the following: __ 1) Depressurize NC System to less than 1905 PSIG using one Pzr PORV. 2) **IF** Pzr PORV will not operate, **THEN** perform the following: a) Align N_2 to Pzr PORVs by opening the following valves: __ • 1NI-438A (Emer N2 From CLA A To 1NC-34A) __ • 1NI-439B (Emer N2 From CLA B To 1NC-32B). b) Depressurize using one of the following Pzr PORVs to less than 1905 PSIG: __ • 1NC-34A (PZR PORV) OR __ • 1NC-32B (PZR PORV). 3) **IF** Pzr PORV available, **THEN** perform the following: __ a) Maintain NC pressure less than 1905 PSIG. __ b) Do not continue until ECCS steam pressure and main steam isolation blocks performed. __ c) **GO TO** Step 15.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 20 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. (Continued)	
f. Depressurize NC System to less than 1905 PSIG using NV aux spray as follows:	__ f. Depressurize NC System to less than 1905 PSIG using one Pzr PORV.
1) Ensure the following valves - OPEN:	
__ • 1NV-312A (Chrg Line Cont Isol)	
__ • 1NV-314B (Chrg Line Cont Isol).	
2) Ensure the following valves - CLOSED:	
__ • 1NC-27 (Pzr Spray Ctrl Frm Loop A)	
__ • 1NC-29 (Pzr Spray Ctrl Frm Loop B)	
__ • 1NV-39A (NV Supply To Loop D Isol)	
__ • 1NV-32B (NV Supply To Loop A Isol).	
__ 3) Maintain charging flow less than 180 GPM.	
__ 4) THROTTLE 1NV-37A (NV Supply To Pzr Aux Spray) and charging flow as required.	

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 21 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

14. (Continued)

5) **IF AT ANY TIME** letdown lost **AND** Pzr PORV available, **THEN** perform the following:

__ a) CLOSE 1NV-37A (NV Supply To Pzr Aux Spray).

b) OPEN one of the following valves:

__ • 1NV-39A (NV Supply To Loop D Isol)

OR

__ • 1NV-32B (NV Supply To Loop A Isol).

__ c) Depressurize NC System to less than 1905 PSIG using one Pzr PORV.

__ g. Do not continue until ECCS steam pressure and main steam isolation blocks performed.

__ 15. **IF AT ANY TIME conditions degrade, THEN manual S/I actuation may be required.**

CAUTION **Failure to monitor Pzr pressure and maintain pressure below P-11 may result in an S/I, main steamline isolation and subsequent loss of feed flow. An operator must monitor Pzr pressure even after leaving this EP.**

__ 16. **Designate operator to continuously monitor and control Pzr pressure. REFER TO Enclosure 4 (Maintaining Pzr pressure below P-11).**

Note to Evaluator:

Enclosure 4 can be found as Attachment 9 in the back of this document.

Op Test No.: 301	Scenario # 3	Event # 6,7,8	Page 81 of 117				
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed						

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 22 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. **Attempt to establish feed flow from CM as follows:**	
NOTE • If feed and bleed has not yet been initiated, it is preferable to depressurize two S/Gs in the next step in order to: • Leave two S/G levels above feed and bleed criteria • Minimize NC System cooldown. • If feed and bleed has been initiated, it is preferable to depressurize just one S/G in the following steps.	
__ a. Depressurize at least one S/G to less than 500 PSIG in the following steps.	
b. Verify condenser available as follows:	__ b. **GO TO** Step 17.i **RNO**.
__ • "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT	
__ • MSIV on S/Gs to be depressurized - OPEN.	
__ c. Place "STM DUMP CTRL" slim station in manual.	
__ d. Ensure steam dumps in pressure mode.	
__ e. **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dump interlock bypass switches in "BYP INTLK".	

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 23 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ f. Verify feed and bleed - PREVIOUSLY ESTABLISHED (Steps 22 through 26 completed).	f. Perform the following: __ 1) CLOSE MSIV on two S/Gs **NOT** to be depressurized. __ 2) Ensure S/G PORV closed or in "AUTO" on two S/Gs **NOT** to be depressurized. __ 3) **GO TO** Step 17.i.
__ g. CLOSE MSIV on three S/Gs **NOT** to be depressurized.	
__ h. Ensure S/G PORV closed or in "AUTO" on three S/Gs **NOT** to be depressurized.	

NOTE
- After low steamline pressure main steam isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation.
- OAC graphic SMRATES to monitor S/G pressure rates can be accessed via a hot button in center of SM graphic.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ i. Dump steam to condenser at maximum rate while attempting to avoid Main Steam Isolation.	i. Perform the following: __ 1) CLOSE MSIV on S/Gs **NOT** to be depressurized. __ 2) CLOSE manual loaders on S/G PORV(s) on S/G(s) **NOT** to be depressurized.
Critical Task #3	__ 3) Dump steam from S/G(s) selected to be depressurized using S/G PORV(s) at maximum rate.

(RNO continued on next page)

Event Description: CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 24 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	N/A 4) **IF** PORV on S/G(s) to be depressurized can **NOT** be operated from Control Room, **THEN** perform the following:
	__ a) Dispatch operator(s) to operate intact S/G(s) PORV. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 10 (Local Operation of S/G PORVs).
	__ b) Obtain sound powered phone from storage box on rear wall of Control Room.
	__ c) Connect sound powered phone to jack on 1MC-11.
	__ d) Monitor sound powered phone for communication from the Doghouse(s).
	N/A 5) **IF** no S/G PORV can be opened, **THEN** perform the following:
	__ a) Evaluate REOPENING MSIVs and dump steam to condenser. **REFER TO** Enclosure 15 (Condenser Dump Operation).
	__ b) Observe Note prior to Step 19 and **GO TO** Step 19.

NOTE Continuing in this step to align feed path open, while waiting for S/G(s) pressure to reach 500 PSIG, will allow S/G(s) to be fed as soon as S/G pressure goes below CF header pressure.

__ j. **WHEN** S/G pressure less than 500 PSIG, **THEN** stabilize S/G pressure less than 500 PSIG.

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 25 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ k. Verify feed and bleed - PREVIOUSLY ESTABLISHED (Steps 22 through 26 completed).	__ k. **GO TO** Step 17.m.
CAUTION **Restoring feed flow at very high flow rates may result in excessive NC System cooldown.**	
__ l. Verify all S/G W/R levels - GREATER THAN 12% (21% ACC).	__ l. Establish feed flow to S/Gs. **GO TO** Enclosure 8 (S/G CM/CF Flow Restoration).
__ m. Verify CF Isolation signal - RESET OR BYPASSED.	m. Perform the following: __ 1) **IF AT ANY TIME** it is determined CF Isolation signal cannot be reset or bypassed, **THEN** observe Note prior to Step 19 and **GO TO** Step 19. __ 2) Do not continue until CF Isolation signal reset or bypassed.
Note to Evaluator: **Only 2 of the valves in the following step are part of the critical task at this time. Only the S/Gs selected to be fed are critical.**	
n. OPEN CF to CA valve on S/G(s) to be fed: __ • 1CA-149 (S/G 1A CF Byp To CA Nozzle) __ • 1CA-150 (S/G 1B CF Byp To CA Nozzle) __ • 1CA-151 (S/G 1C CF Byp To CA Nozzle) __ • 1CA-152 (S/G 1D CF Byp To CA Nozzle). **Critical Task # 3**	n. Perform the following: 1) OPEN CF Cont Isol valve on S/G(s) to be fed: __ • 1CF-33 (S/G 1A CF Cont Isol) __ • 1CF-42 (S/G 1B CF Cont Isol) __ • 1CF-51 (S/G 1C CF Cont Isol) __ • 1CF-60 (S/G 1D CF Cont Isol). __ 2) **IF** flow path cannot be established to at least one depressurized S/G, **THEN** observe Note prior to Step 19 and **GO TO** Step 19.

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	85 of 117
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed						

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 26 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. (Continued)	
__ o. THROTTLE open CF control valve or CF bypass control valve for S/G(s) to be fed. **Critical Task # 3**	__ o. **IF** flow path cannot be established to at least one depressurized S/G, **THEN** observe Note prior to Step 19 and **GO TO** Step 19.
__ p. Verify feedwater flow to depressurized S/G(s) - INDICATING FLOW.	p. Perform the following: __ 1) **IF** depressurized S/G pressure less than 500 PSIG, **THEN** observe Note prior to Step 19 and **GO TO** Step 19. __ 2) **RETURN TO** Step 17.i.
18. **Verify S/G levels as follows:**	
__ a. Verify N/R level in at least one S/G - GREATER THAN 11% (29% ACC).	a. Perform the following: __ 1) **IF** feed flow indicated to at least one S/G, **THEN** maintain feed flow to restore S/G N/R level to greater than 11% (29% ACC). N/A 2) **IF** feed flow not indicated, **THEN** observe Note prior to Step 19 and **GO TO** Step 19.
b. While performing actions in subsequent procedures, maintain the following conditions required for feed flow from CM System: __ • S/G(s) pressure - LESS THAN 500 PSIG __ • Feed flow path - OPEN __ • NC System pressure - LESS THAN 1905 PSIG.	

Op Test No.:	301	Scenario #	3	Event #	6,7,8	Page	86 of 117
Event Description:	CAPT#1 Overspeed Trip / Loss of Heat Sink / Loss of CFPT Vacuum / 1NV-37A Fails Closed						

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK	PAGE NO. 27 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18. (Continued)	
__ c. Verify feed and bleed - PREVIOUSLY ESTABLISHED (Steps 22 through 26 completed)	c. Perform the following: __ 1) **IF** S/I has occurred, **THEN** **RETURN TO** procedure and step in effect. 2) **IF** S/I has not occurred, **THEN** perform the following: __ a) **IF** Pzr level **OR** Pzr pressure low due to operator controlled cooldown in this procedure, **THEN** S/I actuation requirements based on Pzr level or NC pressure do not apply in subsequent procedures unless conditions degrade. b) Control charging as follows: __ • Maintain charging flow less than 180 GPM __ • Restore Pzr level to greater than 11%. __ c) **RETURN TO** procedure and step in effect.
__ d. **GO TO** Step 39.	

Note to Evaluator:

At this time all critical tasks and the scenario are complete. At the discretion of the Lead Evaluator, the scenario may be terminated by having the booth operator place the simulator in FREEZE.

Attachment List

Scenario 3

ATTACHMENT 1 -	Crew Critical Task Summary
ATTACHMENT 2 -	Shift Turnover Information
ATTACHMENT 3 -	AP/1/A/5500/007 Enclosure 1 (Foldout Page)
ATTACHMENT 4 -	EP/1/A/5000/E-0 Enclosure 1 (Foldout Page)
ATTACHMENT 5 -	EP/1/A/5000/ES-0.1 Enclosure 1 (Foldout Page)
ATTACHMENT 6 -	EP/1/A/5000/ES-0.1 Enclosure 2 (NC Temperature Control)
ATTACHMENT 7 -	EP/1/A/5000/FR-H.1 Enclosure 1 (Foldout Page)
ATTACHMENT 8 -	EP/1/A/5000/FR-H.1 Enclosure 9 (S/G BB and NM Valve Checklist)
ATTACHMENT 9	EP/1/A/5000/FR-H.1 Enclosure 4 (Maintaining Pzr pressure below P-11)
ATTACHMENT 10 -	Scenario Specific Technical Specifications

ATTACHMENT 1

CREW CRITICAL TASK SUMMARY			
SAT	**UNSAT**	**CT #**	**CRITICAL TASK**
		1	Control Charging flow to prevent a Reactor Trip on Pressurizer High Level of 92%.
		2	Close Pressurizer PORV prior to reactor trip on Pressurizer low pressure.
		3	Establish feedwater flow to at least one S/G prior to meeting bleed and feed criteria (<24% W/R level in 3 out of 4 S/Gs).

Comments:

ATTACHMENT 2

SHIFT TURNOVER INFORMATION

Unit 1 Status

Power Level	Power History	NCS Boron	Xenon
100 **%**	BOL	1366 PPM	per OAC

Controlling Procedure

- OP/1/A/6100/003 (Controlling Procedure for Unit Operation), Enclosure 4.3 (Operation Between 85 and 100%). The steps through step 3.4 are complete.

Other Information Needed to Assume the Shift

- Unit 1 is at 100% power at the BOL. Unit 2 is at 100% power. 1B CA Pump is removed from service for PMs. 1B CA Pump has been inoperable for 3 hours and is expected to be returned to service in 6 hours. Direction for the crew is to initiate a downpower to ~85% in preparation for performing the Turbine Control Valve Movement PT.

AOs Available

Eight AOs are available as listed on the status board

METEOROLOGICAL CONDITIONS

- Upper wind direction = 125 degrees, speed = 3 mph
- Lower wind direction = 127 degrees, speed = 4.5 mph
- Forecast calls for clear skies over the next 24 hours.

ATTACHMENT 3

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Enclosure 1 - Page 1 of 3 **Foldout Page**	PAGE NO. 84 of 280 Revision 85

1. **SSF Manning Criteria:**

CAUTION **Failure to restore NC pump seal cooling via thermal barrier cooling or NV seal injection within ten minutes will cause damage to NC pump seals resulting in NC System inventory loss.**

IF AT ANY TIME KC **AND** NV seal cooling for any NC pump lost, **THEN** perform the following:

__ a. Dispatch operator to SSF to establish NC pump seal injection. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 19 (Establishing NC Makeup/Seal Injection From The SSF).

b. **IF** 1EMXS de-energized, **THEN** perform the following:

__ 1) Dispatch operator to 1ETA switchgear room to align alternate power supply to 1EMXS. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 20 (Align Alternate Power Supply To 1EMXS).

__ 2) Notify operator at SSF (Ext. 5251 or 5212) an operator has been dispatched to align alternate power supply to 1EMXS.

2. **Containment Air Release Criterion:**

- **IF AT ANY TIME** containment pressure greater than or equal to .25 PSIG, **THEN** perform the following:

__ • Perform normal VQ release. **REFER TO** OP/1/A/6450/017 (Containment Air Release and Addition System)

OR

__ • **IF** VQ fans cannot be started, **THEN REFER TO** Enclosure 13 (VQ Release Without Fans).

__ 3. **IF Containment entry required, THEN coordinate with TSC to defeat 1EMF-17 input to containment evacuation alarm per AM/0/B/5100/009 (Defeating 1EMF-17 or 2EMF-2 Containment Evacuation Alarm Circuit(s)).**

ATTACHMENT 3

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Enclosure 1 - Page 2 of 3 **Foldout Page**	PAGE NO. 85 of 280 Revision 85

4. **NC Pump Trip Criteria:**

IF any of the following NC pump trip criteria met:

__ • #1 Seal outlet temperature - GREATER THAN 235°F

OR

__ • Lower bearing temperature - GREATER THAN 225°F

OR

__ • Motor bearing temperature - GREATER THAN 195°F,

THEN perform the following:

__ a. Ensure steam dumps - IN PRESSURE MODE.

__ b. Ensure "STM DUMP CTRL" - SET AT 1090 PSIG STEAM HEADER PRESSURE.

__ c. Ensure Reactor - TRIPPED.

d. **WHEN** reactor power less than 5%, **THEN** perform the following:

__ 1) Trip all NC pumps.

__ 2) Ensure normal spray valve associated with tripped NC pump(s) - IN MANUAL **AND** CLOSED.

__ e. Secure any dilutions in progress.

f. **IF** reactor trip breakers were closed, **THEN** perform one of the following while continuing with this procedure as time and conditions allow:

__ • **IF** above P-11, **THEN GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection)

OR

__ • **IF** below P-11, **THEN GO TO** AP/1/A/5500/005 (Reactor Trip or Inadvertent S/I Below P-11).

ATTACHMENT 3

CNS AP/1/A/5500/007	LOSS OF NORMAL POWER Enclosure 1 - Page 3 of 3 **Foldout Page**	PAGE NO. 86 of 280 Revision 85

CAUTION **This step should be a high priority if LOCA or other event is anticipated to deplete the FWST.**

NOTE FWST level transmitter heat tracing is **NOT** available in LOOP events.

5. **IF Unit 1 LOOP event in progress and freezing weather exists, THEN perform the following:**

__ • Notify Control Room team FWST level instruments have lost heat tracing and may be inaccurate until external heating established.

• Notify TSC to evaluate the following to provide supplemental freeze protection:

__ • Portable heaters

__ • Additional insulation

NOTE Temporary heat trace equipment is available in the FLEX Dome behind rack "J" in box labeled "Heat Trace".

__ • Portable generators/temporay heat trace.

ATTACHMENT 4

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 1 of 3 **Foldout Page**	PAGE NO. 34 of 49 Revision 46

1. **NC Pump Trip Criteria:**

- **IF** the following conditions satisfied, **THEN** trip all NC pumps while maintaining seal injection flow:
  - Any NV or NI pump - DELIVERING S/I FLOW TO NC SYSTEM
  - NC subcooling based on core exit T/Cs - LESS THAN OR EQUAL TO 0°F
  - Reactor power - LESS THAN 5%.

2. **Open Phase Criteria:**

- **IF** operating NV **AND** KC pumps automatically trip, **THEN** perform the following:
  a. Start the following pumps on opposite train:
     - NV pump
     - KC pumps
     - RN pump.

  b. **IF** pumps do not start, **OR** trip after starting, **THEN** restart pumps on previously operating train.

  c. **IF** all KC pumps off, **THEN** ensure all NC pumps - OFF.

  d. **IF** Unit 2 4160V bus energized by Unit 1 busline, **THEN** immediately notify Unit 2 to perform same actions on Unit 2.

3. **CA Suction Source Switchover Criterion:**

- **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

4. **Position Criteria for 1NV-202B and 1NV-203A (NV Pumps A&B Recirc Isol):**

- **IF** NC pressure less than 1500 PSIG **AND** NV S/I flowpath aligned, **THEN** CLOSE 1NV-202B and 1NV-203A.
- **IF** NC pressure greater than 2000 PSIG, **THEN** OPEN 1NV-202B and 1NV-203A.

ATTACHMENT 4

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 2 of 3 **Foldout Page**	PAGE NO. 35 of 49 Revision 46

NOTE CA flow control valves fail open on CA auto start. Isolating flow with the motor operated isolation valve will not require re-isolation on subsequent CA auto starts.

5. **Ruptured S/G CA Isolation Criteria:**

- **IF** both the following conditions met, **THEN** stop CA flow to affected S/G(s):
  - Level going up in uncontrolled manner or radiation level in that S/G abnormal
  - N/R level - GREATER THAN 11% (29% ACC).

NOTE CA flow control valves fail open on CA auto start. Isolating flow with the motor operated isolation valve will not require re-isolation on subsequent CA auto starts.

6. **Faulted S/G CA isolation Criteria:**

- **IF** all the following conditions met, **THEN** stop CA flow to affected S/G:
  - S/G pressure trends down in uncontrolled manner or completely depressurized
  - Only one S/G diagnosed as faulted
  - Secondary heat sink criteria met:
    - Total CA flow - GREATER THAN 450 GPM

      OR

    - ANY S/G(s) N/R level - GREATER THAN 11% (29% ACC).

ATTACHMENT 4

CNS EP/1/A/5000/E-0	REACTOR TRIP OR SAFETY INJECTION Enclosure 1 - Page 3 of 3 **Foldout Page**	PAGE NO. 36 of 49 Revision 46

7. **NS Pump Trip Criterion:**

- **IF** NS pump in recirc and S/I occurs, **THEN** perform one of the following:
  - **IF** train affected ECCS and D/G load sequencer - RESET, **THEN** stop NS pump

    OR
  - **WHEN** sequencer loading complete, **THEN** perform the following for affected train:

    a. Notify Control Room Supervisor.

    b. Reset ECCS.

    c. Reset D/G load sequencer.

    d. Secure NS pump.

    e. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.

8. **IF AT ANY TIME KC cooling to operating KF pump(s) lost, THEN perform the following:**

- **IF** annunciator 1AD-13, D/6 "KF PUMP A MTR CLR HI TEMP" lit, **THEN** secure 1A KF pump and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).
- **IF** annunciator 1AD-13, D/7 "KF PUMP B MTR CLR HI TEMP" lit, **THEN** secure 1B KF pump and **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).

ATTACHMENT 5

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE Enclosure 1 - Page 1 of 1 **Foldout Page**	PAGE NO. 27 of 41 Revision 48

1. **S/I Actuation Criteria:**

   - **IF** NC subcooling based on core exit T/Cs less than 0°F **OR** Pzr level cannot be maintained greater than 4%, **THEN** perform the following:

     a. Initiate S/I.

     b. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

   - **IF** S/I actuation occurs, **THEN GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

2. **Open Phase Criteria:**

   - **IF** operating NV **AND** KC pumps automatically trip, **THEN** perform the following:

     a. Start the following pumps on opposite train:

        - NV pump
        - KC pumps
        - RN pump.

     b. **IF** pumps do not start, **OR** trip after starting, **THEN** restart pumps on previously operating train.

     c. **IF** all KC pumps off, **THEN** ensure all NC pumps - OFF.

     d. **IF** Unit 2 4160V bus energized by Unit 1 busline, **THEN** immediately notify Unit 2 to perform same actions on Unit 2.

3. **CA Suction Source Switchover Criterion:**

   - **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

ATTACHMENT 6

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE Enclosure 2 - Page 1 of 6 **NC Temperature Control**	PAGE NO. 28 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Verify at least one NC pump - ON.**	**Perform the following:** __ a. Use NC T-Colds to determine NC temperature as required in subsequent steps. __ b. **<u>IF</u>** all MSIVs closed, **<u>THEN</u> <u>GO</u> <u>TO</u>** Step 4. c. Place steam dumps in pressure mode as follows: __ 1) Place steam dumps in pressure mode using "STEAM DUMP SELECT" switch __ 2) Ensure "STM DUMP CTRL" - SET AT 1090 PSIG STEAM HEADER PRESSURE. __ 3) Ensure steam dumps control to maintain steam header pressure - 1090 PSIG. __ d. **<u>GO</u> <u>TO</u>** Step 4.
__ 2. **Use NC T-Avg to determine NC temperature as required in subsequent steps.**	
__ 3. **<u>IF</u> <u>AT</u> <u>ANY</u> <u>TIME</u> NC pumps tripped, <u>THEN</u> use NC T-Colds to determine NC temperature as required in subsequent steps.**	
4. **Verify one of the following:** __ • NC temperature - STABLE AT LESS THAN OR EQUAL TO 557°F OR __ • NC temperature - TRENDING TO 557°F.	__ **<u>GO</u> <u>TO</u> Step 7.**

ATTACHMENT 6

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE Enclosure 2 - Page 2 of 6 **NC Temperature Control**	PAGE NO. 29 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

__ 5. **Continue to monitor NC temperature.**

6. **Do not continue in this enclosure until one of the following occurs:**

- __ NC temperature - GREATER THAN 557°F AND TRENDING UP IN UNCONTROLLED MANNER

  OR

- __ NC temperature - GREATER THAN 557°F AND STABLE

  OR

- __ NC temperature - LESS THAN 557°F AND TRENDING DOWN IN UNCONTROLLED MANNER.

ATTACHMENT 6

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE Enclosure 2 - Page 3 of 6 **NC Temperature Control**	PAGE NO. 30 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 7. **Verify NC temperature - LESS THAN 557°F AND TRENDING DOWN.**	**Perform the following:** a. **IF** NC temperature greater than 557°F **AND** trending up, **THEN** stabilize NC temperature at 557°F as follows: __ 1) **IF** steam dumps available, **THEN** use steam dumps. __ 2) **IF** steam dumps not available, **THEN** use S/G PORVs. b. **IF** the following conditions exist: __ • NC temperature greater than 557°F and stable __ • Time and manpower available, **THEN** stabilize NC temperature at 557°F as follows: __ 1) **IF** steam dumps available, **THEN** use steam dumps __ 2) **IF** steam dumps not available, **THEN** use S/G PORVs. __ c. **GO TO** Step 9.

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE Enclosure 2 - Page 4 of 6 **NC Temperature Control**	PAGE NO. 31 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. **Attempt to stop NC cooldown as follows:**	
a. **IF** steam dumps open **AND** any MSIV open, **THEN** place steam dumps in pressure mode as follows:	
__ 1) Place steam dumps in pressure mode using "STEAM DUMP SELECT" switch	
__ 2) Ensure "STM DUMP CTRL" - SET AT 1090 PSIG STEAM HEADER PRESSURE.	
__ 3) Ensure steam dumps control to maintain steam header pressure - 1090 PSIG.	
__ b. Verify all S/G PORVs - CLOSED.	__ b. **IF** any S/G PORV cannot be closed, **THEN** CLOSE its isolation valve.
__ c. Ensure S/G blowdown isolated.	
d. CLOSE the following valves: __ • 1SM-77A (S/G 1A Otlt Hdr Bldwn C/V) __ • 1SM-76B (S/G 1B Otlt Hdr Bldwn C/V) __ • 1SM-75A (S/G 1C Otlt Hdr Bldwn C/V) __ • 1SM-74B (S/G 1D Otlt Hdr Bldwn C/V).	
e. Verify MSR Second Stage steam supply valves - CLOSED: __ • 1HM-1 (MSRH 1A&1B SSRH Stm Source) __ • 1HM-2 (MSRH 1C&1D SSRH Stm Source).	e. Perform the following: __ 1) CLOSE MSR Second Stage steam supply valve(s). 2) **IF** steam flowpath cannot be isolated from Control Room, **THEN** CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves.

ATTACHMENT 6

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE Enclosure 2 - Page 5 of 6 **NC Temperature Control**	PAGE NO. 32 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. (Continued)	
f. Depress and hold "S/V BEFORE SEAT DRN" "CLOSE" pushbutton (1MC-3) to close the following valves: __ • 1SM-41 (Stop Vlv #1 Before Seat Drn) __ • 1SM-44 (Stop Vlv #2 Before Seat Drn) __ • 1SM-43 (Stop Vlv #3 Before Seat Drn) __ • 1SM-42 (Stop Vlv #4 Before Seat Drn).	
__ g. Verify NC cooldown - STOPPED.	g. **<u>IF</u>** cooldown continues, **<u>THEN</u>** THROTTLE feed flow as follows: 1) **<u>IF</u>** S/G N/R level less than 11% in all S/G's, **<u>THEN</u>** THROTTLE feed flow to achieve the following: __ • Minimize cooldown __ • Maintain total feed flow greater than 450 GPM. 2) **<u>WHEN</u>** N/R level greater than 11% in any S/G, **<u>THEN</u>** THROTTLE feed flow further to achieve the following: __ • Minimize cooldown __ • Maintain at least one S/G N/R level greater than 11%. 3) **<u>IF</u>** cooldown continues, **<u>THEN</u>** CLOSE the following valves: __ • All MSIVs __ • All MSIV bypass valves. __ 4) **<u>IF</u>** cooldown continues **<u>AND</u>** faulted S/G exists, **<u>THEN</u>** stop feeding faulted S/G.

ATTACHMENT 6

CNS EP/1/A/5000/ES-0.1	REACTOR TRIP RESPONSE Enclosure 2 - Page 6 of 6 **NC Temperature Control**	PAGE NO. 33 of 41 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. **Continue to perform actions of this enclosure as required to ensure one of the following:** __ • NC temperature - STABLE AT LESS THAN OR EQUAL TO 557°F OR __ • NC temperature - TRENDING TO 557°F.	

ATTACHMENT 7

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK Enclosure 1 - Page 1 of 1 **Foldout Page**	PAGE NO. 69 of 134 Revision 48

1. **Feed and Bleed Initiation Criteria:**

   - **IF** W/R level in at least 3 S/Gs less than 24% (36% ACC), **THEN GO TO** Section C. (Operator Actions), Step 21.

2. **IF AT ANY TIME a CA pump restored after Step 7 AND prior to meeting Feed and Bleed Initiation Criteria, THEN perform Section C. (Operator Actions), Step 7.**

3. **CA Suction Source Switchover Criterion:**

   - **IF** 1AD-8, B/1 "UST LO LEVEL" lit, **THEN REFER TO** AP/1/A/5500/006 (Loss of S/G Feedwater).

4. **Cold Leg Recirc Switchover Criterion:**

   - **IF** FWST level lowers to 20% (1AD-9, D/8 "FWST 2/4 LO LEVEL"), **THEN GO TO** EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation).

5. **Position Criteria for 1NV-202B and 1NV-203A (NV Pumps A&B Recirc Isol):**

   - **IF** NC pressure less than 1500 PSIG **AND** NV S/I flowpath aligned, **THEN** CLOSE 1NV-202B and 1NV-203A.
   - **IF** NC pressure greater than 2000 PSIG, **THEN** OPEN 1NV-202B and 1NV-203A.

ATTACHMENT 8

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE The following valves are closed to minimize S/G inventory loss.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **Verify the following valves - CLOSED:**	__ **CLOSE valve(s).**
__ • 1BB-56A (S/G 1A Bldwn Cont Isol Insd)	
__ • 1BB-148B (S/G 1A Bldwn Cont Isol Byp)	
__ • 1BB-57B (S/G 1A Bldwn Cont Isol Otsd)	
__ • 1BB-19A (S/G 1B Bldwn Cont Isol Insd)	
__ • 1BB-150B (S/G 1B Bldwn Cont Isol Byp)	
__ • 1BB-21B (S/G 1B Bldwn Cont Isol Otsd)	
__ • 1BB-60A (S/G 1C Bldwn Cont Isol Insd)	
__ • 1BB-149B (S/G 1C Bldwn Cont Isol Byp)	
__ • 1BB-61B (S/G 1C Bldwn Cont Isol Otsd)	
__ • 1BB-8A (S/G 1D Bldwn Cont Isol Insd)	
__ • 1BB-147B (S/G 1D Bldwn Cont Isol Byp)	
__ • 1BB-10B (S/G 1D Bldwn Cont Isol Otsd)	
__ • 1NM-191B (S/G 1A Smpl Hdr Cont Isol)	
__ • 1NM-201A (S/G 1B Smpl Hdr Cont Isol)	
__ • 1NM-211B (S/G 1C Smpl Hdr Cont Isol)	
__ • 1NM-221A (S/G 1D Smpl Hdr Cont Isol)	

ATTACHMENT 8

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. (Continued) __ • 1NM-187A (S/G A UPR Shell Smpl Cont Isol) __ • 1NM-197B (S/G B UPR Shell Smpl Cont Isol) __ • 1NM-207A (S/G C UPR Shell Smpl Cont Isol) __ • 1NM-217B (S/G D UPR Shell Smpl Cont Isol) __ • 1NM-190A (S/G 1A Bldwn Smpl Cont Isol) __ • 1NM-200B (S/G 1B Bldwn Smpl Cont Isol) __ • 1NM-210A (S/G 1C Bldwn Smpl Cont Isol) __ • 1NM-220B (S/G 1D Bldwn Smpl Cont Isol).	

ATTACHMENT 9

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK Enclosure 4 - Page 1 of 3 **Maintaining Pzr pressure below P-11**	PAGE NO. 78 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE **Failure to secure NV aux spray when less than or equal to 1800 PSIG will cause excessive depressurization.**	
__ 1. **Verify 1NV-37A (NV Supply To Pzr Aux Spray) - CLOSED.**	**Perform the following:** a. **WHEN** Pzr pressure less than or equal to 1800 PSIG, **THEN** isolate NV aux spray as follows: 1) Ensure one of the following valves - OPEN: __ • 1NV-39A (NV Supply To Loop D Isol) OR __ • 1NV-32B (NV Supply To Loop A Isol). __ 2) CLOSE 1NV-37A (NV Supply To Pzr Aux Spray). __ 3) Control charging at desired flow rate while maintaining charging flow less than 180 GPM. __ b. Do not continue in this enclosure until 1NV-37A closed in previous step.

ATTACHMENT 9

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK Enclosure 4 - Page 2 of 3 **Maintaining Pzr pressure below P-11**	PAGE NO. 79 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. **IF AT ANY TIME Pzr pressure approaches 1905 PSIG, THEN maintain pressure between 1905 PSIG and 1800 PSIG as follows:**

__ • **IF** normal letdown isolated, **OR** NV aux spray not effective in maintaining pressure below 1905 PSIG, **THEN** cycle Pzr PORV

OR

• **IF** normal letdown in service, **OR** all Pzr PORVs unavailable, **THEN** use NV aux spray as follows:

• **WHEN** required to depressurize to stay less than 1905 PSIG, **THEN** initiate spray as follows:

__ a. THROTTLE 1NV-37A (NV Supply To Pzr Aux Spray) **AND** charging flow as required to control Pzr pressure.

b. Ensure the following valves - CLOSED:

__ • 1NV-32B (NV Supply To Loop A Isol)

__ • 1NV-39A (NV Supply To Loop D Isol).

• **WHEN** Pzr pressure less than or equal to 1800 PSIG, **THEN** isolate NV aux spray as follows:

a. Ensure one of the following valves - OPEN:

__ • 1NV-39A (NV Supply To Loop D Isol)

OR

__ • 1NV-32B (NV Supply To Loop A Isol).

__ b. CLOSE 1NV-37A (NV Supply To Pzr Aux Spray).

ATTACHMENT 9

CNS EP/1/A/5000/FR-H.1	RESPONSE TO LOSS OF SECONDARY HEAT SINK Enclosure 4 - Page 3 of 3 **Maintaining Pzr pressure below P-11**	PAGE NO. 80 of 134 Revision 48

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. **IF AT ANY TIME Pzr pressure inadvertently goes above 1955 PSIG, THEN ensure the following performed again (to reinstate blocks):**

__ a. Ensure another operator prepared to perform blocks in next step as soon as Pzr pressure goes below P-11, to avoid an inadvertent S/I.

b. **WHEN** "P-11 PZR S/I BLOCK PERMISSIVE" status light (1SI-18) lit, **THEN** perform the following:

1) Depress "BLOCK" pushbuttons for the following signals:

__ • ECCS steam pressure

__ • ECCS Pzr pressure.

2) Verify the following status lights (1SI-13) - LIT:

__ • Main Steam Isol

__ • Pzr low pressure S/I.

__ c. Ensure NC System pressure maintained greater than 1845 PSIG until ECCS Pzr pressure blocked.

__ d. **RETURN TO** Step 2 to ensure Pzr pressure maintained below P-11.

ATTACHMENT 10

Event #3
Loss of 1ETA

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level $\leq$ 92% (1656 ft^3); and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group $\geq$ 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours 12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

ATTACHMENT 10

Event #3
Loss of 1ETA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 92% (1656 ft^3).	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is ≥ 150 kW.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.3	Verify required pressurizer heaters are capable of being powered from an emergency power supply.	In accordance with the Surveillance Frequency Control Program

ATTACHMENT 10

Event #3
Loss of 1ETA

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems—Operating

LCO 3.8.9 Train A and Train B AC, four channels of DC, DC Train A and Train B and four AC vital buses electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystem(s) inoperable.	A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours AND 16 hours from discovery of failure to meet LCO
B. One AC vital bus inoperable.	B.1 Restore AC vital bus subsystem to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet LCO

(continued)

ATTACHMENT 10

Event #3

Loss of 1ETA Distribution Systems - Operating
3.8.9

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel of DC electrical power distribution subsystems inoperable.	C.1 Restore channel of DC electrical power distribution subsystems to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet LCO
D. One train of DC electrical power distribution subsystems inoperable.	D.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours AND 16 hours from discovery of failure to meet LCO
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours
F. Two trains with inoperable distribution subsystems that result in a loss of safety function.	F.1 Enter LCO 3.0.3.	Immediately

ATTACHMENT 10

Event #3

Loss of 1ETA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC channel, DC train, and AC vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

ATTACHMENT 10

Event #4
1NC-32B Fails Open

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTES----------
Separate Condition entry is allowed for each PORV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One or two PORVs inoperable and not capable of being manually cycled.	B.1 Close associated block valves.	1 hour
	AND	
	B.2 Remove power from associated block valves.	1 hour
	AND	
	B.3 Restore PORV(s) to OPERABLE status.	72 hours

(continued)

ATTACHMENT 10

Event #4

1NC-32B Fails Open

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One block valve inoperable.	C.1 Place associated PORV in manual control.	1 hour
	AND	
	C.2 Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 4.	12 hours
E. Three PORVs inoperable and not capable of being manually cycled.	E.1 Close associated block valves.	1 hour
	AND	
	E.2 Remove power from associated block valves.	1 hour
	AND	
	E.3 Be in MODE 3.	6 hours
	AND	
	E.4 Be in MODE 4.	12 hours
F. More than one block valve inoperable.	F.1 Place associated PORVs in manual control.	1 hour
	AND	
		(continued)

ATTACHMENT 10

Event #4
1NC-32B Fails Open

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. (continued)	F.2 Restore one block valve to OPERABLE status if three block valves are inoperable. AND F.3 Restore remaining block valve(s) to OPERABLE status.	2 hours 72 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-------------------------------NOTE----------------------------------- Not required to be met with block valve closed in accordance with the Required Action of Condition B or E. -- Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program

(continued)

ATTACHMENT 10

Event #4

1NC-32B Fails Open

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.11.2	-----NOTE----- Required to be performed in MODE 3 or MODE 4 when the temperature of all RCS cold legs is > 200°F. ---------- Perform a complete cycle of each PORV.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.3	-----NOTE----- This SR is not applicable to valve NC-36B. ---------- Verify the nitrogen supply for each PORV is OPERABLE by: a. Manually transferring motive power from the air supply to the nitrogen supply, b. Isolating and venting the air supply, and c. Operating the PORV through one complete cycle.	In accordance with the Surveillance Frequency Control Program

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can effect reactivity. (R.M.)

1.2 Low load operation limits:

1.2.1 The unit can be operated continuously at low loads when exhaust hood temperature is < 175°F. The load shall, however, be increased slowly until the temperature decreases below 125°F before increasing load at normal rate (Multipoint Recorder on 1MC3).

1.2.2 Excessive use of the exhaust hood sprays shall be avoided to prevent accelerated blade erosion.

1.3 Do **NOT** exceed the load, hydrogen pressure, and power factor limits per the Unit One Revised Data Book Figure 43.

1.4 If the limits of the Unit One Revised Data Book Figure 43 (Generator Capability Curves) are exceeded, the Turbine Generator shall be tripped.

1.5 Under certain environmental conditions, indicated condenser vacuum less than 24.3 inches Hg may be reached at full load. Exhaust hood temperatures are a more accurate indicator of true vacuum. It is recommended the turbine **NOT** be operated under the following conditions at full load:

- Exhaust Hood 1A temperature ≥ 136°F
- Exhaust Hood 1B temperature ≥ 129°F
- Exhaust Hood 1C temperature ≥ 124.5 °F

1.6 The maximum differential pressure between adjacent LP shell pressures shall **NOT** exceed 2.0 inches Hg. (main condenser vacuum gauges on 1MC13, OAC points C1P1669 (D/P between A & B Condensers) and C1P1670 (D/P between B & C Condensers) or Main Condenser graphic (CMCOND)).

1.7 A sudden downward trend on an LP turbine's lower extraction temperature shall be investigated as a possible indication of water induction into the turbine. This is indicated on the recorder on the rear of 1MC8 labeled "TURBINE WATER DETECTION", using any of the LP 8th stage lower temperatures.

1.8 A "LOAD RATE" > "6.2 MW/MIN" shall **NOT** be used during normal load changes.

1.9 Differential temperature between adjacent exhaust hoods shall **NOT** exceed 30°F unless evaluated and approved by the responsible engineer (Turbine Generator System Expert). (OAC points C1P1667 (A & B Exhaust Hoods Metal Delta Temp) and C1P1668 (B & C Exhaust Hoods Metal Delta Temp) or Main Condenser graphic (CMCOND)).

1.10 The Main Turbine OIU Work Station has the capability to perform control functions for the Main Turbine, including tripping and resetting of the turbine. If a control function window is inadvertently selected while manipulating the Main Turbine OIU Work Station, the window shall be closed to prevent actuation of the control function.

1.11 To reduce potential for Turbine rubs at low power levels (< 30% Turbine Load) observe the following:

- Steam Seal Header Pressure between 3 and 5 psig.
- Gland Steam Condenser Header vacuum of 10 - 12" H_2O vacuum.
- Condenser Vacuum < 28.0" Hg
- Minimize time that Turbine is at speed no load.
- Minimize time between Turbine Shell Warming and rolling the Turbine.

1.12 Exhaust hood water sprays are used to cool the last-stage buckets and to minimize temporary distortion of the low-pressure hood and shell structures. These sprays have a significant potential for quenching the LP turbine structure, and if they are applied manually should be undertaken very gradually. Large and rapid changes in the temperature of the exhaust hood can also have an impact on bearing alignment and may cause a rub to develop. Excessive use of the sprays may cause unnecessary erosion of the long last-stage buckets during low flow conditions.

2. Initial Conditions

AA Verify Turbine Generator is On Line per Enclosure 4.1 (Turbine Generator Startup).

3. Procedure

CAUTION:

1. The load, hydrogen pressure and power factor limits per the Unit One Revised Data Book Figure 43 shall **NOT** be exceeded.
2. Rate of change of First-Stage Bowl Inner Surface Temperature shall **NOT** exceed 150°F/hr (OAC point C1P1283 (First Stage Metal Temp Rate)).
3. Control valve casing difference, OAC point C1A0961 (Turb Valve Chest Inner Surface Metal Temp) minus C1A0967 (Turb Valve Chest Outer Surface Metal Temp), shall **NOT** exceed curve "Allowable Temp Difference on Turbine Valve Chest" in the Unit 1 OAC Databook.
4. OAC point C1A1140 (Turbine Lower Inner Shell Temp) vs. Percent Steam Flow (OAC point C1P1588 (Design Total Main Steam Flow, Measured (%))) shall be maintained above and to the left of the curve in the Unit One OAC Databook "Load-Changing Recommendations".

NOTE:

- Several of the parameters required for this procedure can be found on OAC graphics, and a list of all OAC points are found on Enclosure 4.8 (Turbine Generator Roll Computer Points).
- Step 3.1 and Step 3.2 may be performed in any order.

N/A 3.1 **IF** increasing turbine generator load, perform the following:

_____ 3.1.1 Ensure the proper reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 3.1.2 **WHILE** increasing Turbine Generator load, perform the following:

_____ 3.1.2.1 **IF AT ANY TIME** Turbine load is < 30%, operate the RC system pumps and fans as required per OP/1/B/6400/001 A (Condenser Circulating Water System) to maintain vacuum in Condenser C < 28" Hg.

_____ 3.1.2.2 **IF** applicable, verify Groups B and C valves on Enclosure 4.6 (Valve Checklist) close at 15% of full load (184 MWe, 107 psig Turbine Impulse Pressure).

_____ 3.1.2.3 **IF** applicable, verify the following valves close at 15% of full load (184 MWe, 107 psig Turbine Impulse Pressure):

- ☐ 1SM-21 (Ctrl Vlv #2 Stm Lead Drn)
- ☐ 1SM-29 (Ctrl Vlv #1 Stm Lead Drn)

_____ 3.1.2.4 **IF** applicable, **WHEN** CV3 comes off of its fully closed seat (65% of full load, 796 MWe), verify 1SM-25 (Ctrl Vlv #3 Stm Lead Drn) closes.

_____ 3.1.2.5 **IF** applicable, **WHEN** CV4 comes off of its fully closed seat (92% of full load, 1127 MWe), verify 1SM-33 (Ctrl Vlv #4 Stm Lead Drn) closes.

CAUTION:

1. Until it is recognized that the first stage shell metal temperature change rate stays below the allowable limit (150°F/hr), the following loading rate shall **NOT** be exceeded:
   - 1/2%/min - First Stage Inner Shell Temperature (1MC3 or OAC point C1A1140 (Turbine Lower Inner Shell Temp)) $\leq$ 350°F
   - 1%/min - First Stage Inner Shell Temperature (1MC3 or OAC point C1A1140 (Turbine Lower Inner Shell Temp)) > 350°F
2. Normal steady-state load changes shall be made without exceeding the limits shown on Enclosure 4.7 (Generator Operating Limits) and in the Unit 1 OAC Databook "Recommended Startup and Loading Curves".
3. Unit One Reactor Operating Data, Section 2.4 shall be referred to for allowable ramp rates. A "LOAD RATE" > 6.2 MW/MIN shall **NOT** be used during normal load changes.

3.1.3 Increase turbine generator load by performing the following:

_____ 3.1.3.1 Select "LOAD RATE" and verify it illuminates.

_____ 3.1.3.2 Input the desired load rate.

_____ 3.1.3.3 Select "ENTER" and verify "LOAD RATE" goes dark.

_____ 3.1.3.4 Select "TARGET" and verify it illuminates.

_____ 3.1.3.5 Input the desired load target.

_____ 3.1.3.6 Select "ENTER" and verify "TARGET" light goes dark.

_____ 3.1.3.7 Verify new load target appears on Target Display.

_____ 3.1.3.8 Select "GO" and verify it illuminates to start load increase.

_____ 3.1.3.9 Coordinate with Secondary Chemistry to adjust S/G blowdown flowrates to obtain maximum blowdown for the appropriate load.

CAUTION:
1. The load, hydrogen pressure and power factor limits per the Unit One Revised Data Book Figure 43 shall **NOT** be exceeded.
2. Rate of change of First-Stage Bowl Inner Surface Temperature shall **NOT** exceed 150°F/hr (OAC point C1P1283 (First Stage Metal Temp Rate)).
3. OAC point C1A1140 (Turbine Lower Inner Shell Temp) vs. Percent Steam Flow (OAC point C1P1588 (Design Total Main Steam Flow, Measured (%)) shall be maintained above and to the left of curve in the Unit One OAC Databook "Load-Changing Recommendations".
4. Control valve casing difference, OAC point C1A0961 (Turb Valve Chest Inner Surface Metal Temp) minus C1A0967 (Turb Valve Chest Outer Surface Metal Temp), shall **NOT** exceed curve "Allowable Temp Difference on Turbine Valve Chest" in the Unit 1 OAC Databook.

_____ 3.2 **IF** decreasing turbine generator load, perform the following:

_____ 3.2.1 Ensure the proper reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 3.2.2 **WHILE** decreasing turbine generator load, perform the following:

_____ 3.2.2.1 **IF AT ANY TIME** Turbine load is < 30%, operate the RC system pumps and fans as required per OP/1/B/6400/001 A (Condenser Circulating Water System) to maintain vacuum in Condenser C < 28" Hg.

_____ 3.2.2.2 **IF** CV4 fully closes (92% of full load, 1127 MWe), verify 1SM-33 (Ctrl Vlv #4 Stm Lead Drn) opens.

_____ 3.2.2.3 **IF** CV3 fully closes (65% of full load, 796 MWe), verify 1SM-25 (Ctrl Vlv #3 Stm Lead Drn) opens.

CAUTION:	1.	Normal steady-state load change shall be made without exceeding limits shown on Enclosure 4.7 (Generator Operating Limits) and in the Unit 1 OAC Databook "Recommended Starting and Loading Curves".
	2.	Unit One Reactor Operating Data, Section 2.4 shall be referred to for allowable ramp rates.

3.2.3 Decrease turbine generator load by performing the following:

_____ 3.2.3.1 Select "LOAD RATE" and verify it illuminates.

_____ 3.2.3.2 Input the desired load rate.

_____ 3.2.3.3 Select "ENTER" and verify "LOAD RATE" goes dark

_____ 3.2.3.4 Select "TARGET" and verify it illuminates.

_____ 3.2.3.5 Input the desired load target.

_____ 3.2.3.6 Select "ENTER" and verify "TARGET" goes dark.

_____ 3.2.3.7 Verify new load target appears on Target Display.

_____ 3.2.3.8 Select "GO" and verify it illuminates to start load decrease.

_____ 3.2.3.9 Coordinate with Secondary Chemistry to adjust S/G blowdown flowrates to obtain maximum blowdown for the appropriate load.

3.3 Do **NOT** file a copy of this enclosure in the designated storage cabinet.

Elapsed Time (Hours)	Power (%FP)	Rod Bank Position (SWD)	Chemical & Volume Control System Changes		Comments
0.0	100.0	CD @ 215	137	Gallons of ACID	includes extra 20 gals ACID
0.5	95.0	CD @ 206	96	Gallons of ACID	
1.0	90.0	CD @ 199	75	Gallons of ACID	flush with 20 gals WATER
1.5	85.0	CD @ 192	168	Gallons of WATER	
2.0	85.0	CD @ 192	108	Gallons of WATER	
2.5	85.0	CD @ 193	127	Gallons of WATER	
3.0	85.0	CD @ 192	105	Gallons of WATER	
3.5	85.0	CD @ 191		--	
4.0	85.0	CD @ 192	62	Gallons of WATER	
4.5	85.0	CD @ 192		--	
5.0	85.0	CD @ 192		--	
5.5	85.0	CD @ 192	45	Gallons of WATER	

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can effect core reactivity by the following: (R.M.)

- Control rod movement
- Turbine load changes
- Feedwater manipulations
- Reactor power changes

1.2 The following Limits and Precautions are Reactivity Management related: (R.M.)

1.2.1 Before returning reactor control to automatic, T-AVG shall be within ± 1°F of T-REF.

1.2.2 Do **NOT** exceed rod insertion limits or temporary rod withdrawal limits.

1.2.3 Automatic control rod withdrawal is blocked when Control Bank D ≥ 200 steps withdrawn.

1.2.4 The difference in boron concentration between the PZR and NC System is desired to be maintained within ± 50 ppm.

1.2.5 Axial Flux Difference (AFD) shall be maintained within the allowable limits as defined in the ROD manual at all power levels above 50% reactor power. (Tech Spec 3.2.3)

1.2.6 During a power change, other indications of reactor power shall be observed along with power range and secondary thermal power indications to aid in determining the reactor power level. Using indications like turbine impulse pressure, CF flow rate, NC loop ΔTs, and others may help in detecting the miscalibration of a nuclear instrument.

1.2.7 During power changes the following alternate indications for Reactor Power are available: (R.M.)

- Thermal Power Best Estimate
- Tavg, Tcold, NC Loop Delta-T
- Intermediate range Channels
- Turbine Impulse Pressure, Megawatt Output
- CF Flowrate

1.3 In the event of an inadvertent power reduction, it is recommended that the power level **NOT** be increased until an investigation has been conducted and corrective action taken.

1.4 If reactor control is in manual, maintain T-AVG within ± 2°F of T-REF to prevent receiving "T-REF/T-AVG HI/LO" alarm.

1.5 Whenever there is a thermal power change greater than or equal to 15% rated thermal power within a one hour period: (OAC point C1L4790 in alarm)

- Notify Chemistry to take an isotopic analysis for iodine within 2 to 6 hours following the last power change that is greater than or equal to 15% rated thermal power within a one hour period. (T.S. SR 3.4.16.2)

- When thermal power has stabilized, notify Radiation Protection to sample and analyze gaseous effluents. (S.L.C. 16.11-6)

1.6 S/G blowdown flowrate shall **NOT** exceed a maximum of 200 GPM per S/G.

1.7 After a thermal power change when plant conditions stabilize, 1EMF-39 setpoints shall be adjusted so the Trip 2 setpoint is set at three times the containment activity and Trip 1 setpoint is set at 70% of Trip 2 setpoint.

1.8 If the RC System condenser inlet temperature drops to less than or equal to 60°F when the reactor is shutdown or less than or equal to 55°F when the reactor is critical, the system shall be aligned as follows:

- One RC pump running (throttled).
- One tower inlet isolated.
- All three riser bypasses open.

1.9 OAC point C1K0628 (CF Flow Venturi Correction Factor) shall be reset to 1.0 when either of the following conditions are met:

- A step load change such as a load rejection greater than 10% rated thermal power,
- A ramp load change of greater than 15% rated thermal power in a one hour period.

1.10 When the Unit is engaged in a power maneuver resulting in a mismatch between OAC point C1P1385 (Reactor Thermal Power, Best) and any excore power channel in excess of 2% refer to Tech Spec Basis for SR 3.3.1.2.

1.11 The insertion of Control Bank D will affect mismatch between OAC point C1P1385 (Reactor Thermal Power, Best) and the excore power range channels. This is due to shielding of the power range detectors by Control Bank D. Therefore, refer to Tech Spec Basis for SR 3.3.1.2 when mismatch between Reactor Thermal Power (Best Estimate) and the excore power range channels shall be observed to be exceeding 2%.

1.12 Engineering normally provides information for planned power maneuvers. The OAC xenon predict program can be used to help anticipate dilution and boration requirements. {PIP C99-0587}

1.13 It is recommended that Primary Chemistry be notified prior to all significant boric acid additions or dilutions to the NC System such that proper pH control may be maintained. Normal boric acid additions and dilutions should be communicated at the Control Room shift briefing. {PIP C-01-665}

1.14 In accordance with INPO best practices when personnel are accessing areas that could experience significant dose rate changes resulting from increasing power, Operations shall maintain Reactor power steady or decreasing.

1.15 The LEFM system shall be functional and providing input to the OAC TPBE or PT/0/A/4220/001 (Manual Calculation of Thermal Power and NC Flow), prior to exceeding 98.3% RTP.

2. Initial Conditions

_____ 2.1 Ensure R1 reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 2.2 Verify Unit is currently operating at greater than or equal to 85% (1041 MWe) turbine load.

3. Procedure

NOTE:
1. Per SOER 07-01 (Reactivity Management) it is recommended that Plant Operators monitor multiple indications such as Thermal Power Best Estimate, Nuclear Instrumentation, Steam Pressure. ΔT's, and Turbine Load when making at power reactivity changes. (R.M.)
2. This enclosure will affect reactivity of core and is therefore designated important to Reactivity Management per guidelines of AD-OP-ALL-0203 (Reactivity Management). (R.M.)

_____ 3.1 **IF** increasing power from 85% to 100% perform the following:

_____ 3.1.1 Ensure DEC-BA (Balancing Authority / SOC) has been notified of the Unit load increase per OP/1/B/6300/001 (Turbine-Generator).

_____ 3.1.2 Continue with load increase to desired power level.

NOTE:	Procedure may continue while performing Step 3.1.3 if required.

_____ 3.1.3 **IF** required due to previous plant conditions at approximately 85% power, align feedwater heater vents as directed by Startup enclosure of OP/1/B/6250/004 (Feedwater Heaters, Vents, Drains and Bleed System).

_____ 3.1.4 **IF** most recent power decrease required Step 3.2.15 of this enclosure to be performed, at approximately 90% reactor thermal power perform following:

3.1.4.1 Close following VI isolation valves associated with listed SP valves:

_____ • 1VI-1298 (VI Supply to 1SP29) (TB-597, 1D-21)

_____ • 1VI-2066 (VI Supply to 1SPSV0330) (TB-598, 1C-19)

_____ • 1VI-1101 (VI Supply to 1SPSV0400) (TB-600, 1C-33)

_____ • 1VI-1282 (VI Supply to 1SP99) (TB-598, 1C-23)

_____ • 1VI-1291 (VI Supply to 1SPSV1230) (TB-606, 1B-22)

NOTE:	The work orders in the following step have been previously issued and closed. These work order numbers can be used by the WCC SRO or SPOC to background new work orders to set the SP valve VI supply regulators to 0.

3.1.4.2 Issue Work Orders to set VI supply regulators to 0 for following SP valves:

_____ • 1SP-29 (1A CFPT HP Drains) (TB-597, 1D-21) (W/O 02187004)

_____ • 1SP-33 (1B CFPT HP Drains) (TB-597, 1D-19) (W/O 02187006)

_____ • 1SP-40 (1A & 1B CFPT HP Drains) (TB-599, 1C-32) (W/O 02187005)

_____ • 1SP-99 (1A CFPT HP Drains) (TB-598, 1C-23) (W/O 02187007)

_____ 3.1.5 **<u>IF</u>** most recent power history required power to be reduced below 90% reactor thermal power, at approximately 90% reactor thermal power, perform following to ensure primary to secondary leakage program inputs are current:

- ☐ Notify RP to ensure EMF-33 background counts in the EMF-33 Background Spreadsheet is current.

  Person Notified____________________

- ☐ Notify Primary Chemistry to ensure the NC Xenon Equivalent in Chemistry Database is current.

  Person Notified____________________

- ☐ Record current air ejector off gas flowrate. __________________________

- ☐ Update "EMF33 Prim/Sec Leakage" program on OAC.

NOTE:	1. For unconditioned fuel, Fuel Maneuvering Limits per ROD Manual Section 2.4 require a total of 10 hrs accumulated holds (±1%) between 70 and 98% power. Engineering tracks these holds per PT/0/A/4150/001 (Controlling Procedure for Startup Physics Testing). 2. To count as a hold, power must be stable (± 1% of average) for at least 3 hours.

_____ 3.1.6 **<u>IF</u>** power increase is being performed with unconditioned fuel prior to reaching 95% Reactor power perform following:

_____ 3.1.6.1 **<u>IF</u>** there have been no holds above 70% power per PT/0/A/4150/001 (Controlling Procedure for Startup Physics Testing), stop power increase at less than 95% and hold for 10 hours.

_____ 3.1.6.2 **<u>IF</u>** there have been one or more holds above 70% power per PT/0/A/4150/001 (Controlling Procedure for Startup Physics Testing) that total less than 10 hours, power increase may continue.

_____ 3.1.7 **<u>IF</u>** required due to Generator/Automatic Voltage Regulator (AVR) testing at approximately 95% turbine load (~1163 MWe), perform following:

_____ 3.1.7.1 **<u>IF</u>** performing Generator/Automatic Voltage Regulator (AVR) testing, **<u>HOLD</u>** until Generator/AVR personnel are ready for Operations to continue with Unit 1 power increase.

_____ 3.1.7.2 Once notified by AVR personnel that AVR testing is complete, at this power, begin power increase.

Person making notification_____________________

_____ 3.1.8 **IF** power increase is being performed with unconditioned fuel prior to reaching 98% Reactor power perform following:

_____ 3.1.8.1 **IF** there have been one or more holds above 70% power per PT/0/A/4150/001 (Controlling Procedure for Startup Physics Testing) that total less than 10 hours, stop power increase at less than 98% and hold until total hold time is at least 10 hours.

_____ 3.1.8.2 **IF** total hold time requirements of PT/0/A/4150/001 (Controlling Procedure for Startup Physics Testing) have been meet procedure may continue.

NOTE: Refer to Unit One R.O.D., Section 2.4 (Fuel Maneuvering Limits) for rate at which power level can be changed.

_____ 3.1.9 **IF** either of below conditions has been met, prior to exceeding 98% thermal power on OAC point C1P1385 (Reactor Thermal Power, Best), verify OAC point C1K0628 (CF Flow Venturi Correction Factor) has been reset to 1.0:

- A step load change such as a load rejection of greater than 10% rated thermal power.
- A ramp load change of greater than 15% rated thermal power in a one hour period.

NOTE:
1. At normal full power operation the plant is **NOT** operated with MW IN in service. Changes in Secondary system efficiency can change MWs causing EHC to adjust control valves to maintain MWs. (R.M.)
2. Placing megawatt feedback loop in and out of service is a bumpless transfer. When MW IN is selected the TARGET and setpoint will become ACTUAL megawatts. When MW OUT is selected the TARGET becomes present valve reference converted to MWs. In either condition, load changes in progress will be seen as complete with TARGET change and GO light extinguished.
3. Previously inputted load rates are **NOT** changed when changing MW feedback status.

_____ 3.1.10 **IF** "MW IN" in service on Main turbine, prior to exceeding 98% thermal power on OAC point C1P1385 (Reactor Thermal Power, Best) perform following:

☐ 3.1.10.1 Ensure Main Turbine is in HOLD.

☐ 3.1.10.2 Depress the "MW IN/MW OUT" pushbutton.

☐ 3.1.10.3 Verify green "MW OUT" light is illuminated.

_____ 3.1.11 Prior to exceeding 98% reactor power verify on OAC graphic LEFMSTAT that TPBE INPUT FLOWS FROM "LEFM INPUTS".

_____ 3.1.12 Continue turbine load increase per OP/1/B/6300/001 (Turbine Generator) enclosure for load changing.

_____ 3.1.13 **WHEN** power has exceeded 98% thermal power on OAC point C1P1385 (Reactor Thermal Power, Best) ensure the rate of power escalation is no greater than 2%/hr to help prevent an inadvertent overpower condition.

NOTE: Intent of the following step is to have a more controlled final approach to 100% power.

_____ 3.1.14 **WHEN** at 99.5% reactor power on C1P1385 (U1 Reactor Thermal Power Best), perform following:

_____ 3.1.14.1 **HOLD** power escalation for at least 10 minutes to allow for Xenon and AFD oscillations to be seen.

_____ 3.1.14.2 **WHEN** at least 10 minutes have elapsed, continue the power increase.

CAUTION: Alternate indications of reactor power shall be monitored to verify reactor power level and help prevent NI miscalibration.

_____ 3.1.15 At 100% thermal power (3469 MWt), compare OAC heat balance point C1P1385 (Reactor Thermal Power, Best) with nuclear instrumentation.

_____ 3.1.16 **IF** this power increase is from Mode 3, notify Secondary Chemistry to perform a primary to secondary leak rate calculation per PT/1/B/4600/028 (Determination of Steam Generator Tube Leak Rate for Unit 1) within 24 hours of reaching 100% power. (NSD 513)

Person notified ________________________________

_____ 3.1.17 **IF** required, notify IAE to adjust nuclear instrumentation per Model W/O #00874628.

Person notified ________________________________

NOTE: Unit is now operating at 100% power.

NOTE:	1. Following steps reduce power to 85% power. 2. If desired power reduction rate is ≥ 15%/hr (3 MW/Min), consider using AP/1/A/5500/009 (Rapid Downpower). 3. During normal plant power reductions, maintaining power reduction rates between 8-12% per hour when power is between 100 – 50% helps prevent steam generator sodium exceeding 0.8 ppb (Chemistry Effectiveness Indicator (CEI) limit). The 0.8 ppb CEI limit does **NOT** apply below 50% power.

_____ 3.2 **IF** decreasing power from 100% to 85% perform the following:

NOTE:	The list below provides examples of items that should be reviewed / evaluated to identify any potential impacts to the power reduction in the following step. Control Room Deficiencies Turnover Sheets ACMPs ODMIs Emergent Work

_____ SRO 3.2.1 **IF** performance of this enclosure is for a planned power reduction with a Risk Special Emphasis Code (SEC) of R1,
THEN perform the following:

_____ SRO 3.2.1.1 Evaluate known degraded plant equipment for potential impacts or complications to the activity.

_____ SRO 3.2.1.2 Determine if any contingency actions from the degraded plant equipment evaluation completed in Step 3.2.1.1 are required.

_____ SRO 3.2.1.3 **IF** contingency actions are required,
THEN implement contingency actions from the degraded plant equipment evaluation.

_____ SRO 3.2.1.4 **IF** the risk evaluation completed in Step 3.2.1.1 results in a decision to cancel **OR** reschedule the planned R1 activity,
THEN generate a CR to document the decision.

_____ 3.2.1.5 Document the following in the narrative Log:

- ☐ Results from the degraded plant equipment evaluation.
- ☐ Any contingency actions that were required to be implemented.

_____ 3.2.2 **<u>IF</u>** this is a T-AVG Coastdown, maintain T-AVG $\geq$ Limiting Curve for Phases 2 and 3 (Enclosure 4.4.1).

_____ 3.2.3 **<u>IF</u>** shutdown is due to Tech Spec, ensure NRC has been notified per RP/0/B/5000/013 (NRC Notification Requirements).
SRO

NOTE:
1. Step 3.2.4 shall be coordinated with Primary Chemistry alignments for PZR steam space purge prior to a refueling outage shutdown. PZR steam space purge alignments normally begin ~ 48 hours prior to beginning initial power decrease.
2. Procedure may continue while performing Step 3.2.4.

_____ 3.2.4 **<u>IF</u>** initiating a shutdown for a refueling outage, perform following to flush and isolate 1EMF-48:

_____ 3.2.4.1 Declare 1EMF-48 inoperable.

_____ 3.2.4.2 Close 1NM-26B (Hot Leg Smpl Hdr Cont Isol).

Record time____________________

_____ 3.2.4.3 Notify Primary Chemistry and RP Shift that sample line will be flushed which could generate highly radioactive CRUD in primary sample piping.

RP Shift person notified_________________________
Primary Chemistry person notified________________________

_____ 3.2.4.4 **<u>WHEN</u>** one hour has passed, open 1NM-26B (Hot Leg Smpl Hdr Cont Isol) to initiate the flush.

Record time_______________________

_____ 3.2.4.5 **<u>WHEN</u>** 15 minutes has passed, close 1NM-26B (Hot Leg Smpl Hdr Cont Isol) to terminate the flush.

NOTE:	During a T-AVG Coastdown when a turbine load reduction or steam pressure increase (safety valve testing) is requested with the turbine control valves at or near wide open, load decrease will **NOT** be linear with valve reference (demand) decrease. This is due to flow characteristic of turbine control valves. Load increase/decrease is linear with demand increase/decrease from approximately 10% to 90% valve reference. AFD oscillations should be expected. (02-3529)

_____ 3.2.5 Notify DEC-BA (Balancing Authority / SOC) prior to reducing load per OP/1/B/6300/001 (Turbine-Generator).

NOTE:	During power reduction, "OTDELTAT - FAIL" alarm on DCS may be received due to inputs over-ranging. This will result in a DCS TROUBLE annunciator on 1AD-2, F/10.

_____ 3.2.6 Begin load reduction to desired power level.

NOTE:

1. Placing megawatt feedback loop in and out of service is a bumpless transfer. When MW IN is selected the TARGET and setpoint will become ACTUAL megawatts. When MW OUT is selected TARGET becomes present valve reference converted to MWs. In either condition, load changes in progress will be seen as complete with TARGET change and GO light extinguished.
2. When MW IN is selected, if MWs drift from TARGET an error signal will be generated to adjust control valves to bring ACTUAL MWs back to TARGET. (R.M.)
3. Previously inputted load rates are **NOT** changed when changing MW feedback status.
4. When MW IN is in service the top half of the associated pushbutton will be illuminated (red). When MW OUT selected the bottom half of the associated pushbutton will be illuminated (green).

_____ 3.2.7 **IF AT ANY TIME** when MW IN in service on the Main Turbine **AND** affects on plant are **NOT** as desired perform following:

_____ 3.2.7.1 Ensure Main Turbine is in HOLD.

_____ 3.2.7.2 Depress "MW IN/MW OUT" pushbutton. (R.M.)

_____ 3.2.7.3 Verify green "MW OUT" light is illuminated.

_____ 3.2.7.4 **IF** desired to continue load decrease with MW IN repeat Steps 3.2.8.1 thru 3.2.8.11 to restart desired Main Turbine load decrease.

_____ 3.2.7.5 **IF** desired to continue load decrease with MW OUT refer to OP/1/B/6300/001 (Turbine Generator) enclosure for load changing to continue desired Main turbine load increase.

_____ 3.2.8 **IF** desired decrease turbine load with MW IN in service on Main Turbine perform the following:

_____ 3.2.8.1 Ensure Main Turbine is in HOLD.

_____ 3.2.8.2 Verify Secondary feedwater and steam systems stable.

_____ 3.2.8.3 Depress "MW IN/MW OUT" pushbutton. (R.M.)

_____ 3.2.8.4 Verify red "MW IN" light is illuminated.

_____ 3.2.8.5 Ensure desired load rate.

_____ 3.2.8.6 Select "TARGET" and verify it illuminates.

_____ 3.2.8.7 Input the desired load target.

_____ 3.2.8.8 Select "ENTER" and verify "TARGET" light goes dark.

_____ 3.2.8.9 Verify new load target appears on Target Display.

_____ 3.2.8.10 Select "GO" and verify it illuminates to start load decrease. (R.M.)

_____ 3.2.8.11 Verify "HOLD" light is dark.

CAUTION:
1. When the Unit is engaged in a power maneuver resulting in a mismatch between OAC point C1P1385 (Reactor Thermal Power, Best) and any excore power channel in excess of 2%, refer to Tech Spec Basis for SR 3.3.1.2.
2. Alternate indications of reactor power shall be monitored to verify reactor power level and help prevent NI miscalibration.

_____ 3.2.9 **IF** a power decrease of more than 20% reactor power is planned, issue Model W/O #00874628 to IAE to prevent mismatch between OAC heat balance point C1P1385 (Reactor Thermal Power, Best) and any excore power channel exceeding 2%.

NOTE:	Following radiochemistry samples are being requested due to suspected failed fuel. This will be used to determine the extent of damage and will allow vendor support to be scheduled during refueling outage, if required. This step may be N/A'd with concurrence from Engineering. {PIP 04-0879}

_____ 3.2.10 **<u>IF</u>** failed fuel is suspected **<u>AND</u>** a power decrease of more than 5% reactor power is planned **<u>AND</u>** Engineering concurs, notify Primary Chemistry to take NC gamma isotopic samples at one, three, and five hour increments after Unit has stabilized at lower power level following power change.

Chemistry person notified ____________________

Engineering contact __________________

NOTE:	Following radiochemistry samples are performed to detect potential fuel defects. This will allow vendor support to be scheduled during refueling outage, if required. This sample will typically be obtained prior to shutdown for refueling outage. This step shall be N/A'd if sample was obtained during a previous power reduction.

_____ 3.2.11 **<u>IF</u>** a power decrease of more than 10% is planned **<u>AND</u>** next scheduled Refueling Outage ≤ 120 days away, notify Primary Chemistry to take an isotopic analysis for iodine within 2 to 6 hours following the last power change.

Chemistry person notified__________________

NOTE:	During a Unit coastdown at EOL, AFD shall be maintained as directed by Engineering.

_____ 3.2.12 **<u>WHILE</u>** reducing power maintain control rods above insertion limit and AFD within its target band by boration or dilution per OP/1/A/6150/009 (Boron Concentration Control). (R.M.)

NOTE:	Procedure may continue while performing Step 3.2.13.

_____ 3.2.13 **<u>IF</u>** shutting down for an outage where condenser vacuum will be broken, at approximately 85% turbine load (~1041 MWe) complete the Heater Vent Orifice Return to Service Valve Alignment enclosure of OP/1/B/6250/004 (Feedwater Heaters, Vents, Drains and Bleed System).

CAUTION: Actuation of steam trap drain valves will result in a power change.

NOTE: When aligned to BYPASS, Manual Bypass Selector Valves will align air directly to drain valve actuator to maintain valve closed. When aligned to ENABLE, Manual Bypass Selector Valves will align air through level switch solenoid valve to allow valve to cycle based on level. Procedure may continue while Step 3.2.14 is being performed

_____ 3.2.14 **IF** reducing power to less than 85% thermal power, ensure following Manual Bypass Selector Valves are in ENABLE position:

- _____ 1SPMV0190 (1SP19 Manual Bypass Selector Valve) (TB-599, 1C-22)
- _____ 1SPMV0230 (1SP23 Manual Bypass Selector Valve) (TB-598, 1C-20)
- _____ 1SPMV0370 (1SP37 Manual Bypass Selector Valve) (TB-598, 1G-22)

_____ 3.2.15 **IF** reducing power to less than 85% thermal power, perform following:

3.2.15.1 Open following VI isolation valves associated with the listed SP valves:

- _____ 1VI-1298 (VI Supply to 1SP29) (TB-597, 1D-21)
- _____ 1VI-2066 (VI Supply to 1SPSV0330) (TB-598, 1C-19)
- _____ 1VI-1101 (VI Supply to 1SPSV0400) (TB-600, 1C-33)
- _____ 1VI-1282 (VI Supply to 1SP99) (TB-598, 1C-23)
- _____ 1VI-1291 (VI Supply to 1SPSV1230) (TB-606, 1B-22)

NOTE: The work orders in the following step have been previously issued and closed. These work order numbers can be used by the WCC SRO or SPOC to background new work orders to restore the SP valve VI supply regulators to normal.

3.2.15.2 Issue Work Orders to restore VI supply regulators to normal for following SP valves:

- _____ 1SP-29 (1A CFPT HP Drains) (TB-597, 1D-21) (W/O 02187004)
- _____ 1SP-33 (1B CFPT HP Drains) (TB-597, 1D-19) (W/O 02187006)
- _____ 1SP-40 (1A & 1B CFPT HP Drains) (TB-599, 1C-32) (W/O 02187005)
- _____ 1SP-99 (1A CFPT HP Drains) (TB-598, 1C-23) (W/O 02187007)

_____ 3.2.16 **IF** reducing power to less than 85% thermal power, go to Enclosure 4.2 (Power Decrease).

_____ 3.2.17 **IF** return to 100% thermal power desired, start a new Enclosure 4.3 (Unit Operation Between 85% and 100% Power).

3.3 File a copy of this enclosure in the designated cabinet.

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can affect core reactivity by changing control rod position. (R.M.)

1.2 The following Limits and Precautions are Reactivity Management related: (R.M.)

1.2.1 When rods are being moved, observe "RODS IN/RODS OUT" light for proper direction.

1.2.2 When rods are being moved, observe the demand position and actual (digital) position to verify proper operation of the Rod Control System.

1.2.3 Adjusting T-Avg ± 1°F of T-Ref before transferring rod control to "AUTO" will prevent undesired rod movement.

1.2.4 Monitor startup rate continuously during any rod motion to ensure < 0.5 DPM stable startup rate.

1.3 Automatic rod control shall **NOT** be used when less than 15% (184 MW_e) turbine power.

1.4 Individual control bank positions on "CRD BANK SELECT" switch shall not be used to position rods manually. (The automatic overlap feature is disabled.)

1.5 After releasing Rod Motion lever, waiting 2 seconds before attempting to move rods again will allow all signals to clear the firing cards.

1.6 A rod motion demand below zero steps may result in the movable grippers **NOT** properly engaging the drive shaft.

2. Initial Conditions

AA 2.1 Ensure Reactivity Management controls established per AD-OP-ALL-0203 (Reactivity Management. (RM)

AA 2.2 Verify Unit 1 is **NOT** in an EP or AP.

AA 2.3 Verify one of the following exist:

☑ Control Bank movement required to increase/decrease Reactor Power

☐ Control Bank movement required to increase/decrease Tavg

☐ Control Bank movement required to maintain AFD

☐ Control Bank manual control required to support testing/maintenance activity

3. Procedure

NOTE:	Steps 3.1 through 3.6 may be signed off as time allows ensuring operator maintains proper focus on reactivity management.

AA 3.1 Monitor the following:

- Tavg/Tref
- Demand Counter positions
- DRPI rod positions
- ROD MOTION RODS-IN/RODS-OUT Light
- ROD MOTION DEMAND SIGNALS - TEMP ERROR/POWER MISMATCH
- Power Range instruments
- IR SUR (Startup Rate)

AA 3.2 **IF** MANUAL ROD movement is desired, perform the following:

- Verify the "ALM" LED on circuit card A206 in the left side of 1ERCC0006 (Rod Control Logic Cabinet) is **NOT** illuminated.
- Verify one GRP select light is illuminated on each power cabinet.

AA 3.3 **IF** plant conditions require, place the "CRD BANK SELECT" switch in "MAN".

_____ 3.4 **IF** withdrawing Control Banks, pull and hold the "ROD MOTION" lever "OUT" as required until control rods are in the desired position. (R.M.)

_____ 3.5 **IF** inserting Control Banks, push and hold the "ROD MOTION" lever "IN" as required until control rods are in the desired position. (R.M.)

_____ 3.6 **IF** automatic rod control is desired, perform the following:

_____ 3.6.1 Verify Unit 1 Reactor Power is ≥ 15% RTP.

_____ 3.6.2 **WHEN** Tavg is within 1°F of Tref, place "CRD BANK SELECT" in "AUTO".

3.7 Do **NOT** file this enclosure.

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can affect core reactivity by changing boron concentration. (R.M.)

1.2 The following Limits and Precautions are Reactivity Management related: (R.M.)

1.2.1 If the boron concentration is being increased in the NC System, at least one NC pump or one ND pump shall be in operation, recirculating the NC System.

1.2.2 If the unit has operated continuously for several months, significant Boron 10 depletion may have occurred. The effective boron concentration of the NC System may be lower than indicated by Chemistry samples.

1.3 Maintaining VCT pressure as low as practical during large makeups will minimize gas absorption. VCT pressure can be reduced by diverting letdown or by VCT purge.

1.4 Due to Electromagnetic Interference within the Unit 1 Reactor Makeup Control System, the Unit 1 Boric Acid Counter may sporadically count up during dilution activities. OFF indications for the Boric Acid Xfer Pumps and Closed indication for valve 1NV-238A can be used by the Reactor Operators to validate that sporadic counts are indication only. (NCR 02081372).

2. Initial Conditions

AA 2.1 **IF** in Mode 1, 2 or 3, ensure R2 reactivity management controls established AD-OP-ALL-0203 (Reactivity Management). (R.M.)

AA 2.2 Verify the NV System is in operation per OP/1/A/6200/001 (Chemical and Volume Control System).

AA 2.3 Verify sufficient RHT volume is available to receive the reactor coolant displaced during the planned boration operation.

N/A 2.4 **IF** NC System boron concentration will be changed by $\geq$ 50 ppm, initiate PZR spray to equalize the boron concentration throughout the system by operating backup heaters per OP/0/A/6200/055 (Miscellaneous Component Operation).

3. Procedure

NOTE:	This enclosure will affect reactivity of the core and is therefore designated important to Reactivity Management per the guidelines of AD-OP-ALL-0203 (Reactivity Management). (R.M.)

AA 3.1 Ensure valves are aligned per Enclosure 4.8 (Valve Checklist).

3.2 Ensure the following valve control switches in "AUTO":

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

_____ 3.3 Ensure 1NV-238A (B/A Xfer Pmp To Blender Ctrl) controller in auto.

_____ 3.4 Ensure at least one boric acid transfer pump is in "AUTO" or "ON".

☐ 3.5 Record the desired volume of boric acid to be added. _________ gallons

_____ 3.6 Adjust the boric acid counter to the desired volume of boric acid to be added. (R.M.)

_____ 3.7 **<u>IF</u>** the blender is set up for automatic makeup per Enclosure 4.1 (Automatic Makeup), record the setpoint of the controller for 1NV-238A (B/A Xfer Pmp To Blender Ctrl). _______gpm

_____ 3.8 Place the "NC MAKEUP MODE SELECT" switch in "BORATE".

NOTE:	Boric Acid flow rates > 32 gpm may result in a boric acid flow deviation annunciator.

_____ 3.9 **<u>IF</u>** required, adjust the controller for 1NV-238A (B/A Xfer Pmp To Blender Ctrl) to the desired flow.

_____ 3.10 **<u>IF AT ANY TIME</u>** it is desired to divert letdown to the RHT manually operate 1NV-172A (3-Way Divert To VCT-RHT) as follows:

_____ 3.10.1 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) to the "RHT" position.

_____ 3.10.2 Ensure VCT level is monitored continuously while diverting to the RHT.

NOTE:	Procedure may continue while performing the following step.

3.10.3 **<u>WHEN</u>** desired VCT level is reached return 1NV-172A (3-Way Divert To VCT-RHT) to auto as follows:

_____ 3.10.3.1 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) in the "VCT" position.

_____ 3.10.3.2 Place the control switch for 1NV-172A (3-Way Divert To VCT-RHT) in the "AUTO" position.

_____ 3.11 **<u>IF AT ANY TIME</u>** during the makeup it becomes necessary to change the makeup flow rate, adjust the setpoint for 1NV-238A (B/A Xfer Pmp To Blender Ctrl) as necessary to achieve the desired flow.

_____ 3.12 **IF AT ANY TIME** while boration is in progress it becomes necessary **OR** it is desired to stop the boration, perform the following:

_____ 3.12.1 Place the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.12.2 Ensure the following valves close: (R.M.)

- _____ 1NV-238A (B/A To Blender Ctrl Vlv)
- _____ 1NV-186A (B/A Blender Otlt To VCT Otlt)

☐ 3.12.3 Record boric acid volume added as indicated on the Boric Acid counter.
_____________ gallons

3.12.4 **WHEN** conditions allow resuming the boration, perform the following:

☐ 3.12.4.1 Determine remaining volume to be added by subtracting the amount previously added (Step 3.12.3) from the desired volume to be added (Step 3.5).

___________ - ___________ = ____________ gallons
(Step 3.5) (Step 3.12.3)

_____ 3.12.4.2 Adjust boric acid counter to the volume of boric acid determined in Step 3.12.4.1. (R.M.)

_____ 3.12.4.3 Place the "NC MAKEUP CONTROL" switch in the "START" position. (R.M.)

_____ 3.12.4.4 Verify the following:

☐ 1NV-238A (B/A To Blender Ctrl Vlv) modulates to establish desired flow
☐ 1NV-186A (B/A Blender Otlt To VCT Otlt) opens

_____ 3.12.4.5 **IF** in "AUTO", verify the boric acid pump starts.

_____ 3.13 **WHILE** makeup is in progress, monitor the following for expected results:

☐ Control rod motion
☐ NC System Tavg
☐ Reactor Power

NOTE:	If a small makeup is being performed, placekeeping for Steps 3.14 through 3.17 may be performed after Step 3.18 is performed.

_____ 3.14 Place the "NC MAKEUP CONTROL" switch to the "START" position. (R.M.)

_____ 3.15 Verify the following:

- ☐ 1NV-238A (B/A To Blender Ctrl Vlv) modulates to establish desired flow
- ☐ 1NV-186A (B/A Blender Otlt To VCT Otlt) opens

_____ 3.16 **IF** in "AUTO", verify the boric acid transfer pump starts.

☐ 3.17 Verify proper flow by observing the Boric Acid Counter. {PIP 96-0137}

NOTE:	The boric acid counter may count up 1 - 5 gallons after termination.

3.18 **WHEN** the desired volume of boric acid is reached on the boric acid counter, ensure the following valves close: (R.M.)

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

NOTE:	If additional borations will be performed over the course of the shift, flushing the makeup line is **NOT** recommended.

_____ 3.19 **IF** desired, flush the makeup line as follows:

☐ 3.19.1 Record the setpoint on 1NV-242A (RMWST To B/A Blender Ctrl): ___________gpm

_____ 3.19.2 Place controller for 1NV-242A (RMWST To B/A Blender Ctrl) in manual.

_____ 3.19.3 Increase demand on controller for 1NV-242A (RMWST To B/A Blender Ctrl) to full open.

NOTE:	It is essential for the operator to read and understand the following steps before initiating a flush of the makeup line. If a reactor makeup water pump is currently on, the following step will initiate flow to the makeup line. Steps 3.19.4, 3.19.5, and 3.19.6 may be performed prior to signing off the steps.

3.19.4 Open the following valves:

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

_____ 3.19.5 Ensure one reactor makeup water pump is in "ON".

NOTE:	Valves in the following step shall be positioned as sequenced to preclude unanticipated additional reactor makeup water flow due to seat leak by on 1NV-186A.

3.19.6 **WHEN** ~ 20 gallons of makeup water have been flushed through the makeup line, close the following valves:

_____ 3.19.6.1 1NV-242A (RMWST To B/A Blender Ctrl)

_____ 3.19.6.2 1NV-186A (B/A Blender Otlt To VCT Otlt)

3.19.7 Place the following valve control switches in "AUTO":

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

_____ 3.19.8 Ensure controller for 1NV-242A (RMWST To B/A Blender Ctrl) is set to the value recorded in Step 3.19.1. (R.M.)

_____ 3.19.9 Place controller for 1NV-242A (RMWST To B/A Blender Ctrl) in auto.

_____ 3.19.10 **<u>IF</u>** **<u>NOT</u>** required for current plant operation, place the reactor makeup water pump started in Step 3.19.5 in "AUTO".

_____ 3.20 **<u>IF</u>** automatic makeup is desired, perform one of the following:

_____ 3.20.1 **<u>IF</u>** it is desired to change the blender outlet boron concentration, refer to Enclosure 4.1 (Automatic Makeup).

OR

_____ 3.20.2 **<u>IF</u>** makeup at the previous concentration is acceptable **<u>AND</u>** the system was previously aligned per Enclosure 4.1 (Automatic Makeup), perform the following:

_____ 3.20.2.1 Ensure the controller for 1NV-238A (B/A Xfer Pmp To Blender Ctrl) is set to the value recorded in Step 3.7. (R.M.)

_____ 3.20.2.2 Place the "NC MAKEUP MODE SELECT" switch in "AUTO".

_____ 3.20.2.3 Place the "NC MAKEUP CONTROL" switch to the "START" position. (R.M.)

3.21 Do **<u>NOT</u>** file this enclosure.

Catawba Nuclear Station
JPM A
2021 NRC Exam

JPM A

Catawba Nuclear Station
JPM A
2021 NRC Exam

EVALUATION SHEET

Task: Emergency Borate the Reactor Coolant System

Alternate Path: Yes

Facility JPM #: NV-017

Safety Function: 1 **Title:** Reactivity Control

K/A 004 A2.14 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency Boration

Rating(s): 3.8 / 3.9 **CFR:** 41.5 / 43.5 / 45.3 / 45.5

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator __X__ In-**Pl**ant ________ **P**erform __X__ **S**imulate ________

References: EP/1/A/5000/FR-S.1 (Nuclear Power Generation/ATW) rev. 022

Task Standard: One NV pump running with its suction aligned to the FWST and isolated from the VCT.

Validation Time: 5 minutes **Time Critical:** Yes _____ No __X__

===

Applicant: Time Start: __________
NAME____________________________ Docket #_______________ Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

===

COMMENTS

Catawba Nuclear Station
JPM A
2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. ENSURE NRC Examination Security has been established.
2. Reset to IC #141
3. Enter the password.
4. Select “Yes” on the INITIAL CONDITION RESET pop-up window.
5. Ensure simulator setup per table below.
6. Place simulator in RUN and acknowledge any alarms.
7. ENSURE “Extra Operator” is present in the simulator.
8. Place simulator in FREEZE until Examiner cue is given.

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event
	MAL-IPX003A (REACTOR TRIP BKR A FAILURE)	ACTIVE				
	MAL-IPX003B (REACTOR TRIP BKR B FAILURE)	ACTIVE				
	VLV-NV043F (NV236B BORIC ACID TO CHG PMP VLV FAIL TO POSITION)	0				
	MAL-MT007 (LOSS OF TURBINE LUBE OIL PRESSURE)	ACTIVE				
	Instructor will act as the OATC and be manually inserting control rods when the simulator is placed in RUN.					

Catawba Nuclear Station
JPM A
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **A valid reactor trip signal has been received.**
- **The reactor will NOT trip automatically or manually.**
- **A Red Path for Subcriticality is in effect.**
- **The OATC is inserting rods manually.**

INITIATING CUES:

The Control Room Supervisor instructs you to initiate emergency boration, per EP/1/A/5000/FR-S.1, (Nuclear Power Generation/ATWS) step 4.

EXAMINER NOTE: After reading cue, provide the applicant with a copy of EP/1/A/5000/FR-S.1.

STEP/STANDARD	SAT/UNSAT

START TIME: _______

STEP 1: 4. **Initiate emergency boration of NC System as follows:** a. Ensure at least one NV pump - ON. STANDARD: Applicant verifies red "ON" light lit for "NV PMP 1A" or "1B" (1MC-10). COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 2 4. b. OPEN 1NV-236B (Boric Acid To NV Pumps Suct). STANDARD: Applicant depresses the red "OPEN" pushbutton for 1NV-236B and verifies the red "OPEN" light remains dark and the green "CLSD" light remains lit on 1MC-10. 1NV-236B remains closed. **Examiner Note: This begins the alternate path of this JPM.** COMMENTS:	___ **SAT** ___**UNSAT**

STEP 3 4. c. Ensure both boric acid transfer pump switches - IN THE "ON" POSITION. STANDARD: Applicant rotates the switches for "B/A XFER PMP 1A" and "B/A XFER PMP 1B" to the "ON" position. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 4 4. d. Verify emergency boration flow - GREATER THAN OR EQUAL TO 30 GPM. STANDARD: Applicant verifies "EMER BORATE FLOW" (1NVP5440) indicates 0 gpm (1MC-5) and transitions to the RNO. COMMENTS:	___ SAT ___ UNSAT
STEP 5 4. d. RNO d. Align NV pump suction to FWST as follows: 1) OPEN the following valves: • 1NV-252A (NV Pumps Suct From FWST) • 1NV-253B (NV Pumps Suct From FWST). STANDARD: Applicant depresses the red "OPEN" pushbuttons for 1NV-252A and 1NV-253B and verifies the red "OPEN" light is lit and green "CLSD" light is dark for both valves. **Examiner Note: This step is critical to align borated water to the suction of the charging pumps. Only ONE of the valves opened meets the Critical Step criteria.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 6 4. d. RNO d. 2) CLOSE the following valves: • 1NV-188A (VCT Otlt Isol) • 1NV-189B (VCT Otlt Isol). STANDARD: Applicant depresses the green “CLOSE” pushbutton for 1NV-188A and 1NV-189B and verifies the green “CLSD” light lit and red “OPEN” light dark for both valves. **Examiner Note: This step is critical to prevent borated water from going to the VCT instead of the suction of the charging pumps as long as one of the valves is closed. Closing only ONE of the valves meets the intent of the Critical Step criteria.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 7 4. e. Verify the following charging line isolation valves - OPEN: • 1NV-312A (Chrg Line Cont Isol) follows: • 1NV-314B (Chrg Line Cont Isol). STANDARD: Applicant verifies the red “OPEN” lights lit and green “CLSD” lights dark on 1NV-312A and 1NV-314B. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 8 4. f. Verify Pzr pressure - LESS THAN 2335 PSIG. STANDARD: Applicant verifies PZR pressure instruments (1NCP5161, 1NCP5150, 1NCP5170 and 1NCP5171) indicate less than 2335 psig (1MC-10). COMMENTS: **END OF TASK**	___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **A valid reactor trip signal has been received.**
- **The reactor will NOT trip automatically or manually.**
- **A Red Path for Subcriticality is in effect.**
- **The OATC is inserting rods manually.**

INITIATING CUES:

The Control Room Supervisor instructs you to initiate emergency boration, per EP/1/A/5000/FR-S.1, (Nuclear Power Generation/ATWS), step 4.

A. Purpose

This procedure provides actions to add negative reactivity to a core which is observed to be critical when expected to be shutdown.

B. Symptoms or Entry Conditions

This procedure is entered from:

a. EP/1/A/5000/E-0 (Reactor Trip or Safety Injection), when reactor trip is not verified and manual trip is not effective.

b. EP/1/A/5000/F-0 (Critical Safety Function Status Trees), (SUBCRITICALITY) on either a RED or ORANGE condition.

c. EP/1/A/5000/FR-S.2 (Response to Loss of Core Shutdown), Step 1, when I/R or W/R power trending up.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. <u>Operator Actions</u>

<u>CAUTION</u> NC pumps should <u>NOT</u> be tripped with reactor power greater than 5% to prevent fuel damage.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **Verify Reactor Trip:**	**Perform the following:**
__ • All rod bottom lights - LIT	✓ a. Trip reactor.
__ • All reactor trip and bypass breakers - OPEN	✓ b. **<u>IF</u>** reactor will not trip, **<u>THEN</u>** insert rods.
__ • I/R power - TRENDING DOWN.	
2. **Verify Turbine Trip:**	**Perform the following:**
✓ • All turbine stop valves - CLOSED	__ a. Trip turbine.
	b. **<u>IF</u>** turbine will not trip, **<u>THEN</u>** perform the following:
	__ 1) Depress "MANUAL" pushbutton on turbine control panel.
	__ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons.
	3) **<u>IF</u>** control valves will not close, **<u>THEN</u>** CLOSE the following:
	__ • All MSIVs
	__ • All MSIV bypass valves.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. **Verify CA pumps running as follows:**	
✓ a. Motor driven CA pumps - ON.	__ a. Start motor driven CA pump(s).
✓ b. 3 S/G N/R levels - GREATER THAN 11%.	__ b. Ensure CA Pump #1 - RUNNING.
4. **Initiate Emergency Boration of NC System as follows:**	
__ a. Ensure at least one NV pump - ON.	
__ b. OPEN 1NV-236B (Boric Acid To NV Pumps Suct).	
__ c. Ensure both boric acid transfer pump switches - IN THE "ON" POSITION.	
__ d. Verify emergency boration flow - GREATER THAN OR EQUAL TO 30 GPM.	d. Align NV pump suction to FWST as follows: 1) OPEN the following valves: __ • 1NV-252A (NV Pumps Suct From FWST) __ • 1NV-253B (NV Pumps Suct From FWST). 2) CLOSE the following valves: __ • 1NV-188A (VCT Otlt Isol) __ • 1NV-189B (VCT Otlt Isol).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

4. (Continued)

e. Verify the following charging line isolation valves - OPEN:

__ • 1NV-312A (Chrg Line Cont Isol)
__ • 1NV-314B (Chrg Line Cont Isol).

e. Perform the following:

1) Align NV pump suction to FWST as follows:

a) OPEN the following valves:

__ • 1NV-252A (NV Pumps Suct From FWST)

__ • 1NV-253B (NV Pumps Suct From FWST).

b) CLOSE the following valves:

__ • 1NV-188A (VCT Otlt Isol)
__ • 1NV-189B (VCT Otlt Isol).

2) Ensure the following valves - OPEN:

__ • 1NI-9A (NV Pmp C/L Inj Isol)
__ • 1NI-10B (NV Pmp C/L Inj Isol).

__ f. Verify Pzr pressure - LESS THAN 2335 PSIG.

f. Perform the following:

1) Verify the following valves - OPEN.

__ • All Pzr PORVs
__ • All Pzr PORV isolation valves.

__ 2) **IF** any Pzr PORV(s) **OR** isolation valves closed, **THEN** OPEN Pzr PORV(s) and isolation valves as required to reduce Pzr pressure to less than 2135 PSIG.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify the following Monitor Light Panel Group 5 St lights on energized train - LIT:** __ • I/2 __ • I/3 __ • I/10 __ • H/11.	**Ensure the following VQ isolation valves - CLOSED:** __ • 1VQ-2A (VQ Fan Suct From Cont Isol) __ • 1VQ-3B (VQ Fan Suct From Cont Isol) __ • 1VQ-15B (Cont Air Add Cont Isol) __ • 1VQ-16A (Cont Air Add Cont Isol).
6. **Verify S/I status as follows:**	
__ a. "SAFETY INJECTION ACTUATED" status light (1SI-13) - LIT.	a. Perform the following: __ 1) **IF AT ANY TIME** S/I signal exists **OR** occurs while in this procedure, **THEN** perform Step 6.b. __ 2) **GO TO** Step 7.
b. **WHEN** manpower **AND** time permits, **THEN** verify proper system alignments as follows: __ • **REFER TO** Enclosure 1 (System Verification Following S/I Actuation) __ • Notify Unit 2 operator to perform Enclosure 2 (Opposite Unit Ventilation Verification).	

JPM B

Catawba Nuclear Station
JPM B
2021 NRC Exam

EVALUATION SHEET

Task: Isolate Cold Leg Accumulators Following a Shutdown LOCA

Alternate Path: Yes

Facility JPM #: N/A

Safety Function: 3 **Title:** Emergency Core Cooling System (ECCS)

K/A 006 A1.13 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration)

Rating(s): 3.5 / 3.7 **CFR:** 41.5 / 45.5

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator __X__ In-**P**lant ________ **P**erform __X__ **S**imulate ________

References: AP/1/A/5500/027 (Shutdown LOCA), Enclosure 17 (Isolating Cold Leg Accumulators)

Task Standard: AP/1/A/5500/027 (Shutdown LOCA) Enclosure 17 (Isolating Cold Leg Accumulators) is performed and 1B & 1C Cold Leg Accumulators are isolated and 1NI-83 fully opened to vent 1A & 1D Cold Leg Accumulators to containment.

Validation Time: 15 minutes **Time Critical:** Yes _____ No __X__

===

Applicant: Time Start: __________

NAME____________________________ Docket #_______________ Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

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COMMENTS

Catawba Nuclear Station
JPM B
2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. ENSURE NRC Examination Security has been established.
2. Reset to IC # 142.
3. Enter the password.
4. Select "Yes" on the INITIAL CONDITION RESET pop-up window.
5. Ensure simulator setup per table below.
6. Place simulator in RUN and acknowledge any alarms.
7. ENSURE "Extra Operator" is present in the simulator.
8. Place simulator in FREEZE until Examiner cue is given.

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event
	ANN-AD11-B03 (TRANSFORMER A TROUBLE)	ON				
	ANN-AD11-E03 (TRANSFORMER B TROUBLE)	ON				
	MAL-NC013B (NC COLD LEG B LEAK)	0.5				
	VLV-NI005F (NI54A ACCUM ISOL VLV FAIL TO POSITION)	1				
	VLV-NI014F (NI88B ACCUM ISOL VLV FAIL TO POSITION)	1				

Catawba Nuclear Station
JPM B
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is in Mode 4.**
- **Unit 1 shutdown was in progress for a refueling outage, when pressurizer pressure and level began to decrease uncontrollably.**
- **The CRS has entered AP/1/A/5500/027 (Shutdown LOCA) to address the reactor coolant system leak.**
- **Power to all CLA discharge isolation valves has been restored per EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 9 (Power Alignment for CLA Valves).**

INITIATING CUES:

- **The CRS has directed you to isolate the Unit 1 Cold Leg Accumulators by performing AP/1/A/5500/027 (Shutdown LOCA) Enclosure 17 (Isolating Cold Leg Accumulators).**

EXAMINER NOTE: After reading the cue, provide the applicant with a copy of AP/1/A/5500/027 Enclosure 17.

Catawba Nuclear Station
JPM B
2021 NRC Exam

START TIME: _______

STEP/STANDARD	SAT/UNSAT
STEP 1: 1. **Dispatch operator to restore power to all CLA discharge isolation valves. REFER TO EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 9 (Power Alignment for CLA Valves).** STANDARD: Per initiating cue, the applicant should realize that this step is complete. Also valve indication for each of the CLA discharge isolation valves is available, due to power already being aligned by the AO. COMMENTS:	**___ SAT** **___ UNSAT**

STEP 2: 2. **Ensure S/I - RESET** a. ECCS. b. D/G load sequencers. c. **IF AT ANY TIME** a B/O occurs, **THEN** restart S/I equipment previously on. STANDARD: Applicant verifies that the yellow ECCS and D/G load sequencer RESET lights are lit. Applicant acknowledges the If at any time statement. COMMENTS:	**___ SAT** **___UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 3: 3. **WHEN power is aligned, THEN perform the following:** a. CLOSE the following valves: • 1NI-54A (C-Leg Accum A Disch Isol) • **1NI-65B (C-Leg Accum B Disch Isol)** • **1NI-76A (C-Leg Accum C Disch Isol)** • 1NI-88B (C-Leg Accum D Disch Isol) STANDARD: Applicant depresses the green CLOSE pushbutton for the valves listed, and verifies the green CLSD light lit and red OPEN light dark for valves 1NI-65B & 1NI-76A. Applicant also verifies the red OPEN light lit and green CLSD light dark for valves 1NI-54A & 1NI-88B and transitions to the RNO. **The critical part of this step is to close isolation valves 1NI-65B & 1NI-76A. The other 2 Cold Leg Accumulators will be vented to containment in subsequent steps in the RNO.** **EXAMINER NOTE: This begins the alternate path of this JPM.** **EXAMINER NOTE: The applicant should determine from the initiating cue that power has been aligned. If necessary, inform "Power has been aligned".** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 4: 3.RNOa. Perform the following: 1) Ensure containment isolation signals – RESET: • Phase A • Phase B STANDARD: Applicant verifies that the yellow RESET lights are lit for both trains Phase A and Phase B isolations. COMMENTS:	___ SAT ___ UNSAT
STEP 5: 3.RNOa. 2) Ensure 1VI-77B (VI Cont Isol) - OPEN STANDARD: Applicant determines that the red OPEN light lit and green CLSD light dark for 1VI-77B. COMMENTS:	___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 6: 3.RNOa. 3) **IF** VI pressure is less than 85 PSIG, **THEN** dispatch operator to ensure proper VI compressor operation. STANDARD: Applicant determines that VI pressure is ~ 90 PSIG. This step is N/A. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 7: 3.RNOa.4) Vent any CLA which cannot be isolated as follows: a) OPEN 1NI-47A (C-Leg Accum N2 Sup Cont Isol) STANDARD: Applicant depresses the red OPEN pushbutton for 1NI-47A and verifies the red OPEN light lit and green CLSD light dark. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 8: 3.RNOa.4) b) Place breaker 1CB-1 (behind 1MC-6)(Key #11) to - ON STANDARD: Applicant lifts breaker 1CB-1 fully up to the ON position. **Examiner Cue: If asked, "Key #11 has been obtained."** **This step is critical to place power on Cold Leg Accumulator Nitrogen supply isolation valves for 1B and 1C CLAs which will be opened in the next step to allow venting the accumulators to containment.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 9: 3.RNOa.4) c) OPEN valve for CLA(s) to be vented: • **1NI-50 (C-Leg Accum A N2 Supply Isol)** • 1NI-61 (C-Leg Accum B N2 Supply Isol) • 1NI-72 (C-Leg Accum C N2 Supply Isol) • **1NI-84 (C-Leg Accum D N2 Supply Isol)** STANDARD: Applicant depresses the red OPEN pushbutton and verifies the red OPEN light lit and green CLSD light dark for 1NI-50 and 1NI-84. **This step is critical due to not being able to isolate these CLAs. If these CLAs are not vented to reduce the N2 overpressure, it could cause a hard bubble to form in the reactor coolant system as reactor coolant pressure continues to decrease.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 10: 3.RNOa.4) d) CLOSE 1NI-47A. STANDARD: Applicant depresses the green CLOSE pushbutton and verifies the green CLSD light lit and red OPEN light dark for 1NI-47A. **This step is critical to allow venting the 1B and 1C CLAs.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 11: 3.RNOa.4) e) OPEN 1NI-83 (C-Leg Accums N2 Vent Ctrl) to depressurize affected CLA(s). STANDARD: Applicant rotates potentiometer for 1NI-83 clockwise to full open to begin venting the 1A and 1D CLAs. **This step is critical to vent the 1A and 1D CLAs.** **NOTE TO EVALUATOR: The time to fully vent the 1A & 1D CLAs would be approximately 30 minutes. The critical steps for this JPM have been met at this point and the JPM may be terminated at your discretion.** **EVALUATOR CUE: "Another operator will continue to vent the 1A and 1D Cold Leg Accumulators. This JPM is complete."** COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is in Mode 4.**
- **Unit 1 shutdown was in progress for a refueling outage, when pressurizer pressure and level began to decrease uncontrollably.**
- **The CRS has entered AP/1/A/5500/027 (Shutdown LOCA) to address the reactor coolant system leak.**
- **Power to all CLA discharge isolation valves has been restored per EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 9 (Power Alignment for CLA Valves).**

INITIATING CUES:

- **The CRS has directed you to isolate the Unit 1 Cold Leg Accumulators by performing AP/1/A/5500/027 (Shutdown LOCA) Enclosure 14 (Isolating Cold Leg Accumulators).**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Dispatch operator to restore power to all CLA discharge isolation valves. REFER TO EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 9 (Power Alignment for CLA Valves).**	
2. **Ensure S/I - RESET:**	
__ a. ECCS.	__ a. Locally reset ECCS. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 4 (ECCS Master Reset).
__ b. D/G load sequencers.	b. Dispatch operator to open affected sequencer(s) control power breaker: __ • 1EDE-F01F (Diesel Generator Load Sequencer Panel 1DGLSA) (AB-577, BB-46, Rm 496) __ • 1EDF-F01F (Diesel Generator Load Sequencer Panel 1DGLSB) (AB-560, BB-46, Rm 372).
__ c. **IF AT ANY TIME** B/O occurs, **THEN** restart S/I equipment previously on.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. **WHEN power aligned, THEN perform the following:**

a. CLOSE the following valves:

__ • 1NI-54A (C-Leg Accum A Disch Isol)

__ • 1NI-65B (C-Leg Accum B Disch Isol)

__ • 1NI-76A (C-Leg Accum C Disch Isol)

__ • 1NI-88B (C-Leg Accum D Disch Isol).

RNO:

a. Perform the following:

1) Ensure containment isolation signals - RESET:

__ • Phase A

__ • Phase B.

__ 2) Ensure 1VI-77B (VI Cont Isol) - OPEN.

3) **IF** VI pressure less than 85 PSIG, **THEN** perform the following:

__ a) Dispatch operator to ensure proper VI compressor operation.

__ b) Restore VI while continuing with this procedure. **REFER TO** AP/0/A/5500/022 (Loss of Instrument Air).

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. (Continued)

4) Vent any unisolated CLA as follows:

__ a) OPEN 1NI-47A (C-Leg Accum N2 Sup Cont Isol).

__ b) Place breaker 1CB-1 (behind 1MC-6) (Key #11) to - ON.

c) OPEN valve for CLA(s) to be vented:

__ • 1NI-50 (C-Leg Accum A N2 Supply Isol)

__ • 1NI-61 (C-Leg Accum B N2 Supply Isol)

__ • 1NI-72 (C-Leg Accum C N2 Supply Isol)

__ • 1NI-84 (C-Leg Accum D N2 Supply Isol).

__ d) CLOSE 1NI-47A.

__ e) OPEN 1NI-83 (C-Leg Accums N2 Vent Ctrl) to depressurize affected CLA(s).

__ f) **IF** any unisolated CLA(s) cannot be vented, **THEN** consult Station Management for further actions.

__ b. Notify dispatched operator to remove power from all CLA isolation valves. **REFER TO** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 9 (Power Alignment for CLA Valves).

JPM C

Catawba Nuclear Station
JPM C
2021 NRC Exam

EVALUATION SHEET

Task: Restore CA Flow Following Bleed and Feed

Alternate Path: Yes

Facility JPM #: NEW

Safety Function: 4S **Title:** Loss of Secondary Heat Sink

K/A EPE05 EA1.1 Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Rating(s): 4.1 / 4.0 **CFR:** 41.7 / 45.5 / 45.6

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator __X__ In-**P**lant ________ **P**erform __X__ **S**imulate ________

References: EP/1/A/5000/FR-H.1 (Loss of Secondary Heat Sink) Rev 48

Task Standard: CA flow restored to 1A and/or 1B Steam Generators from the 1A CA pump with suction aligned to the RN system.

Validation Time: 15 minutes **Time Critical:** Yes _____ No __X__

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Applicant:
NAME______________________________ Docket #________________ Time Start: ___________
Time Finish: ___________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ________________________________ ________________________________/________
NAME SIGNATURE DATE

==

COMMENTS

Catawba Nuclear Station
JPM C
2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. ENSURE NRC Examination Security has been established.
2. Reset to IC # 143.
3. Enter the password.
4. Select “Yes” on the INITIAL CONDITION RESET pop-up window.
5. Ensure simulator setup per table below.
6. Place simulator in RUN and acknowledge any alarms.
7. ENSURE “Extra Operator” is present in the simulator.
8. Place simulator in FREEZE until Examiner cue is given.

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event
	MAL-CA004A (FAILURE OF CA PUMP A TO START)	AUTO				
	MAL-CA005 (CA PUMP OVERSPEED TRIP)	MECH-ANICAL				
	MAL-CF001A (LOSS OF CFPT 1A VACUUM)	100				
	MAL-CF001B (LOSS OF CFPT 1B VACUUM)	100				
	MAL-EHC002 (TURBINE TRIP FAILURE)	BOTH				
	VLV-CA002F (CA4 CA PMP SUCT FROM UST HDR ISOL FAIL TO POSITION)	0				

Catawba Nuclear Station
JPM C
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is in Mode 3 following a reactor trip due to a loss of all feedwater.**
- **Bleed and Feed has been established per EP/1/A/5000/FR-H.1.**
- **CA flow control valves have been closed per Step 37.**
- **Report from AO and Maintenance in the field that 1A CA pump is ready to be started has just been received.**

INITIATING CUES:

- **The CRS instructs you to perform Step 7 to establish CA flow from 1A CA pump.**

Examiner Note: After reading cue, provide the applicant with a copy of EP/1/A/5000/FR-H.1 Step 7.

STEP/STANDARD	SAT/UNSAT

START TIME: _______

STEP 1 7. **Attempt to establish CA flow to at least one S/G as follows:** a. Verify 1AD-8, B/1 "UST LO LEVEL" – DARK. STANDARD: Applicant determines that 1AD-8, B/1 is dark. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 2 7.b Verify 1CA-4 (CA Pmps Suct From UST) – OPEN. STANDARD: Applicant determines that 1CA-4 is closed and transitions to the RNO. COMMENTS:	___ **SAT** ___**UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 3 7.b.RNO Perform the following: 1) OPEN 1CA-4. STANDARD: Applicant depresses the red OPEN pushbutton and verifies the red OPEN light remains dark and the green CLSD light remains lit. Applicant determines the valve will not open and continues in the RNO. **Examiner Note: This begins the alternate path of this JPM.** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 4 7.b.RNO.2) **IF** no suction source can be aligned, **THEN** OPEN the following valves: • **1RN-250A (RN Hdr A to CA Pmp Suct Isol)** • 1CA-116A (CA Pump #1 Suct Frm RN Hdr A) • **1CA-15A (CA Pump 1A Suct Frm RN Isol)** • 1CA-85B (CA Pump #1 Suct Frm RN Hdr B) • 1CA-18B (CA Pump 1B Suct Frm RN Isol) • 1CA-310B (RN Hdr B To CA Pmp Suct Isol) STANDARD: Applicant rotates switches for the listed valves clockwise to the OPEN position and verifies the red OPEN lights lit and green CLOSED lights dark. **Examiner Note: This step is critical to align a suction flowpath to the CA pumps. Only 1RN-250A and 1CA-15A are required to be opened to meet this critical step due to the 1A CA pump being the only available source for CA flow.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 5 7.c Verify proper CA pump status as follows: 1) Power to both motor driven CA pumps – AVAILABLE. STANDARD: Applicant determines that power is not available to 1B CA pump by observing the indicating lights dark and transitions to the RNO. COMMENTS:	___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 6 7.c.1) Perform the following: • **IF** 1ETA **OR** 1ETB de-energized, **THEN** restore power to affected essential bus. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power). • **IF** essential bus energized, **THEN** dispatch operator to determine cause of breaker failure. STANDARD: Applicant determines that power is available to 1ETB and dispatches an operator to determine cause of breaker failure. **Examiner Cue: Once contacted as an AO, "Operator dispatched to determine cause of breaker failure."** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 7 7.c.2) 1AD-5, F/3 "CAPT MECH OS TRIP" – DARK. STANDARD: Applicant determines 1AD-5, F/3 is lit and transitions to the RNO. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 8 7.c.2) RNO Perform the following: a) Dispatch operator to reset CAPT trip and throttle valve. b) **IF AT ANY TIME** CAPT trip and throttle valve reset prior to reaching feed and bleed criteria, **THEN** perform step 7. c) GO TO Step 7.d STANDARD: Applicant determines that the CAPT T/V CTRL is closed and transitions to the RNO. **Examiner Cue: "Operator dispatched to open the trip and throttle valve."** COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 9 7.d Ensure all CA isolation valves – OPEN. STANDARD: Applicant verifies all CA isolation valve red OPEN lights lit and green CLSD lights dark. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 10 7.e Verify all CA flow control valves – OPEN. STANDARD: Applicant verifies all CA flow control valves are closed and transitions to the RNO. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 11 7.e RNO Perform the following: 1) **IF** valve(s) closed as required by Step 37, **THEN GO TO** Step 7.f. 2) OPEN affected valve(s). STANDARD: Applicant determines from cue sheet that the CA flow control valves were closed per step 37 and proceeds to Step 7.f. **Examiner Cue: If asked why the CA flow control valves are closed, reply "CA flow control valves were closed per Step 37."** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 12 7.f Start all available CA pumps. STANDARD: Applicant depresses the 1A CA pump red ON pushbutton and verifies the red ON light lit and green OFF light dark. **Examiner Note: This step is critical to provide CA flow to the 1A and/or 1B S/Gs to meet the JPM standard.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 13 7.g Verify total CA flow – GREATER THAN 450 GPM. STANDARD: Applicant determines that total CA flow is < 450 GPM and transitions to the RNO. COMMENTS:	___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 14 7.g RNO Perform the following: 1) **IF** only one motor driven CA pump on, **AND** its discharge path cannot be aligned to associated S/Gs, **THEN** evaluate aligning flow to other S/Gs through motor driven CA train A/B cross-tie alignment. **REFER TO** Enclosure 3 (Motor Driven CA Pump Train A/B Cross-Tie Alignment). STANDARD: Applicant determines that this step is not applicable and continues in the RNO. **Examiner Note: A copy of Encl. 3 has been provided if the applicant asks for it.** COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 15 7.g RNO 2) **IF** any CA pump on, **AND** Step 37 has been implemented, **THEN GO TO** Enclosure 7 (S/G CA Flow Restoration). STANDARD: Applicant determines from cue sheet that Step 37 has been implemented and transitions to Enclosure 7. **Examiner Note: Provide applicant with a copy of FR-H.1 Encl. 7. The following steps are from Encl. 7.** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT

NOTE:
- It may be preferable to feed 1B or 1C S/G first, to maintain steam supply to CAPT.
- Selecting S/G with the highest level will reduce risk of thermal shock to S/G when reestablishing feed flow.
- The available feed source will also determine which S/G can be fed.

STEP 16 1. Select one S/G to be fed. STANDARD: Applicant determines that either 1A or 1B S/G will be fed since the 1A CA pump is the only source of feed at this time. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 17 2. Verify core exit T/Cs – STABLE OR TRENDING DOWN. STANDARD: Applicant determines from the OAC or plasma display that CETs are trending down. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 18 3. THROTTLE open CA flow control valve to selected S/G to establish feed flow rate less than or equal to 100 GPM. STANDARD: Applicant turns flow control knob clockwise to establish flow to either 1A or 1B S/G. **Examiner Note: This step is critical to establish CA flow to either 1A or 1B S/G.** **Examiner Cue: If asked, "Containment pressure has remained less than 3 PSIG."** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 19 4. Maintain feed flow rate less than or equal to 100 GPM until W/R S/G level greater than 12% (21% ACC). STANDARD: Applicant determines that W/R level is > 12% on both 1A and 1B S/Gs. COMMENTS:	___ **SAT** ___ **UNSAT**

CAUTION Feed flow rates should be controlled to raise S/G level and prevent excessive NC System cooldown.

STEP/STANDARD	SAT/UNSAT
STEP 20 5. **WHEN** W/R S/G level greater than 12% (21% ACC), **THEN** feed flow can be raised to desired rate. STANDARD: Applicant determines that feed flow can be raised to desired rate. COMMENTS:	___ SAT ___ UNSAT
STEP 21 6. Slowly establish flow to any intact S/G with W/R level greater than 12% (21% ACC). STANDARD: Applicant determines that W/R level is > 12% on both 1A and 1B S/Gs and establishes flow to the S/G that has not been fed to this point by rotating the flow control knob in the clockwise direction. COMMENTS:	___ SAT ___ UNSAT
STEP 22 7. Verify the following: • NC T-Hot associated with S/G(s) being fed - TRENDING DOWN • Core Exit T/Cs – TRENDING DOWN. STANDARD: Applicant determines that the temperatures are trending down. COMMENTS:	___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 23 8. Verify S/G(s) being fed – INTACT. STANDARD: Applicant determines that both 1A and 1B S/Gs are intact. COMMENTS:	___ SAT ___ UNSAT
STEP 24 9. **GO TO** Section C. (Operator Actions), Step 39. STANDARD: Applicant proceeds to Step 39. **Examiner Cue: "Another operator will continue with FR-H.1. This JPM is complete."** COMMENTS: **END OF TASK**	___ SAT ___ UNSAT

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

- **Unit 1 is in Mode 3 following a reactor trip due to a loss of all feedwater.**
- **Bleed and Feed has been established per EP/1/A/5000/FR-H.1.**
- **CA flow control valves have been closed per Step 37.**
- **Report from AO and Maintenance in the field that 1A CA pump is ready to be started has just been received.**

INITIATING CUES:

- **The CRS instructs you to perform Step 7 to establish CA flow from 1A CA pump.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify NC System feed and bleed required as follows:**	
__ a. W/R level in at least 3 S/Gs - LESS THAN 24% (36% ACC).	a. Perform the following: __ 1) Monitor feed and bleed initiation criteria. **<u>REFER</u> <u>TO</u>** Enclosure 1 (Foldout Page). __ 2) **<u>WHEN</u>** criteria satisfied, **<u>THEN</u> <u>GO</u> <u>TO</u>** Step 21. __ 3) **<u>GO</u> <u>TO</u>** Step 6.
__ b. **<u>GO</u> <u>TO</u>** Step 21.	
__ 6. **Ensure S/G BB and NM valves closed. <u>REFER</u> <u>TO</u> Enclosure 9 (S/G BB and NM Valve Checklist).**	
7. **Attempt to establish CA flow to at least one S/G as follows:**	
__ a. Verify 1AD-8, B/1 "UST LO LEVEL" - DARK.	a. Perform the following: __ 1) **<u>REFER</u> <u>TO</u>** AP/1/A/5500/006 (Loss of S/G Feedwater). __ 2) **<u>GO</u> <u>TO</u>** Step 7.c.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	
__ b. Verify 1CA-4 (CA Pmps Suct From UST) - OPEN.	b. Perform the following: __ 1) OPEN 1CA-4. 2) **IF** no suction source can be aligned, **THEN** OPEN the following valves: __ • 1RN-250A (RN Hdr A To CA Pmp Suct Isol) __ • 1CA-116A (CA Pump #1 Suct Frm RN Hdr A) __ • 1CA-15A (CA Pump 1A Suct Frm RN Isol) __ • 1CA-85B (CA Pump #1 Suct Frm RN Hdr B) __ • 1CA-18B (CA Pump 1B Suct Frm RN Isol) __ • 1RN-310B (RN Hdr B To CA Pmp Suct Isol).
c. Verify proper CA pump status as follows:	
__ 1) Power to both motor driven CA pumps - AVAILABLE.	1) Perform the following: __ • **IF** 1ETA **OR** 1ETB de-energized, **THEN** restore power to affected essential bus. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power). __ • **IF** essential bus energized, **THEN** dispatch operator to determine cause of breaker failure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	
__ 2) 1AD-5, F/3 "CAPT MECH OS TRIP" - DARK.	2) Perform the following: __ a) Dispatch operator to reset CAPT trip and throttle valve. __ b) **<u>IF</u> <u>AT</u> <u>ANY</u> <u>TIME</u>** CAPT trip and throttle valve reset prior to reaching feed and bleed criteria, **<u>THEN</u>** perform Step 7. __ c) **<u>GO</u> <u>TO</u>** Step 7.d.
__ 3) "CAPT TRIP T/V CTRL" - OPEN.	3) Perform the following: __ a) OPEN valve. __ b) **<u>IF</u>** valve will not open, **<u>THEN</u>** dispatch operator to open CAPT trip and throttle valve.
4) Verify the following valves - OPEN: __ • 1SA-2 (S/G 1B SM To CAPT) __ • 1SA-5 (S/G 1C SM To CAPT).	__ 4) Place CA Pump #1 to "ON".
__ d. Ensure all CA isolation valves - OPEN.	
__ e. Verify all CA flow control valves - OPEN.	e. Perform the following: __ 1) **<u>IF</u>** valve(s) closed as required by Step 37, **<u>THEN</u> <u>GO</u> <u>TO</u>** Step 7.f. __ 2) OPEN affected valve(s).
__ f. Start all available CA pumps.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

7. (Continued)

__ g. Verify total CA flow - GREATER THAN 450 GPM.

g. Perform the following:

__ 1) **IF** only one motor driven CA pump on, **AND** its discharge path cannot be aligned to associated S/Gs, **THEN** evaluate aligning flow to other S/Gs through motor driven CA train A/B cross-tie alignment. **REFER TO** Enclosure 3 (Motor Driven CA Pump Train A/B Cross-Tie Alignment).

__ 2) **IF** any CA pump on, **AND** Step 37 has been implemented, **THEN GO TO** Enclosure 7 (S/G CA Flow Restoration).

3) **IF** any feed flow to at least one S/G verified, **THEN** perform the following:

__ a) Maintain flow to restore N/R level in at least one S/G to greater than 11% (29% ACC).

b) **IF AT ANY TIME** N/R level in at least one S/G trends up to greater than 11% (29% ACC) prior to reaching feed and bleed initiation criteria, **THEN** perform the following:

(1) **IF** NC System **OR** S/G depressurization in progress to feed S/G(s) from CM, **THEN** stabilize the following:

__ • NC pressure

__ • S/G pressure.

__ (2) **RETURN TO** procedure and step in effect.

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

7. (Continued)

__ c) Dispatch operator to verify proper CA valve alignment. **REFER TO** Enclosure 2 (Local CA Flowpath Restoration).

__ d) **IF AT ANY TIME** CA flow restored greater than 450 GPM prior to meeting feed and bleed initiation criteria, **THEN** perform Step 7.

4) **IF** no CA flow indicated, **THEN** perform the following:

__ a) **IF** no CA pump can be started, **THEN** dispatch operator and maintenance to CA pumps to attempt to restore one CA pump to service. **REFER TO** EM/1/A/5200/007 (Troubleshooting Cause For CA Pump(s) Failing to Start).

__ b) Dispatch operator to verify proper CA valve alignment. **REFER TO** Enclosure 2 (Local CA Flowpath Restoration).

__ c) **IF AT ANY TIME** CA flow restored prior to meeting feed and bleed initiation criteria, **THEN** perform Step 7.

__ 5) **GO TO** Step 8.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. (Continued)	
__ h. Verify feed and bleed - PREVIOUSLY ESTABLISHED (Steps 22 through 26 completed).	h. Perform the following: 1) **<u>IF</u>** NC System **<u>OR</u>** S/G depressurization in progress to feed S/G(s) from CM, **<u>THEN</u>** stabilize the following: __ • NC pressure __ • S/G pressure. __ 2) **<u>RETURN</u> <u>TO</u>** procedure and step in effect.
__ i. **<u>GO</u> <u>TO</u>** Step 39.	
8. **Verify Condenser Steam Dump as follows:**	
a. Verify condenser available as follows: __ • "C-9 COND AVAILABLE FOR STM DUMP" status light (1SI-18) - LIT __ • Any MSIV - OPEN.	__ a. **<u>GO</u> <u>TO</u>** Step 9.
__ b. Verify steam dumps in - T-AVG MODE.	__ b. **<u>GO</u> <u>TO</u>** Step 9.
c. Place steam dumps in pressure mode as follows: __ 1) Place steam dumps in - PRESSURE MODE. __ 2) Ensure "STM DUMP CTRL" - SET AT 1090 PSIG STEAM HEADER PRESSURE.	
__ 9. **Stop all NC pumps.**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- It may be preferable to feed 1B or 1C S/G first, to maintain steam supply to CAPT.
- Selecting S/G with highest level will reduce risk of thermal shock to S/G when reestablishing feed flow.
- The available feed source will also determine which S/G can be fed.

__ 1. **Select one S/G to be fed.**

__ 2. **Verify core exit T/Cs - STABLE OR TRENDING DOWN.**

Perform the following:

__ a. THROTTLE open CA flow control valve to establish flow rate required to lower core exit T/Cs temperature.

__ b. **IF** core exit T/Cs continue to trend up, **THEN** THROTTLE open CA flow control valves to feed other S/Gs as needed to lower core exit T/Cs temperature.

c. **IF** CA flow cannot be established to at least one S/G, **THEN** perform the following:

__ 1) Dispatch operator to verify proper CA valve alignment. **REFER TO** Enclosure 2 (Local CA Flowpath Restoration).

__ 2) **GO TO** Section C. (Operator Actions), Step 37.

__ d. **GO TO** Step 6.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 3. **THROTTLE open CA flow control valve to selected S/G to establish feed flow rate less than or equal to 100 GPM.**	**IF CA flow cannot be established to at least one S/G, THEN perform the following:** __ a. Dispatch operator to verify proper CA valve alignment. **REFER TO** Enclosure 2 (Local CA Flowpath Restoration). __ b. **GO TO** Section C. (Operator Actions), Step 37.
__ 4. **Maintain feed flow rate less than or equal to 100 GPM until W/R S/G level greater than 12% (21% ACC).**	

CAUTION **Feed flow rates should be controlled to raise S/G level and prevent excessive NC System cooldown.**

__ 5. **WHEN W/R S/G level greater than 12% (21% ACC), THEN feed flow can be raised to desired rate.**	
__ 6. **Slowly establish flow to any intact S/G with W/R level greater than 12% (21% ACC).**	
7. **Verify the following:** __ • NC T-Hot associated with S/G(s) being fed - TRENDING DOWN __ • Core Exit T/Cs - TRENDING DOWN.	__ **Do not continue in this enclosure until both conditions met.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 8. **Verify S/G(s) being fed - INTACT.**	**Perform the following:** __ a. **IF** S/G being fed faulted **OR** ruptured, **THEN** establish feedwater flow to another S/G observing previous flowrate requirements. __ b. **IF** an intact S/G not available to be fed, **THEN** establish feed flow to ruptured or faulted S/G. __ c. **WHEN** ruptured **OR** faulted S/G no longer required for heat sink, **THEN** isolate feed flow to ruptured or faulted S/G.
__ 9. **GO TO Section C. (Operator Actions), Step 39.**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

<u>CAUTION</u> Aligning single motor driven CA pump to more than two S/Gs can lead to pump runnout if control valves are opened too far or if they fail open.

<u>NOTE</u> 1A CA pump is normally aligned to 1A and 1B S/Gs. 1B CA pump is normally aligned to 1C and 1D S/Gs. Opening train A/B cross-tie will allow either motor driven CA pump to feed any S/G.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Verify 1A CA pump - ON.**	__ **<u>GO</u> <u>TO</u> Step 4.**
2. **CLOSE the following CA flow control valves:** __ • 1CA-44 (CA Pump 1B Flow To S/G 1C) __ • 1CA-40 (CA Pump 1B Flow To S/G 1D).	**Perform the following:** a. Dispatch operator to close the following valves. **<u>REFER</u> <u>TO</u>** Enclosure 14 (Unit 1 Local CA Flow Control Valve Operation): __ • 1CA-44 (CA Pump 1B Flow To S/G 1C) (AB-552, CC-DD, 52-53, Rm 250) (Ladder needed) __ • 1CA-40 (CA Pump 1B Flow To S/G 1D) (AB-553, BB-49, Rm 250) (Ladder needed). __ b. Do not continue until valves in previous step - CLOSED.
__ 3. **<u>GO</u> <u>TO</u> Step 6.**	
__ 4. **Verify 1B CA pump - ON.**	__ **Exit this enclosure.**

Motor Driven CA Pump Train A/B Cross-Tie Alignment

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **CLOSE the following CA flow control valves:** __ • 1CA-60 (CA Pump 1A Flow To S/G 1A) __ • 1CA-56 (CA Pump 1A Flow To S/G 1B).	**Perform the following:** a. Dispatch operator to close the following valves. **<u>REFER</u> <u>TO</u>** Enclosure 14 (Unit 1 Local CA Flow Control Valve Operation): __ • 1CA-60 (CA Pump 1A Flow To S/G 1A) (AB-551, BB-CC, 49-50, Rm 250) (Ladder needed) __ • 1CA-56 (CA Pump 1A Flow To S/G 1B) (AB-552, DD-52, Rm 250) (Ladder needed). __ b. Do not continue until valves in previous step - CLOSED.
__ 6. **Limit motor driven CA pump flow in the following steps to less than 500 GPM.**	
7. **Dispatch operator to unlock and open the following valves while monitoring CA flow:** __ • 1CA-111 (1A & 1B CA Pump Disch X-over To S/G Isol) (AB-552, BB-50, Rm 250) __ • 1CA-112 (1A & 1B CA Pump Disch X-over To S/G Isol) (AB-552, BB-50, Rm 250).	**<u>IF</u> valve(s) will not open, <u>THEN</u> perform the following to reduce ΔP across valves:** __ a. Stop motor driven CA pump. __ b. Notify dispatched operator to open valves prior to starting pump. __ c. Start motor driven CA pump.
__ 8. **THROTTLE open CA flow control valves to desired flow rate.**	__ **Notify dispatched operator to throttle local valves as required. <u>REFER</u> <u>TO</u> Enclosure 14 (Unit 1 Local CA Flow Control Valve Operation).**

JPM D

Catawba Nuclear Station
JPM D
2021 NRC Exam

EVALUATION SHEET

Task: Perform E-0 Actions To Ensure A Complete Containment Isolation

Alternate Path: Yes

Facility JPM #: N/A

Safety Function: 2 **Title:** Engineering Safety Features Actuation System

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

Rating(s): 4.5 / 4.8 **CFR:** 41.7 / 45.5 to 45.8

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator __X__ In-**Pl**ant ________ **P**erform __X__ **S**imulate ________

References: EP/1/A/5000/E-0 (Reactor Trip or Safety Injection) Rev 46

Task Standard: At least one train of Phase A containment isolation initiated and either 1NV-10A or 1NV-15B manually closed and either 1WL-805A or 1WL-807B manually closed.

Validation Time: 5 minutes **Time Critical:** Yes _____ No __X__

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Applicant: Time Start: __________

NAME______________________________ Docket #________________ Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/_________

NAME SIGNATURE DATE

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COMMENTS

Catawba Nuclear Station
JPM D
2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. ENSURE NRC Examination Security has been established.
2. Reset to IC # 144.
3. Enter the password.
4. Select “Yes” on the INITIAL CONDITION RESET pop-up window.
5. Ensure simulator setup per table below.
6. Place simulator in RUN.

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event
	MAL-NC013D (NC COLD LEG D LEAK)	27.5	:10			1
	MAL-ISE003A (AUTO PHASE A ISOL SIGNAL TRN A)	BLOCK				
	MAL-ISE003B (AUTO PHASE A ISOL SIGNAL TRN B)	BLOCK				
	VLV-NV009A (NV10A L/D ORIFICE 1B ISOL CNTRL FAIL AUTO ACTIONS)	ACTIVE				
	VLV-NV012A (NV15B L/D ISOL OUTSIDE CNMT VLV FAIL AUTO ACTIONS)	ACTIVE				
	VLV-WL007A (WL805A NCDT PMPS DISCH CONT ISOL IN FAIL AUTO ACTIONS)	ACTIVE				
	VLV-WL008A (WL807B NCDT PMPS DISCH CONT ISOL OUT FAIL AUTO ACTIONS)	ACTIVE				
	When applicant is ready, INSERT EVENT 1.					

Catawba Nuclear Station
JPM D
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **You are the OATC**
- **The BOP has stepped out of the control room**
- **The CRS is performing an IPTE brief on Unit 2**

INITIATING CUES:

- **Monitor your control boards**

EXAMINER NOTE: **After reading cue, and applicant has walked down the control boards, have the simulator operator INSERT EVENT 1. This will cause a safety injection on Unit 1. All E-0 immediate action steps will be met without any operator action. Once the applicant has announced that the Immediate Actions are complete, hand them a copy of E-0 beginning at step 6.**

EXAMINER CUE: "The CRS has validated that the immediate actions are complete and directs you to continue performance of E-0, beginning at Step 6."

STEP/STANDARD	SAT/UNSAT

START TIME: _______

STEP 1: 6. **Announce "Unit 1 Safety Injection".** STANDARD: Applicant uses the plant paging system to make the announcement. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 2 7. **Determine required notifications:** • **REFER TO** AD-EP-ALL-0111 (Control Room Activation of the ERO) • **REFER TO** AD-LS-ALL-0006 (Notification/Reportability Evaluation) STANDARD: Applicant acknowledges the step and continues with E-0. COMMENTS:	___ **SAT** ___**UNSAT**
STEP 3 8. **Verify all Feedwater Isolation status lights (1SI-5) – LIT.** STANDARD: Applicant verifies all Feedwater Isolation status lights lit on 1SI-5. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 4 9. **Verify Phase A Containment Isolation status as follows:** a. Phase A “RESET” lights – DARK. STANDARD: Applicant determines that the Phase A RESET lights are lit and transitions to the RNO. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 5 9.a.RNO. Initiate Phase A Isolation. STANDARD: Applicant depresses both of the red INITIATE pushbuttons for Train A and Train B and verifies the yellow RESET lights extinguish. **Examiner Note: It is critical for the applicant to at least initiate one train of Phase A to ensure that at least one valve in every penetration other than from the letdown penetration is isolated. The letdown penetration will have to be manually isolated in the RNO for the following step.** **Examiner Note: This begins the alternate path for this JPM.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 6 9.b. Monitor Light Panel Group 5 St lights on energized train(s) – LIT. STANDARD: Applicant determines that Group 5 St lights B/1, B/12, C/1, and C/12 are dark and transitions to the RNO. **Examiner Note: It may take up to 60 seconds for all other valves to indicate isolated.** COMMENTS:	___ SAT ___ UNSAT
STEP 7 9.b.RNO. Align valves as necessary to ensure each penetration isolated by at least one isolation valve. STANDARD: Applicant rotates the switch for 1NV-10A counterclockwise to the CLOSE position and verifies the green closed light is lit and red open light dark. Applicant depresses the green CLOSE pushbutton for 1NV-15B and verifies the green CLSD light lit and red OPEN light dark. Applicant depresses the green CLOSE pushbutton for 1WL-805A and 1WL-807B and verifies the green CLSD light lit and red OPEN light dark on both valves. **Examiner Note: It is critical for the applicant to close 1 of the valves in each penetration (either 1NV-10A or 1NV-15B and either 1WL-805A or 1WL-807B) to ensure the penetration is isolated.** **Examiner Cue: Once valves have been closed, "This JPM is complete."** COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

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INITIAL CONDITIONS:

- **You are the OATC**
- **The BOP has stepped out of the control room**
- **The CRS is performing an IPTE brief on Unit 2**

INITIATING CUES:

- **Monitor your control boards**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. <u>Operator Actions</u>

✓ 1. **Monitor Enclosure 1 (Foldout Page).**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
(2.) **Verify Reactor Trip:**	**Perform the following:**
✓ • All rod bottom lights - LIT	__ a. Trip reactor.
✓ • All reactor trip and bypass breakers - OPEN	b. **<u>IF</u>** reactor will not trip, **<u>THEN</u>** concurrently perform the following:
✓ • I/R power - TRENDING DOWN.	__ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees)
	__ • **<u>GO TO</u>** EP/1/A/5000/FR-S.1 (Response to Nuclear Power Generation/ATWS).
(3.) **Verify Turbine Trip:**	**Perform the following:**
✓ • All turbine stop valves - CLOSED.	__ a. Trip turbine.
	b. **<u>IF</u>** turbine will not trip, **<u>THEN</u>** perform the following:
	__ 1) Depress "MANUAL" pushbutton on turbine control panel.
	__ 2) Rapidly CLOSE control valves by simultaneously depressing "CONTROL VALVE LOWER" and "FAST RATE" pushbuttons.
	3) **<u>IF</u>** control valves will not close, **<u>THEN</u>** CLOSE the following valves:
	__ • All MSIVs
	__ • All MSIV bypass valves.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. Verify 1ETA and 1ETB - ENERGIZED.	**Perform the following:** __ a. **IF** 1ETA **AND** 1ETB de-energized, **THEN GO TO** EP/1/A/5000/ECA-0.0 (Loss of All AC Power). __ b. **WHEN** time allows, **THEN** attempt to restore power to de-energized switchgear while continuing with this procedure. **REFER TO** AP/1/A/5500/007 (Loss of Normal Power).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Verify S/I actuated:**	
✓ a. "SAFETY INJECTION ACTUATED" status light (1SI-13) - LIT.	a. Perform the following: 1) Verify conditions requiring S/I: __ • Pzr pressure - LESS THAN 1845 PSIG OR __ • Containment pressure - GREATER THAN 1.2 PSIG. __ 2) **IF** S/I required, **THEN** initiate S/I. 3) **IF** S/I not required, **THEN** concurrently perform the following: • **IF** 1ETA **OR** 1ETB de-energized, **THEN** ensure the following pumps running on energized bus: __ • NV pump __ • KC pumps __ • RN pump. __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees). __ • **GO TO** EP/1/A/5000/ES-0.1 (Reactor Trip Response).
✓ b. Both E/S load sequencer actuated status lights (1SI-14) - LIT.	__ b. Initiate S/I.
__ 6. **Announce "Unit 1 Safety Injection".**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Determine required notifications:**	
__ • **<u>REFER TO</u>** AD-EP-ALL-0111 (Control Room Activation of the ERO)	
__ • **<u>REFER TO</u>** AD-LS-ALL-0006 (Notification/Reportability Evaluation).	
__ 8. **Verify all Feedwater Isolation status lights (1SI-5) - LIT.**	**Perform the following:** __ a. Initiate Feedwater Isolation. __ b. **<u>IF</u>** proper status light indication not obtained, **<u>THEN</u>** CLOSE valves.
9. **Verify Phase A Containment Isolation status as follows:**	
__ a. Phase A "RESET" lights - DARK.	__ a. Initiate Phase A Isolation.
__ b. Monitor Light Panel Group 5 St lights on energized train(s) - LIT.	__ b. Align valves as necessary to ensure each penetration isolated by at least one isolation valve.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

10. **Verify Phase B actuation as follows:**

__ a. Verify containment pressure - HAS REMAINED LESS THAN 3 PSIG.

a. Perform the following:

1) Verify Phase B Isolation actuated as follows:

__ a) Phase B Isolation "RESET" lights - DARK.

__ b) **<u>IF</u>** Phase B Isolation "RESET" lights lit, **<u>THEN</u>** initiate Phase B Isolation.

c) Verify following monitor light panel lights on energized train(s) - LIT:

__ • Group 1 Sp lights

__ • Group 5 Sp lights

__ • Group 5 St light L/11.

__ d) **<u>IF</u>** monitor light panel not in correct alignment, **<u>THEN</u>** ensure correct alignment.

__ 2) Stop all NC pumps.

__ 3) Maintain seal injection flow.

__ 4) Energize H_2 igniters.

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

10. (Continued)

5) Dispatch operator to perform the following:

__ a) Secure all ice condenser air handling units. **<u>REFER</u>** **<u>TO</u>** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 11 (Securing All Ice Condenser Units).

__ b) Place containment H_2 analyzers in service. **<u>REFER</u>** **<u>TO</u>** OP/1/A/6450/010 (Containment Hydrogen Control Systems).

__ 6) **<u>WHEN</u>** 9 minutes elapsed, **<u>THEN</u>** verify proper VX System operation. **<u>REFER</u>** **<u>TO</u>** Enclosure 5 (VX System Operation).

__ 7) **<u>GO</u>** **<u>TO</u>** Step 11.

__ b. **<u>IF</u>** **<u>AT</u>** **<u>ANY</u>** **<u>TIME</u>** containment pressure exceeds 3 PSIG while in this procedure, **<u>THEN</u>** perform Step 10.a.

Catawba Nuclear Station
JPM E
2021 NRC Exam

JPM E

Catawba Nuclear Station
JPM E
2021 NRC Exam

EVALUATION SHEET

Task: Align the NS System for Cold Leg Recirculation

Alternate Path: Yes

Facility JPM #: NS-106

Safety Function: 5 **Title:** Containment Integrity

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

Rating(s): 4.5 / 4.3 **CFR:** 41.7 / 45.5 to 45.8

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator __X__ In-**P**lant ________ **P**erform __X__ **S**imulate ________

References: EP/1/A/5000/ES-1.3 (Transfer to Cold Leg Recirculation) Enclosure 2. Rev.031

Task Standard: Aligns NS system for Cold Leg Recirculation with 1B NS Pump running and proper RN flow to the 1B NS Heat Exchanger.

Validation Time: 15 minutes **Time Critical:** Yes _____ No __X__

==

Applicant:
NAME____________________________ Docket #________________ Time Start: __________
Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

Catawba Nuclear Station
JPM E
2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. ENSURE NRC Examination Security has been established.
2. Reset to IC # 145.
3. Enter the password.
4. Select "Yes" on the INITIAL CONDITION RESET pop-up window.
5. Ensure simulator setup per table below.
6. Place simulator in RUN and acknowledge any alarms.
7. ENSURE "Extra Operator" is present in the simulator.
8. Place simulator in FREEZE until Examiner cue is given.

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event
	MAL-NC013A (NC COLD LEG A LEAK)	27.5				
	VLV-NS009F (NS18A NS PMP A SUCT FROM CNMT SUMP FAIL TO POSITION)	0				

Catawba Nuclear Station
JPM E
2021 NRC Exam

READ TO APPLICANT

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INITIAL CONDITIONS:

- **A LOCA has occurred on Unit 1.**
- **EP/1/A/5000/ES-1.3, (Transfer to Cold Leg Recirculation) has been implemented.**

INITIATING CUES:

The CRS instructs you to align NS to Cold Leg Recirculation per Enclosure 2 of ES-1.3.

Examiner Note: After reading cue, provide the applicant with a copy of EP/1/A/5000/ES-1.3 Enclosure 2.

STEP/STANDARD	SAT/UNSAT

START TIME: _______

STEP 1: 1. **Verify both NS pumps - OFF.** STANDARD: Applicant verifies that 1A and 1B NS pumps are off . COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 2 2. **CLOSE the following valves:** • 1NS-20A (NS Pump 1A Suct From FWST) • 1NS-3B (NS Pump 1B Suct From FWST). STANDARD: Applicant depresses the green CLOSE pushbuttons for 1NS-20A and 1NS-3B. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 3 3. **Verify containment pressure - GREATER THAN 3 PSIG** STANDARD: Applicant verifies containment pressure is greater than 3 psig on 1NSP5040, 1NSP5050, 1NSP5060 or 1NSP5070 on 1MC-11 or on the OAC or on any chart recorder containing containment pressure. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 4 4. **Verify at least one of the following annunciators - LIT:** • 1AD-20, B/3 "CONT. SUMP LEVEL >3.3 ft" OR • 1AD-21, B/3 "CONT. SUMP LEVEL >3.3 ft". STANDARD: Applicant verifies that at least 1AD-20, B/3 or 1AD-21, B/3 is lit on 1MC-7 COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 5 5. **Align NS Train 1A to containment sump as follows:** a. Verify NS Pump 1A - AVAILABLE TO RUN. STANDARD: Applicant verifies that NS Pump 1A is available to run. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 6 5. b. Verify 1NI-185A (ND Pump 1A Cont Sump Suct) - OPEN. STANDARD: Applicant verifies that 1NI-185A is open. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 7 5. c. Verify NS Pump 1B – OFF STANDARD: Applicant verifies that NS Pump 1B is off. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 8 5. d. OPEN 1NS-29A (NS Spray Hdr 1A Cont Isol). STANDARD: Applicant depresses the red OPEN pushbutton for 1NS-29A on 1MC-11. COMMENTS:	___ SAT ___ UNSAT
STEP 9 5. e. OPEN 1NS-32A (NS Spray Hdr 1A Cont Isol). STANDARD: Applicant depresses the red OPEN pushbutton for 1NS-32A on 1MC-11. COMMENTS:	___ SAT ___ UNSAT
STEP 10 5. f. Verify 1NS-20A (NS Pump 1A Suct From FWST) - CLOSED. STANDARD: Applicant verifies that 1NS-20A is closed. COMMENTS:	___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 11 5. g. OPEN 1NS-18A (NS Pmp A Suct From Cont Sump). STANDARD: Applicant depresses the red OPEN pushbutton for 1NS-18A on 1MC-11. The valve will not open. No RNO step available for this step. Applicant proceeds to next step. **Examiner Note: This begins the alternate path for this JPM.** COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 12 5. h. Verify the following valves - OPEN: • 1NS-29A (NS Spray Hdr 1A Cont Isol) • 1NS-32A (NS Spray Hdr 1A Cont Isol) • 1NS-18A (NS Pmp A Suct From Cont Sump). STANDARD: Applicant verifies that 1NS-29A and 1NS-32A are open and that 1NS-18A is closed and transitions to the RNO. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 13 5.h RNO. **IF** any valve remains CLOSED or INTERMEDIATE for over 25 seconds, **THEN GO TO** Step 6. STANDARD: Applicant proceeds to Step 6 in order to place 1B NS train in service. COMMENTS:	___ SAT ___ UNSAT
STEP 14 6. **Align NS Train 1B to containment sump as follows:** a. Verify 1NI-184B (ND Pump 1B Cont Sump Suct) - OPEN. STANDARD: Applicant verifies that 1NI-184B is open. COMMENTS:	___ SAT ___ UNSAT
STEP 15 6. b. Verify NS Pump 1A – OFF STANDARD: Applicant verifies that NS Pump 1A is off. COMMENTS:	___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 16 6.c OPEN 1NS-15B (NS Spray Hdr 1B Cont Isol). STANDARD: Applicant depresses the red OPEN pushbutton for 1NS-15B and verifies the red open light is lit and green CLSD light is dark. **Examiner Note: This step is critical to provide a flowpath from the pump to the containment spray rings.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 17 6.d OPEN 1NS-12B (NS Spray Hdr 1B Cont Isol). STANDARD: Applicant depresses the red OPEN pushbutton for 1NS-12B on 1MC-11 and verifies the red OPEN light lit and green CLSD light dark. **Examiner Note: This step is critical to provide a flowpath from the pump to the containment spray rings.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 18 6.e Verify 1NS-3B (NS Pump 1B Suct From FWST) - CLOSED. STANDARD: Applicant verifies green CLSD light lit and red OPEN light dark for 1NS-3B. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 19 6.f. OPEN 1NS-1B (NS Pmp B Suct From Cont Sump). STANDARD: Applicant depresses the red OPEN pushbutton and verifies the red OPEN light lit and green CLSD light dark for 1NS-1B. **Examiner Note: This step is critical to provide a suction path from the containment sump to the NS pump.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 20 6.g. Verify the following valves – OPEN: • 1NS-15B (NS Spray Hdr 1B Cont Isol) • 1NS-12B (NS Spray Hdr 1B Cont Isol) • 1NS-1B (NS Pmp B Suct From Cont Sump) STANDARD: Applicant verifies that listed valves are open. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 21 6.h. Verify containment pressure – GREATER THAN 1 PSIG. STANDARD: Applicant verifies that containment pressure is > 1 PSIG. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 22 6.i Start NS Pump 1B. STANDARD: Applicant depresses the red ON pushbutton for 1B NS pump and verifies the red ON light lit and green OFF light dark. **Examiner Note: This step is critical to initiate containment spray flow.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

CAUTION: Exceeding 4650 GPM RN flow through an NS Hx will cause damage to the Hx tubes.

STEP 23 6.j.1) Align RN to NS Hx 1B as follows: Verify at least one of the following: • All Unit 1 and Unit 2 RN pumps – ON OR • RN System – ALIGNED FOR SINGLE SUPPLY HEADER OPERATION STANDARD: Applicant verifies that all Unit 1 and Unit 2 RN pumps are on. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 24 6.j.2) OPEN 1RN-225B (NS Hx 1B Inlet Isol). STANDARD: Applicant depresses the red OPEN pushbutton for 1RN-225B and verifies the red OPEN light lit and green CLSD light dark. **Examiner Note: This step is critical to provide cooling of containment spray.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 25 6.j.3). **WHEN** 1RN-225B begins to open, **THEN** OPEN 1RN-229B (NS Hx 1B Otlt Isol) STANDARD: Applicant depresses and holds the red OPEN pushbutton for 1RN-229B and verifies the red OPEN light lit and green CLSD light dark. **Examiner Note: This step is critical to provide cooling of containment spray.** **Examiner Cue: Once cooling water has been established: "Another operator will continue this enclosure. JPM complete."** COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **A LOCA has occurred on Unit 1.**
- **EP/1/A/5000/ES-1.3, (Transfer to Cold Leg Recirculation) has been implemented.**

INITIATING CUES:

The CRS instructs you to align NS to Cold Leg Recirculation per Enclosure 2 of ES-1.3.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Verify both NS pumps - OFF.**	**Perform the following:** a. <u>**IF**</u> all the following conditions met: __ • NS in service __ • NS suction aligned to containment sump __ • RN established to associated NS Hx, __ <u>**THEN**</u> <u>**RETURN**</u> <u>**TO**</u> procedure section and step in effect. __ b. Ensure both NS pumps - OFF.
2. **CLOSE the following valves:** __ • 1NS-20A (NS Pump 1A Suct From FWST) __ • 1NS-3B (NS Pump 1B Suct From FWST).	
__ 3. **Verify containment pressure - GREATER THAN 3 PSIG.**	**Perform the following:** __ a. Wait up to 20 seconds for 1NS-20A and 1NS-3B to close. __ b. OPEN 1NS-18A (NS Pmp A Suct From Cont Sump). __ c. OPEN 1NS-1B (NS Pmp B Suct From Cont Sump). __ d. <u>**IF**</u> <u>**AT**</u> <u>**ANY**</u> <u>**TIME**</u> containment pressure goes above 3 PSIG, <u>**THEN**</u> perform Enclosure 2 (Aligning NS for Recirculation). __ e. <u>**RETURN**</u> <u>**TO**</u> procedure section and step in effect.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. **Verify at least one of the following annunciators - LIT:** __ • 1AD-20, B/3 "CONT. SUMP LEVEL >3.3 ft" OR __ • 1AD-21, B/3 "CONT. SUMP LEVEL >3.3 ft".	**Perform the following:** __ a. **<u>WHEN</u>** at least one "CONT. SUMP LEVEL >3.3 ft" annunciator - LIT, **<u>THEN</u> <u>GO</u> <u>TO</u>** Step 5. __ b. Do not continue in this enclosure until at least one annunciator - LIT.
5. **Align NS train 1A to containment sump as follows:**	
__ a. Verify NS pump 1A - AVAILABLE TO RUN.	__ a. **<u>GO</u> <u>TO</u>** Step 6.
__ b. Verify 1NI-185A (ND Pump 1A Cont Sump Suct) - OPEN.	__ b. **<u>GO</u> <u>TO</u>** Step 6.
__ c. Verify NS pump 1B - OFF.	c. **<u>IF</u>** NS pump 1B running **<u>AND</u>** RN established to NS Hx 1B, **<u>THEN</u>** perform the following: __ 1) Ensure 1NS-20A (NS Pump 1A Suct From FWST) - CLOSED. __ 2) Ensure 1NS-18A (NS Pmp A Suct From Cont Sump) - OPEN. __ 3) **<u>GO</u> <u>TO</u>** Step 7.
__ d. OPEN 1NS-29A (NS Spray Hdr 1A Cont Isol).	__ d. **<u>GO</u> <u>TO</u>** Step 6.
__ e. OPEN 1NS-32A (NS Spray Hdr 1A Cont Isol).	__ e. **<u>GO</u> <u>TO</u>** Step 6.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. (Continued)	
__ f. Verify 1NS-20A (NS Pump 1A Suct From FWST) - CLOSED.	__ f. **IF** 1NS-20A remained OPEN or INTERMEDIATE for over 20 seconds, **THEN GO TO** Step 6.
__ g. OPEN 1NS-18A (NS Pmp A Suct From Cont Sump).	
h. Verify the following valves - OPEN: __ • 1NS-29A (NS Spray Hdr 1A Cont Isol) __ • 1NS-32A (NS Spray Hdr 1A Cont Isol) __ • 1NS-18A (NS Pmp A Suct From Cont Sump).	__ h. **IF** any valve remains CLOSED or INTERMEDIATE for over 25 seconds, **THEN GO TO** Step 6.
__ i. Verify containment pressure - GREATER THAN 1 PSIG.	i. Perform the following: 1) CLOSE the following valves: __ • 1NS-29A (NS Spray Hdr 1A Cont Isol) __ • 1NS-32A (NS Spray Hdr 1A Cont Isol). __ 2) **IF AT ANY TIME** containment pressure exceeds 1 PSIG, **THEN RETURN TO** Step 4. __ 3) **GO TO** Step 7.
__ j. Start NS pump 1A.	__ j. **GO TO** Step 6.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. (Continued)

CAUTION **Exceeding 4650 GPM RN flow through an NS Hx will cause damage to the Hx tubes.**

k. Align RN to NS Hx 1A as follows:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1) Verify at least one of the following: __ • All Unit 1 and Unit 2 RN pumps - ON **OR** __ • RN System - ALIGNED FOR SINGLE SUPPLY HEADER OPERATION.	1) Perform the following to support NS Hx cooling flow: __ a) **IF** only one A train RN pump on, **THEN** CLOSE Unit 2 2RN-48B (RN Supply X-Over Isol). b) **IF** only A train RN pumps on, **THEN** CLOSE one of the following Unit 2 valves: __ • 2RN-47A (RN Supply X-Over Isol) OR __ • 2RN-48B (RN Supply X-Over Isol).
__ 2) OPEN 1RN-144A (NS Hx 1A Inlet Isol).	
__ 3) **WHEN** 1RN-144A begins to open, **THEN** OPEN 1RN-148A (NS Hx 1A Otlt Isol).	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. **Align NS train 1B to containment sump as follows:**	
__ a. Verify 1NI-184B (ND Pump 1B Cont Sump Suct) - OPEN.	__ a. **<u>GO</u> <u>TO</u>** Step 7.
__ b. Verify NS pump 1A - OFF.	b. **<u>IF</u>** NS pump 1A running **<u>AND</u>** RN established to NS Hx 1A, **<u>THEN</u>** perform the following: __ 1) Ensure 1NS-3B (NS Pump 1B Suct From FWST) - CLOSED. __ 2) Ensure 1NS-1B (NS Pmp B Suct From Cont Sump) - OPEN. __ 3) **<u>GO</u> <u>TO</u>** Step 7.
__ c. OPEN 1NS-15B (NS Spray Hdr 1B Cont Isol).	__ c. **<u>GO</u> <u>TO</u>** Step 7.
__ d. OPEN 1NS-12B (NS Spray Hdr 1B Cont Isol).	__ d. **<u>GO</u> <u>TO</u>** Step 7.
__ e. Verify 1NS-3B (NS Pump 1B Suct From FWST) - CLOSED.	__ e. **<u>IF</u>** 1NS-3B remained OPEN or INTERMEDIATE for over 20 seconds, **<u>THEN</u> <u>GO</u> <u>TO</u>** Step 7.
__ f. OPEN 1NS-1B (NS Pmp B Suct From Cont Sump).	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
g. Verify the following valves - OPEN: __ • 1NS-15B (NS Spray Hdr 1B Cont Isol) __ • 1NS-12B (NS Spray Hdr 1B Cont Isol) __ • 1NS-1B (NS Pmp B Suct From Cont Sump).	__ g. **IF** any valve remains CLOSED or INTERMEDIATE for over 25 seconds, **THEN GO TO** Step 7.
__ h. Verify containment pressure - GREATER THAN 1 PSIG.	h. Perform the following: 1) CLOSE the following valves: __ • 1NS-15B (NS Spray Hdr 1B Cont Isol) __ • 1NS-12B (NS Spray Hdr 1B Cont Isol). __ 2) **IF AT ANY TIME** containment pressure exceeds 1 PSIG, **THEN RETURN TO** Step 4. __ 3) **GO TO** Step 7.
__ i. Start NS pump 1B.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6. (Continued)	
CAUTION **Exceeding 4650 GPM RN flow through an NS Hx will cause damage to the Hx tubes.**	
j. Align RN to NS Hx 1B as follows:	
1) Verify at least one of the following: __ • All Unit 1 and Unit 2 RN pumps - ON **OR** __ • RN System - ALIGNED FOR SINGLE SUPPLY HEADER OPERATION.	1) Perform the following to support NS Hx cooling flow: __ a) **IF** only one B train RN pump on, **THEN** CLOSE Unit 2 2RN-47A (RN Supply X-Over Isol). b) **IF** only B Train RN pumps on, **THEN** CLOSE one of the following Unit 2 valves: __ • 2RN-48B (RN Supply X-Over Isol) OR __ • 2RN-47A (RN Supply X-Over Isol).
__ 2) OPEN 1RN-225B (NS Hx 1B Inlet Isol).	
__ 3) **WHEN** 1RN-225B begins to open, **THEN** OPEN 1RN-229B (NS Hx 1B Otlt Isol).	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7. **Verify proper NS alignment as follows:**	
__ a. Verify 1NS-18A (NS Pmp A Suct From Cont Sump) - OPEN.	a. Perform the following: __ 1) **<u>IF</u>** 1NI-185A (ND Pump 1A Cont Sump Suct) open **<u>AND</u>** 1NS-20A (NS Pump 1A Suct From FWST) closed, **<u>THEN</u>** OPEN 1NS-18A. __ 2) DO **<u>NOT</u>** start 1A NS pump until aligned to containment sump.
__ b. Verify 1NS-1B (NS Pmp B Suct From Cont Sump) - OPEN.	b. Perform the following: __ 1) **<u>IF</u>** 1NI-184B (ND Pump 1B Cont Sump Suct) open **<u>AND</u>** 1NS-3B (NS Pump 1B Suct From FWST) closed, **<u>THEN</u>** OPEN 1NS-1B. __ 2) DO **<u>NOT</u>** start 1B NS pump until aligned to containment sump.
__ c. Verify NS pump 1A - ON.	c. Ensure the following valves - CLOSED: __ • 1NS-29A (NS Spray Hdr 1A Cont Isol) __ • 1NS-32A (NS Spray Hdr 1A Cont Isol).
__ d. Verify NS pump 1B - ON.	d. Ensure the following valves - CLOSED: __ • 1NS-15B (NS Spray Hdr 1B Cont Isol) __ • 1NS-12B (NS Spray Hdr 1B Cont Isol).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. **IF AT ANY TIME NS flow lost OR RN flow lost to operating NS Hx, THEN start other NS pump as follows:**	
__ a. Ensure affected NS pump - OFF.	
b. CLOSE the following valves for affected train: • A Train: __ • 1NS-29A (NS Spray Hdr 1A Cont Isol) __ • 1NS-32A (NS Spray Hdr 1A Cont Isol) __ • 1RN-148A (NS Hx 1A Otlt Isol) __ • 1RN-144A (NS Hx 1A Inlet Isol). • B Train: __ • 1NS-12B (NS Spray Hdr 1B Cont Isol) __ • 1NS-15B (NS Spray Hdr 1B Cont Isol) __ • 1RN-229B (NS Hx 1B Otlt Isol) __ • 1RN-225B (NS Hx 1B Inlet Isol).	
c. Verify both the following Unit 2 valves - OPEN: __ • 2RN-47A (RN Supply X-Over Isol) __ • 2RN-48B (RN Supply X-Over Isol).	__ c. **IF** affected valve closed to support NS Hx cooling flow, **THEN** ensure valve - RETURNED TO PREVIOUS ALIGNMENT.
__ d. **RETURN TO** Step 4 in this enclosure.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9. **Verify adequate RN heat sink as follows:**

__ • RN System - SUCTION ALIGNED TO LAKE WYLIE

• RN essential header temperatures at one of the following locations - LESS THAN OR EQUAL TO 93°F:

__ • 1MC-9

OR

__ • RO Logbook.

Perform the following:

a. Ensure the following valves - OPEN:

__ • 1RN-3A (RN P/H Pit A Isol From SNSWP)

__ • 1RN-4B (RN P/H Pit B Isol From SNSWP)

__ • 1RN-58B (RN Hdr B Ret To SNSWP)

__ • 1RN-63A (RN Hdr A Ret To SNSWP)

__ • 1RN-846A (D/G 1A Hx Ret To SNSWP)

__ • 1RN-848B (D/G 1B Hx Ret To SNSWP)

__ • 2RN-846A (D/G 2A Hx Ret To SNSWP)

__ • 2RN-848B (D/G 2B Hx Ret To SNSWP).

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9. (Continued)	b. Ensure the following valves - CLOSED: __ • 1RN-1A (RN P/H Pit A Isol From Lake) __ • 1RN-2B (RN P/H Pit A Isol From Lake) __ • 1RN-5A (RN P/H Pit B Isol From Lake) __ • 1RN-6B (RN P/H Pit B Isol From Lake) __ • 1RN-53B (Station RN Disch Hdr X-Over) __ • 1RN-54A (Station RN Disch Hdr X-Over) __ • 1RN-57A (Station RN Disch To RL Sys) __ • 1RN-843B (Station RN Disch To RL Sys) __ • 1RN-847A (D/G 1A Hx Ret To Lake) __ • 1RN-849B (D/G 1B Hx Ret To Lake) __ • 2RN-847A (D/G 2A Hx Ret To Lake) __ • 2RN-849B (D/G 2B Hx Ret To Lake).
__ 10. **Verify any NS pump - ON.**	__ **Exit this enclosure.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

__ 11. **Notify Control Room Supervisor this enclosure shall remain in effect until current or subsequent procedures provide alternate guidance.**

12. **IF AT ANY TIME containment pressure less than 1 PSIG, THEN perform the following:**

__ a. Ensure NS pump - OFF.

b. CLOSE the following valves for affected train:

- A Train:
  - __ 1NS-29A (NS Spray Hdr 1A Cont Isol)
  - __ 1NS-32A (NS Spray Hdr 1A Cont Isol)
  - __ 1RN-148A (NS Hx 1A Otlt Isol)
  - __ 1RN-144A (NS Hx 1A Inlet Isol).
- B Train:
  - __ 1NS-12B (NS Spray Hdr 1B Cont Isol)
  - __ 1NS-15B (NS Spray Hdr 1B Cont Isol)
  - __ 1RN-229B (NS Hx 1B Otlt Isol)
  - __ 1RN-225B (NS Hx 1B Inlet Isol).

c. Verify both the following Unit 2 valves - OPEN:

- __ 2RN-47A (RN Supply X-Over Isol)
- __ 2RN-48B (RN Supply X-Over Isol).

RESPONSE NOT OBTAINED:

__ c. **IF** affected valve closed to support NS Hx cooling flow, **THEN** ensure valve - RETURNED TO PREVIOUS ALIGNMENT.

__ d. **IF AT ANY TIME** containment pressure exceeds 3 PSIG, **THEN RETURN TO** Step 4 in this enclosure.

JPM F

Catawba Nuclear Station
JPM F
2021 NRC Exam

EVALUATION SHEET

Task: Manually align Essential power to 1FTB from 1ETB per AP/1/A/5500/007 Case 1 Step 10

Alternate Path: No

Facility JPM #: EP-018

Safety Function: 6 **Title:** A.C. Electrical Distribution

K/A 062A4.01 Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard)

Rating(s): 3.3 / 3.1 **CFR:** 41.7 / 45.5 to 45.8

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator __X__ In-**P**lant ________ **P**erform __X__ **S**imulate ________

References: AP/1/A/5500/007 Case I

Task Standard: Applicant manually energizes 1FTB from 1ETB.

Validation Time: 10 minutes **Time Critical:** Yes _____ No __X__

===

Applicant: Time Start: __________
NAME____________________________ Docket #_____________ Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

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COMMENTS

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Catawba Nuclear Station
JPM F
2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. **ENSURE NRC Examination Security has been established.**
2. **Reset to IC # 146.**
3. **Enter the password.**
4. **Select "Yes" on the INITIAL CONDITION RESET pop-up window.**
5. **Ensure simulator setup per table below.**
6. **Place simulator in RUN and acknowledge any alarms.**
7. **ENSURE "Extra Operator" is present in the simulator.**
8. **Place simulator in FREEZE until Examiner cue is given.**

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event
	LOA-EP095 (600VLXH BKR LXH-4B)	OPEN				1
	LOA-EP095 (600V LC LXH BKR LXH-4B)	CLOSE				2
	MAL-EQB001B (D/G 1B LOAD SEQUENCER FAILURE)	R1				

Catawba Nuclear Station
JPM F
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is operating at 100% power.**

INITIATING CUES:

- **Unit 1 has experienced a B Train blackout due to a failure of 1ATD. The 1B D/G is supplying 1ETB. The crew has implemented AP/1/A/5500/07 Case 1 (Loss of Normal Power to an Essential Train). The CRS has directed you to perform step 10.**

EXAMINER NOTE: After reading Initiating Cue, provide the applicant with a copy of AP/1/A/5500/007 Case 1 step 10.

Catawba Nuclear Station
JPM F
2021 NRC Exam

START TIME: _______

STEP/STANDARD	SAT/UNSAT
STEP 1: 10. **Verify B/O busses are energized as follows:** a. 1AD-11, K/3 "4KV B/O BUS FTA VOLTAGE LO" - DARK. STANDARD: Applicant will verify 1AD-11, K/3 is DARK COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 2 10.b 1AD-11, K/4 "4KV B/O BUS FTB VOLTAGE LO" - DARK. STANDARD: Applicant will note that 1AD-11, K/4 is LIT and will transition to the RNO column. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 3 10.b.RNO b. Perform the following: 1) Ensure breaker "FTB B/O NORM FDR FRM ATD" - OPEN. STANDARD: Applicant will verify breaker is open COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 4 10.b. RNO b. 2) Dispatch operator to open 1LXH-4B (Incoming Breaker Fed From Xfmr TXH) (SB-594, U-30). STANDARD: Applicant will contact the booth and dispatch an operator to open 1LXH-4B **Booth Operator: Insert Event 1 to open 1LXH-4B.** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 5 10.b. RNO b. 3) **IF** S/I has actuated, **THEN** ensure "ECCS TRN B" reset. STANDARD: Applicant will determine that this step does not apply. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 6 10.b RNO b. 4) Reset "D/G 1B LOAD SEQ RESET". STANDARD: Applicant will depress the “D/G 1B LOAD SEQ REST” pushbutton **Examiner Note: This step is critical to gain manual control of the breakers in subsequent steps.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 7 10.b. RNO b. 5) **WHEN** notified by dispatched operator that 1LXH-4B is open, **THEN** perform the following: a) Close breaker "FTB B/O ALT FDR FRM ETB". STANDARD: Once notified, applicant will close breaker **Booth Operator: Contact Control Room and inform operator that 1LXH-4B is open.** **Examiner Note: This step is critical to energize 1FTB from 1ETB.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 8 10.b. RNO b.5) b) Close breaker "ETB ALT FDR TO FTB". STANDARD: Applicant will close the breaker. **Examiner Note: This step is critical to energize 1FTB from 1ETB.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 9 10.b. RNO b.5) c) Notify dispatched operator to close 1LXH-4B (Incoming Breaker Fed From Xfmr 1TXH) (SB-594, U-30). STANDARD: Applicant will contact the booth and notify dispatched operator to close 1LXH-4B. **Booth Operator : Insert Event 2 to close 1LXH-4B.** **Examiner Cue: Following breaker closure, "Another operator will continue steps of this procedure. JPM complete."** COMMENTS: **END OF TASK**	___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is operating at 100% power.**

INITIATING CUES:

- **Unit 1 has experienced a B Train blackout due to a failure of 1ATD. The 1B D/G is supplying 1ETB. The crew has implemented AP/1/A/5500/07 Case 1 (Loss of Normal Power to an Essential Train). The CRS has directed you to perform step 10.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Verify B/O busses energized as follows:**	
__ a. 1AD-11, K/3 "4KV B/O BUS FTA VOLTAGE LO" - DARK.	a. Perform the following:
	NOTE Both ND Hx Bypass valves fail closed on loss of 1LXI (1FTA).
	1) **IF** ND pump 1A operating in Residual Heat Removal Mode, **THEN** perform the following:
	__ a) Place "PWR DISCON FOR 1NI173A" in "THROT".
	__ b) THROTTLE 1NI-173A (ND Hdr 1A To Cold Legs C&D) to stabilize NC temperature.
	2) **IF** ND pump 1B operating in Residual Heat Removal Mode, **THEN** perform the following:
	__ a) Place "PWR DISCON FOR 1NI178B" in "THROT".
	__ b) THROTTLE 1NI-178B (ND Hdr 1B To Cold Legs A&B) to stabilize NC temperature.
	__ 3) Ensure breaker "FTA B/O NORM FDR FRM ATC" - OPEN.
	__ 4) Dispatch operator to open 1LXI-4B (Incoming Breaker Fed From Xfmr 1TXI) (SB-594, U-V, 29-30).
	__ 5) **IF** S/I actuated, **THEN** ensure "ECCS TRN A" reset.
	__ 6) Reset "D/G 1A LOAD SEQ RESET".

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. (Continued)	
	7) **WHEN** notified by dispatched operator 1LXI-4B open, **THEN** perform the following: __ a) CLOSE breaker "FTA B/O ALT FDR FRM ETA". __ b) CLOSE breaker "ETA ALT FDR TO FTA". __ c) Notify dispatched operator to close 1LXI-4B (Incoming Breaker Fed From Xfmr 1TXI) (SB-594, U-V, 29-30).
__ b. 1AD-11, K/4 "4KV B/O BUS FTB VOLTAGE LO" - DARK.	b. Perform the following: __ 1) Ensure breaker "FTB B/O NORM FDR FRM ATD" - OPEN. __ 2) Dispatch operator to open 1LXH-4B (Incoming Breaker Fed From Xfmr 1TXH) (SB-594, U-30). __ 3) **IF** S/I actuated, **THEN** ensure "ECCS TRN B" reset. __ 4) Reset "D/G 1B LOAD SEQ RESET". 5) **WHEN** notified by dispatched operator 1LXH-4B open, **THEN** perform the following: __ a) CLOSE breaker "FTB B/O ALT FDR FRM ETB". __ b) CLOSE breaker "ETB ALT FDR TO FTB". __ c) Notify dispatched operator to close 1LXH-4B (Incoming Breaker Fed From Xfmr 1TXH) (SB-594, U-30).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **Verify B/O loads in service as follows:**	
__ a. Maintain D/G load less than 5750 KW.	
b. Ensure proper B/O sequencer(s) loading as follows:	
__ • **REFER TO** Enclosure 2 (Blackout Loads)	
__ • Dispatch operator to ensure all required in plant loads energized **OR** on. **REFER TO** Enclosure 3 (Local Blackout Loads).	
__ c. Ensure Spent Fuel Pool cooling established. **REFER TO** AP/1/A/5500/041 (Loss of Spent Fuel Cooling or Level).	
__ 12. **Verify VI pressure - GREATER THAN 85 PSIG AND STABLE.**	__ **REFER TO AP/0/A/5500/022 (Loss of Instrument Air).**

JPM G

Catawba Nuclear Station
JPM G
2021 NRC Exam
EVALUATION SHEET

Task: Reset Radiation Monitor Trip Setpoints

Alternate Path: No

Facility JPM #: WE-EMF-001

Safety Function: 7 **Title:** Process Radiation Monitoring System

K/A 073 A4.02 Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel

Rating(s): 3.7 / 3.7 **CFR:** 41.7 / 45.5 to 45.8

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator __X__ In-**Pl**ant ________ **P**erform __X__ **S**imulate __________

References: OP/0/A/6500/080 (EMF RP86A Output Modules) rev. 18, Enclosure 4.2 (EMF RP86A Trip Setpoint Adjustment)

Task Standard: EMF50L TRIP 1 setpoint set to 6.3 E3 CPM and the TRIP 2 setpoint set to 9.0 E3 CPM per OP/0/A/6500/080 (EMF RP86A Output Modules) Enclosure 4.2 (EMF RP86A Trip Setpoint Adjustment).

Validation Time: 10 minutes **Time Critical:** Yes _____ No __X__

==

Applicant: Time Start: __________
NAME___________________________ Docket #_______________ Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

Catawba Nuclear Station
JPM G
2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. **ENSURE NRC Examination Security has been established.**
2. **Reset to IC # 147.**
3. **Enter the password.**
4. **Select “Yes” on the INITIAL CONDITION RESET pop-up window.**
5. **Ensure simulator setup per table below.**
6. **Place simulator in RUN and acknowledge any alarms.**
7. **ENSURE “Extra Operator” is present in the simulator.**
8. **Ensure copy of EMF-50 setpoint log page has been replaced.**
9. **Place simulator in FREEZE until Examiner cue is given.**

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event

Catawba Nuclear Station
JPM G
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is at 100% power.**

INITIATING CUES:

Following a discussion with Glenn from RP concerning a premature gaseous release termination, the Control Room Supervisor directs you to set EMF 50L setpoints using OP/0/A/6500/080 (EMF RP86A Output Modules) Enclosure 4.2 (EMF RP86A and RM1000 Trip Setpoint Adjustment) to the following values:

- **Trip 1 = 6300 CPM**
- **Trip 2 = 9000 CPM**

OAC Program EMFLIB is currently not available.

EXAMINER NOTE: After reading the cue, provide the applicant with a copy of OP/0/A/6500/080 (EMF RP86A Output Modules) Enclosure 4.2 (EMF RP86A and RM1000 Trip Setpoint Adjustment).

STEP/STANDARD	SAT/UNSAT

START TIME: _______

NOTE: 1. If desired, EMF setpoints adjustments may be performed from the "EMF SETPOINT" screen of OAC EMF Library (EMFLIB) Application. EMFLIB is user friendly, no procedure instructions are provided for this application.
2. The Trip Lamps can only be cleared if the activity level has decreased below the Trip Setpoint.
3. The setpoints given on release permits are already rounded to 3 significant digits and are entered into the EMF as is. Setpoints for non-release conditions are rounded up or down to 3 significant digits using standard mathematical rules for rounding.

STEP 1 3.1 **IF** necessary, press the clear key [CLR] to reset trip lamps STANDARD: Applicant verifies trip lamps dark or depresses the [CLR] key to clear alarms. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 2 3.2 Press the function key [FUN] to bring up the "SELECT FUNCTION" screen. STANDARD: Applicant depresses the [FUN] key to bring up the "SELECT FUNCTION" screen. **This step is critical, due to being the only way to get to the select function screen, which is required to input new Trip 1 and Trip 2 values.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 3 3.3 Adjust Trip 1 Setpoint as follows: 3.3.1 Press [1] for Trip 1 setting display screen. STANDARD: Applicant depresses the [1] to bring up the Trip 1 setting display screen. **This step is critical to bring up the screen that the new Trip 1 setting will be input into.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 4 3.3.2 Key in the desired Trip 1 setpoint. STANDARD: Applicant keys in 6300 CPM using the numeric keypad. **This step is critical to input the new Trip 1 setpoint required to meet the task standard. This new setpoint is checked in the next step. The critical requirement is for the new trip setpoint of 6300 CPM to be entered by the end of this JPM.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 5 3.3.3 Ensure the following: 3.3.3.1 Setpoint is correctly displayed in the "ENTER" block on the setpoint display screen. 3.3.3.2 Setpoint is greater than the current EMF reading. STANDARD: Applicant ensures 6300 CPM is displayed in the ENTER block and that the setpoint is greater than the current reading on the EMF. COMMENTS:	___ **SAT** ___ **UNSAT**
NOTE: Once the enter key [ENT] is pressed, the change in Trip 1 alarm setpoint is active.	
STEP 6 3.3.4 Press the enter key [ENT]. This value is now displayed under "TRIP 1" and the "ENTER" block is cleared. STANDARD: Applicant presses the [ENT] key and verifies the correct value under the "Trip 1" on the display. **This step is critical to input the new Trip 1 setpoint. Again the critical requirement is to have the Trip 1 setpoint set to 6300 CPM by the end of the JPM.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 7 3.3.5 Press the clear key [CLR] to return to the "SELECT FUNCTION" screen. STANDARD: Applicant presses the [CLR] key to return to the "SELECT FUNCTION" screen. COMMENTS:	___ SAT ___ UNSAT
STEP 8 3.4 Adjust Trip 2 Setpoint as follows: 3.4.1 Press [2] for Trip 2 setting display screen. STANDARD: Applicant presses [2] to bring up the Trip 2 setting display screen. **This step is critical to get to the required screen to input the new Trip 2 setpoint.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT
STEP 9 3.4.2 Key in the desired Trip 2 setpoint. STANDARD: Applicant enters 9000 CPM using the numeric keypad. **This step is critical to input the new Trip 2 setpoint required to meet the task standard. This new setpoint is checked in the next step. The critical requirement is for the new trip setpoint of 9000 CPM to be entered by the end of this JPM.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 10 3.4.3 Ensure the following: 3.4.3.1 Setpoint is correctly displayed in the "ENTER" block on the setpoint display screen. 3.4.3.2 Setpoint is greater than the current EMF reading. STANDARD: Applicant ensures 9000 CPM is displayed in the ENTER block and that the entered setpoint is greater than the current reading on the EMF. COMMENTS:	___ **SAT** ___ **UNSAT**
NOTE: Once the enter key [ENT] is pressed, the change in Trip 2 alarm setpoint is active.	
STEP 11 3.4.4 Press the enter key [ENT]. This value is now displayed under "TRIP 2" and the "ENTER" block is cleared. STANDARD: Applicant presses the [ENT] key and verifies the correct value under the "Trip 2" on the display. **This step is critical to input the new Trip 2 setpoint. Again the critical requirement is to have the Trip 2 setpoint set to 9000 CPM by the end of the JPM.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 12 3.5 Press the clear key [CLR] twice to return to the normal display screen. STANDARD: Applicant presses the [CLR] key twice to return to the normal display screen. **EXAMINER CUE: "Another operator will complete the procedure. This JPM is complete."** COMMENTS: **END OF TASK**	___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is at 100% power.**

INITIATING CUES:

Following a discussion with Glenn from RP concerning a premature gaseous release termination, the Control Room Supervisor directs you to set EMF 50L setpoints using OP/0/A/6500/080 (EMF RP86A Output Modules) Enclosure 4.2 (EMF RP86A and RM1000 Trip Setpoint Adjustment) to the following values:

- **Trip 1 = 6300 CPM**
- **Trip 2 = 9000 CPM**

OAC Program EMFLIB is currently not available.

EMF RP86A and RM1000 Trip Setpoint Adjustment
Information Use

1. Limits and Precautions

1.1 The EMF RP86A and RM1000 green "OPERATE" light goes dark and the failure relay de-energizes under any of the following conditions:

- The operate/calibrate switch is set to calibrate
- Loss of high voltage
- Loss of signal (0 counts in 2 minutes)
- Safety loop open
- Loss of power

1.2 If an EMF RP86A and RM1000 Trip 1 is set higher than Trip 2, the entered value will be accepted.

2. Initial Conditions

Verify a need to adjust the EMF setpoints.

3. Procedure

NOTE:
1. If desired, EMF setpoints adjustments may be performed from the "EMF SETPOINT" screen of OAC EMF Library (EMFLIB) Application. EMFLIB is user friendly, no procedure instructions are provided for this application.
2. The Trip Lamps can only be cleared if the activity level has decreased below the Trip Setpoint.
3. The setpoints given on release permits are already rounded to 3 significant digits and are entered into the EMF as is. Setpoints for non-release conditions are rounded up or down to 3 significant digits using standard mathematical rules for rounding.

3.1 **IF** necessary, press the clear key [CLR] to reset trip lamps.

3.2 Press the function key [FUN] to bring up the "SELECT FUNCTION" screen.

3.3 Adjust Trip 1 Setpoint as follows:

3.3.1 Press [1] for Trip 1 setting display screen.

3.3.2 Key in the desired Trip 1 setpoint.

3.3.3 Ensure the following:

3.3.3.1 Setpoint is correctly displayed in the "ENTER" block on the setpoint display screen.

3.3.3.2 Setpoint is greater than the current EMF reading.

NOTE:	Once the enter key [ENT] is pressed, the change in Trip 1 alarm setpoint is active.

3.3.4 Press the enter key [ENT]. This value is now displayed under "TRIP 1" and the "ENTER" block is cleared.

3.3.5 Press the clear key [CLR] to return to the "SELECT FUNCTION" screen.

3.4 Adjust Trip 2 Setpoint as follows:

3.4.1 Press [2] for Trip 2 setting display screen.

3.4.2 Key in the desired Trip 2 setpoint.

3.4.3 Ensure the following:

3.4.3.1 Setpoint is correctly displayed in the "ENTER" block on the setpoint display screen.

3.4.3.2 Setpoint is greater than the current EMF reading.

NOTE:	Once the enter key [ENT] is pressed, the changes in Trip 2 alarm setpoint is active.

3.4.4 Press the enter key [ENT]. This value is now displayed under "TRIP 2" and the "ENTER" block is cleared.

3.5 Press the clear key [CLR] twice to return to the normal display screen.

3.6 Enter the new EMF setpoints on the Control Room EMF Setpoint Log.

3.7 Sign the Control Room EMF Setpoint Log in the appropriate box.

3.8 **IF** applicable, document the RP personnel that supplied the setpoints in the Control Room EMF Setpoint Log.

3.9 A separate Operator shall perform the following:

3.9.1 Verify that the correct setpoints are entered in the Control Room EMF Setpoint Log.

3.9.2 Sign the IV BY block on the Control Room EMF Setpoint Log.

Enclosure 4.6

Control Room EMF Setpoint Log

Use Enclosure 4.7 for EMF35 or EMF38 and Enclosure 4.8 for EMF-71, EMF-72, EMF-73, or EMF-74.

Unit # 0 EMF # 50L

DATE/TIME	**Present Reading** Circle one: cpm,mrem/hr,rem/hr,gpd	**TRIP 2**	**TRIP 1**	PERFORMED BY (SIGNATURE)	IV BY (SIGNATURE)	COMMENTS (or RP name):

IV = Independent Verification

Do **NOT** File this enclosure in the Control Copy folder of this procedure.

Catawba Nuclear Station
JPM H
2021 NRC Exam

JPM H

Catawba Nuclear Station
JPM H
2021 NRC Exam

EVALUATION SHEET

Task: Place KC in Parallel Operation per OP/1/A/6400/005

Alternate Path: No

Facility JPM #: KC-085

Safety Function: 8 **Title:** Plant Service Systems

K/A 008 A4.01 Ability to manually operate and/or monitor in the control room: CCW indications and controls

Rating(s): 3.3 / 3.1 **CFR:** 41.7 / 45.5

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator __X__ In-**P**lant ________ **P**erform __X__ **S**imulate ________

References: OP/1/A/6400/005 (Component Cooling System) Rev. 132 Enclosure 4.4 (Operation of Additional KC Pumps/Parallel Operation)

Task Standard: KC Pump 1A1 is started and 1A KC HX OTLT MODE switch is placed in "KC TEMP" position to place KC in parallel operation.

Validation Time: 10 minutes **Time Critical:** Yes _____ No __X__

==

Applicant: Time Start: __________
NAME____________________________ Docket #_______________ Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

Catawba Nuclear Station
JPM H
2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. **ENSURE NRC Examination Security has been established.**
2. **Reset to IC # 148**
3. **Ensure 1B2 KC pump in operation**
4. **Place simulator in RUN and acknowledge any alarms.**
5. **ENSURE “Extra Operator” is present in the simulator.**
6. **Place simulator in FREEZE until Examiner cue is given.**

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event

Catawba Nuclear Station
JPM H
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. **Unit 1 is at 100% power.**
2. **A work list item has been generated to place Unit 1 KC in parallel operation in preparation for Aux Safeguards Testing early next shift.**

INITIATING CUES:

1. **The CRS instructs you to place KC in parallel operation by performing Encl. 4.4 of OP/1/A/6400/005. You are to start 1A1 KC pump. Pre-start pump checkout has been successfully completed. Initial conditions have previously been verified and signed off. You are to begin at step 3.4.**
2. **CV is waived for this JPM.**

Examiner Note: After reading cue, provide the applicant with a copy of OP/1/A/6400/005 Encl. 4.4.

Catawba Nuclear Station
JPM H
2021 NRC Exam

START TIME: _______

STEP/STANDARD	SAT/UNSAT
STEP 1: 3.4 **IF** placing KC train 1A in parallel operation with KC Train 1B with the trains cross connected, complete the following steps: 3.4.1 Complete the following steps to ensure the RN system has miniflow protection: 3.4.1.1 **IF** a Unit 2 KC Hx discharge valve is in the "MINIFLOW" position, perform the following: A. Ensure the associated inlet valve is open: • 2RN-287A (KC Hx 2A Inlet Isol) • 2RN-347B (KC Hx 2B Inlet Isol) B. Ensure a complete RN flow path exists from the RN pumps through the applicable Hx to the discharge. STANDARD: Applicant asks Unit 2 operator to verify which KC heat exchanger is in service and whether its respective heat exchanger inlet valve is open. Applicant determines RN system is in normal lineup and has a flowpath through 2B KC heat exchanger to the discharge by looking at the OAC graphic. **Examiner Cue: "2B KC heat exchanger is in the "MINIFLOW" position and 2RN-347B is open."** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 2: 3.1.4.2 **IF** no Unit 2 KC Hxs are available for RN miniflow, establish miniflow per OP/0/A/6400/006 C (Nuclear Service Water System) as necessary to maintain RN flow ≥ 8600 GPM per operating RN pump. STANDARD: Applicant determines this step is N/A. 2B KC heat exchanger is available for miniflow. COMMENTS:	**___ SAT** **___UNSAT**

STEP 3: 3.4.2 Ensure 1RN-287A (KC Hx 1A Inlet Isol) is open. STANDARD: Applicant determines that the red OPEN light is lit and green CLSD light is dark on 1RN-287A. COMMENTS:	**___ SAT** **___ UNSAT**

STEP 4: 3.4.3 Ensure “KC HX A OTLT MODE” is in “KC TEMP”. STANDARD: Applicant rotates “KC HX A OTLT MODE” switch counter clockwise to the “KC TEMP” position. **Examiner Note: This step is critical in order to maintain consistent temperature in the 1A KC train and prevent reactivity excursions due to flowrate changes (and associated letdown temperature changes) following the start of 1A1 KC Pump.** COMMENTS:	**CRITICAL STEP** **___ SAT** **___ UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 5: 3.4.4 IF letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System) perform the following: 3.4.4.1 Verify the Cation Bed Demineralizer is NOT in service per OP/1/A/6200/001 (Chemical and Volume Control System). STANDARD: Applicant requests information concerning status of the Cation Bed Demineralizer. **Examiner Cue: "The Cation Bed Demineralizer is not in service."** COMMENTS:	___ SAT ___ UNSAT
STEP 6: 3.4.4.2 Record position of 1NV-153A (Letdn Hx Otlt 3-Way Vlv) Recorded valve position ________________ STANDARD: Applicant records position of 1NV-153A to be the AUTO/DEMIN position. COMMENTS:	___ SAT ___ UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 7: 3.4.4.3 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers will be bypassed while shifting KC pumps. Person notified ____________________ STANDARD: Applicant contacts Primary Chemistry and informs them that the demineralizers will be bypassed. **Examiner Cue: "This is Steve in Primary Chemistry. I understand that the letdown demineralizers will be bypassed while shifting KC pumps."** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 8: 3.4.4.4 IF letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers will be bypassed while shifting KC Pumps. Person notified ______________________ STANDARD: Applicant contacts Radiation Protection and informs them that the demineralizers will be bypassed. **Examiner Cue: "This is Gary in Radiation Protection. I understand that the letdown demineralizers will be bypassed while shifting KC pumps."** COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 9: 3.4.4.5 Place 1NV-153A (Letdn Hx Otlt 3-Way Vlv) in the "VCT" position. STANDARD: Applicant rotates switch for 1NV-153A counter clockwise to the VCT position and verifies the white light lit and red light dark. COMMENTS:	___ **SAT** ___ **UNSAT**

CAUTION: 5700 GPM discharge header flow per operating KC Pump shall NOT be exceeded

STEP/STANDARD	SAT/UNSAT
STEP 10: 3.4.5 Start either KC Train 1A pump: • "KC PUMP A1" • OR • "KC PUMP A2" STANDARD: Per initiating cue, applicant will start 1A1 KC pump by depressing the red ON pushbutton and verifying the red ON light lit and green OFF light dark. **Examiner Note: This step is critical because on 'A' train KC pump has to be started to place KC in parallel operation per the JPM standard.** **Examiner Cue: Following pump start, "1A1 KC pump post start check is complete – Good for continued operation."** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 11: 3.4.6 **IF** KC flow requirement in the Train 1A header is > 5700 gpm, perform the following: STANDARD: Applicant determines this step is not applicable. COMMENTS:	___ **SAT** ___ **UNSAT**

NOTE: One pump running is preferred as long as flow is < 5700 gpm.

STEP/STANDARD	SAT/UNSAT

CAUTION:

- **The time two KC pumps in a train are operating with header flow less than 5700 gpm shall be minimized.**
- **Operating two KC pumps in a train with header flow less than 4850 gpm has the potential for a "strong pump" versus "weak pump" interaction, which can impact minimum flow capacity.**

STEP 12: 3.4.7 **IF** KC flow requirement in the 1A header is < 5700 gpm **AND** it is desirable to place the second Train 1A pump in service, perform the following: STANDARD: Applicant determines this step is not applicable. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 13: 3.4.8 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System), **WHEN** KC flow and temperature have stabilized, perform the following: 3.4.8.1 **IF** 1NV-153A (Letdn Hx Otlt 3-Way Vlv) position was recorded as "DEMIN" in Step 3.7.4.2 **AND** no other reason exists for it to remain in the "VCT" position, return it to "AUTO" as follows: A. Place 1NV-153A in the "DEMIN" position. (RM) STANDARD: Applicant will rotate switch for 1NV-153A clockwise to the DEMIN position, verifying the red light lit and white light dark, and then return the switch to the AUTO position. **Examiner Cue: "Using time compression, KC flow and temperature have stabilized."** COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 14: 3.4.8.1.B Verify 1NV-153A returns to "AUTO". STANDARD: Applicant verifies switch position in "AUTO". COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 15: 3.4.8.2 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers have been restored to service. Person notified ____________________ STANDARD: Applicant contacts Primary Chemistry and lets them know that the demineralizers have been returned to service. **Examiner Cue: "This is Steve with Primary Chemistry. I understand that the letdown demineralizers have been returned to service."** COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 16: 3.4.8.3 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers have been restored to service. Person notified ____________________ STANDARD: Applicant contacts Radiation Protection and lets them know that the demineralizers have been returned to service. **Examiner Cue: "This is Gary with Radiation Protection. I understand that the letdown demineralizers have been returned to service."** **Examiner Cue: "Another operator will finish the procedure. This JPM is complete."** COMMENTS: **END OF TASK**	___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

1. **Unit 1 is at 100% power.**
2. **A work list item has been generated to place Unit 1 KC in parallel operation in preparation for Aux Safeguards Testing early next shift.**

INITIATING CUES:

1. **The CRS instructs you to place KC in parallel operation by performing Encl. 4.4 of OP/1/A/6400/005. You are to start 1A1 KC pump. Pre-start pump checkout has been successfully completed. Initial conditions have previously been verified and signed off. You are to begin at step 3.4.**
2. **CV is waived for this JPM.**

1. Limits and Precautions

1.1 This procedure is Reactivity Management related because it controls activities that can affect core reactivity by changing letdown temperature. (R.M.)

1.2 The maximum discharge header flow for each operating KC pump is 5700 gpm.

1.3 When manually operating any motor operated valve, minimize the torque applied to the handwheel.

1.4 After manual operation, maintenance or packing adjustment of any safety related motor operated valve, it shall be cycled electrically to ensure reliable automatic operation.

1.5 KC pump minimum flow is 1100 gpm.

1.6 While running just one KC Pump, making KC flow adjustments in a gradual manner will minimize total KC System flow fluctuations. (PIP 96-1894)

1.7 The time two KC pumps in a train are operating with header flow less than 5700 gpm shall be minimized.

1.8 Operating two KC pumps in a train with header flow less than 4850 gpm has the potential for a "strong pump" versus "weak pump" interaction, which can impact minimum flow capacity.

2. Initial Conditions

AA 2.1 Verify one train of KC is operating per Enclosure 4.1 (System Startup).

AA 2.2 Notify Radwaste of the intent to change the current KC system pump lineup.

AA 2.3 **IF** in Mode 1 or 2, ensure R3 reactivity management controls established per AD-OP-ALL-0203 (Reactivity Management). (R.M.)

3. Procedure

NOTE: Steps 3.1 through 3.9 are independent of each other and may be performed in any order.

N/A 3.1 **IF** an operating KC Train has a single pump in operation with the idle pump in that train available **AND** it is desired to start the additional pump in the train, **THEN** perform one of the following:

_____ 3.1.1 **IF** KC flow requirements on the train are > 5700 gpm, perform the following:

_____ 3.1.1.1 Ensure the appropriate miniflow valve is closed:

_____ _____ • 1KC-C37A (Train A Miniflow Isol)

OR

_____ _____ • 1KC-C40B (Train B Miniflow Isol)

3.1.1.2 Start the idle pump in the train:

_____ • "KC PUMP A1"

OR

_____ • "KC PUMP A2"

OR

_____ • "KC PUMP B1"

OR

_____ • "KC PUMP B2"

NOTE:	One pump running is preferred as long as flow is < 5700 gpm.

CAUTION:
- The time two KC pumps in a train are operating with header flow less than 5700 gpm shall be minimized.
- Operating two KC pumps in a train with header flow less than 4850 gpm has the potential for a "strong pump" versus "weak pump" interaction, which can impact minimum flow capacity.

_____ 3.1.2 **IF** KC flow requirement in the train is < 5700 gpm **AND** it is desired to start the additional pump in the train, perform the following:

3.1.2.1 Start the idle pump in the train:

_____ • "KC PUMP A1"
OR
_____ • "KC PUMP A2"

OR

_____ • "KC PUMP B1"
OR
_____ • "KC PUMP B2"

_____ 3.1.2.2 Ensure minimum flow requirements are met.

NOTE: One pump running is preferred as long as flow is < 5700 gpm.

N/A 3.2 **IF** both KC pumps in an operating train are running **AND** flow requirement in that operating train is < 5700 gpm, perform the following:

_____ 3.2.1 **IF** required, throttle KC flow to the inservice KF heat exchanger as necessary to prevent KC pump runout:

_____ • 1KC-149 (KF Hx 1A Cool Wtr Otlt)

_____ • 1KC-156 (KF Hx 1B Cool Wtr Otlt)

_____ 3.2.2 **IF AT ANY TIME** KC Train flow approaches 5700 gpm while performing the next step, ensure the appropriate miniflow valve is closed:

_____ _____ • 1KC-C37A (Train A Miniflow Isol)
OR
_____ _____ • 1KC-C40B (Train B Miniflow Isol)

3.2.3 Stop one of the operating pumps:

_____ • "KC PUMP A1"
OR
_____ • "KC PUMP A2"
OR
_____ • "KC PUMP B1"
OR
_____ • "KC PUMP B2"

3.2.4 Perform the following for the KF cooling loops that are in service:

_____ 3.2.4.1 **IF** 1A KF Cooling Loop is in service, adjust 1KC-149 (KF Hx 1A Cool Wtr Otlt) flow controller on 1MC11 to 3000 gpm or as necessary to maintain Spent Fuel Pool temperature < 125°F.

_____ 3.2.4.2 **IF** 1B KF Cooling Loop is in service, adjust 1KC-156 (KF Hx 1B Cool Wtr Otlt) flow controller on 1MC11 to 3000 gpm or as necessary to maintain Spent Fuel Pool temperature < 125°F.

AA 3.3 **IF** additional KC flow is needed **AND** both pumps in the operating loop are running **OR** KC is in single pump operation with the idle pump in the operating train **NOT** available, place KC in parallel operation per Step 3.4, 3.5, 3.7 or 3.8 as applicable.

_____ 3.4 **IF** placing KC Train 1A in parallel operation with KC Train 1B with the trains cross-connected, complete the following steps:

NOTE: RN System minimum flow protection is normally established using an idle KC train.

3.4.1 Complete the following steps to ensure the RN System has miniflow protection:

_____ 3.4.1.1 **IF** a Unit 2 KC Hx discharge valve is in the "MINIFLOW" position, perform the following:

A. Ensure the associated inlet valve is open:

_____ _____ • 2RN-287A (KC Hx 2A Inlet Isol)

_____ _____ • 2RN-347B (KC Hx 2B Inlet Isol)

_____ B. Ensure a complete RN flow path exists from the RN Pumps through the applicable Hx to the discharge.

_____ 3.4.1.2 **IF** no Unit 2 KC Hxs are available for RN miniflow, establish miniflow per OP/0/A/6400/006 C (Nuclear Service Water System) as necessary to maintain RN flow ≥ 8600 gpm per operating RN Pump.

_____ _____ 3.4.2 Ensure 1RN-287A (KC Hx 1A Inlet Isol) is open.

_____ 3.4.3 Ensure "KC HX 1A OTLT MODE" is in "KC TEMP".

_____ 3.4.4 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System) perform the following: (R.M.)

_____ 3.4.4.1 Verify the Cation Bed Demineralizer is **NOT** in service per OP/1/A/6200/001 (Chemical and Volume Control System).

_____ 3.4.4.2 Record position of 1NV-153A (Letdn Hx Otlt 3-Way Vlv).
Recorded valve position_________________

_____ 3.4.4.3 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers will be bypassed while shifting KC Pumps.
Person notified ______________________________

_____ 3.4.4.4 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers will be bypassed while shifting KC Pumps.
Person notified ______________________________

_____ 3.4.4.5 Place 1NV-153A (Letdn Hx Otlt 3-Way Vlv) in the "VCT" position.

CAUTION: 5700 gpm discharge header flow per each operating KC pump shall **NOT** be exceeded.

3.4.5 Start either KC Train 1A pump:

- _____ "KC PUMP A1"
  OR
- _____ "KC PUMP A2"

_____ 3.4.6 **IF** KC flow requirement in the Train 1A header is > 5700 gpm, perform the following:

_____ _____ 3.4.6.1 Ensure 1KC-C37A (Train A Miniflow Isol) is closed.

3.4.6.2 Start the remaining KC Train 1A pump:

- _____ "KC PUMP A1"
  OR
- _____ "KC PUMP A2"

NOTE: One pump running is preferred as long as flow is < 5700 gpm.

CAUTION:
- The time two KC pumps in a train are operating with header flow less than 5700 gpm shall be minimized.
- Operating two KC pumps in a train with header flow less than 4850 gpm has the potential for a "strong pump" versus "weak pump" interaction, which can impact minimum flow capacity.

_____ 3.4.7 **IF** KC flow requirement in the Train 1A header is < 5700 gpm **AND** it is desired to place the second Train 1A pump in service, perform the following:

3.4.7.1 Start the remaining KC Train 1A pump:

- _____ "KC PUMP A1"
  OR
- _____ "KC PUMP A2"

_____ 3.4.7.2 Ensure minimum flow requirements are met.

_____ 3.4.8 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System), **WHEN** KC flow and temperature have stabilized perform the following: (R.M.)

_____ 3.4.8.1 **IF** 1NV-153A (Letdn Hx Otlt 3-Way Vlv) position was recorded as "DEMIN" in Step 3.4.4.2 **AND** no other reason exists for it to remain in the "VCT" position, return it to "AUTO" as follows:

_____ A. Place 1NV-153A in the "DEMIN" position. (R.M.)

☐ B. Verify 1NV-153A returns to "AUTO".

_____ 3.4.8.2 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers have been restored to service.
Person notified ______________________________

_____ 3.4.8.3 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers have been restored to service.
Person notified _____________________________

NOTE: At this point, KC Train 1A and 1B are in parallel service.

_____ 3.4.9 **IF** RN miniflow was established per Step 3.4.1.2, **WHEN** no longer needed, secure unneeded flow paths.

_____ 3.4.10 **IF** Train 1A is to be secured, leaving Train 1B in service as per Enclosure 4.1 (System Startup), go to Step 3.6.

_____ 3.5 **IF** placing KC Train 1A in parallel operation with KC Train 1B with the trains **NOT** cross-connected, complete the following steps:

NOTE: RN System minimum flow protection is normally established using an idle KC train.

3.5.1 Complete the following steps to ensure the RN System has miniflow protection:

_____ 3.5.1.1 **IF** a Unit 2 KC Hx discharge valve is in the "MINIFLOW" position, perform the following:

A. Ensure the associated inlet valve is open:

_____ _____ • 2RN-287A (KC Hx 2A Inlet Isol)

_____ _____ • 2RN-347B (KC Hx 2B Inlet Isol)

_____ B. Ensure a complete RN flow path exists from the RN Pumps through the applicable Hx to the discharge.

_____ 3.5.1.2 **IF** no Unit 2 KC Hxs are available for RN miniflow, establish miniflow per OP/0/A/6400/006 C (Nuclear Service Water System) as necessary to maintain RN flow ≥ 8600 gpm per operating RN Pump.

_____ _____ 3.5.2 Ensure 1RN-287A (KC Hx 1A Inlet Isol) is open.

_____ 3.5.3 Ensure "KC HX 1A OTLT MODE" is in "KC TEMP".

_____ _____ 3.5.4 Ensure 1KC-56A (KC To ND Hx 1A Sup Isol) is closed.

CAUTION: 5700 gpm discharge header flow per each operating KC pump shall **NOT** be exceeded.

3.5.5 Start either KC Train 1A pump:

_____ • "KC PUMP A1"
OR
_____ • "KC PUMP A2"

_____ _____ 3.5.6 Ensure 1KC-C37A (Train A Miniflow Isol) opens.

_____ 3.5.7 **IF** KC flow requirement in the Train 1A header is > 5700 gpm, perform the following:

_____ _____ 3.5.7.1 Ensure 1KC-C37A (Train A Miniflow Isol) is closed.

_____ 3.5.7.2 **IF** Train 1A header flow is > 5700 gpm, start the remaining KC Train 1A pump:

- _____ "KC PUMP A1"
  OR
- _____ "KC PUMP A2"

NOTE: One pump running is preferred as long as flow is < 5700 gpm.

CAUTION:
- The time two KC pumps in a train are operating with header flow less than 5700 gpm shall be minimized.
- Operating two KC pumps in a train with header flow less than 4850 gpm has the potential for a "strong pump" versus "weak pump" interaction, which can impact minimum flow capacity.

_____ 3.5.8 **IF** KC flow requirement in the Train 1A header is < 5700 gpm **AND** it is desired to place the second Train 1A pump in service, perform the following:

3.5.8.1 Start the remaining KC Train 1A pump:

- _____ "KC PUMP A1"
  OR
- _____ "KC PUMP A2"

_____ 3.5.8.2 Ensure minimum flow requirements are met.

_____ 3.5.9 **IF** RN miniflow was established per Step 3.5.1.2, **WHEN** no longer needed, secure unneeded flow paths.

NOTE: At this point, KC Train 1A and 1B are in parallel service with KC Train 1A isolated from the Aux and Rx Bldg Non-Ess Headers.

_____ 3.6 **IF** Train 1A is to be secured, leaving Train 1B in service as per Enclosure 4.1 (System Startup), complete the following steps:

_____ 3.6.1 Notify Radwaste of the intent to change the current KC system pump lineup.

_____ 3.6.2 **IF** KC Trains 1A and 1B are **NOT** cross-connected, ensure that any component required to support unit operation is **NOT** being cooled by KC Train 1A.

_____ 3.6.3 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System), perform the following: (R.M.)

_____ 3.6.3.1 Verify the Cation Bed Demineralizer is **NOT** in service per OP/1/A/6200/001 (Chemical and Volume Control System).

_____ 3.6.3.2 Record position of 1NV-153A (Letdn Hx Otlt 3-Way Vlv).
Recorded valve position _______________

_____ 3.6.3.3 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers will be bypassed while shifting KC Trains.
Person notified ____________________________

_____ 3.6.3.4 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers will be bypassed while shifting KC Trains.
Person notified ____________________________

_____ 3.6.3.5 Place 1NV-153A (Letdn Hx Otlt 3-Way Vlv) in the "VCT" position.

3.6.4 Adjust the following flow controllers on 1MC11 to zero gpm flow:

_____ • 1KC-149 (KF Hx 1A Cool Wtr Otlt)

_____ • 1KC-156 (KF Hx 1B Cool Wtr Otlt)

3.6.5 Stop all KC Train 1A pumps:

_____ • "KC PUMP A1"

_____ • "KC PUMP A2"

_____ 3.6.6 Place "KC HX 1A OTLT MODE" in "MINIFLOW".

3.6.7 Perform the following for the KF cooling loops that are in service:

_____ 3.6.7.1 **IF** 1A KF Cooling Loop is in service, adjust 1KC-149 (KF Hx 1A Cool Wtr Otlt) flow controller on 1MC11 to 3000 gpm or as necessary to maintain Spent Fuel Pool temperature < 125°F.

_____ 3.6.7.2 **IF** 1B KF Cooling Loop is in service, adjust 1KC-156 (KF Hx 1B Cool Wtr Otlt) flow controller on 1MC11 to 3000 gpm or as necessary to maintain Spent Fuel Pool temperature < 125°F.

_____ _____ 3.6.8 **IF AT ANY TIME** KC Train 1B flow approaches 5700 gpm while performing the next step, ensure 1KC-C40B (Train B Miniflow Isol) is closed.

NOTE:	One pump running is preferred as long as flow is < 5700 gpm.

_____ 3.6.9 **IF** KC flow requirements are < 5700 gpm **AND** both KC Train 1B pumps are running, stop either KC Train 1B pump:

_____ • "KC PUMP B1"
OR
_____ • "KC PUMP B2"

NOTE:	At this point, KC Train 1B is in service as per Enclosure 4.1 (System Startup).

_____ 3.6.10 **IF** RN flow has been established through components other than the Unit 2 KC Hx's for RN miniflow, secure unneeded flow paths.

_____ 3.6.11 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System), **WHEN** KC flow and temperature have stabilized, perform the following: (R.M.)

_____ 3.6.11.1 **IF** 1NV-153A (Letdn Hx Otlt 3-Way Vlv) position was recorded as "DEMIN" in Step 3.6.3.2 **AND** no other reason exists for it to remain in the "VCT" position, return it to "AUTO" as follows:

_____ A. Place 1NV-153A in the "DEMIN" position. (R.M.)

☐ B. Verify 1NV-153A returns to "AUTO".

_____ 3.6.11.2 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers have been restored to service.
Person notified ______________________________

_____ 3.6.11.3 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers have been restored to service.
Person notified ______________________________

_____ 3.7 **IF** placing KC Train 1B in parallel operation with KC Train 1A with the trains cross-connected, complete the following steps:

NOTE:	RN System minimum flow protection is normally established using an idle KC train.

3.7.1 Complete the following steps to ensure the RN System has miniflow protection:

_____ 3.7.1.1 **IF** a Unit 2 KC Hx discharge valve is in the "MINIFLOW" position, perform the following:

A. Ensure the associated inlet valve is open:

_____ _____ • 2RN-287A (KC Hx 2A Inlet Isol)

_____ _____ • 2RN-347B (KC Hx 2B Inlet Isol)

_____ B. Ensure a complete RN flow path exists from the RN Pumps through the applicable Hx to the discharge.

_____ 3.7.1.2 **IF** no Unit 2 KC Hxs are available for RN miniflow, establish miniflow per OP/0/A/6400/006 C (Nuclear Service Water System) as necessary to maintain RN flow ≥ 8600 gpm per operating RN Pump.

_____ _____ 3.7.2 Ensure 1RN-347B (KC Hx 1B Inlet Isol) is open.

_____ 3.7.3 Ensure "KC HX 1B OTLT MODE" is in "KC TEMP".

_____ 3.7.4 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System) perform the following: (R.M.)

_____ 3.7.4.1 Verify the Cation Bed Demineralizer is **NOT** in service per OP/1/A/6200/001 (Chemical and Volume Control System).

_____ 3.7.4.2 Record position of 1NV-153A (Letdn Hx Otlt 3-Way Vlv).
Recorded valve position_________________

_____ 3.7.4.3 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers will be bypassed while shifting KC Pumps.
Person notified ______________________________

_____ 3.7.4.4 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers will be bypassed while shifting KC Pumps.
Person notified ______________________________

_____ 3.7.4.5 Place 1NV-153A (Letdn Hx Otlt 3-Way Vlv) in the "VCT" position.

CAUTION: 5700 gpm discharge header flow per operating KC Pump shall **NOT** be exceeded.

3.7.5 Start either KC Train 1B pump:

_____ • "KC PUMP B1"
OR
_____ • "KC PUMP B2"

_____ 3.7.6 **IF** KC flow requirement in the Train 1B header is > 5700 gpm, perform the following:

_____ _____ 3.7.6.1 Ensure 1KC-C40B (Train B Miniflow Isol) is closed.

3.7.6.2 Start the remaining KC Train 1B pump:

_____ • "KC PUMP B1"
OR
_____ • "KC PUMP B2"

NOTE: One pump running is preferred as long as flow is < 5700 gpm.

CAUTION:
- The time two KC pumps in a train are operating with header flow less than 5700 gpm shall be minimized.
- Operating two KC pumps in a train with header flow less than 4850 gpm has the potential for a "strong pump" versus "weak pump" interaction, which can impact minimum flow capacity.

_____ 3.7.7 **IF** KC flow requirement in the Train 1B header is < 5700 gpm **AND** it is desired to place the second Train 1B pump in service, perform the following:

3.7.7.1 Start the remaining KC Train 1B pump:

_____ • "KC PUMP B1"
OR
_____ • "KC PUMP B2"

_____ 3.7.7.2 Ensure minimum flow requirements are met.

_____ 3.7.8 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System), **WHEN** KC flow and temperature have stabilized perform the following: (R.M.)

_____ 3.7.8.1 **IF** 1NV-153A (Letdn Hx Otlt 3-Way Vlv) position was recorded as "DEMIN" in Step 3.7.4.2 **AND** no other reason exists for it to remain in the "VCT" position, return it to "AUTO" as follows:

_____ A. Place 1NV-153A in the "DEMIN" position. (R.M.)

☐ B. Verify 1NV-153A returns to "AUTO".

_____ 3.7.8.2 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers have been restored to service. Person notified ____________________________

_____ 3.7.8.3 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers have been restored to service. Person notified ___________________________

NOTE:	At this point, KC Train 1A and 1B are in parallel service.

_____ 3.7.9 **IF** RN miniflow was established per Step 3.7.1.2, **WHEN** no longer needed, secure unneeded flow paths.

_____ 3.7.10 **IF** Train 1B is to be secured, leaving Train 1A in service as per Enclosure 4.1 (System Startup), go to Step 3.9.

_____ 3.8 **IF** placing KC Train 1B in parallel operation with KC Train 1A with the trains **NOT** cross-connected, complete the following steps:

> **NOTE:** RN System minimum flow protection is normally established using an idle KC train.

3.8.1 Complete the following steps to ensure the RN System has miniflow protection:

_____ 3.8.1.1 **IF** a Unit 2 KC Hx discharge valve is in the "MINIFLOW" position, perform the following:

A. Ensure the associated inlet valve is open:

_____ _____ • 2RN-287A (KC Hx 2A Inlet Isol)

_____ _____ • 2RN-347B (KC Hx 2B Inlet Isol)

_____ B. Ensure a complete RN flow path exists from the RN Pumps through the applicable Hx to the discharge.

_____ 3.8.1.2 **IF** no Unit 2 KC Hxs are available for RN miniflow, establish miniflow per OP/0/A/6400/006 C (Nuclear Service Water System) as necessary to maintain RN flow ≥ 8600 gpm per operating RN Pump.

_____ _____ 3.8.2 Ensure 1RN-347B (KC Hx 1B Inlet Isol) is open.

_____ 3.8.3 Ensure "KC HX 1B OTLT MODE" is in "KC TEMP".

_____ _____ 3.8.4 Ensure 1KC-81B (KC To ND Hx 1B Sup Isol) is closed.

> **CAUTION:** 5700 gpm discharge header flow per operating KC Pump shall **NOT** be exceeded.

3.8.5 Start either KC Train 1B pump:

_____ • "KC PUMP B1"
OR
_____ • "KC PUMP B2"

_____ _____ 3.8.6 Ensure 1KC-C40B (Train B Miniflow Isol) opens

_____ 3.8.7 **IF** KC flow requirement in the Train 1B header is > 5700 gpm, perform the following:

_____ _____ 3.8.7.1 Ensure 1KC-C40B (Train B Miniflow Isol) is closed.

_____ 3.8.7.2 **IF** Train 1B header flow is > 5700 gpm, start the remaining KC Train 1B pump:

- _____ "KC PUMP B1"
  OR
- _____ "KC PUMP B2"

NOTE:	One pump running is preferred as long as flow is < 5700 gpm.

CAUTION:	· The time two KC pumps in a train are operating with header flow less than 5700 gpm shall be minimized. · Operating two KC pumps in a train with header flow less than 4850 gpm has the potential for a "strong pump" versus "weak pump" interaction, which can impact minimum flow capacity.

_____ 3.8.8 **IF** KC flow requirement in the Train 1B header is < 5700 gpm **AND** it is desired to place the second Train 1B Pump in service, perform the following:

3.8.8.1 Start the remaining KC Train 1B pump:

- _____ "KC PUMP B1"
  OR
- _____ "KC PUMP B2"

_____ 3.8.8.2 Ensure minimum flow requirements are met.

_____ 3.8.9 **IF** RN miniflow was established per Step 3.8.1.2, **WHEN** no longer needed, secure unneeded flow paths.

NOTE:	At this point, KC Train 1A and 1B are in parallel service with KC Train 1B isolated from the Aux and Rx Bldg Non-Ess Headers.

_____ 3.9 **IF** Train 1B is to be secured, leaving Train 1A in service as per Enclosure 4.1 (System Startup), complete the following steps.

_____ 3.9.1 Notify Radwaste of the intent to change the current KC system pump lineup.

_____ 3.9.2 **IF** KC Trains 1A and 1B are **NOT** cross-connected, ensure that any component required to support unit operation is **NOT** being cooled by KC Train 1B.

_____ 3.9.3 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System) perform the following: (R.M.)

_____ 3.9.3.1 Verify the Cation Bed Demineralizer is **NOT** in service per OP/1/A/6200/001 (Chemical and Volume Control System).

_____ 3.9.3.2 Record position of 1NV-153A (Letdn Hx Otlt 3-Way Vlv).
Recorded valve position_________________

_____ 3.9.3.3 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers will be bypassed while shifting KC Pumps.
Person notified ______________________________

_____ 3.9.3.4 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers will be bypassed while shifting KC Pumps.
Person notified ______________________________

_____ 3.9.3.5 Place 1NV-153A (Letdn Hx Otlt 3-Way Vlv) in the "VCT" position.

3.9.4 Adjust the following flow controllers on 1MC11 to zero gpm flow:

_____ • 1KC-149 (KF Hx 1A Cool Wtr Otlt)

_____ • 1KC-156 (KF Hx 1B Cool Wtr Otlt)

3.9.5 Stop all KC Train 1B pumps:

_____ • "KC PUMP B1"

_____ • "KC PUMP B2"

_____ 3.9.6 Place "KC HX 1B OTLT MODE" in "MINIFLOW".

3.9.7 Perform the following for the KF cooling loops that are in service:

_____ 3.9.7.1 **IF** 1A KF Cooling Loop is in service, adjust 1KC-149 (KF Hx 1A Cool Wtr Otlt) flow controller on 1MC11 to 3000 gpm or as necessary to maintain Spent Fuel Pool temperature < 125°F.

_____ 3.9.7.2 **IF** 1B KF Cooling Loop is in service, adjust 1KC-156 (KF Hx 1B Cool Wtr Otlt) flow controller on 1MC11 to 3000 gpm or as necessary to maintain Spent Fuel Pool temperature < 125°F.

_____ _____ 3.9.8 **IF** KC Train 1A flow approaches 5700 gpm while performing the next step, ensure 1KC-C37A (Train A Miniflow Isol) is closed.

NOTE:	One pump running is preferred as long as flow is < 5700 gpm.

_____ 3.9.9 **IF** KC flow requirements are < 5700 gpm **AND** both KC Train 1A pumps are running, stop either KC Train 1A pump:

_____ • "KC PUMP A1"
OR
_____ • "KC PUMP A2"

NOTE:	At this point, KC Train 1A is in service as per Enclosure 4.1 (System Startup).

_____ 3.9.10 **IF** RN flow has been established through components other than the Unit 2 KC Hx's for RN miniflow, secure unneeded flow paths.

_____ 3.9.11 **IF** letdown is in service per OP/1/A/6200/001 (Chemical and Volume Control System), **WHEN** KC flow and temperature have stabilized perform the following: (R.M.)

_____ 3.9.11.1 **IF** 1NV-153A (Letdn Hx Otlt 3-Way Vlv) position was recorded as "DEMIN" in Step 3.9.3.2 **AND** no other reason exists for it to remain in the "VCT" position, return it to "AUTO" as follows:

_____ A. Place 1NV-153A in the "DEMIN" position. (R.M.)

☐ B. Verify 1NV-153A returns to "AUTO".

_____ 3.9.11.2 **IF** letdown flow is through the demineralizers, notify Primary Chemistry that the demineralizers have been restored to service. Person notified ____________________________

_____ 3.9.11.3 **IF** letdown flow is through the demineralizers, notify Radiation Protection that the demineralizers have been restored to service. Person notified ___________________________

3.10 File this enclosure in the designated storage cabinet.

Catawba Nuclear Station
JPM I
2021 NRC Exam

JPM I

Catawba Nuclear Station
JPM I
2021 NRC Exam

EVALUATION SHEET

Task: Local ESPS Alignment to 1ETA through 1ATC

Alternate Path: No

Facility JPM #: NEW

Safety Function: 6 **Title:** Loss of Offsite and Onsite Power (Station Blackout)

K/A EPE 055 EA2.03 Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power.

Rating(s): 3.9 / 4.7 **CFR:** 43.5 / 45.13

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ In-**P**lant __X__ **P**erform __X__ **S**imulate ______

References: EP/2/A/5000/ECA-0.0 (Loss of All AC Power) Encl. 52 (Local ESPS Alignment to 2ETB through 2ATD)

Task Standard: Both ESPS D/Gs started with ESPS Switchgear supply breaker to Unit 1 and Unit 2 6.9Kv switchgear and 2TD-12 closed.

Validation Time: 10 minutes **Time Critical:** Yes _____ No __X__

==

Applicant:
NAME________________________ Docket #_____________ Time Start: ___________
Time Finish: ___________

Performance Time ______

Performance Rating:

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

Catawba Nuclear Station
JPM I
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 2 is in Mode 3 following a Loss of All AC power**
- **EP/2/A/5000/ECA-0.0 has been entered**
- **Neither of the Emergency D/Gs could be started**
- **Management has determined that power will be restored to 2ETB from ESPS through 2ATD**

INITIATING CUES:

- **ECA-0.0 Enclosure 52 (Local ESPS Alignment to 2ETB through 2ATD) has been entered and Steps 1 & 2 are complete. The Control Room Supervisor has instructed you to perform ECA-0.0 Enclosure 52 beginning at step 3.**

Examiner Note: Provide applicant with a copy of EP/2/A/5000/ECA-0.0, Enclosure 52.

Catawba Nuclear Station
JPM I
2021 NRC Exam

START TIME _________

STEP/STANDARD	SAT/UNSAT
STEP 1: 3. Perform the following on 0ELCP0091 (ESPS Diesel Generator Emergency Control Panel): a. Ensure the following breakers – OPEN: • "ESPS SWGR SUPPLY BKR TO U1 & U2 6.9 KV SWGR" • "1TA-13 SWGR INCOMING FDR FROM ESPS SWGR" • "1TB-2 SWGR INCOMING FDR FROM ESPS SWGR" • "2TC-13 SWGR INCOMING FDR FROM ESPS SWGR" • "2TD-12 SWGR INCOMING FDR FROM ESPS SWGR" **Examiner Cue: As applicant describes verifying breaker status for ESPS SWGR SUPPLY BKR and 2TD-12, "green light is lit." For 1TA-13, 1TB-2, and 2TC-13, "Breaker green flag is shown."** STANDARD: Applicant verifies the green OPEN lights lit and red CLOSE lights dark or breaker green flag showing for the listed breakers. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 2: 3.b Verify the following lights – LIT: • "D/G #1 READY TO START" • "D/G #2 READY TO START" **Examiner Cue: As applicant describes verifying the lights lit, "Light is lit."** STANDARD: Applicant verifies both D/G ready to start lights lit. COMMENTS:	___ SAT ___UNSAT

STEP 3: 3.c Start ESPS DGs by depressing the following pushbuttons: • "D/G #1 START" pushbutton • "D/G #2 START" pushbutton **Examiner Cue: As applicant describes depressing the D/G start pushbuttons, "Button has been depressed and D/G red running lights lit."** STANDARD: Applicant describes depressing the D/G start pushbuttons. **Examiner Note: This step is critical to allow power restoration to 2ETB from ESPS.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

NOTE

- Up to two minutes may elapse from pressing the "START" pushbuttons to "READY TO LOAD" indicating lights illuminate.
- While "TROUBLE" light may momentarily illuminate during start up.

STEP/STANDARD	SAT/UNSAT
STEP 4: 3.d Verify the following lights: • "D/G # 1 RUNNING" red light – LIT • "D/G # 2 RUNNING" red light – LIT • "ESPS GENERATOR #1 OUTPUT BREAKER" red light – LIT • "ESPS GENERATOR #2 OUTPUT BREAKER" red light – LIT • "D/G #1 READY TO LOAD" amber light – LIT • "D/G #2 READY TO LOAD" amber light – LIT **Examiner Cue: As applicant describes verifying each light, "Light is LIT."** STANDARD: Applicant describes verifying the listed lights lit. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 5: 4. Contact Control Room for the following: a. Verify the following switches in "OFF" • "HTWL PUMP 2C" • "CM BSTR PUMP 2C". b. Verify 2ETB load shed – COMPLETE. c. Verify 2ETB aligned to ATD. **Examiner Cue: Once applicant contacts the control room, "2C hotwell and condensate booster pump switches are in the "OFF" position, 2ETB load shed is complete, and 2ETB is aligned to ATD."** STANDARD: Applicant contacts the control room for the required information. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 6: 5. CLOSE "ESPS SWGR SUPPLY BKR to U1 & U2 6.9 KV SWGR". **Examiner Cue: As applicant describes rotating pistol grip for the listed breaker to the CLOSE position, "Pistol grip rotated clockwise to the CLOSE position and red light is lit and green light is dark."** STANDARD: Applicant describes depressing the breaker close pushbutton. **Examiner Note: This step is critical to allow power restoration to 2ETB from ESPS.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

<table>
<tr><td>
STEP 7: 6. CLOSE “2TD-12 SWGR INCOMING FDR FROM ESPS SWGR”.

Examiner Cue: As applicant describes rotating pistol grip for the listed breaker, “Pistol grip rotated clockwise to the CLOSE position and red light is lit and green light is dark.”

STANDARD:

Applicant describes rotating the pistol grip for breaker clockwise to the CLOSE position.

Examiner Note: This step is critical to allow power restoration to 2ETB from ESPS.

COMMENTS:
</td><td>
CRITICAL STEP

___ SAT

___ UNSAT
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STEP 8: 7. WHEN 2ATD aligned to 2ETB, THEN notify Control Room to close “4KV XFMR 2ATD FDR” on 1MC-11.

Examiner Cue: As applicant describes contacting the control room, “4KV XFMR 2ATD FDR has been closed. Another operator will continue with the procedure. This JPM is complete.”

STANDARD:

Applicant describes contacting the control room to close the appropriate breaker.

COMMENTS:

END OF TASK
</td><td>
___ SAT

___ UNSAT
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STOP TIME________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 2 is in Mode 3 following a Loss of All AC power**
- **EP/2/A/5000/ECA-0.0 has been entered**
- **Neither of the Emergency D/Gs could be started**
- **Management has determined that power will be restored to 2ETB from ESPS through 2ATD**

INITIATING CUES:

- **ECA-0.0 Enclosure 52 (Local ESPS Alignment to 2ETB through 2ATD) has been entered and Steps 1 & 2 are complete. The Control Room Supervisor has instructed you to perform ECA-0.0 Enclosure 52 beginning at step 3.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE Steps 1 and 2 can be performed in any order or concurrently.

✓ 1. **Ensure 2GTB (Incoming Feeder From Xfmr 2ATD) (TB-568, 2F-17) - OPEN.**

2. **Align 2TD as follows:**

NOTE Steps 2.a through 2.e may be performed in any order or concurrently.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Verify the following 2TD lockout relays reset: ✓ • 86T/2TD (2TD-04 cubicle) ✓ • 86BNA/2TD05 (2TD-05 cubicle) ✓ • 86NA1/2TD05 (2TD-05 cubicle) ✓ • 86BS/2TD07 (2TD-07 cubicle) ✓ • 86S/2TD07 (2TD-07 cubicle) ✓ • 86BNB/2TD09 (2TD-09 cubicle) ✓ • 86NB1/2TD09 (2TD-09 cubicle) ✓ • 86NA2/2SCPD (2SCPD panel) ✓ • 86NB2/2SCPD (2SCPD panel).	a. Perform the following: __ 1) Notify Control Room Supervisor. **NOTE** It is acceptable to continue in this enclosure if discrepancies are expected to be resolved. __ 2) Do not continue attempts to energize 2TD until all lockouts evaluated and cleared.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	
b. Verify the following breakers - RACKED OUT:	b. Rack out affected breaker(s) as follows:
✓ • 2TD-05 (Normal B Incoming Feeder From XFMR 2T1B)	__ 1) Open sliding door in compartment door.
✓ • 2TD-09 (Normal A Incoming Feeder From Xfmr 2T1A)	__ 2) Verify breaker "OPEN" per mechanical "OPEN-CLOSE" flag.
✓ • 2TD-10 (2D RC Pump Motor).	__ 3) Engage racking tool.
	__ 4) Push racking unlocking lever left and rotate racking crank counterclockwise ¼ turn.
	__ 5) Release racking unlocking lever.
	__ 6) Continue cranking counterclockwise until unlocking lever snaps back and racking mechanism automatically stops in "TEST".
	__ 7) Push racking unlocking lever to left and rotate racking crank counterclockwise ¼ turn.
	__ 8) Release racking unlocking lever.
	__ 9) Continue cranking counterclockwise until unlocking lever snaps back and racking mechanism automatically stops in "DISCONNECT".
	__ 10) Remove racking tool.
	__ 11) Close sliding door in compartment door.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. (Continued)

✓ c. Verify 2TD-04 (6900/4160 VAC XFMR 2ATD) - RACKED IN.

c. Rack in breaker as follows:

__ 1) Verify breaker "OPEN" per mechanical "OPEN-CLOSE" flag.

__ 2) Ensure closing springs toggle switch in "ON" position.

__ 3) Verify breaker in "DISCONNECT".

__ 4) Ensure compartment doors closed.

__ 5) Open sliding door in compartment door.

__ 6) Engage racking tool.

__ 7) Push racking unlocking lever to left and rotate racking crank clockwise ¼ turn.

__ 8) Release racking unlocking lever.

__ 9) Continue cranking clockwise until unlocking lever snaps back and racking mechanism automatically stops in "TEST".

__ 10) Push rotating unlocking lever to left and rotate racking crank clockwise ¼ turn.

__ 11) Release racking unlocking lever.

__ 12) Continue cranking clockwise until unlocking lever snaps back and racking mechanism automatically stops in "CONNECT".

__ 13) Remove racking tool.

__ 14) Close sliding door in compartment door.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. (Continued)

CAUTION **The 6.9 KV bus must be de-energized while racking the breaker from "TEST" to "CONNECT".**

✓ d. Verify 2TD-12 (2TD SWGR INCOMING FDR FROM ESPS SWGR) - RACKED IN.

d. Rack in 2TD-12 (2TD SWGR INCOMING FDR FROM ESPS SWGR) as follows:

__ 1) Open sliding door in compartment door.

__ 2) Verify breaker "OPEN" per mechanical "OPEN-CLOSE" flag.

__ 3) Verify breaker in "DISCONNECT".

__ 4) Engage racking tool.

__ 5) Push racking unlocking lever to left and rotate racking crank clockwise ¼ turn.

__ 6) Release racking unlocking lever.

__ 7) Continue cranking clockwise until unlocking lever snaps back and racking mechanism automatically stops in "TEST".

__ 8) Push rotating unlocking lever to left and rotate racking crank clockwise ¼ turn.

__ 9) Release racking unlocking lever.

__ 10) Continue cranking clockwise until unlocking lever snaps back and racking mechanism automatically stops in "CONNECT".

__ 11) Remove racking tool.

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. (Continued)	
	__ 12) Close sliding door in compartment door.
	NOTE If the "7 KV BUS 2TD MODE SEL" switch is in "AUTO", the breaker will auto close when taken to "CONNECT".
✓ e. Verify 2TD-07 (7KV Bus 2TD Tie Bkr) - RACKED IN.	e. Perform the following: 1) Rack in 2TD-07 (7KV Bus 2TD Tie Bkr) as follows: __ a) Verify breaker "OPEN" per mechanical "OPEN-CLOSE" flag. __ b) Ensure power control toggle switch in "ON" position. __ c) Verify breaker in "DISCONNECT". __ d) Ensure compartment doors closed. __ e) Open sliding door in compartment door. __ f) Engage racking tool. __ g) Push racking unlocking lever to left and rotate racking crank clockwise ¼ turn. __ h) Release racking unlocking lever.

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. (Continued)

__ i) Continue cranking clockwise until unlocking lever snaps back and racking mechanism automatically stops in "TEST".

__ j) Push rotating unlocking lever to left and rotate racking crank clockwise ¼ turn.

__ k) Release racking unlocking lever.

__ l) Continue cranking clockwise until unlocking lever snaps back and racking mechanism automatically stops in "CONNECT".

__ m) Remove racking tool.

__ n) Close sliding door in compartment door.

__ 2) Notify Control Room to place "7 KV BUS 2TD MODE SEL" switch in "AUTO" to close 2TD-07.

3) **<u>IF</u>** 2TD-07 cannot be closed in Control Room, **<u>THEN</u>** perform the following:

__ a) Obtain breaker manual pull cord from break glass station located near breaker racking tool storage location.

__ b) Attach pull cord to "CLOSE" lever at bottom of breaker.

__ c) Do not stand in front of breaker cubicle.

__ d) Pull cord to close breaker.

__ e) Notify Control Room of status of 2TD-07.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. (Continued)

f. Ensure the following breakers - OPEN:

✓ • 2TD-03 (6900 VAC Switchgear RCP2D Feeder)

✓ • 2TD-04 (6900/4160 VAC XFMR 2ATD)

✓ • 2TD-06 (6900/600 VAC Xfrmr 2TXD)

✓ • 2TD-08 (2C2 Heater Drain Tank Pump Motor)

✓ • 2TD-11 (2C Condensate Booster Pump Motor)

✓ • 2TD-13 (2C Hotwell Pump Motor)

✓ • 2TD-12 (2TD SWGR Incoming FDR From ESPS SWGR)

✓ • 2TD-14 (6900/600 VAC Xfrmr 2STXD)

✓ • 2TD-15 (6900/600 VAC Xfmr 2TXF).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. **Perform the following on 0ELCP0091 (ESPS Diesel Generator Emergency Control Panel) (SERV-594, U-31):**

a. Ensure the following breakers - OPEN:

__ • "ESPS SWGR SUPPLY BKR TO U1 & U2 6.9 KV SWGR"

__ • "1TA-13 SWGR INCOMING FDR FROM ESPS SWGR"

__ • "1TB-2 SWGR INCOMING FDR FROM ESPS SWGR"

__ • "2TC-13 SWGR INCOMING FDR FROM ESPS SWGR"

__ • "2TD-12 SWGR INCOMING FDR FROM ESPS SWGR".

NOTE ESPS cannot be started from the ESPS Diesel Generator Emergency Control Panel if the ESPS Mode select switch is in the test position.

b. Verify the following white lights - LIT:

__ • "DG #1 READY TO START"

__ • "DG #2 READY TO START".

b. Perform the following:

__ 1) Ensure 0EQSCA3003CSMODE (ESPS SWGR MODE SELECT SWITCH) on 0EQSCA3003 (ESPS SWGR CONTROL PANEL) set to "EMERGENCY" mode (YRD/E-594, 44-X, 56-Y, Bldg 77163).

__ 2) **IF** DG start lights **NOT** lit, **THEN** notify Control Room.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. (Continued)

   c. Start ESPS DGs by depressing the following pushbuttons:

   __ • "DG #1 START" pushbutton

   __ • "DG #2 START" pushbutton.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3. (Continued)

NOTE
- Up to two minutes may elapse from pressing the "START" pushbuttons to "READY TO LOAD" indicating lights illuminate.
- White "TROUBLE" light may momentarily illuminate during start up.

d. Verify the following lights:

__ • "D/G # 1 RUNNING" red light - LIT

__ • "D/G # 2 RUNNING" red light - LIT

__ • "ESPS GENERATOR #1 OUTPUT BREAKER" red light - LIT

__ • "ESPS GENERATOR #2 OUTPUT BREAKER" red light - LIT

__ • "DG #1 READY TO LOAD" amber light - LIT

__ • "DG #2 READY TO LOAD" amber light - LIT.

RNO:

d. Perform the following:

1) **IF** both "DG #1 READY TO LOAD" and "DG #2 READY TO LOAD" lights dark, **THEN** perform the following:

__ a) Notify Control Room Supervisor that 2ETB cannot be energized by ESPS.

__ b) **GO TO** Step 13 to secure ESPS.

2) **IF** either "DG #1 READY TO LOAD" **OR** "DG #2 READY TO LOAD" lights lit, **THEN** perform the following:

__ a) Notify Control Room Supervisor to **REFER TO** Enclosure 55 (ESPS Operation With One D/G) to evaluate if 2ETB should be energized with one ESPS D/G.

(RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. (Continued)	b) Perform one of the following based on Control Room Supervisor evaluation: __ • **IF** Control Room Supervisor determines ESPS should be used with one D/G, **THEN GO TO** Step 4. **OR** __ • **IF** Control Room Supervisor determines ESPS should **NOT** be used, **THEN GO TO** Step 13 to secure ESPS.
4. **Contact Control Room for the following:** a. Verify the following switches in "OFF": __ • "HTWL PUMP 2C" __ • "CM BSTR PUMP 2C". __ b. Verify 2ETB load shed - COMPLETE. __ c. Verify 2ETB aligned to 2ATD.	
__ 5. **CLOSE "ESPS SWGR SUPPLY BKR to U1 & U2 6.9 KV SWGR".**	
__ 6. **CLOSE "2TD-12 SWGR INCOMING FDR FROM ESPS SWGR".**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 7. **WHEN 2ATD aligned to 2ETB, THEN notify Control Room to close "4KV XFMR 2ATD FDR" on 2MC-11.**	**IF "4KV XFMR 2ATD FDR" cannot be closed in Control Room, THEN CLOSE 2TD-04 (6900/4160 VAC XFMR 2ATD) as follows:** __ a. Obtain breaker manual pull cord from break glass station located near breaker racking tool storage location. __ b. Attach pull cord to "CLOSE" lever at bottom of breaker. __ c. Do not stand in front of breaker cubicle. __ d. Pull cord to close breaker. __ e. Notify Control Room of status of 2TD-04.
8. **IF notified by Control Room to close 2ETB-03 (Normal Incoming Feeder From Xfmr 2ATD), THEN perform the following:** __ a. Obtain breaker manual pull cord from break glass station located near breaker racking tool storage location. __ b. Attach pull cord to "CLOSE" lever at bottom of breaker. __ c. Do not stand in front of breaker cubicle. __ d. Pull cord to close breaker. __ e. Notify Control Room of status of 2ETB-03.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

9. **WHEN notified by Control Room, THEN CLOSE the following load center normal incoming breakers from 2ETB:**

   __ • 2ELXB-4B (Normal Incoming Breaker Fed From Xfmr 2ETXB) (AB-560, AA-67, Rm 362)

   __ • 2ELXD-4B (Normal Incoming Breaker Fed from Xfmr 2ETXD) (AB-560, AA-68, Rm 362).

10. **WHEN time allows, THEN tag the following breakers:**

    __ • 2TD-05 (Normal B Incoming Feeder From XFMR 2T1B)

    __ • 2TD-09 (Normal A Incoming Feeder From Xfmr 2T1A)

    __ • 2TD-10 (2D RC Pump Motor).

__ 11. **Notify Control Room personnel of status.**

__ 12. **Exit this enclosure.**

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

13. **Secure ESPS as follows:**

a. Perform the following on 0ELCP0091 (ESPS Diesel Generator Emergency Control Panel) (SERV-594, U-31):

NOTE
- The stop button must be depressed even if ESPS D/G is off to remove the run permissive.
- D/G engine goes into a cool down cycle for approximately five minutes after stop buttons are depressed. The D/G ready to load light will go dark immediately, but the running light will remain lit for 5 minutes.

1) Stop ESPS D/Gs by depressing the following pushbuttons:

__ • "DG #1 STOP" pushbutton

__ • "DG #2 STOP" pushbutton.

__ 2) Ensure "ESPS SWGR SUPPLY BKR to U1 & U2 6.9 KV SWGR" - OPEN.

__ 3) Ensure "2TD-12 SWGR INCOMING FDR FROM ESPS SWGR" - OPEN.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

13. (Continued)

CAUTION **The 6.9 KV bus must be de-energized while racking the breaker from "CONNECT" to "TEST".**

b. Rack out 2TD-12 (2TD SWGR Incoming Fdr From ESPS SWGR) as follows:

__ 1) Open sliding door in compartment door.

__ 2) Verify breaker "OPEN" per mechanical "OPEN-CLOSE" flag.

__ 3) Engage racking tool.

__ 4) Push racking unlocking lever to left and rotate racking crank counterclockwise ¼ turn.

__ 5) Release racking unlocking lever.

__ 6) Continue cranking counterclockwise until unlocking lever snaps back and racking mechanism automatically stops in "TEST".

__ 7) Push racking unlocking lever to left and rotate racking crank counterclockwise ¼ turn.

__ 8) Release racking unlocking lever.

__ 9) Continue cranking counterclockwise until unlocking lever snaps back and racking mechanism automatically stops in "DISCONNECT".

__ 10) Remove racking tool.

__ 11) Close sliding door in compartment door.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

13. (Continued)

c. Rack in the following breakers per Step 13.d:

__ 1) 2TD-05 (Normal B Incoming Feeder From XFMR 2T1B)

__ 2) 2TD-09 (Normal A Incoming Feeder From Xfmr 2T1A)

__ 3) 2TD-10 (2D RC Pump Motor).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

13. (Continued)

d. Rack in each breaker as follows:

__ 1) Verify breaker "OPEN" per mechanical "OPEN-CLOSE" flag.

__ 2) Ensure closing springs toggle switch in "ON" position.

__ 3) Verify breaker in "DISCONNECT".

__ 4) Ensure compartment doors closed.

__ 5) Open sliding door in compartment door.

__ 6) Engage racking tool.

__ 7) Push racking unlocking lever to left and rotate racking crank clockwise ¼ turn.

__ 8) Release racking unlocking lever.

__ 9) Continue cranking clockwise until unlocking lever snaps back and racking mechanism automatically stops in "TEST".

__ 10) Push rotating unlocking lever to left and rotate racking crank clockwise ¼ turn.

__ 11) Release racking unlocking lever.

__ 12) Continue cranking clockwise until unlocking lever snaps back and racking mechanism automatically stops in "CONNECT".

__ 13) Remove racking tool.

__ 14) Close sliding door in compartment door.

__ e. Notify Control Room personnel of status.

JPM J

Catawba Nuclear Station
JPM J
2021 NRC Exam

EVALUATION SHEET

Task: Place 1B Hydrogen Recombiner in Service

Alternate Path: No

Facility JPM #: VX-025

Safety Function: 5 **Title:** Hydrogen Recombiner and Purge Control System

K/A 028 A2.01 Malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Hydrogen Recombiner power setting, determined by using plant data book

Rating(s): 3.4 / 3.6 **CFR:** 41.5 / 43.5 / 45.3 / 45.13

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ In-**P**lant ___X____ **P**erform _________ **S**imulate ___X_____

References: OP/1/A/6450/010 (Containment Hydrogen Control Systems) rev. 44, Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA)
Unit 1 Revised Data Book, Figure 10

Task Standard: Hydrogen Recombiner 1B in service with Power Adjust "POT" set to approximately 58 KW per Figure 10 of the Unit 1 Revised Data Book.

Validation Time: 15 minutes **Time Critical:** Yes _____ No ___X___

===

Applicant: Time Start: ___________
NAME_____________________________ Docket #________________ Time Finish: ___________

Performance Rating: Performance Time ______

SAT ____ UNSAT ____

Examiner: ________________________________ ________________________________/_________
NAME SIGNATURE DATE

===

COMMENTS

____

____

____

____

____

____

Catawba Nuclear Station
JPM J
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **A LOCA has occurred on Unit 1.**

INITIATING CUES:

- **The CRS instructs you to place Hydrogen Recombiner 1B in service at the required power per OP/1/A/6450/010 (Containment Hydrogen Control Systems), Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA), steps 3.1 through 3.3.14.**
- **All initial conditions are complete.**
- **Containment pressure is 4.3 psig.**
- **Containment hydrogen concentration is 5% as indicated on 1MC-7.**
- **Hydrogen Recombiner 1A is tagged for maintenance.**

Examiner Note: After reading cue, provide the applicant with a copy of OP/1/A/6450/010 rev. 44, Enclosure 4.10 signed off through step 2.2 and a copy of Unit 1 Revised Data Book Figure 10.

STEP/STANDARD	SAT/UNSAT

START TIME: _______

STEP 1 3.1 Ensure the H2 Skimmer Fans running per Enclosure 4.13 (Emergency Manual Operation of the H2 Skimmer Fans) **Examiner Cue**: **"Hydrogen Skimmer Fans 1A & 1B are running per Enclosure 4.13."** STANDARD: Applicant acknowledges cue and signs off the step. COMMENTS:	___ SAT ___ UNSAT

CAUTION: Hydrogen Recombiners are NOT operated with hydrogen concentration $\geq$ 6% without TSC approval.

NOTE:
1. If desired to place both Hydrogen Recombiners in service, Steps 3.2 and 3.3 may be performed in conjunction.
2. If desired to place both Hydrogen Recombiners in service, use additional Enclosure 4.11 (Hydrogen Recombiner Heater Temperature Log).
3. Placing Hydrogen Recombiner 1A in service is preferred for ALARA consideration.

STEP 2 3.2 **IF** placing Hydrogen Recombiner 1A in service, perform the following at 1ELCP0139 (1A Hydrogen Recombiner Control Panel) (AB-577, DD-52, Rm 494): STANDARD: Applicant determines that per the cue sheet, Hydrogen Recombiner 1A is tagged out for maintenance, and that this step is N/A. COMMENTS:	___ SAT ___UNSAT

STEP/STANDARD	SAT/UNSAT
STEP 3 3.3 **IF** placing Hydrogen Recombiner 1B in service, perform the following at 1ELCP0140 (1B Hydrogen Recombiner Control Panel) (AB-560, DD-52, Rm 370): 3.3.1 Ensure the "POWER OUT SWITCH" is in the "OFF" position. STANDARD: Applicant locates the Power Out Switch and ensures it is in the OFF position. **Examiner Cue: "The "POWER OUT SWITCH" is in the "OFF" position."** COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 4 3.3.2 Ensure the "POWER ADJUST" potentiometer is set to zero (000). STANDARD: Applicant verifies the "POWER ADJUST" potentiometer is set to 000. **Examiner Cue: "The "POWER ADJUST" pot is set to zero (000)."** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 5 3.3.3 **IF** the "POWER IN AVAILABLE" light is DARK, ensure 1EMXL-F07C (1B Electric Hydrogen Recombiner Power Supply Panel) (AB-560, BB-47) is in the "ON" position. **Examiner Cue**: **"The "POWER IN AVAILABLE" light is lit."** STANDARD: Applicant determines that with the POWER IN AVAILABLE light being lit, this step is N/A. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 6 3.3.4 Place the "POWER OUT SWITCH" in the "ON" position. **Examiner Cue: After applicant describes placing the POWER OUT SWITCH up to the ON position, "The POWER OUT SWITCH is in the "ON" position."** STANDARD: Applicant places the POWER OUT SWITCH up to the ON position. **Examiner Note: This step is critical to place the hydrogen recombiner in service.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 7 3.3.5 Verify that the red indicating light is lit. **Examiner Cue: After finding the indicating light on the switch plate inform the applicant - "The RED light on the switch plate is lit."** STANDARD: Applicant finds the red indicating light on the switch plate and verifies that it is lit. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 8 3.3.6 Slowly turn the "POWER ADJUST" potentiometer clockwise until 5 KW is indicated on the "POWER OUT" meter. **Examiner Cue: After explaining operation of the potentiometer in the clockwise direction, inform applicant – "The POWER OUT meter rises to 5 KW".** STANDARD: Applicant describes turning the potentiometer clockwise to increase the POWER OUTPUT meter reading to 5 KW. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 9 3.3.7 Maintain a 5 KW output for 10 minutes. **Examiner Cue**: **“Using time compression, 10 minutes has elapsed.”** STANDARD: Applicant describes maintaining this power output for 10 minutes. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 10 3.3.8 Slowly advance the “POWER ADJUST” setting until an output of 10 KW is obtained on the “POWER OUT” meter. **Examiner Cue**: **After explaining the operation of the potentiometer in the clockwise direction inform applicant - “The POWER OUT meter rises to 10 KW.”** STANDARD: Applicant describes turning the potentiometer clockwise to increase the POWER OUTPUT meter reading to 10 KW. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 11 3.3.9 Maintain a 10 KW output for 10 minutes. **Examiner Cue**: **“Using time compression, 10 minutes has elapsed.”** STANDARD: Applicant describes maintaining this power output for 10 minutes. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 12 3.3.10 Advance the "POWER ADJUST" setting until an output of 20 KW is obtained on the "POWER OUT" meter. **Examiner Cue**: **After explaining the operation of the potentiometer in the clockwise direction inform applicant - "The POWER OUT meter rises to 20 KW."** STANDARD: Applicant describes turning the potentiometer clockwise to increase the POWER OUTPUT meter reading to 20 KW. COMMENTS:	___ **SAT** ___ **UNSAT**
STEP 13 3.3.11 Maintain a 20 KW output for 5 minutes. **Examiner Cue**: **"Using time compression, 5 minutes has elapsed."** STANDARD: Applicant describes maintaining this power output for 5 minutes. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 14 3.3.12 Determine Hydrogen Recombiner 1B power setting as follows: 3.3.12.1 Determine KW value from Figure 10 of the Unit 1 Revised Data Book. **Examiner Note: The applicant will use the copy of Unit 1 Revised Data Book Figure 10 provided with the initiating cue.** STANDARD: Applicant determines that the KW value from Figure 10 is 54 KW. **Examiner Note: This step is critical in determining the proper power setting for the given containment conditions.** COMMENTS:	**CRITICAL STEP** **___ SAT** **___ UNSAT**
STEP 15 3.3.12.2 H_2 concentration (1MC-7) ________ % STANDARD: Applicant determines from cue sheet that hydrogen concentration is 5%. COMMENTS:	**___ SAT** **___ UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 16 3.3.12.3 **IF** H_2 concentration is > 3.5%, add 4 KW to calculation. STANDARD: Applicant determines that H2 concentration is > 3.5% and adds 4 KW to the calculation. **Examiner Note: This step is critical in determining the proper power setting for the given containment conditions.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 17 3.3.12.4 Calculate KW as follows: 54 (Step 3.3.12.1) + 4 (Step 3.3.12.3 or N/A) = 58 KW STANDARD: Applicant calculates the required power setting to be 58 KW. **Examiner Note: This step is critical in determining the proper power setting for the given containment conditions.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 18 3.3.13 Advance the "POWER ADJUST" setting until the "POWER OUT" meter indicates the value calculated in 3.3.12.4. Adjust "POWER ADJUST" as necessary to maintain this output. **Examiner Cue: After explaining the operation of the potentiometer in the clockwise direction inform applicant - "The POWER OUT meter rises to 58 KW."** STANDARD: Applicant describes turning the potentiometer clockwise to increase the POWER OUTPUT meter reading to 58 KW. **Examiner Note: This step is critical to set the necessary power output setting for the given containment conditions.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**
STEP 19 3.3.14 Notify the NCO that Hydrogen Recombiner 1B is now in service. Person notified ________________________ STANDARD: Applicant calls the control room and notifies them that 1B Hydrogen Recombiner is in service. **Examiner Cue: "Unit 1 Control Room, this is Greg." Repeat any additional information given.** COMMENTS: **END OF TASK**	___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)
May 2015 NRC Initial License Exam
JPM I

INITIAL CONDITIONS:

- **A LOCA has occurred on Unit 1.**

INITIATING CUES:

- **The CRS instructs you to place Hydrogen Recombiner 1B in service at the required power per OP/1/A/6450/010 (Containment Hydrogen Control Systems), Enclosure 4.10 (Operation of the Hydrogen Recombiners Following a LOCA), steps 3.1 through 3.3.14.**
- **All initial conditions are complete.**
- **Containment pressure is 4.3 psig.**
- **Containment hydrogen concentration is 5% as indicated on 1MC-7.**
- **Hydrogen Recombiner 1A is tagged for maintenance.**

1. Limits and Precautions

1.1 Hydrogen concentrations greater than 3.5% are combustible.

1.2 The maximum electric hydrogen recombiners heater temperature is 1400°F.

1.3 Hydrogen Recombiners and Hydrogen Ignitors are **NOT** operated with hydrogen concentration ≥ 6% without TSC approval.

2. Initial Conditions

AA 2.1 Verify operation of the Hydrogen Recombiner is required per appropriate emergency procedures.

AA 2.2 Request RP coverage due to increasing radiation levels at Hydrogen Recombiner panels during a LOCA.
Person notified Gary Johnson

3. Procedure

_____ 3.1 Ensure the H_2 Skimmer Fans running per Enclosure 4.13 (Emergency Manual Operation of the H_2 Skimmer Fans).

CAUTION: Hydrogen Recombiners are **NOT** operated with hydrogen concentration ≥ 6% without TSC approval.

NOTE:
1. If desired to place both Hydrogen Recombiners in service, Steps 3.2 and 3.3 may be performed in conjunction.
2. If desired to place both Hydrogen Recombiners in service, use additional Enclosure 4.11 (Hydrogen Recombiner Heater Temperature Log).
3. Placing Hydrogen Recombiner 1A in service is preferred for ALARA consideration.

_____ 3.2 **IF** placing Hydrogen Recombiner 1A in service, perform the following at 1ELCP0139 (1A Hydrogen Recombiner Control Panel) (AB-577, DD-52, Rm 494):

☐ 3.2.1 Ensure the "POWER OUT SWITCH" is in the "OFF" position.

☐ 3.2.2 Ensure the "POWER ADJUST" potentiometer is set to zero (000).

_____ _____ 3.2.3 **IF** the "POWER IN AVAILABLE" light is DARK, ensure 1EMXK-F07C (1A Electric Hydrogen Recombiner Power Supply Panel) (AB-577, BB-47) is in the "ON" position:

☐ 3.2.4 Place the "POWER OUT SWITCH" in the "ON" position

☐ 3.2.5 Verify that the red indicating light is lit.

☐ 3.2.6 Slowly turn the "POWER ADJUST" potentiometer clockwise until 5 KW is indicated on the "POWER OUT" meter.

☐ 3.2.7 Maintain a 5 KW output for 10 minutes.

☐ 3.2.8 Slowly advance the "POWER ADJUST" setting until an output of 10 KW is obtained on the "POWER OUT" meter.

☐ 3.2.9 Maintain a 10 KW output for 10 minutes.

☐ 3.2.10 Advance the "POWER ADJUST" setting until an output of 20 KW is obtained on the "POWER OUT" meter.

☐ 3.2.11 Maintain a 20 KW output for 5 minutes.

3.2.12 Determine Hydrogen Recombiner 1A power setting as follows:

_____ 3.2.12.1 Determine KW value from Figure 10 of the Unit One Revised Data Book. ______ KW VALUE

_____ 3.2.12.2 H_2 concentration (1MC7) ________%

_____ 3.2.12.3 **<u>IF</u>** H_2 concentration is > 3.5%, add 4KW to calculation.

_____ 3.2.12.4 Calculate KW as follows:
_________ + __________ = ______ KW
Step 3.2.12.1 Step 3.2.12.3 or N/A

_____ 3.2.13 Advance the "POWER ADJUST" setting until the "POWER OUT" meter indicates the value calculated in 3.2.12.4. Adjust "POWER ADJUST" as necessary to maintain this output.

_____ 3.2.14 Notify NCO that Hydrogen Recombiner 1A is now in service.
Person notified ___________________________________

CAUTION: The maximum electric hydrogen recombiner heater temperature is 1400°F.

NOTE:
1. Temperature stabilization may take up to 5 hours. The heater temperatures are considered stabilized when the channels are within 60°F of each other and the average temperature is ≥ 1225°F. (CNM-1399.36-0010)
2. Hydrogen Recombiner Heater 1A Temperature is monitored from 1VXCR5000 (Train A Hydrogen Recombiner Heater Temperature Recorder) located on 1ELCP0299 (AB-577, DD-52, Rm 494).
3. 1VXCR5000 is a touch screen video monitor. Touching the screen while in screen saver mode will display the main menu. The thermocouple temperatures will be displayed when the "Digital" icon is selected. The "Return" icon on the lower left of the screen returns to the main menu.

☐ 3.2.15 Record hourly the Hydrogen Recombiner 1A Heater Temperature for each of the three thermocouples on Enclosure 4.11 (Hydrogen Recombiner Heater Temperature Log) until stabilized.

☐ 3.2.16 **WHEN** the Hydrogen Recombiner Heater 1A Temperature has stabilized per Enclosure 4.11 (Hydrogen Recombiner Heater Temperature Log), perform the following:

_____ 3.2.16.1 **IF** the thermocouples are inaccurate, proceed to Step 3.2.17.

☐ 3.2.16.2 Adjust "POWER ADJUST" potentiometer as necessary to maintain recombination temperature of 1225-1400°F as read on 1VXCR5000 (Train A Hydrogen Recombiner Heater Temperature Recorder).

☐ 3.2.16.3 Verify the "POWER OUT" meter indicates ≥ the value calculated in Step 3.2.12.4

3.2.17 Every 24 hours, measure containment hydrogen concentration **AND** adjust recombiner power for the duration of recombiner operation as follows:

_____ 3.2.17.1 Determine KW value from Figure 10 of the Unit One Revised Data Book. ______ KW VALUE

_____ 3.2.17.2 H_2 concentration (1MC7) ________%

_____ 3.2.17.3 **IF** H_2 concentration has increased by .5% **OR** is > 3.5%, add 4KW to calculation.

_____ 3.2.17.4 Calculate KW as follows:

_________ + __________ = ______ KW

Step 3.2.17.1 Step 3.2.17.3 or N/A

_____ 3.2.17.5 Advance the "POWER ADJUST" setting until the "POWER OUT" meter indicates the value calculated in 3.2.17.4. Adjust "POWER ADJUST" as necessary to maintain this output.

☐ 3.2.17.6 Monitor Hydrogen Recombiner Heater Temperature per Steps 3.2.15 and 3.2.16 to prevent temperature from exceeding 1400°F.

_____ 3.3 **IF** placing Hydrogen Recombiner 1B in service, perform the following at 1ELCP0140 1B Hydrogen Recombiner Control Panel) (AB-560, DD-52, Rm 370):

☐ 3.3.1 Ensure the "POWER OUT SWITCH" is in the "OFF" position.

☐ 3.3.2 Ensure the "POWER ADJUST" potentiometer is set to zero (000).

_____ _____ 3.3.3 **IF** the "POWER IN AVAILABLE" light is DARK, ensure 1EMXL-F07C (1B Electric Hydrogen Recombiner Power Supply Panel) (AB-560, BB-47) is in the "ON" position:

☐ 3.3.4 Place the "POWER OUT SWITCH" in the "ON" position.

☐ 3.3.5 Verify that the red indicating light is lit.

☐ 3.3.6 Slowly turn the "POWER ADJUST" potentiometer clockwise until 5 KW is indicated on the "POWER OUT" meter.

☐ 3.3.7 Maintain a 5 KW output for 10 minutes.

☐ 3.3.8 Slowly advance the "POWER ADJUST" setting until an output of 10 KW is obtained on the "POWER OUT" meter.

☐ 3.3.9 Maintain a 10 KW output for 10 minutes.

☐ 3.3.10 Advance the "POWER ADJUST" setting until an output of 20 KW is obtained on the "POWER OUT" meter.

☐ 3.3.11 Maintain a 20 KW output for 5 minutes.

3.3.12 Determine Hydrogen Recombiner 1B power setting as follows:

_____ 3.3.12.1 Determine KW value from Figure 10 of the Unit One Revised Data Book.

_____ 3.3.12.2 H_2 concentration (1MC7) ________%

_____ 3.3.12.3 **<u>IF</u>** H_2 concentration is > 3.5%, add 4KW to calculation.

_____ 3.3.12.4 Calculate KW as follows:

_________ + __________ = ______ KW
Step 3.3.12.1 Step 3.3.12.3 or N/A

_____ 3.3.13 Advance the "POWER ADJUST" setting until the "POWER OUT" meter indicates the value calculated in 3.3.12.4. Adjust "POWER ADJUST" as necessary to maintain this output.

_____ 3.3.14 Notify NCO that Hydrogen Recombiner 1B is now in service.
Person notified ___________________________________

CAUTION: The maximum electric hydrogen recombiner heater temperature is 1400°F.

NOTE:
1. Temperature stabilization may take up to 5 hours. The heater temperatures are considered stabilized when the channels are within 60°F of each other and the average temperature is ≥ 1225°F. (CNM-1399.36-0010)
2. Hydrogen Recombiner Heater 1B Temperature is monitored from 1VXCR5500 (Train B Hydrogen Recombiner Heater Temperature Recorder) located on 1ELCP0300 (AB-560, DD-52, Rm 370).
3. 1VXCR5500 is a touch screen video monitor. Touching the screen while in screen saver mode will display the main menu. The thermocouple temperatures will be displayed when the "Digital" icon is selected. The "Return" icon on the lower left of the screen returns to the main menu.

☐ 3.3.15 Record hourly the Hydrogen Recombiner Heater 1B Temperature for each of the three thermocouples on Enclosure 4.11 (Hydrogen Recombiner Heater Temperature Log), until stabilized.

☐ 3.3.16 **WHEN** the Hydrogen Recombiner Heater 1B Temperature has stabilized per Enclosure 4.11 (Hydrogen Recombiner Heater Temperature Log), perform the following:

_____ 3.3.16.1 **IF** the thermocouples are inaccurate, proceed to Step 3.3.17.

☐ 3.3.16.2 Adjust "POWER ADJUST" potentiometer as necessary to maintain recombination temperature of 1225-1400°F as read on1VXCR5500 (Train B Hydrogen Recombiner Heater Temperature Recorder).

☐ 3.3.16.3 Verify the "POWER OUT" meter indicates $\geq$ the value calculated in Step 3.3.12.4.

3.3.17 Every 24 hours, measure containment hydrogen concentration **AND** adjust recombiner power for the duration of recombiner operation as follows:

_____ 3.3.17.1 Determine KW value from Figure 10 of the Unit One Revised Data Book. ______ KW VALUE

_____ 3.3.17.2 H_2 concentration (1MC7) ________%

_____ 3.3.17.3 **IF** H_2 concentration has increased by .5% **OR** is > 3.5%, add 4KW to calculation.

_____ 3.3.17.4 Calculate KW as follows:

_________ + __________ = ______ KW

Step 3.3.17.1 Step 3.3.17.3 or N/A

_____ 3.3.17.5 Advance the "POWER ADJUST" setting until the "POWER OUT" meter indicates the value calculated in 3.3.17.4. Adjust "POWER ADJUST" as necessary to maintain this output.

☐ 3.3.17.6 Monitor Hydrogen Recombiner Heater Temperature per Steps 3.3.15 and 3.3.16 to prevent temperature from exceeding 1400°F.

_____ 3.4 **IF** operation of Hydrogen Recombiner 1A is no longer required, perform the following on panel 1ELCP0139 (1A Hydrogen Recombiner Power Control Panel):

_____ 3.4.1 Turn the "POWER ADJUST" potentiometer on the control panel to zero (000).

_____ 3.4.2 Place the "POWER OUT SWITCH" on the control panel in the "OFF" position.

_____ 3.5 **IF** operation of Hydrogen Recombiner 1B is no longer required, perform the following on panel 1ELCP0140 (1B Hydrogen Recombiner Power Control Panel):

_____ 3.5.1 Turn the "POWER ADJUST" potentiometer on the control panel to zero (000).

_____ 3.5.2 Place the "POWER OUT SWITCH" on the control panel in the "OFF" position.

3.6 Do **NOT** file a copy of this enclosure in the designated storage cabinet.

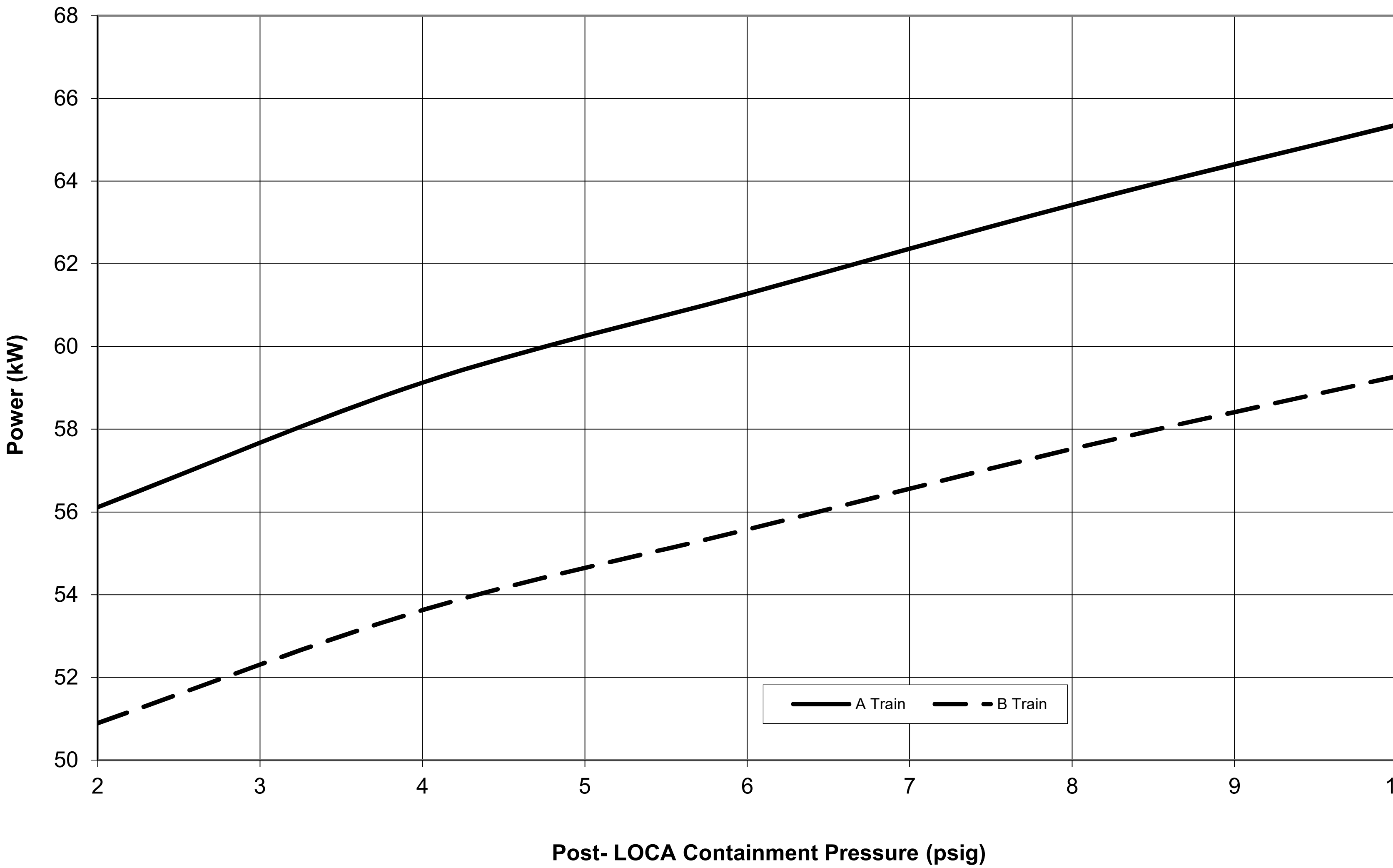

Figure 10 - Unit 1 Hydrogen Recombiners Post-LOCA Power Consumption

Catawba Nuclear Station
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JPM K

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2021 NRC Exam

EVALUATION SHEET

Task: Break Condenser Vacuum Locally

Alternate Path: No

Facility JPM #: CA-084

Safety Function: 4S **Title:** Main Turbine Generator (MT/G) System

K/A 045 A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of secondary plant parameters following T/G trip

Rating(s): 3.3 / 3.7 **CFR:** 41.5 / 45.5

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ In-**P**lant __X__ **P**erform __X__ **S**imulate ________

References: AP/1/A/5500/006 (Loss of S/G Feedwater) Rev 45 Enclosure 3

Task Standard: Enclosure 3 has been completed with the first vacuum breaker opened within 10 minutes.

Validation Time: 8 minutes **Time Critical:** Yes __X__ No ______

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Applicant:
NAME________________________ Docket #______________

Time Start: ____________
Time Finish: ____________

Time Critical (<10 minutes)
Time Start: ___________
Time Finish: ___________

Performance Rating:

SAT ____ UNSAT ____

Performance Time ______

Examiner: ________________________________ ________________________________/_________

NAME SIGNATURE DATE

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COMMENTS

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JPM K
2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is in Mode 3 following a reactor trip.**

INITIATING CUES:

- **The Control Room Supervisor instructs you to perform AP/1/A/5500/006 (Loss of S/G Feedwater) Enclosure 3 (Local Actions to Break Condenser Vacuum).**

This JPM is TIME CRITICAL; time begins when you acknowledge the task.

EXAMINER NOTE: **Provide applicant with a copy of AP/1/A/5500/006, Enclosure 3.**

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JPM K
2021 NRC Exam

<u>**Critical Time Start:**</u> **Record Time that applicant acknowledges the task __________.**

STEP/STANDARD	SAT/UNSAT
CAUTION **High air flow rates will exist when vacuum breakers are first opened. Stay clear of pipe end.**	**CRITICAL STEP**
<u>STEP 1</u>: 1 **Break condenser vacuum by opening the following valves:** • 1CM-368 (2A Main Cond Shell Vacuum Bkr) (TB1-600,1F- 26) (Ladder needed) • 1CM-369 (2B Main Cond Shell Vacuum Bkr) (TB1-603, 1F- 24) (Ladder needed) • 1CM-370 (2C Main Cond Shell Vacuum Bkr) (TB1-605, 1F-22) (Ladder needed). **Examiner Note:** **The critical end time is when the applicant describes opening the first valve. Due to the height of the valves, no fall protection will be required.** **Examiner Cue:** **When applicant describes engaging lever and rotating handwheel counter clockwise to open the following valve then: "A large volume of airflow is heard."** **Examiner Note:** **This step is critical in order to open correct valves for breaking vacuum.** **Critical Time End: __________________** <u>STANDARD</u>: Applicant will describe opening the valves: 1CM-368, 1CM-369, 1CM-370. <u>COMMENTS:</u>	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 2 2 **Secure steam to CSAEs as follows:** a. Close the following valves: • 1SA-22 (Main Steam To CSAE) (TB1-594, 1M-32) • 1SA-27 (Aux Steam To CSAE) (TB-594, 1M-27). STANDARD: Applicant will describe closing 1SA-22 and 1SA-27. **Examiner Cue:** **As applicant properly describes closing the valves give cue as appropriate, "Valve turns until resistance is felt."** **Examiner Note:** **This step is critical, because if it is not performed, the CSAEs will continue to pull vacuum** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___**UNSAT**

STEP 3 b. **WHEN** time and manpower permit, **THEN** complete the shutdown of the CSAEs. **REFER TO** OP/1/B/6300/006 (Main Vacuum). STANDARD: Applicant will read the step **Examiner Cue:** **The Control Room Supervisor has instructed another operator to complete the shutdown of the CSAEs.** COMMENTS:	___ **SAT** ___ **UNSAT**

STEP/STANDARD	SAT/UNSAT
STEP 4 3. **WHEN requested by Control Room Supervisor, THEN verify condenser vacuum broken as follows:** a. Inspect each vacuum breaker for absence of air flow into condenser. b. Notify Control Room Supervisor of results. STANDARD: Applicant will inspect each vacuum breaker for the absence of air flow into the condenser and will report to the Control Room Supervisor. **Examiner Cue:** **After each inspection, "No air flow into condenser."** COMMENTS:	___ SAT ___ UNSAT

STOP TIME________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is in Mode 3 following a reactor trip.**

INITIATING CUES:

- **The Control Room Supervisor instructs you to perform AP/1/A/5500/006 (Loss of S/G Feedwater) Enclosure 3 (Local Actions to Break Condenser Vacuum).**

This JPM is TIME CRITICAL; time begins when you acknowledge the task.

CAUTION **High air flow rates will exist when vacuum breakers are first opened. Stay clear of pipe end.**

1. **Break condenser vacuum by opening the following valves:**

__ • 1CM-368 (1A Main Cond Shell Vacuum Bkr) (TB1-600,1F-26) (Ladder needed)

__ • 1CM-369 (1B Main Cond Shell Vacuum Bkr) (TB1-603, 1F-24) (Ladder needed)

__ • 1CM-370 (1C Main Cond Shell Vacuum Bkr) (TB1-605, 1F-22) (Ladder needed).

2. **Secure steam to CSAEs as follows:**

a. CLOSE the following valves:

__ • 1SA-22 (Main Steam To CSAE) (TB1-594, 1M-32)

__ • 1SA-27 (Aux Steam To CSAE) (TB-594, 1M-27).

__ b. **WHEN** time and manpower permit, **THEN** complete the shutdown of CSAEs. **REFER TO** OP/1/B/6300/006 (Main Vacuum).

3. **WHEN requested by Control Room Supervisor, THEN verify condenser vacuum broken as follows:**

__ a. Inspect each vacuum breaker for absence of air flow into condenser.

__ b. Notify Control Room Supervisor of results.

JPM A.1-1R

RO

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Admin. JPM A.1-1R
September 2021 NRC Exam

EVALUATION SHEET

Task: Calculate Reactor Vessel Head Venting Time

Alternate Path: N/A

Facility JPM #: New

Safety Function: N/A

K/A G 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Importance: 4.3 / 4.4 **CFR:** 41.10 / 43.5 / 45.2 / 45.6

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom **X** Perform **X** Simulate ________

References: EP/1/A/5000/FR-I.3 Response to Voids in Reactor Vessel, Enclosure 2 (rev 19) and images of control room gauges needed to perform head vent calculation.

Task Standard: Reactor vessel head maximum allowable vent time calculated as between 2.1 minutes and 4.4 minutes.

Validation Time: 15 minutes **Time Critical:** Yes _____ No **X**

Applicant:
NAME____________________________ Docket #______________ Time Start: __________
Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

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COMMENTS

Catawba Nuclear Station
Admin. JPM A.1-1R
September 2021 NRC Exam

SIMULATOR OPERATOR INSTRUCTIONS:

1. **ENSURE NRC Examination Security has been established.**
2. **Insert the following malfunctions:**

✓	Instructor Action	Final	Delay	Ramp	Delete In	Event
	XMT-VV010 (TVV_5090 LOWER CNT AIR TEMP A MTR)	145				
	XMT-VV011 (TVV_5110 LOWER CNT AIR TEMP B MTR)	145				
	XMT-VV014 (TVV_5170 LOWER CNT AIR TEMP C MTR)	145				
	XMT-VV015 (TVV_5190 LOWER CNT AIR TEMP D MTR)	145				
	XMT-VX003 (XMI_5320 CNT TRN A H2 ANAL MTR)	2				
	XMT-VX004 (XMI_5330 CNT TRN B H2 ANAL MTR)	2				
	XMT-CNT009 (PNS_5090 CNT PRESS MTR (PI-937))	3.5				
	XMT-CNT008 (PNS_5060 CNT PRESS MTR (PI-936))	3.5				
	XMT-CNT007 (PNS_5050 CNT PRESS MTR (PI-935))	3.5				
	XMT-CNT006 (PNS_5040 CNT PRESS MTR (PI-934))	3.5				
	XMT-CNT011 (PNS_5380 CNT TRN B PRESS MTR)	3.5				
	XMT-CNT010 (PNS_5370 CNT TRN A PRESS MTR)	3.5				
	IND-NC023 (PNC_5120 LOOP B HOT LEG W/R PRESS MTR (PI-405))	900				
	IND-NC024 (PNC_5140 LOOP C HOT LEG W/R PRESS MTR (PI-403))	900				

3. **Take digital photographs of the gauges listed above or provide control board mimics from the instructor station.**

NOTE TO EVALUATOR: These have been provided as part of the JPM package for each applicant.

Catawba Nuclear Station
Admin. JPM A.1-1R
September 2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **A LOCA is in progress on Unit 1**

INITIATING CUES:

- **The CRS directs you to calculate and record the maximum reactor vessel head venting time per EP/1/A/5000/FR-I.3 (Response to Voids in Reactor Vessel) Enclosure 2 (Allowable Hydrogen Venting Time)**

EXAMINER NOTE: Each applicant should receive a copy of FR-I.3 Enclosure 2 as well as 4 pictures of control room gauges for:

- **H2 Analyzers**
- **Containment Pressure**
- **Lower Containment Air Temperatures**
- **Loop B and C Hot Leg W/R Pressure**

START TIME: _______

STEP / STANDARD	CRITICAL STEP
STEP 1: Step 1 **Calculate A where** $A = 9500 \times \frac{(P + 14.7)}{14.7} \times \frac{492}{(T+460)}$ **Where: P = Containment pressure (PSIG)** **T = Lower Containment temperature (°F)** STANDARD: Calculate A: Determine containment pressure as 3.4 psig to 3.6 psig. Determine Lower Containment Temperature as 140°F - 150°F. Using 3.4, 140 $A = 9500 \times \frac{(3.4 + 14.7)}{14.7} \times \frac{492}{(140+460)}$ = **9591.7** Using 3.4, 150 $A = 9500 \times \frac{(3.4 + 14.7)}{14.7} \times \frac{492}{(150+460)}$ = **9434.5** Using 3.6, 140 $A = 9500 \times \frac{(3.6 + 14.7)}{14.7} \times \frac{492}{(140+460)}$ = **9697.7** Using 3.6, 150 $A = 9500 \times \frac{(3.6 + 14.7)}{14.7} \times \frac{492}{(150+460)}$ = **9538.7** **NOTE TO EXAMINER: Acceptable band for value of A is 9434 – 9698.** **This step is critical to properly perform this task and to meet the JPM standard to calculate the maximum head vent time for the containment conditions given.** COMMENTS:	___ **SAT** ___ **UNSAT**

<table>
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STEP 2: Step 2

Calculate B where B = (3 - H) X A

Where H = Containment Hydrogen Concentration (%)

STANDARD:

Determine H2 concentration as between 1.75% and 2.25%.

Using bounding values of A (9434 to 9698):

B = (3 – 1.75) X 9434 = **11792.5**

B = (3 – 1.75) X 9698 = **12122.5**

B = (3 – 2.25) X 9434 = **7075.5**

B = (3 – 2.25) X 9698 = **7273.5**

NOTE TO EXAMINER: Acceptable range for value of B is 7075 – 12123.

This step is critical to properly perform this task and to meet the JPM standard to calculate the maximum head vent time for the containment conditions given.

COMMENTS:

</td><td>

CRITICAL STEP

___ SAT

___UNSAT

</td></tr>
</table>

STEP 3: Step 3 **Determine C from the curve for the current NC system pressure.** STANDARD: Determines NC pressure from pictures as **875** psig to **925** psig. Determines C (hydrogen flow rate) as between **2,750** – **3,250** SCFM. **This step is critical to properly perform this task and to meet the JPM standard to calculate the maximum head vent time for the conditions given.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP 4 Enclosure 5 Step 4 **Calculate T:** **T= B / C = Venting time in minutes** STANDARD: Using bounding values for B and C: T = 7075 / 2750 = **2.57** minutes T = 7075 / 3250 = **2.18** minutes T = 12123 / 2750 = **4.40** minutes T = 12123 / 3250 = **3.73** minutes **NOTE TO EXAMINER: Maximum allowable venting time acceptable band is 2.1 – 4.4 minutes (2 minutes 6 seconds to 4 minutes 24 seconds).** **This step is critical to properly perform this task and to meet the JPM standard to calculate the maximum head vent time for the containment conditions given.** COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

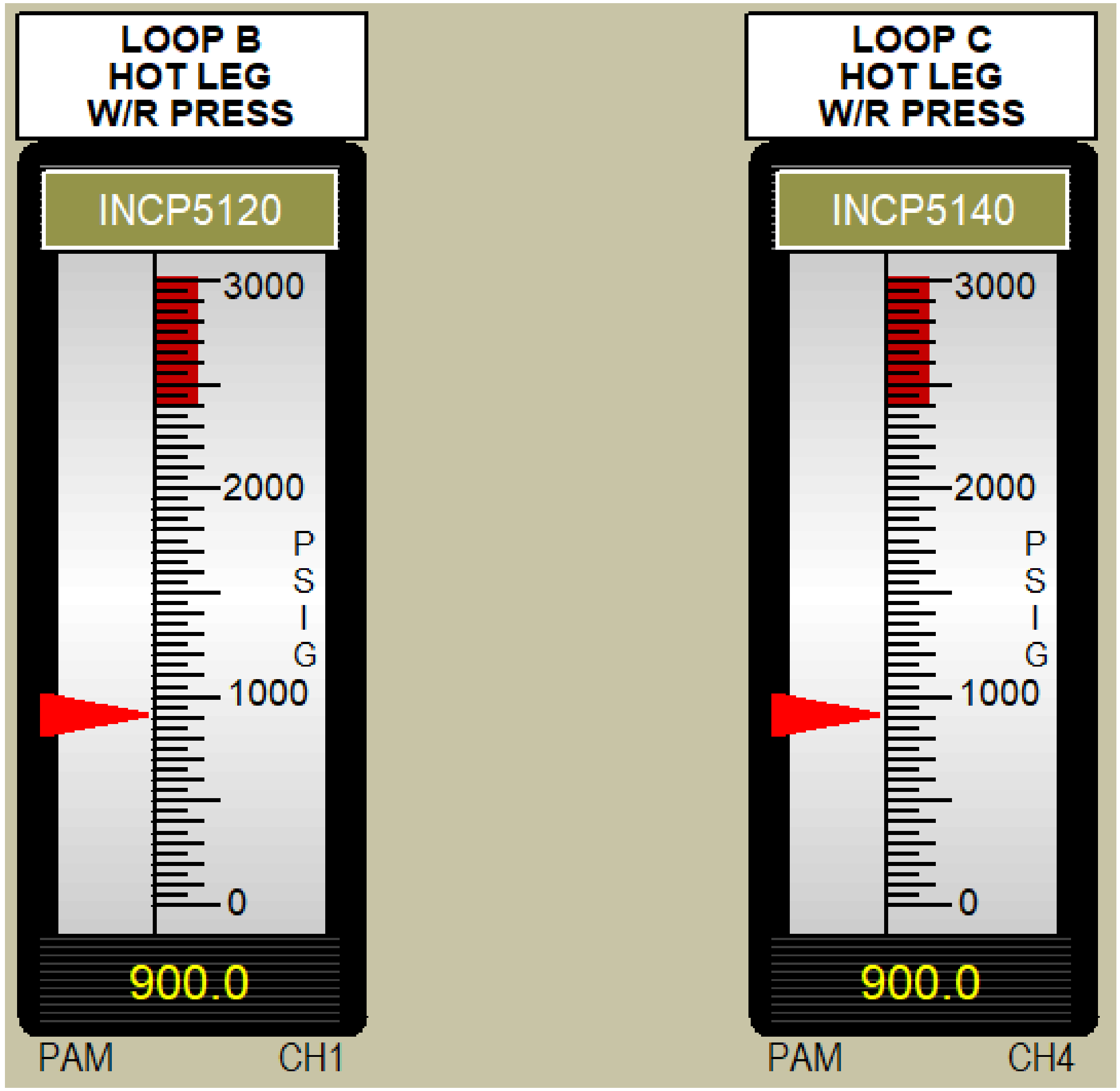
(RETURN TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

- **A LOCA is in progress on Unit 1**

INITIATING CUES:

- **The CRS directs you to calculate and record the maximum reactor vessel head venting time per EP/1/A/5000/FR-I.3 (Response to Voids in Reactor Vessel) Enclosure 2 (Allowable Hydrogen Venting Time)**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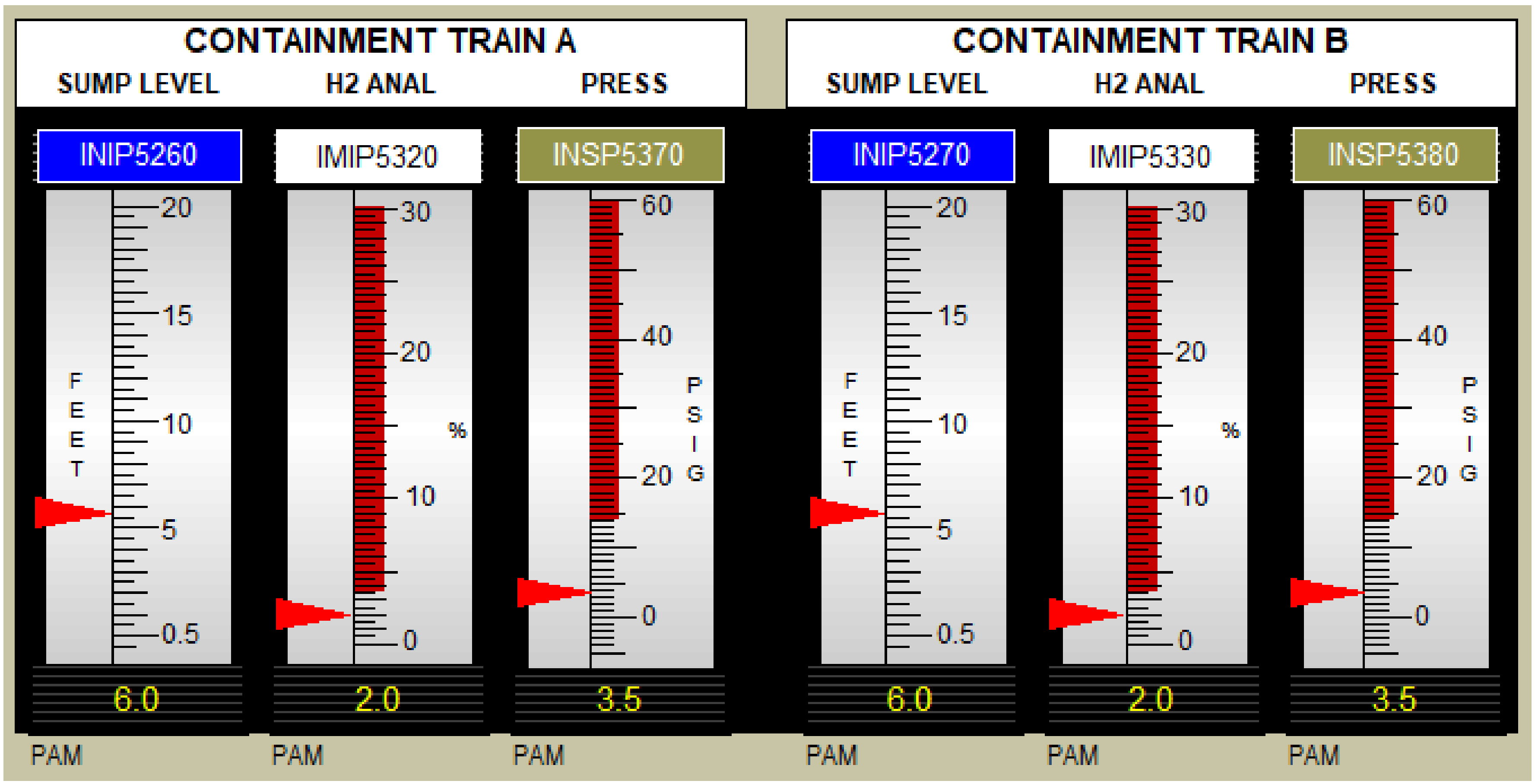
HOT LEG
W/R PRESS
INCP5120
3000
2000
1000
0
PSIG
900.0
PAM
CH1
LOOP C
HOT LEG
W/R PRESS
INCP5140
3000
2000
1000
0
PSIG
900.0
PAM
CH4


CONTAINMENT TRAIN A
SUMP LEVEL
H2 ANAL
PRESS
INIP5260
IMIP5320
INSP5370
FEET
%
PSIG
6.0
2.0
3.5
PAM
PAM
PAM
CONTAINMENT TRAIN B
SUMP LEVEL
H2 ANAL
PRESS
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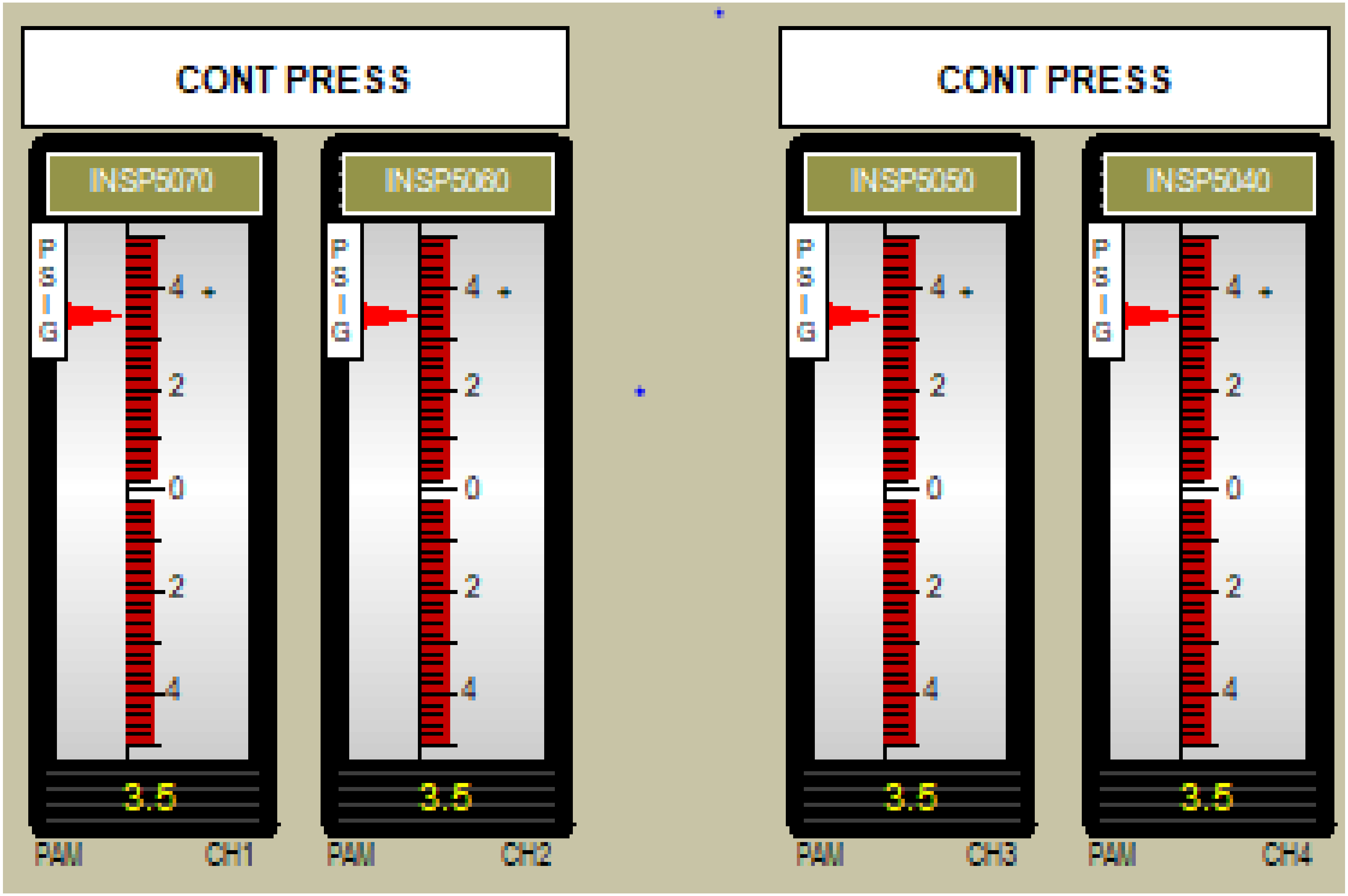
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PSIG
3.5
PAM
CH1
INSP5060
PSIG
3.5
PAM
CH2
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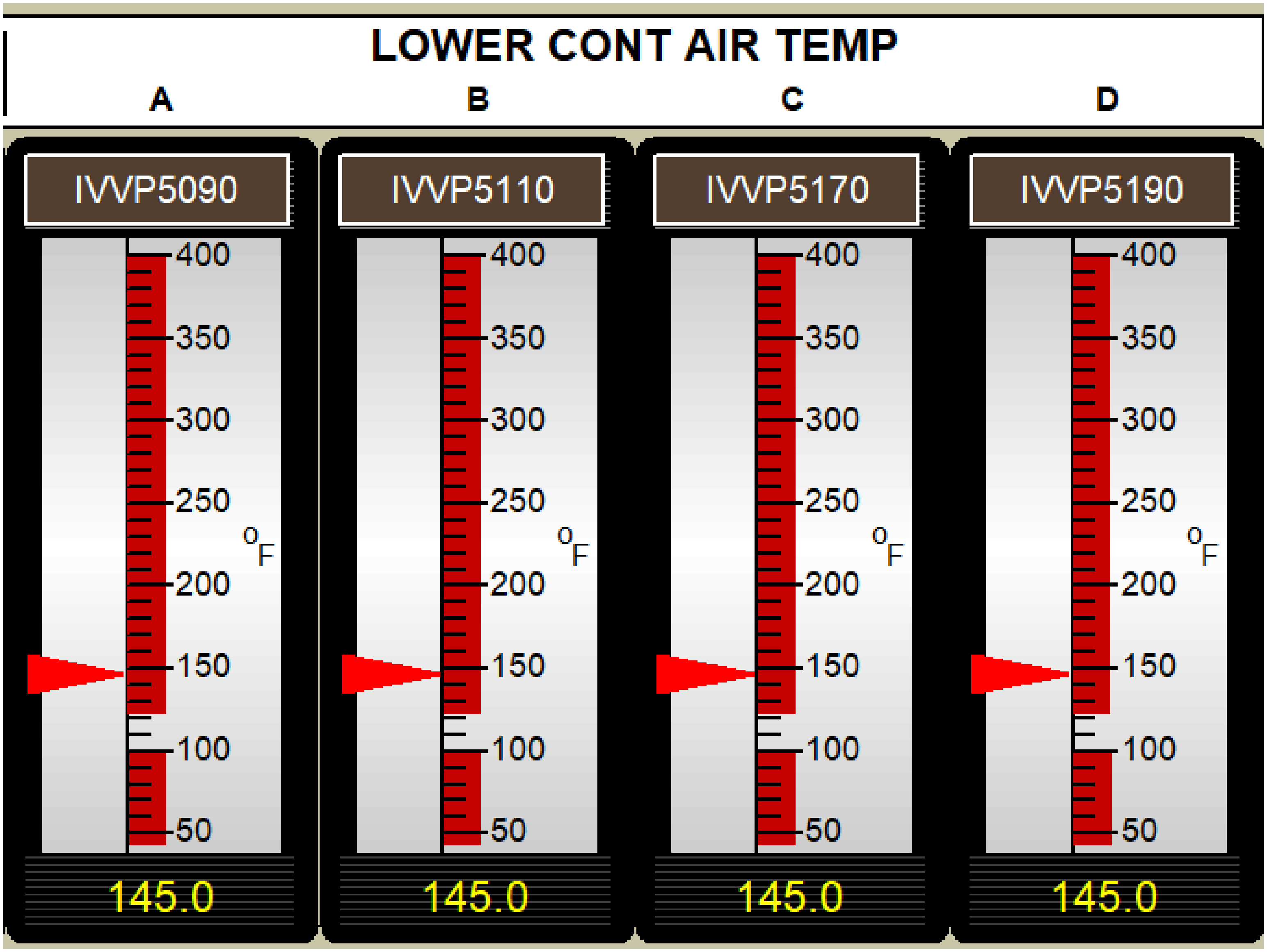
PSIG
3.5
PAM
CH3
INSP5040
PSIG
3.5
PAM
CH4


LOWER CONT AIR TEMP
A
B
C
D
IVVP5090
IVVP5110
IVVP5170
IVVP5190
400
350
300
250
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145.0
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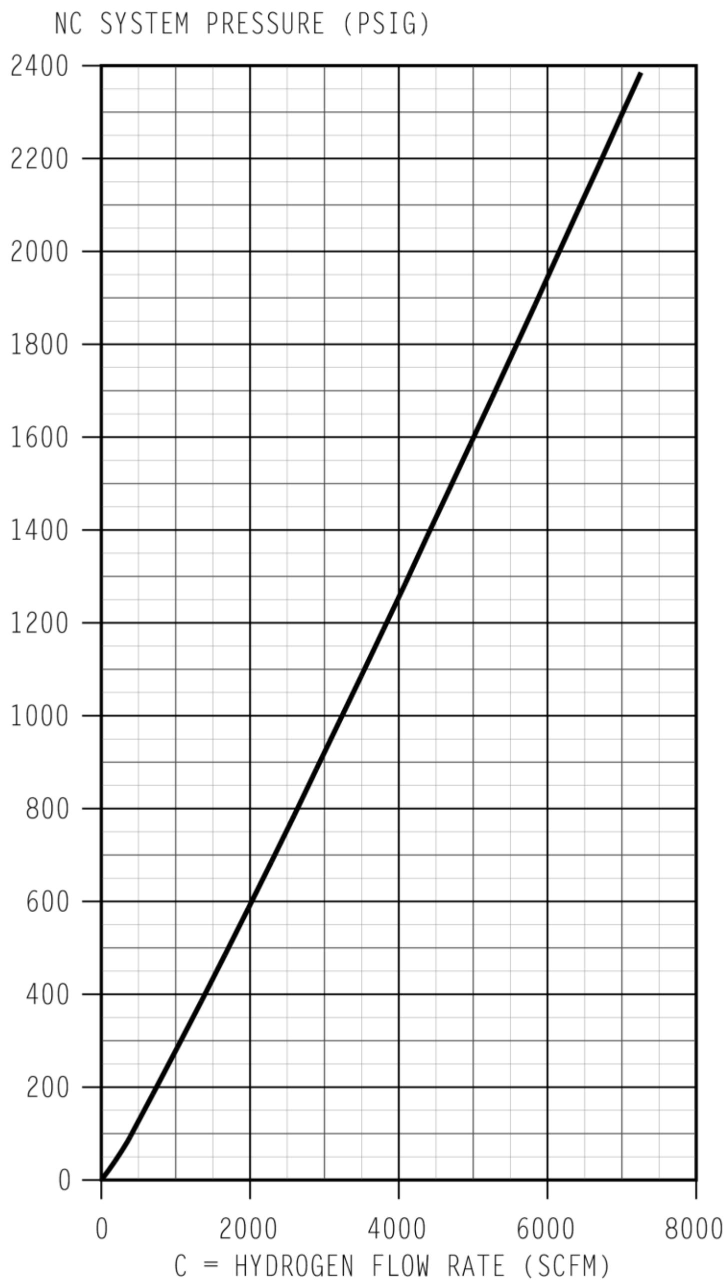
145.0



CALCULATION OF MAXIMUM ALLOWABLE VENTING TIME

STEP 1: Calculate A

$$A = 9{,}500 \times \frac{(P + 14.7)}{14.7} \times \frac{492}{(T + 460)}$$

where: P = Containment pressure (PSIG)
T = Lower containment temperature (deg F)

STEP 2: Calculate B

B = (3 - H) x A

where: H = Containment hydrogen concentration (%)

STEP 3: Determine C from the curve for the current NC system pressure.

STEP 4: Calculate T

T = B/C = Venting time in minutes

NC pressure ______ PSIG. Maximum allowable venting time ______ minutes.

JPM A.1-2R

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September 2021 NRC Exam

EVALUATION SHEET

Task: Determine Rod Insertion Limit Boration

Alternate Path: N/A

Facility JPM #: New

Safety Function: N/A

K/A 2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Importance: 4.1 / 4.3 **CFR:** 41.10 / 43.6 / 45.6

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom __X__ Perform __X__ Simulate ________

References: AP/1/A/5500/003 (Load Rejection) Rev 47 Enclosure 3 (Rod Insertion Limit Boration)

Task Standard: Applicant determines the required amount of boric acid to add to the NC system to restore rods to 10 steps above RIL is 411 - 515 gallons.

Validation Time: 10 minutes **Time Critical:** Yes _____ No __X__

Applicant:
NAME____________________________ Docket #______________ Time Start: __________
Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

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COMMENTS

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Catawba Nuclear Station
Admin. JPM A.1-2R
September 2021 NRC Exam

<u>READ TO APPLICANT</u>

<u>DIRECTION TO APPLICANT:</u>

I will explain the initial conditions and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

<u>INITIAL CONDITIONS:</u>

- Unit 1 has experienced a runback from 85% RTP following a Zone A Lockout
- Current reactor power is 48% RTP
- Unit 1 is at 250 EFPD
- Annunciator 1AD-2, B/9 (CONTROL ROD BANK LO-LO LIMIT) is LIT
- Control Bank D rods are currently 20 steps withdrawn
- Current NC System Boron concentration is 900 ppm

<u>INITIATING CUES:</u>

- The CRS has directed you to borate the NC system as necessary to maintain control rods above the insertion limit per AP/1/A/5500/003 (Load Rejection) Enclosure 3 (Rod Insertion Limit Boration) step 2.
- You are to determine the amount of Boric Acid required to restore control rods to **<u>10 STEPS ABOVE</u>** the required rod insertion limit and record below.

Boric Acid addition required: ____________________ gallons

<u>EXAMINER NOTE</u>:
After reading cue, provide applicant with a copy of AP/1/A/5500/003 Enclosure 3 and the ROD book for Unit 1

START TIME: _______

STEP / STANDARD	SAT / UNSAT
STEP 1: 2.a **IF** initial reactor power was 100%, **THEN** borate NC System as required to restore control rods above insertion limits. **REFER TO** Unit 1 R.O.D. book, section 4.8 (Reactivity Data Sheet). STANDARD: Applicant determines that initial reactor power was 85% per the initiating cue and that this step is not applicable. COMMENTS:	___ SAT ___ UNSAT

STEP 2: 2.b **IF** initial reactor power was less than 100% **OR** Unit 1 R.O.D. book, section 4.8 (Reactivity Data Sheet) is **NOT** available, **THEN** perform the following as required to restore control rods above the insertion limits:	**CRITICAL STEP**
NOTE OAC point C1P1448 (Low Bank Insertion Limit Margin) and R.O.D. Book (Section 2.2) provide rod insertion limit indication. 1) Determine control rod insertion limit. ________ STANDARD: Applicant determines that the rod insertion limit for 48% power is approximately 40 steps withdrawn on Control Bank D (Acceptable Range 39-42 steps). **This step is critical in order to properly determine the correct amount of boric acid to add to restore control rods to above the insertion limit.** COMMENTS:	___ SAT ___ UNSAT

STEP 3: 2.b.2) Calculate "A" (reactivity difference between required rod position and current rod position). **REFER TO** Unit 1 R.O.D. book section 5.6.3. R = Required rod position IRW ______ PCM P = Current rod position IRW _______ PCM (R – P = A _________ PCM).	**CRITICAL STEP**
STANDARD: Applicant determines R for 50 steps withdrawn (40 steps RIL + 10 steps = 50 steps) is 805 PCM (Acceptable Range is 796 PCM – 823 PCM). Applicant determines P for 20 steps withdrawn is 1084 PCM. Applicant then calculates A to be 279 PCM (Acceptable Range is 261 PCM – 288 PCM). **This step is critical in order to properly determine the correct amount of boric acid to add to restore control rods to above the insertion limit.** COMMENTS:	___ SAT ___ UNSAT

STEP 4: 2.b.3) Determine "B" (differential boron worth). **REFER TO** Unit 1 R.O.D. book section 5.5 _________ PCM/PPM. STANDARD: Applicant determines from ROD book section 5.5 that differential boron worth for 250 EFPD is -6.47 PCM/PPM. **This step is critical in order to properly determine the correct amount of boric acid to add to restore control rods to above the insertion limit.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 5: 2.b.4) Calculate "C" (difference in reactivity) as follows: A / B = C ________ PPM. STANDARD: Applicant calculates C to be 43.12 PPM (279 PCM / 6.47 PCM/PPM = 42.97 PPM). (Acceptable Range is 40.34 PPM – 44.51 PPM) **This step is critical in order to properly determine the correct amount of boric acid to add to restore control rods to above the insertion limit.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 6: 2.b.5) Calculate "D" (required boron concentration) as follows: E = Current Boron Concentration _______ PPM. E + C = D ________ PPM STANDARD: Applicant calculates D to be 943 PPM. (900 + 43 = 943 PPM) (Acceptable Range is 940 PPM – 945 PPM) **This step is critical in order to properly determine the correct amount of boric acid to add to restore control rods to above the insertion limit.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP 7: 2.b.6) Determine required boric acid needed to raise NC System boron concentration to value "D" calculated in Step 2.b.5. **REFER TO** Unit 1 R.O.D. book table 4.1 or REACT Boration/Dilution module. ________. STANDARD: Applicant determines that 442 gallons of boric acid are needed to raise NC system boron concentration from 900 PPM to 943 PPM. (Acceptable range is 411 – 515 gallons). **This step is critical in order to properly determine the correct amount of boric acid to add to restore control rods to above the insertion limit.** COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- Unit 1 has experienced a runback from 85% RTP following a Zone A Lockout
- Current reactor power is 48% RTP
- Unit 1 is at 250 EFPD
- Annunciator 1AD-2, B/9 (CONTROL ROD BANK LO-LO LIMIT) is LIT
- Control Bank D rods are currently 20 steps withdrawn
- Current NC System Boron concentration is 900 ppm

INITIATING CUES:

- The CRS has directed you to borate the NC system as necessary to maintain control rods above the insertion limit per AP/1/A/5500/003 (Load Rejection) Enclosure 3 (Rod Insertion Limit Boration) step 2.
- You are to determine the amount of Boric Acid required to restore control rods to **10 STEPS ABOVE** the required rod insertion limit and record below.

Boric Acid addition required: ___________________ gallons

CAUTION **Failure to initiate boration within one hour of exceeding rod insertion limits may violate Tech Spec 3.1.6.**

NOTE OAC point C1L4409 (Ctrl Bank Tech Spec Insertion Lmt Reached) and R.O.D Book (Section 2.2) provide rod insertion limit indication.

1. **IF control rods cannot be maintained above rod insertion limits, THEN perform the following:**

   a. Stop any dilutions in progress.

   b. Ensure control rods restored above insertion limits within 2 hours of exceeding limits.

   c. Ensure compliance with Tech Spec 3.1.6 (Control Bank Insertion Limits).

2. **Perform one of the following to restore control rods above insertion limits:**

   a. **IF** initial reactor power was 100%, **THEN** borate NC System as required to restore control rods above insertion limits. **REFER TO** Unit 1 R.O.D. book, section 4.8 (Reactivity Data Sheet).

2. (Continued)

b. **IF** initial reactor power was less than 100% **OR** Unit 1 R.O.D. book, section 4.8 (Reactivity Data Sheet) is **NOT** available, **THEN** perform the following as required to restore control rods above insertion limits:

NOTE OAC point C1P1448 (Low Bank Insertion Limit Margin) and R.O.D Book (Section 2.2) provide rod insertion limit indication.

1) Determine control rod insertion limit. ________.

2) Calculate "A" (reactivity difference between required rod position and current rod position). **REFER TO** Unit 1 R.O.D. book section 5.6.3.

   R = Required rod position IRW _________ PCM

   P = Current rod position IRW _________ PCM

   (R - P = A __________ PCM).

3) Determine "B" (differential boron worth). **REFER TO** Unit 1 R.O.D. book section 5.5 ______________ PCM/PPM.

4) Calculate "C" (difference in reactivity) as follows:

   A / B = C ______________ PPM.

5) Calculate "D" (required boron concentration) as follows:

   E = Current Boron concentration _________ PPM.

   E + C = D__________ PPM.

6) Determine required boric acid needed to raise NC System boron concentration to value "D" calculated in Step 2.b.5. **REFER TO** Unit 1 R.O.D. book table 4.1 or REACT Boration/Dilution module. ___________.

NOTE

- The boric acid added to the NC System should be added in several increments within the first hour of the runback.
- Due to the post transient Xenon build-in rate, the total boric acid value calculated in Step 2.b.6, may not need to be added to restore control rods above insertion limits.

7) Borate NC System as required to restore control rods above insertion limits.

JPM A.2R

RO

Catawba Nuclear Station
Admin. JPM A.2R
September 2021 NRC Exam

EVALUATION SHEET

Task: Determine NC Subcooling on a loss of OAC

Alternate Path: N/A

Facility JPM #: CCM-003

Safety Function: N/A

K/A G 2.2.12 Knowledge of surveillance procedures.

Importance: 3.7 / 4.1 **CFR:** 41.10 / 45.13

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom __X__ Perform __X__ Simulate ________

References: PT/1/A/4600/009 (Loss of Operator Aid Computer), Unit 1 Revised Databook Figures 57 & 58

Task Standard: Determines Subcooling is 13°F – 28°F and fills out Enclosure 13.8 (Subcooling Data Sheet) per the key. Determines the Acceptance Criteria is NOT met.

Validation Time: 15 minutes **Time Critical:** Yes _____ No __X__

Applicant:
NAME_____________________________ Docket #_______________ Time Start: __________
Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

Catawba Nuclear Station
Admin. JPM A.2R
September 2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 is in Mode 3 and has experienced a loss of OAC**
- **PT/1/A/4600/009 (Loss of Operator Aid Computer) is in progress**
- **Both trains of the plasma display monitor are inoperable**

INITIATING CUES:

- **The CRS directs you to complete Enclosure 13.8 (Subcooling Data Sheet) to determine the °F Subcooled and if this subcooling margin meets the acceptance criteria.**

	T cold	T hot	T ave	D/T		
Loop A	445	448	447	3		
Loop B	446	450	448	4		
Loop C	442	444	443	2		
Loop D	443	446	445	3		
5 Highest Core Exit T/C	454	450	453	451	452	
	Chan 1	Chan 2	Chan 3	Chan 4	W/R B	W/R C
Pressure	1700	1700	1700	1700	600	620

EXAMINER NOTE: **Each applicant should receive a copy of PT/1/A/4600/009 Enclosure 13.8 and a copy of the Unit 1 Revised Data Book.**

START TIME: _______

STEP 1: Record lowest indicated system pressure. STANDARD: Applicant determines that 600 psig is the lowest pressure and records this value on the table in Enclosure 13.8. COMMENTS:	___ **SAT** ___ **UNSAT**

STEP 2: T-Sat – Using NC pressure, determine saturation temperature from the Unit 1 Revised Data Book Figure 57 or Figure 58. STANDARD: Applicant determines T-Sat is 465°F-480°F and records this value on the table in Enclosure 13.8. **This step is critical in determining the actual amount of subcooling on Unit 1 to meet the JPM standard.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___**UNSAT**

STEP 3: Highest NC Temp – Determine the highest NC Temp:
 	
• In Modes 1 and 2, use Loop Thot
• In Modes 3-6:
  ○ Compare the average of the 5 highest reading operable core exit T/Cs to Loop Thot.
  OR
  ○ Use the operating train(s) of ND inlet temperature, Loop Thot and/or the operable core exit T/Cs.
 	
STANDARD:
 	
Applicant determines that the average of the 5 highest core exit T/Cs is the hottest and calculates it to be 452°F and records this value on the table in Enclosure 13.8.
 	
This step is critical in determining the actual amount of subcooling on Unit 1 to meet the JPM standard.
 	
COMMENTS:
	CRITICAL STEP
	
	___ **SAT**
	
	___ **UNSAT**

STEP 4 °F Subcooled – Calculate by subtracting “HIGHEST NC TEMP” from “T-Sat”. STANDARD: Applicant calculates °F Subcooled to be 13°F-28°F and records this value on the table in Enclosure 13.8. **This step is critical in determining the actual amount of subcooling on Unit 1 to meet the JPM standard.** COMMENTS:	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STEP 5 Determine if the acceptance criteria is met. STANDARD: Applicant determines that °F Subcooled is < the required of 30°F while shutdown and therefore the acceptance criteria is NOT met. **This step is critical in determining that the amount of subcooling present does not meet acceptance criteria for the current mode.** COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ **SAT** ___ **UNSAT**

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

INITIAL CONDITIONS:

- **Unit 1 is in Mode 3 and has experienced a loss of OAC**
- **PT/1/A/4600/009 (Loss of Operator Aid Computer) is in progress**
- **Both trains of the plasma display monitor are inoperable**

INITIATING CUES:

- **The CRS directs you to complete Enclosure 13.8 (Subcooling Data Sheet) to determine the °F Subcooled and if this subcooling margin meets the acceptance criteria.**

	T cold	T hot	T ave	D/T		
Loop A	445	448	447	3		
Loop B	446	450	448	4		
Loop C	442	444	443	2		
Loop D	443	446	445	3		
5 Highest Core Exit T/C	454	450	453	451	452	
	Chan 1	Chan 2	Chan 3	Chan 4	W/R B	W/R C
Pressure	1700	1700	1700	1700	600	620

°F Subcooled: ___________

Acceptance Criteria met (Yes/No): ___________

Enclosure 13.8

Subcooling Data Sheet

CALCULATION SHEET FOR NC SYSTEM DEGREES SUBCOOLED

Date/Time	NC Press	T-SAT	Highest NC Temp	°F Subcooled	Initials

ACCEPTANCE CRITERIA:

Subcool limit is 10°F while at power.
30°F while shutdown.

INFORMATION:

NC Pressure - Record lowest indicated system pressure.

T-SAT - Using NC pressure, determine saturation temperature from the Unit One Revised Data Book Figure 57 or Figure 58.

Highest NC Temp - Determine the highest NC Temp:

- In Modes 1 and 2, use Loop T_{HOT}.
- In Modes 3-6:
  - Compare the average of the 5 highest reading operable core exit T/Cs to Loop T_{HOT}. (This is the preferred method)
    OR
  - Use the operating train(s) of ND inlet temperature, Loop T_{HOT} and/or the operable core exit T/Cs.

°F Subcooled - Calculate by subtracting "HIGHEST NC TEMP" from "T-SAT".

JPM A.3R

RO

Catawba Nuclear Station
Admin. JPM A.3R
September 2021 NRC Exam

EVALUATION SHEET

Task: Determine Radiation Protection Requirements for an activity

Alternate Path: N/A

Facility JPM #: 2019 NRC Exam Admin JPM A.3R

Safety Function: N/A

K/A G 2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Importance: 3.4 / 3.8 **CFR:** 41.12 / 43.4 / 45.10

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom __X__ Perform __X__ Simulate ______

References: Radiation Work Permit # 5021 Task 1, Room 105 (ND Pump 1A) Survey Map

Task Standard: Correctly determine that total dose received for the job is 13 mR and maximum additional time allowed at Low Exposure Waiting Area is 42 minutes prior to exceeding 80% of RWP dose limits.

Validation Time: 10 minutes **Time Critical:** Yes _____ No __X__

Applicant:
NAME____________________________ Docket #______________ Time Start: __________
Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

COMMENTS

Catawba Nuclear Station
Admin. JPM A.3R
September 2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Unit 1 has entered AP/1/A/5500/019 (Loss of Residual Heat Removal)**
- **The CRS has dispatched an AO to 1A ND pump room to vent the 1A ND pump casing when directed.**
- **The AO has staged himself to minimize dose while waiting.**
- **The following is the timeline for the venting evolution:**

0730 – AO enters 1A ND Pump Room.
0800 – AO is directed to vent using 1ND-88 (1A ND Pump Seal Water Hx Inlet Vent).
0812 – Venting is complete. AO returns to the LEWA to await further instruction.

INITIATING CUES:

Based on the above time line, RWP # 5021, and Survey Maps provided and discounting any dose received during transit:

1. **State the amount of dose that has been received thus far.**

   Amount of dose received - ________________ mR

2. **Following completion of venting activity, calculate how much longer the AO can remain in the room until required to exit (based on exceeding 80% of RWP allowable dose).**

   Allowable time in room following venting activity – ___________ .

EXAMINER NOTE: Each applicant should receive a copy of RWP # 5021 (Task 1) and the Room Survey for rooms 105 and 110.

Catawba Nuclear Station
Admin. JPM A.3R
September 2021 NRC Exam

START TIME: _______

STEP / STANDARD	SAT / UNSAT
STEP 1 Determine dose received for the venting evolution: AO is to stage themselves at the Low Exposure Waiting Area (LEWA) where they wait for 30 minutes (.5 hours) LEWA dose: **10 mR/hr X .5 hr = 5 mR** Venting time is 12 minutes (.2 hours) Area dose: **40 mR/hr X .2 hr = 8 mR** Total amount of dose received = **5 mR + 8 mR = 13 mR** STANDARD: Applicant determines the total amount of dose received for the evolution to be 13 mR. **This step is critical to meet the task requirements and standard for this JPM to determine the total amount of dose received.** COMMENTS:	**CRITICAL STEP** **___ SAT** **___ UNSAT**

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>
<u>STEP 2</u> Determine allowable time in room following venting activity.

The RWP limits total exposure to 25 mR

25 mR x 80% = 20 mR total allowable dose

20 mR – 13 mR (dose following venting) = 7 mR (remaining)

7 mR ÷ 10 mR/hr = 0.7 hr (42 minutes)

<u>STANDARD</u>:

Applicant determines they can remain in the room another 42 minutes, or until 0854.

This step is critical to meet the task requirements and standard for this JPM to determine the latest time that venting could complete without exceeding RWP dose requirements.

<u>COMMENTS:</u>

END OF TASK
</td><td>
CRITICAL STEP

___ SAT

___UNSAT
</td></tr>
</table>

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

<u>DIRECTION TO APPLICANT:</u>

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<u>INITIAL CONDITIONS:</u>

- **Unit 1 has entered AP/1/A/5500/019 (Loss of Residual Heat Removal)**
- **The CRS has dispatched an AO to 1A ND pump room to vent the 1A ND pump casing when directed.**
- **The AO has staged himself to minimize dose while waiting.**
- **The following is the timeline for the venting evolution:**

0730 – AO enters 1A ND Pump Room.
0800 – AO is directed to vent using 1ND-88 (1A ND Pump Seal Water Hx Inlet Vent).
0812 – Venting is complete. AO returns to the LEWA to await further instruction.

<u>INITIATING CUES:</u>

Based on the above time line, RWP # 5021, and Survey Maps provided and discounting any dose received during transit:

1. State the amount of dose that has been received thus far.

Amount of dose received - ________________ mR

2. Following completion of venting activity, calculate how much longer the AO can remain in the room until required to exit (based on exceeding 80% of RWP allowable dose).

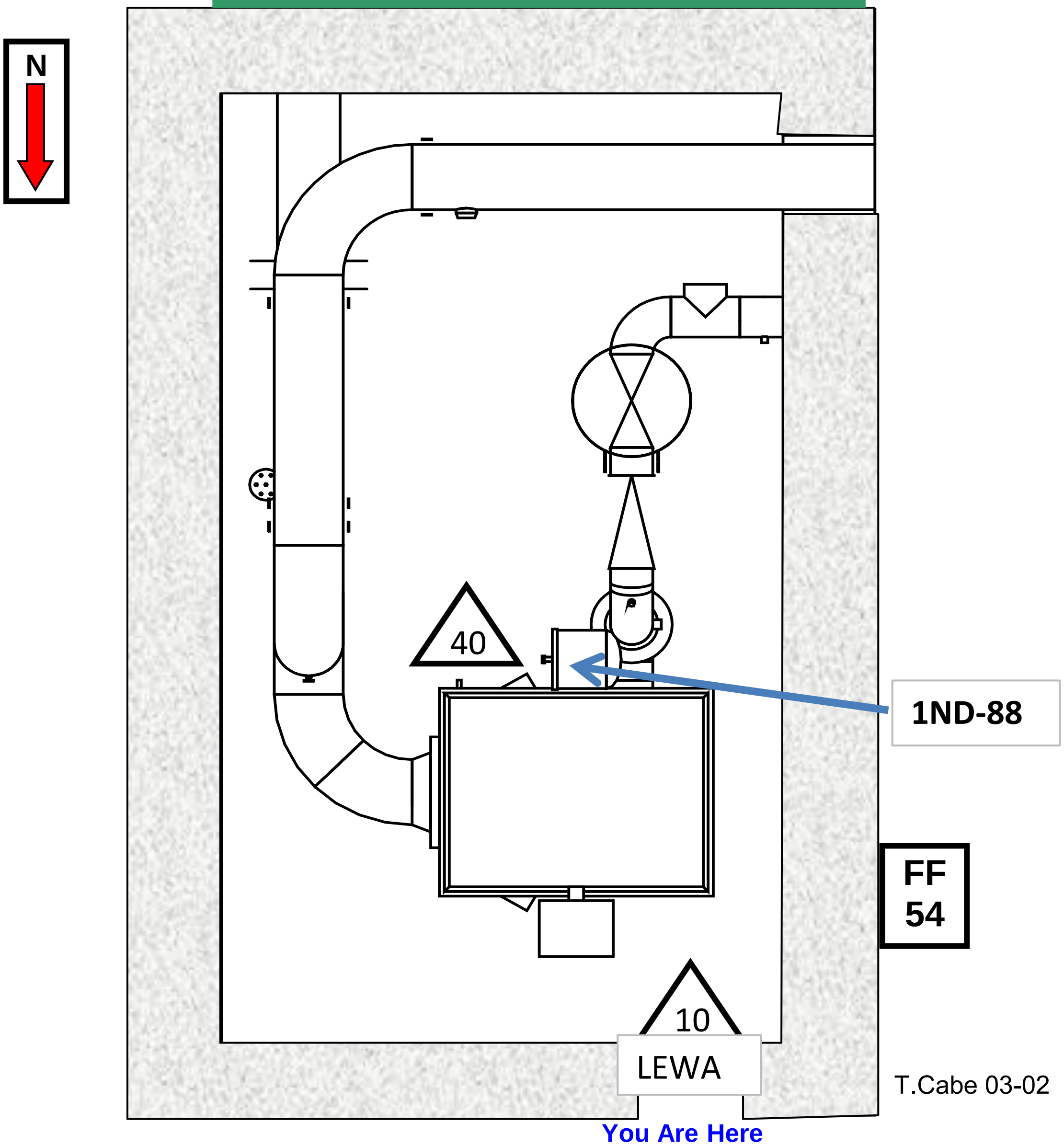
Allowable time in room following venting activity – ___________ .

CATAWBA NUCLEAR STATION
MONTHLY ROUTINE SURVEY SHEET

PERFORMED BY: ______________________ / ______________ DATE/TIME: ______________ / ________
(print name) (initials)

Instrument/#(s) ______________________ βγ Counter/#: ______________ Reviewed By: ________ (initials)

Room 105 ND Pump 1A



HIGHEST GA*: ____________ **HIGHEST CONTACT*:** ____________ **HIGHEST dpm/100cm²:** ____________ K βγ

HIGHEST MASSLINN CCPM: ____________

*** Radiation readings in mRem/hr unless otherwise noted.**

LEGEND: **LEWA** = LOW EXPOSURE WAITING AREA **HS** = HOT SPOT --x----x----x-- = RCZ BOUNDARY

= DIRECT FRISK = GENERAL AREA > = CONTACT = SMEAR (dpm/100cm²) = MASSLINN

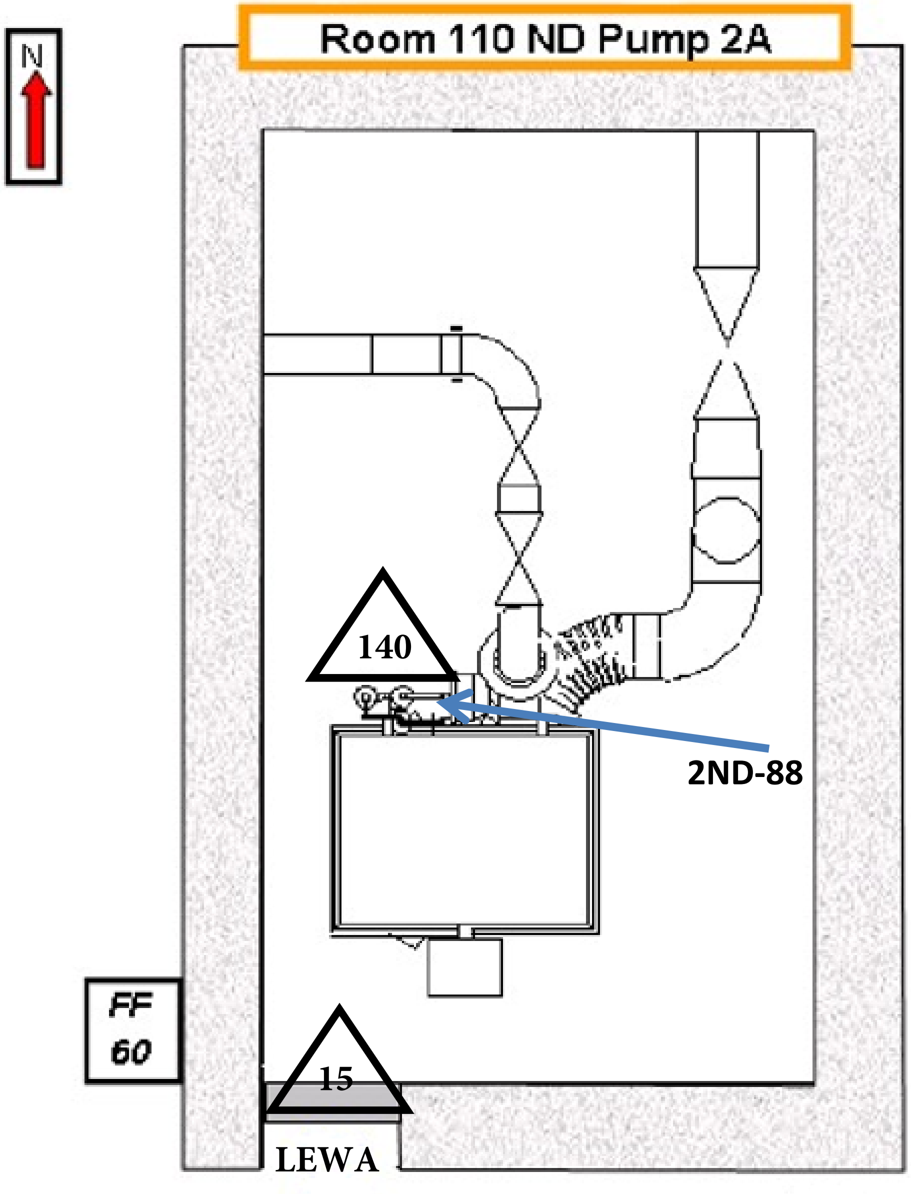
Comments: ____

RWP#15

CATAWBA NUCLEAR STATION
MONTHLY ROUTINE SURVEY SHEET

PERFORMED BY: ______________________ / ____________ DATE/TIME: ______________ / ________
(print name) (initials)

Instrument/#(s) ______________________ βγ Counter/#: __________ Reviewed By: ________
(initials)



You Are Here

HIGHEST GA*: __________ **HIGHEST CONTACT*:** __________ **HIGHEST dpm/100cm^2:** __________ K βγ
HIGHEST MASSLINN CCPM: __________

*** Radiation readings in mRem/hr unless otherwise noted.**

LEGEND: **LEWA** = LOW EXPOSURE WAITING AREA **HS** = HOT SPOT --x----x----x-- = RCZ BOUNDARY

= DIRECT FRISK = GENERAL = CONTACT = SMEAR **(dpm/100cm^2)** = MASSLINN

Comments: ______________________________

RWP#15

INFORMATION USE ONLY

Catawba Nuclear Station
Radiation Work Permit

INFORMATION USE ONLY

AUX BUILDING ENTRY INTO PENTRATION ROOMS HEAT EXCHANGER ROOMS AND PUMP ROOMS TO PERFORM VENTING ON ECCS SYSTEMS | **RWP # 5021** | **Rev: 11**

Task # 1

AUX BUILDING ENTRY INTO PENTRATION ROOMS, HEAT EXCHANGER ROOMS AND PUMP ROOMS TO PERFORM VENTING ON ECCS SYSTEMS

ED Alarm Set Points:

Dose Alarm: 25 mrem **Dose Rate Alarm: 50 mrem/hr**

RWP Requirements

Dress Category/Work Description

- Dress Category "F" 1. Complete protection of skin and clothing is NOT required. 2. Radioactive material is handled and/or transported AND the potential for loose surface contamination >1000 dpm/100cm2 exists AND durability of surgical gloves is sufficient.
- E Dress Category "G" Entry into dry contaminated areas <25,000 dpm/100cm2 with NO climbing or physical / strenuous work. NO brushing, grinding, lapping, etc. is allowed.
- Dress Category "H" Entry into dry contaminated areas <25,000 dpm/100cm2 with NO climbing or physical / strenuous work. NO brushing, grinding, lapping, etc. is allowed.
- Dress Category "I" Work in dry contaminated areas <50,000 dpm/100cm2 with a risk of puncturing or tearing gloves or beta dose concerns to the hands. Work may involve brushing, grinding, lapping, etc. (1) (3) (4)
- Dress Category "N" Wet work, hot particle controls are required or work in highly contaminated areas. Work may involve brushing, grinding, lapping, etc. (1)
- Dress Category "Z" Special dress. See Additional Instructions.
- Modesty garments, top & bottom, are required under protective clothing where personal outer clothing is not worn
- (1) Cloth coverall are acceptable for use when allowed by RP. Gloves and booties must be secured (e.g. taped, elastic cuff) when wearing cloth coveralls
- (2) IF double SOP is not used when wearing double PCs, remove the outer layers at the source
- (3) Skull caps may be substituted for a hood when approved by RP and NO hands on work is to be performed.
- (4) For activities requiring crawling, kneeling, etc., review need for additional barrier to prevent contamination events, e.g., additional protective clothing, knee pads, use of floor covering, etc.

Protective Clothing

- F - Lab coat, glove liners, rubber gloves OR surgical gloves, booties and shoe covers
- G - Hood, coveralls, glove liners and rubber gloves, booties, and shoe covers over personal clothing (NO modesty clothing required).
- H - Hood, coveralls, glove liners and rubber gloves, booties and shoe covers, NO personal outer clothing.
- I - Hood, coveralls, glove liners, 2 pair rubber gloves, booties and shoe covers, no personal outer clothing.
- N - Hood, coveralls, water resistant/water proof suit, glove liners, 2 pair rubber gloves, booties, 2 pair shoe covers, no personal outer clothing.
- Z - Special dress

Contamination Control

- Wipe down AND bag all tools and equipment prior to removal from a contaminated area as directed by RP
- Utilize facial protection (e.g. face shield, hood sock, power visor) as directed by RP
- Install catch containments OR drain rigs to prevent spills if draining components
- If installing a drain rig, use hose clamps to secure hose OR tubing connections
- If installing a drain rig, secure hose OR tubing to floor drain
- Wear disposable (plastic) booties inside of orex booties for work in wet conditions
- Change outer rubber gloves often when handling highly contaminated material as directed by RP

INFORMATION USE ONLY

Catawba Nuclear Station
Radiation Work Permit

INFORMATION USE ONLY

AUX BUILDING ENTRY INTO PENTRATION ROOMS HEAT EXCHANGER ROOMS AND PUMP ROOMS TO PERFORM VENTING ON ECCS SYSTEMS **RWP # 5021** **Rev: 11**

Task # 1

AUX BUILDING ENTRY INTO PENTRATION ROOMS, HEAT EXCHANGER ROOMS AND PUMP ROOMS TO PERFORM VENTING ON ECCS SYSTEMS

ED Alarm Set Points:

Dose Alarm: 25 mrem

Dose Rate Alarm: 50 mrem/hr

RWP Requirements

- Use surgical gloves in lieu of rubber gloves for the manipulation of small or specialty items as directed by RP

RP Job Coverage

- Start of Job, Intermittent or No Coverage In Radiation Areas or Less
- RP Coverage Required To Transport Material > 5 mrem/hr at 30 cm
- Pre-job briefing required
- Continuous RP Coverage for aggressive work in Alpha Level III areas or Alpha Level II areas with beta-gamma to alpha ratios less than 3000:1 or where conditions could change

Dosimetry Requirements

- Monitor ED periodically while inside the RCA/RCZ (once or twice per hour in low dose rate areas). Monitor more frequently in higher dose rate areas, for example every 10 to 15 minutes.
- If dress requirements prevent the monitoring of ED, and RP is not remotely monitoring (via teledose & communications), place ED external to the outmost layer of protective clothing for monitoring

Respiratory Protection

- If weighted DAC-Hours are expected to result in greater than or equal to 4 DAC-Hours per person, perform a TEDE/ALARA evaluation
- Full Face Particulate (Additional Hood Required) IF warranted by TEDE ALARA Evaluation OR directed by RP
- Personal (lapel) air samplers required for Alpha Level III areas or Alpha Level II areas with beta-gamma to alpha ratios of less than 3000:1

RP Hold Points

- Breaching Contaminated System
- RP Survey Required Prior to Handling Debris or Foreign Material
- RP survey required after removal of items from contaminated systems. Decon may be necessary (as directed by RP)
- Notify RP prior to reaching OR entry into the overhead (8 feet and above)
- Accumulated Dose Higher than Expected
- Notify RP Prior to Start of Work
- A change in Alpha Level (AL I to AL II or AL III; AL II to AL III) requires additional planning for alpha considerations

Stop Work Criteria

- Dose Alarm
- Unexpected dose rate alarm
- Airborne conditions higher than expected
- Actual dose rates are higher than the expected levels written on this RWP task
- Actual contamination levels are higher than the expected levels written on this RWP task

INFORMATION USE ONLY

Catawba Nuclear Station
Radiation Work Permit

INFORMATION USE ONLY

AUX BUILDING ENTRY INTO PENTRATION ROOMS HEAT EXCHANGER ROOMS AND PUMP ROOMS TO PERFORM VENTING ON ECCS SYSTEMS

RWP # 5021

Rev: 11

Task # 1

AUX BUILDING ENTRY INTO PENTRATION ROOMS, HEAT EXCHANGER ROOMS AND PUMP ROOMS TO PERFORM VENTING ON ECCS SYSTEMS

ED Alarm Set Points:

Dose Alarm: 25 mrem

Dose Rate Alarm: 50 mrem/hr

RWP Requirements

- Unexpected wet conditions
- Work scope changes
- If monitoring of the ED indicates that the dose alarm set point will be exceeded prior to completing the job, leave the area and contact RP. Do not wait to receive an alarm before exiting the area
- Failure of OR sweat soaked protective clothing

Expected Radiological Conditions

Expected radiological conditions:
General Area Dose Rates: <0.1 mrem/hr - 50 mrem/hr
High Contact Dose Rates: <0.1 mrem/hr - 1000 mrem/hr

Contamination Levels: < 1000 dpm/100cm2- 100,000 dpm/100cm2

Additional Instructions

Electronic Dosimeter rate alarms are established based on general area dose rates. If personnel are positioned in close proximity to primary piping and equipment they may anticipate receiving a dose rate alarm.

Z Dress - Orex Coveralls only. This is for use in clean areas only due to potential for contamination from wearing a fall harness.

JPM A.1-1S

SRO

Catawba Nuclear Station
Admin. JPM A.1-1S
Sept 2021 NRC Exam

EVALUATION SHEET

Task: Calculate Boric Acid and Water Addition to FWST and determine Tech Spec actions.

Alternate Path: N/A

Facility JPM #: 2017 NRC Exam JPM A.1-2S

Safety Function: N/A

K/A 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Importance: 4.3 / 4.4 **CFR:** 41.10 / 43.5 / 45.2 / 45.6

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom __X__ Perform __X__ Simulate ________

References: OP/1/A/6200/014 (Refueling Water System), Enclosure 4.4 (FWST Makeup From Blender)

Task Standard: Applicant determines required boric acid addition of 368.8 gallons (368-370 acceptable) and reactor makeup water addition of 3402.2 gallons (3401-3403 acceptable) to complete the required FWST makeup and determines the following SLC/TS required LCO entries: 1000 - SLC 16-9.12 Condition D and TS 3.5.4 Condition B, 1100 – SLC 16-9.12 Condition D and TS 3.5.4 Condition B and Condition C.

Validation Time: 30 minutes **Time Critical:** Yes _____ No __X__

Applicant:
NAME____________________________ Docket #_______________ Time Start: __________
Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

===

COMMENTS

____

____

____

____

____

____

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Admin. JPM A.1-1S
Sept 2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit 1 is in Mode 3 during a refueling outage. The following events have occurred at the given times:

- **0945**: 1AD-9 B/8 “FWST AT MAKEUP LEVEL” annunciator received. Operators are dispatched to investigate decreasing Unit 1 FWST level.
- **1000**: Unit 1 BOP reports that FWST volume is 377,536.9 gallons.
- **1015**: An improperly executed FW system clearance is discovered and determined to be the cause of FWST level decrease.
  Current conditions are as follows:
  - The valve lineup has been immediately corrected securing the level decrease.
  - Current FWST level is 95%.
  - Current FWST boron concentration is 2740 ppmB.
  - Unit 1 BAT boron concentration is 7500 ppmB.
  - Unit 2 BAT boron concentration is 7300 ppmB.
  - Unit 1 RMWST boron concentration is 4 ppmB.
  - Total Blender Makeup flowrate = 90 gpm

INITIATING CUES:

Using OP/1/A/6200/014 (Refueling Water System), perform an FWST makeup from the blender as follows:

- **Consider any initial conditions complete.**
- **Final FWST level of 96% at 2720 ppmB.**
- **S. Jackson (Primary Chemistry) has been notified and directed use of the Unit 1 BAT.**
- **Concurrent Verification is waived for this task.**

Boric Acid __________________ **RMWST Water** __________________

1. **Assuming the required makeup begins at time 1030, determine which Active <u>SLC and TS</u> LCO entries (if any) will be required at 1000 and 1100.**

1000: ____

1100: ____

Catawba Nuclear Station
Admin. JPM A.1-1S
Sept 2021 NRC Exam

<u>EXAMINER NOTE</u>:
After reading cue, provide applicant with a copy of OP/1/A/6200/014.

STEP / STANDARD	SAT / UNSAT

QUESTION 1

START TIME: _______

STEP / STANDARD	SAT / UNSAT
NOTE: To prevent an inadvertent reduction in the shutdown margin during NC fill evolutions, makeup to the FWST shall be performed with a water source having a Boron concentration which is greater than or equal to the required NC system Boron concentration. STEP 1: 3.1 Notify Chemistry that the following Boric Acid Tank will be used for Tech Spec blending calculations {PIP 91-0449}: Person notified ________________________ _____________ Boric Acid Tank #1 _____________ Boric Acid Tank #2 STANDARD: From the cue, applicant records S. Jackson as Chemistry person notified and checks Boric Acid Tank #1. COMMENTS:	___ SAT ___ UNSAT

STEP / STANDARD	SAT / UNSAT
STEP 2: 3.2 Document current boron concentration in the Boric Acid Tank as specified by Chemistry in Step 3.1 {PIP 91-0449}: _____________ ppmB STANDARD: From the cue, applicant determines that BAT #1 will be used and records 7500 ppmB. COMMENTS:	___ SAT ___ UNSAT
NOTE The final boron concentration used in the next step shall comply with COLR requirements for the current mode of operation. STEP 3: 3.3 Determine the boron concentration of the makeup water to the FWST (C_f) to attain the final boron concentration, C_{FW}. 3.3.1 Initial volume of FWST _______ = V_i. STANDARD: Using supplied information applicant enters the initial level of 373,875 gallons. COMMENTS:	___ SAT ___ UNSAT

STEP / STANDARD	SAT / UNSAT
STEP 4: 3.3.2 Final volume of FWST after makeup _______ = V_{FW}. STANDARD: Using supplied information applicant enters the final level of 377,646 gallons. COMMENTS:	___ SAT ___ UNSAT
STEP 5: 3.3.3 Compute the total gallons of makeup water to be added to the FWST _________ = V_f.. STANDARD: Applicant calculates, and records, makeup volume of 3,771 gallons. 377,646 - 373,875 = 3,771 **Examiner Note: This step is critical to ensure accurate result for final calculation.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP / STANDARD	SAT / UNSAT
STEP 6: 3.3.4 Initial boron concentration of water in the FWST _______ = C_i. STANDARD: From the cue, applicant records initial FWST boron concentration of 2740 ppmB. COMMENTS:	___ SAT ___ UNSAT
STEP 7: 3.3.5 Solve for the desired makeup water boron concentration, Cf. $C_f = \dfrac{C_{FW} V_{FW} - C_i V_i}{V_f}$ $C_f =$ ___________ STANDARD: Applicant calculates desired concentration of makeup water to be 737.1 ppmB $C_f =$ (2720 x 377646) – (2740 x 373875) / 3771 = 737.1 ppmB (737-738 ppmB acceptable) **Examiner Note: This step is critical to ensure accurate result for final calculation.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP / STANDARD	SAT / UNSAT
<u>STEP 8:</u> 3.4 Determine the amount of boric acid and reactor makeup water to add as follows: 3.4.1 Boron Concentration of water in BAT _________ = C_1. <u>STANDARD</u>: From the cue, applicant records Unit 1 BAT boron concentration of 7500 ppmB. <u>COMMENTS:</u>	___ SAT ___ UNSAT
<u>STEP 9:</u> 3.4.2 Boron concentration of water in RMWST __________ = C_2.: <u>STANDARD</u>: From the cue, applicant records Unit 1 RMWST boron concentration of 4 ppmB. <u>COMMENTS:</u>	___ SAT ___ UNSAT

STEP / STANDARD	SAT / UNSAT
STEP 10: 3.4.3 Total gallons of makeup water to be added to FWST ________ = V_f from Step 3.3.3.: STANDARD: Applicant records total makeup volume of 3,771 from step 3.3.3. COMMENTS:	___ SAT ___ UNSAT
STEP 11: 3.4.4 Solve for the amount of boric acid to be added (V_1) using C_f obtained in Step 3.3.5. $V_1 = \frac{V_f (C_f - C_2)}{C_1 - C_2}$ V_1 = ____________ STANDARD: Applicant calculates boric acid total of 368.8 gallons. V_1 = 3771 (737.1 – 4) / (7500 – 4) = 368.8 (368-370 acceptable) **Examiner Note: This step is critical to ensure accurate result for final calculation.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP / STANDARD	SAT / UNSAT
STEP 12: 3.4.5 If V_1 is negative, contact the OWPM Staff for instruction on adjusting the boron concentration in the FWST. STANDARD: Applicant determines that this step does not apply. COMMENTS:	___ SAT ___ UNSAT

STEP 13: 3.4.6 Solve for the amount of RMWST water to be added (V_2). $V_2 = V_f - V_1$ $V_2 =$ __________ gal. STANDARD: Applicant calculates required RMWST water to be 3402.2 gallons. 3771 – 368.8 = 3402.2 (3401-3403 acceptable) **Examiner Note: This step is critical to ensure accurate result for final calculation.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP / STANDARD	SAT / UNSAT
QUESTION 2	**CRITICAL STEP**
1000: SLC 16-9.12 Condition D / LCO 3.5.4, Condition B The minimum volume requirement of SLC 16-9.12 (Boration Systems Borated Water Sources – Operating) is not met. Therefore, Condition D (RWST non-functional) entry will be required. The Required Action associated with this condition is to declare the RWST inoperable and enter the applicable conditions of TS 3.5.4 Immediately. The requirement of SR 3.5.4.2 (Verify RWST borated water volume is ≥ 377,537 gallons) is not met. Therefore, TS 3.5.4, Condition B (RWST inoperable for reasons other than Condition A) entry will be required. The Required Action associated with this condition is to Restore RWST to OPERABLE status within 1 hour. **NOTE:** The applicant may also determine SLC 16-9.8 (Boration Systems Flow Paths - Operating), Condition A (One required Boration System Flow Path non-functional) applies if familiar with the associated plant testing procedure. This procedure requires the FWST to be functional in order to meet the acceptance criteria. However, the surveillance requirements of this SLC do not list this particular requirement so this determination will not be critical to this task. **1100: SLC 16-9.12 Condition D / LCO 3.5.4, Conditions B & C** With an initial volume of 373,875 gallons and an available makeup rate of 90 gpm (beginning at 1030), the total FWST volume at 1100 will be 376,575 gallons which remains below the required minimum of SR 3.5.4.2. 373,875 (initial volume) + {90 gpm x 30 min} = 376,575 gallons Therefore, SLC 16-9.12 Condition D and TS 3.5.4 Condition B will remain in effect **AND** TS 3.5.4 Condition C (Required Action and associated Completion Time not met) must be entered. **Examiner Note: Determination of SLC 16-9.12 and TS 3.5.4 required actions (only) are critical to ensure proper actions are entered to address this adverse condition.** **END OF TASK**	___ SAT ___ UNSAT

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

Unit 1 is in Mode 3 during a refueling outage. The following events have occurred at the given times:

- **0945**: 1AD-9 B/8 "FWST AT MAKEUP LEVEL" annunciator received. Operators are dispatched to investigate decreasing Unit 1 FWST level.
- **1000**: Unit 1 BOP reports that FWST volume is 377,536.9 gallons.
- **1015**: An improperly executed FW system clearance is discovered and determined to be the cause of FWST level decrease.

  Current conditions are as follows:
  - The valve lineup has been immediately corrected securing the level decrease.
  - Current FWST level is 95%.
  - Current FWST boron concentration is 2740 ppmB.
  - Unit 1 BAT boron concentration is 7500 ppmB.
  - Unit 2 BAT boron concentration is 7300 ppmB.
  - Unit 1 RMWST boron concentration is 4 ppmB.
  - Total Blender Makeup flowrate = 90 gpm

INITIATING CUES:

1. **Using OP/1/A/6200/014 (Refueling Water System), determine the required amount of Boric Acid and RMWST water necessary to perform an FWST makeup from the blender as follows:**
   - **Consider any initial conditions complete.**
   - **Final FWST level of 96% at 2720 ppmB.**
   - **S. Jackson (Primary Chemistry) has been notified and directed use of the Unit 1 BAT.**
   - **Concurrent Verification is waived for this task.**

   Boric Acid ________________ **RMWST Water** ________________

2. **Assuming the required makeup begins at time 1030, determine which Active <u>SLC and TS</u> LCO entries (if any) will be required at 1000 and 1100.**

1000: ____

1100: ____

Catawba Nuclear Station
Admin. JPM A.1-1S
Sept 2021 NRC Exam

REFERENCE HANDOUT

ONLINE DATABOOK CALCULATOR: XY

FILENAME : REFUEL_T.XYP

TITLE : FW Refueling Water Storage Tank

SUB TITLE : Source: CNC-1223.21-00-0004

	VALUES	
	MINIMUM	MAXIMUM
X:Level (%)	0.00000	100.000
Y:Volume (GALLONS)	15584.6	392732.

ENTER ONE VARIABLE AND PRESS CALCULATE

X VALUE	Y VALUE
95.0	373875.

CALCULATE | CLEAR | CANCEL

ONLINE DATABOOK CALCULATOR: XY

FILENAME : REFUEL_T.XYP

TITLE : FW Refueling Water Storage Tank

SUB TITLE : Source: CNC-1223.21-00-0004

	VALUES	
	MINIMUM	MAXIMUM
X:Level (%)	0.00000	100.000
Y:Volume (GALLONS)	15584.6	392732.

ENTER ONE VARIABLE AND PRESS CALCULATE

X VALUE	Y VALUE
96.0	377646.

CALCULATE | CLEAR | CANCEL

1. Limits and Precautions

1.1 Misaligning the FW System can result in draining the Refueling Water Storage Tank (FWST).

1.2 Miscalculating an FWST addition can result in boron concentration being lower than that required by the COLR.

1.3 If a Safety Injection occurs with the FW Pump in operation, the pump will lose its flowpath due to its suction valves receiving a close signal.

1.4 If either of the following occurs, a release concern may be created:

- A large quantity (more than normal makeup) of non-degassed water is used for makeup to the FWST.

  OR
- Any quantity of water with high activity (back leakage from NC into FWST) is added to the FWST.

2. Initial Conditions

_____ 2.1 Ensure Reactivity Management controls for an R3 evolution are established per AD-OP-ALL-0203 (Reactivity Management).

_____ 2.2 Verify FWST operation per one of the following:

- ☐ Normal operation per Enclosure 4.1 (FWST Normal Operation)
  OR
- ☐ Purification per Enclosure 4.5 (FWST Purification)
  OR
- ☐ Recirculation per Enclosure 4.6 (FWST Recirculation)

_____ 2.3 Verify boron concentration control systems are available per OP/1/A/6150/009 (Boron Concentration Control).

_____ 2.4 **IF** charging has been established per OP/1/A/6200/001 (Chemical and Volume Control System), verify VCT level is adequate to allow boric acid blender alignment to the FWST for the duration of the FWST makeup.

3. Procedure

NOTE:	To prevent an inadvertent reduction in the shutdown margin during NC fill evolutions, makeup to the FWST shall be performed with a water source having a Boron concentration which is greater than or equal to the required NC system Boron concentration.

_____ 3.1 Notify Chemistry that the following Boric Acid Tank will be used for Tech Spec blending calculations {PIP 91-0449}:

Person notified ___________________________________

_____________ Boric Acid Tank #1
_____________ Boric Acid Tank #2

_____ 3.2 Document current boron concentration in the Boric Acid Tank as specified by Chemistry in Step 3.1 {PIP 91-0449}:

_____________ ppmB

NOTE:	The final boron concentration used in the next step shall comply with COLR requirements for the current mode of operation.

_____ _____ 3.3 Determine the boron concentration of the makeup water to the FWST (C_f) to attain the final boron concentration, C_{FW}.

3.3.1 Initial volume of FWST _______ = V_i.

3.3.2 Final volume of FWST after makeup _______ = V_{FW}.

3.3.3 Compute the total gallons of makeup water to be added to the FWST _________ = V_f.

$V_f = V_{FW} - V_i$

3.3.4 Initial boron concentration of water in the FWST ______ = C_i.

3.3.5 Solve for the desired makeup water boron concentration, C_f.

$$C_f = \frac{C_{FW} V_{FW} - C_i V_i}{V_f} \qquad C_f = \_\_\_\_\_\_\_\_\_$$

_____ _____ 3.4 Determine the amount of boric acid and reactor makeup water to add as follows:

3.4.1 Boron Concentration of water in BAT _________ = C_1.

3.4.2 Boron concentration of water in RMWST __________ = C_2.

3.4.3 Total gallons of makeup water to be added to FWST ________ = V_f from Step 3.3.3.

3.4.4 Solve for the amount of boric acid to be added (V_1) using C_f obtained in Step 3.3.5.

$$V_1 = \frac{V_f (C_f - C_2)}{C_1 - C_2} \qquad V_1 = \_\_\_\_\_\_\_\_\_$$

_____ 3.4.5 **<u>IF</u>** V_1 is negative, contact the OWPM Staff for instructions on adjusting the boron concentration in the FWST.

3.4.6 Solve for the amount of RMWST water to be added (V_2).

$V_2 = V_f - V_1$ V_2 = _________ gal.

NOTE:	Step 3.5 assumes boric acid for FWST makeup will be supplied from BAT #1. If BAT #2 is aligned to Unit 1 per OP/1/A/6150/009 (Boron Concentration Control), Step 3.5 may be N/A'd since both BATs are non-functional while in that alignment.

_____ 3.5 Determine effect on BAT #1 level as follows:

_____ 3.5.1 Record volume of BAT #1. (OAC Point C1P5645 (BAT Volume))
_____________ gallons

_____ 3.5.2 Determine final BAT volume by subtracting amount of acid to be added to FWST (Step 3.4.4) from initial volume (Step 3.5.1).

( ) - ( ) = _____________ gallons
Step 3.5.1 Step 3.4.4

_____ 3.5.3 **<u>IF</u>** final BAT volume will be less than COLR requirements, perform <u>one</u> of the following:

- _____ Place BAT #2 in service for Unit 1 per OP/1/A/6150/009 (Boron Concentration Control).
  OR
- _____ (SRO) Declare BAT #1 non-functional per the following:
  - Unit 1 boric acid flow path SLC 16.9-7 or 16.9-8
  - Unit 1 borated water source SLC 16.9-11 or 16.9-12

_____ _____ 3.6 Determine the final counter readings as follows:

3.6.1 Initial TOTAL MAKEUP COUNTER reading ________ = T_{mi}.

3.6.2 Initial BORIC ACID COUNTER reading ________ = T_{Bi}.

3.6.3 Solve for final TOTAL MAKEUP COUNTER reading (T_{mf}) using V_f obtained in Step 3.3.3.

$T_{mf} = T_{mi} + V_f$ T_{mf} = ____________

3.6.4 Solve for final BORIC ACID COUNTER reading (T_{Bf}) using V_1 obtained in Step 3.4.4.

$T_{Bf} = T_{Bi} + V_1$ T_{Bf} = ____________

3.7 Place the switches for the following valves in the "CLOSE" position:

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

3.8 Set the following blender flow controllers to achieve the desired boron concentration and volume:

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

3.9 Ensure the selector switches for the following valves are in "AUTO":

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

3.10 Set the following counters to achieve the desired boron concentration and volume:

_____ • BORIC ACID

_____ • TOTAL MAKEUP

_____ _____ 3.11 Verify 1NI-96B (C-Leg Accum Chk Vlv Tst Isol) is closed.

_____ 3.12 Verify 1NB-5 (Unit 1 VCT To NB Evap Feed Demin Isol) (1ELCC0024) (AB-560, MM-52) is closed.

_____ 3.13 Close 1NV-187 (Boric Acid Blender Outlet To VCT Outlet Isol) (AB-583, KK-50, Rm 419).

3.14 Open the following valves:

_____ • 1NV-183 (Boric Acid Blender Outlet To FWST & RHT Isol) (AB-585, KK-51, Rm 419)

_____ • 1NV-185 (Boric Acid To FWST Isol) (AB-581, KK-51, Rm 419)

_____ 3.15 **<u>IF</u>** the FWST is in recirculation alignment per <u>one</u> of the following enclosures:

☐ Enclosure 4.6 (FWST Recirculation)

<u>OR</u>

☐ Enclosure 4.5 (FWST Purification)

THEN perform the following:

_____ 3.15.1 Secure the "FW PUMP".

Record FW Pump stop time. ____________

3.15.2 Start the FW recirc pump secured in Enclosure 4.5 (FWST Purification) **OR** Enclosure 4.6 (FWST Recirculation).

_____ • FW Recirc Pump 1A

_____ • FW Recirc Pump 1B

_____ 3.16 Place "NC MAKEUP MODE SELECT" switch in the "MANUAL" position.

3.17 Ensure one of the following Reactor Makeup Water Pumps is in "AUTO":

_____ • "RX M/U WTR PUMP 1A"
OR
_____ • "RX M/U WTR PUMP 1B"

3.18 Ensure the Reactor Makeup Water Pump **NOT** selected to "AUTO" in Step 3.17 is in "OFF":

_____ • "RX M/U WTR PUMP 1A"
OR
_____ • "RX M/U WTR PUMP 1B"

3.19 Ensure at least one of the following Boric Acid Transfer Pumps is in "AUTO":

_____ • "B/A XFER PUMP 1A"
AND/OR
_____ • "B/A XFER PUMP 1B"

_____ 3.20 **IF** a Boric Acid Transfer Pump was **NOT** selected to "AUTO" in Step 3.19, ensure it is in "OFF":

_____ • "B/A XFER PUMP 1A"
OR
_____ • "B/A XFER PUMP 1B"

_____ 3.21 **IF** Unit 1 is in Mode 1, 2, 3 or 4, perform the following:

_____ 3.21.1 Step 3.21.2 is an action to maintain the FWST within analyzed conditions during a Safety Injection. The designated individuals shall sign in the indicated places to document understanding of responsibilities. All required actions are to be completed prior to reaching FWST Lo Level (20%) to ensure no air entrainment occurs due to vortex formation in the FWST.

Operator in Control Room ______________________________

Shift Manager approval _________________________________

_____ 3.21.2 **IF AT ANY TIME** while FWST makeup is in progress a Safety Injection occurs on Unit 1, terminate the makeup by performing the following:

_____ 3.21.2.1 Turn the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.21.2.2 Ensure the following:

- _____ Selected Reactor Makeup Water Pump stops.
- _____ 1NV-238A (B/A To Blender Ctrl Vlv) closes.
- _____ 1NV-242A (RMWST To B/A Blender Ctrl) closes.

_____ 3.22 Energize makeup control circuit by turning the "NC MAKEUP CONTROL" switch to the "START" position.

3.23 Ensure the following:

- _____ Selected Reactor Makeup Water Pump starts.
- _____ Selected Boric Acid Transfer Pump starts.
- _____ 1NV-238A (B/A To Blender Ctrl Vlv) positions to produce desired boric acid flow.
- _____ 1NV-242A (RMWST To B/A Blender Ctrl) positions to produce desired total makeup flow.

_____ 3.24 **WHEN** makeup is complete, turn the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.25 Ensure the following:

_____ • Selected Reactor Makeup Water Pump stops.
_____ • Selected Boric Acid Transfer Pump stops.
_____ • 1NV-238A (B/A To Blender Ctrl Vlv) closes.
_____ • 1NV-242A (RMWST To B/A Blender Ctrl) closes.

3.26 Flush the flowpath for one minute as follows:

3.26.1 Place both Boric Acid Transfer Pumps in "OFF".

_____ • "B/A XFER PUMP 1A"
_____ • "B/A XFER PUMP 1B"

_____ 3.26.2 Place the selector switch for 1NV-238A (B/A To Blender Ctrl Vlv) in the "CLOSED" position.

_____ 3.26.3 Place the selector switch for 1NV-242A (RMWST To B/A Blender Ctrl) in the "OPEN" position.

_____ 3.26.4 Turn the "NC MAKEUP CONTROL" switch to the "START" position.

3.26.5 Ensure the following:

_____ • Selected Reactor Makeup Water Pump starts.
_____ • Total makeup flow indicates flow.

_____ 3.26.6 **<u>WHEN</u>** flushing is completed, turn the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.26.7 Ensure the following:

_____ • Selected Reactor Makeup Water Pump stops.
_____ • Total makeup flow indicates no flow.

3.27 Place the selector switches for the following valves in "AUTO":

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)
_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

3.28 Close the following valves:

_____ • 1NV-183 (Boric Acid Blender Outlet To FWST & RHT Isol) (AB-585, KK-51, Rm 419)

_____ • 1NV-185 (Boric Acid To FWST Isol) (AB-581, KK-51, Rm 419)

_____ 3.29 Open 1NV-187 (Boric Acid Blender Outlet To VCT Outlet Isol) (AB-583, KK-50, Rm 419).

3.30 Place at least <u>one</u> of the following Boric Acid Transfer Pumps in "AUTO":

_____ • "B/A XFER PUMP 1A"
AND/OR
_____ • "B/A XFER PUMP 1B"

_____ 3.31 **<u>IF</u>** needed to support plant conditions, place desired Boric Acid Transfer Pump in "ON":

_____ • "B/A XFER PUMP 1A"

_____ • "B/A XFER PUMP 1B"

_____ 3.32 **<u>IF</u>** desired, place the second Reactor Makeup Water Pump in "AUTO":

_____ • "RX M/U WTR PUMP 1A"

_____ • "RX M/U WTR PUMP 1B"

_____ 3.33 **<u>IF</u>** needed to support plant conditions, place desired Reactor Makeup Water Pump in "ON":

_____ • "RX M/U WTR PUMP 1A"

_____ • "RX M/U WTR PUMP 1B"

3.34 Place the selector switches for the following valves in "AUTO":

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

_____ 3.35 Align for makeup to the VCT per OP/1/A/6150/009 (Boron Concentration Control).

_____ 3.36 **IF** Step 3.15 was performed,
THEN perform the following:

_____ 3.36.1 Secure the FW recirc pump started in Step 3.15.2.

_____ 3.36.2 Start the "FW PUMP".

Record FW Pump start time. ____________

NOTE:	FWST sample is required within 24 hours of completion of makeup.

_____ 3.37 **IF** either of the following conditions exists, recirculate the FWST using an NS pump:

- ☐ A time constraint exists that prohibits the use of the normal recirculation of the FWST.
- ☐ Unit/WCC SRO determines that NS System Availability is **NOT** a concern, and that recirculation via the NS is desired.

_____ 3.37.1 Place the FWST in recirculation via an NS pump per OP/1/A/6200/007 (Containment Spray System) at a flow rate of 900 GPM.

_____ 3.37.2 Notify Chemistry of the following:
Person notified ___________________________________

- FWST is in recirculation with an NS Pump.
- FWST is to be sampled per OP/1/A/6200/027 (Sampling Local Primary Sample Points).

_____ 3.37.3 **WHEN** notified by Chemistry that they are ready to obtain the sample, secure the NS pump per OP/1/A/6200/007 (Containment Spray System).

NOTE:	• FWST sample is required within 24 hours of completion of makeup. • IF Step 3.36 was performed, Step 3.38.1 may be N/A'd.

_____ 3.38 **IF** FWST was **NOT** recirculated with the NS pump, perform the following:

_____ 3.38.1 Place the FWST in recirculation per one of the following:

_____ • Enclosure 4.6 (FWST Recirculation).

OR

_____ • Enclosure 4.5 (FWST Purification).

_____ 3.38.2 Notify Chemistry of the following:

Person notified ______________________________________

- FWST is in recirculation with the FW Pump.
- FWST is to be sampled per OP/1/A/6200/027 (Sampling Local Primary Sample Points).

3.39 Do **NOT** file this enclosure.

1. Limits and Precautions

1.1 Misaligning the FW System can result in draining the Refueling Water Storage Tank (FWST).

1.2 Miscalculating an FWST addition can result in boron concentration being lower than that required by the COLR.

1.3 If a Safety Injection occurs with the FW Pump in operation, the pump will lose its flowpath due to its suction valves receiving a close signal.

1.4 If either of the following occurs, a release concern may be created:

- A large quantity (more than normal makeup) of non-degassed water is used for makeup to the FWST.

  OR
- Any quantity of water with high activity (back leakage from NC into FWST) is added to the FWST.

2. Initial Conditions

_____ 2.1 Ensure Reactivity Management controls for an R3 evolution are established per AD-OP-ALL-0203 (Reactivity Management).

_____ 2.2 Verify FWST operation per <u>one</u> of the following:

- ☐ Normal operation per Enclosure 4.1 (FWST Normal Operation)

  OR
- ☐ Purification per Enclosure 4.5 (FWST Purification)

  OR
- ☐ Recirculation per Enclosure 4.6 (FWST Recirculation)

_____ 2.3 Verify boron concentration control systems are available per OP/1/A/6150/009 (Boron Concentration Control).

_____ 2.4 **<u>IF</u>** charging has been established per OP/1/A/6200/001 (Chemical and Volume Control System), verify VCT level is adequate to allow boric acid blender alignment to the FWST for the duration of the FWST makeup.

3. Procedure

NOTE:	To prevent an inadvertent reduction in the shutdown margin during NC fill evolutions, makeup to the FWST shall be performed with a water source having a Boron concentration which is greater than or equal to the required NC system Boron concentration.

_____ 3.1 Notify Chemistry that the following Boric Acid Tank will be used for Tech Spec blending calculations {PIP 91-0449}:

Person notified **Steph Jackson**

X Boric Acid Tank #1

_____ Boric Acid Tank #2

_____ 3.2 Document current boron concentration in the Boric Acid Tank as specified by Chemistry in Step 3.1 {PIP 91-0449}:

7500 ppmB

NOTE:	The final boron concentration used in the next step shall comply with COLR requirements for the current mode of operation.

_____ _____ 3.3 Determine the boron concentration of the makeup water to the FWST (C_f) to attain the final boron concentration, C_{FW}.

3.3.1 Initial volume of FWST **373,875** = V_i.

3.3.2 Final volume of FWST after makeup **377,646** = V_{FW}.

3.3.3 Compute the total gallons of makeup water to be added to the FWST **3,771** = V_f.

$V_f = V_{FW} - V_i$

3.3.4 Initial boron concentration of water in the FWST **2,740** = C_i.

3.3.5 Solve for the desired makeup water boron concentration, C_f.

$$C_f = \frac{C_{FW} V_{FW} - C_i V_i}{V_f}$$

$$C_f = \frac{(2,720)(277,646)-(2,740)(373,875)}{3771}$$

=737.1

_____ _____ 3.4 Determine the amount of boric acid and reactor makeup water to add as follows:

3.4.1 Boron Concentration of water in BAT **7,500** = C_1.

3.4.2 Boron concentration of water in RMWST **4** = C_2.

3.4.3 Total gallons of makeup water to be added to FWST **3,771** = V_f from Step 3.3.3.

3.4.4 Solve for the amount of boric acid to be added (V_1) using C_f obtained in Step 3.3.5.

$$V_1 = \frac{V_f\,(C_f - C_2)}{C_1 - C_2} \qquad V_1 = \frac{3771(737.1-4)}{(7500-4)} = 368.8$$

_____ 3.4.5 **IF** V_1 is negative, contact the OWPM Staff for instructions on adjusting the boron concentration in the FWST.

3.4.6 Solve for the amount of RMWST water to be added (V_2).

$V_2 = V_f - V_1$ V_2 = **3402.2** gal.

3771-368.8

NOTE:	Step 3.5 assumes boric acid for FWST makeup will be supplied from BAT #1. If BAT #2 is aligned to Unit 1 per OP/1/A/6150/009 (Boron Concentration Control), Step 3.5 may be N/A'd since both BATs are non-functional while in that alignment.

_____ 3.5 Determine effect on BAT #1 level as follows:

_____ 3.5.1 Record volume of BAT #1. (OAC Point C1P5645 (BAT Volume))
_____________ gallons

_____ 3.5.2 Determine final BAT volume by subtracting amount of acid to be added to FWST (Step 3.4.4) from initial volume (Step 3.5.1).

(   ) - (   ) = _____________ gallons
Step 3.5.1   Step 3.4.4

_____ 3.5.3 **<u>IF</u>** final BAT volume will be less than COLR requirements, perform <u>one</u> of the following:

_____ • Place BAT #2 in service for Unit 1 per OP/1/A/6150/009 (Boron Concentration Control).
OR

_____ (SRO) • Declare BAT #1 non-functional per the following:

- Unit 1 boric acid flow path SLC 16.9-7 or 16.9-8
- Unit 1 borated water source SLC 16.9-11 or 16.9-12

_____ _____ 3.6 Determine the final counter readings as follows:

3.6.1 Initial TOTAL MAKEUP COUNTER reading ________ = T_{mi}.

3.6.2 Initial BORIC ACID COUNTER reading ________ = T_{Bi}.

3.6.3 Solve for final TOTAL MAKEUP COUNTER reading (T_{mf}) using V_f obtained in Step 3.3.3.

$T_{mf} = T_{mi} + V_f$ T_{mf} = ____________

3.6.4 Solve for final BORIC ACID COUNTER reading (T_{Bf}) using V_1 obtained in Step 3.4.4.

$T_{Bf} = T_{Bi} + V_1$ T_{Bf} = ____________

3.7 Place the switches for the following valves in the "CLOSE" position:

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

3.8 Set the following blender flow controllers to achieve the desired boron concentration and volume:

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

3.9 Ensure the selector switches for the following valves are in "AUTO":

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

3.10 Set the following counters to achieve the desired boron concentration and volume:

_____ • BORIC ACID

_____ • TOTAL MAKEUP

_____ _____ 3.11 Verify 1NI-96B (C-Leg Accum Chk Vlv Tst Isol) is closed.

_____ 3.12 Verify 1NB-5 (Unit 1 VCT To NB Evap Feed Demin Isol) (1ELCC0024) (AB-560, MM-52) is closed.

_____ 3.13 Close 1NV-187 (Boric Acid Blender Outlet To VCT Outlet Isol) (AB-583, KK-50, Rm 419).

3.14 Open the following valves:

_____ • 1NV-183 (Boric Acid Blender Outlet To FWST & RHT Isol) (AB-585, KK-51, Rm 419)

_____ • 1NV-185 (Boric Acid To FWST Isol) (AB-581, KK-51, Rm 419)

_____ 3.15 **<u>IF</u>** the FWST is in recirculation alignment per <u>one</u> of the following enclosures:

☐ Enclosure 4.6 (FWST Recirculation)

<u>OR</u>

☐ Enclosure 4.5 (FWST Purification)

<u>THEN</u> perform the following:

_____ 3.15.1 Secure the "FW PUMP".

Record FW Pump stop time. ____________

3.15.2 Start the FW recirc pump secured in Enclosure 4.5 (FWST Purification) **<u>OR</u>** Enclosure 4.6 (FWST Recirculation).

- _____ FW Recirc Pump 1A
- _____ FW Recirc Pump 1B

_____ 3.16 Place "NC MAKEUP MODE SELECT" switch in the "MANUAL" position.

3.17 Ensure <u>one</u> of the following Reactor Makeup Water Pumps is in "AUTO":

- _____ "RX M/U WTR PUMP 1A"
  OR
- _____ "RX M/U WTR PUMP 1B"

3.18 Ensure the Reactor Makeup Water Pump **<u>NOT</u>** selected to "AUTO" in Step 3.17 is in "OFF":

- _____ "RX M/U WTR PUMP 1A"
  OR
- _____ "RX M/U WTR PUMP 1B"

3.19 Ensure at least <u>one</u> of the following Boric Acid Transfer Pumps is in "AUTO":

- _____ "B/A XFER PUMP 1A"
  AND/OR
- _____ "B/A XFER PUMP 1B"

_____ 3.20 **<u>IF</u>** a Boric Acid Transfer Pump was **<u>NOT</u>** selected to "AUTO" in Step 3.19, ensure it is in "OFF":

- _____ "B/A XFER PUMP 1A"
  OR
- _____ "B/A XFER PUMP 1B"

_____ 3.21 **IF** Unit 1 is in Mode 1, 2, 3 or 4, perform the following:

_____ 3.21.1 Step 3.21.2 is an action to maintain the FWST within analyzed conditions during a Safety Injection. The designated individuals shall sign in the indicated places to document understanding of responsibilities. All required actions are to be completed prior to reaching FWST Lo Level (20%) to ensure no air entrainment occurs due to vortex formation in the FWST.

Operator in Control Room ________________________________

Shift Manager approval ___________________________________

_____ 3.21.2 **IF AT ANY TIME** while FWST makeup is in progress a Safety Injection occurs on Unit 1, terminate the makeup by performing the following:

_____ 3.21.2.1 Turn the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.21.2.2 Ensure the following:

- _____ Selected Reactor Makeup Water Pump stops.
- _____ 1NV-238A (B/A To Blender Ctrl Vlv) closes.
- _____ 1NV-242A (RMWST To B/A Blender Ctrl) closes.

_____ 3.22 Energize makeup control circuit by turning the "NC MAKEUP CONTROL" switch to the "START" position.

3.23 Ensure the following:

- _____ Selected Reactor Makeup Water Pump starts.
- _____ Selected Boric Acid Transfer Pump starts.
- _____ 1NV-238A (B/A To Blender Ctrl Vlv) positions to produce desired boric acid flow.
- _____ 1NV-242A (RMWST To B/A Blender Ctrl) positions to produce desired total makeup flow.

_____ 3.24 **WHEN** makeup is complete, turn the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.25 Ensure the following:

_____ • Selected Reactor Makeup Water Pump stops.

_____ • Selected Boric Acid Transfer Pump stops.

_____ • 1NV-238A (B/A To Blender Ctrl Vlv) closes.

_____ • 1NV-242A (RMWST To B/A Blender Ctrl) closes.

3.26 Flush the flowpath for one minute as follows:

3.26.1 Place both Boric Acid Transfer Pumps in "OFF".

_____ • "B/A XFER PUMP 1A"

_____ • "B/A XFER PUMP 1B"

_____ 3.26.2 Place the selector switch for 1NV-238A (B/A To Blender Ctrl Vlv) in the "CLOSED" position.

_____ 3.26.3 Place the selector switch for 1NV-242A (RMWST To B/A Blender Ctrl) in the "OPEN" position.

_____ 3.26.4 Turn the "NC MAKEUP CONTROL" switch to the "START" position.

3.26.5 Ensure the following:

_____ • Selected Reactor Makeup Water Pump starts.

_____ • Total makeup flow indicates flow.

_____ 3.26.6 **WHEN** flushing is completed, turn the "NC MAKEUP CONTROL" switch to the "STOP" position.

3.26.7 Ensure the following:

_____ • Selected Reactor Makeup Water Pump stops.

_____ • Total makeup flow indicates no flow.

3.27 Place the selector switches for the following valves in "AUTO":

_____ • 1NV-238A (B/A To Blender Ctrl Vlv)

_____ • 1NV-242A (RMWST To B/A Blender Ctrl)

3.28 Close the following valves:

_____ • 1NV-183 (Boric Acid Blender Outlet To FWST & RHT Isol) (AB-585, KK-51, Rm 419)

_____ • 1NV-185 (Boric Acid To FWST Isol) (AB-581, KK-51, Rm 419)

_____ 3.29 Open 1NV-187 (Boric Acid Blender Outlet To VCT Outlet Isol) (AB-583, KK-50, Rm 419).

3.30 Place at least <u>one</u> of the following Boric Acid Transfer Pumps in "AUTO":

_____ • "B/A XFER PUMP 1A"
AND/OR
_____ • "B/A XFER PUMP 1B"

_____ 3.31 **<u>IF</u>** needed to support plant conditions, place desired Boric Acid Transfer Pump in "ON":

_____ • "B/A XFER PUMP 1A"

_____ • "B/A XFER PUMP 1B"

_____ 3.32 **<u>IF</u>** desired, place the second Reactor Makeup Water Pump in "AUTO":

_____ • "RX M/U WTR PUMP 1A"

_____ • "RX M/U WTR PUMP 1B"

_____ 3.33 **<u>IF</u>** needed to support plant conditions, place desired Reactor Makeup Water Pump in "ON":

_____ • "RX M/U WTR PUMP 1A"

_____ • "RX M/U WTR PUMP 1B"

3.34 Place the selector switches for the following valves in "AUTO":

_____ • 1NV-181A (B/A Blender Otlt To VCT)

_____ • 1NV-186A (B/A Blender Otlt To VCT Otlt)

_____ 3.35 Align for makeup to the VCT per OP/1/A/6150/009 (Boron Concentration Control).

_____ 3.36 **IF** Step 3.15 was performed,
THEN perform the following:

_____ 3.36.1 Secure the FW recirc pump started in Step 3.15.2.

_____ 3.36.2 Start the "FW PUMP".

Record FW Pump start time. ____________

NOTE:	FWST sample is required within 24 hours of completion of makeup.

_____ 3.37 **IF** either of the following conditions exists, recirculate the FWST using an NS pump:

- ☐ A time constraint exists that prohibits the use of the normal recirculation of the FWST.
- ☐ Unit/WCC SRO determines that NS System Availability is **NOT** a concern, and that recirculation via the NS is desired.

_____ 3.37.1 Place the FWST in recirculation via an NS pump per OP/1/A/6200/007 (Containment Spray System) at a flow rate of 900 GPM.

_____ 3.37.2 Notify Chemistry of the following:
Person notified ____________________________________

- FWST is in recirculation with an NS Pump.
- FWST is to be sampled per OP/1/A/6200/027 (Sampling Local Primary Sample Points).

_____ 3.37.3 **WHEN** notified by Chemistry that they are ready to obtain the sample, secure the NS pump per OP/1/A/6200/007 (Containment Spray System).

NOTE:	• FWST sample is required within 24 hours of completion of makeup. • IF Step 3.36 was performed, Step 3.38.1 may be N/A'd.

_____ 3.38 **IF** FWST was **NOT** recirculated with the NS pump, perform the following:

_____ 3.38.1 Place the FWST in recirculation per one of the following:

_____ • Enclosure 4.6 (FWST Recirculation).
OR
_____ • Enclosure 4.5 (FWST Purification).

_____ 3.38.2 Notify Chemistry of the following:

Person notified ____________________________________

- FWST is in recirculation with the FW Pump.
- FWST is to be sampled per OP/1/A/6200/027 (Sampling Local Primary Sample Points).

3.39 Do **NOT** file this enclosure.

JPM A.1-2S

SRO

Catawba Nuclear Station
Admin. JPM A.1-2S
Sept 2021 NRC Exam

EVALUATION SHEET

Task: Determine License Status

Alternate Path: N/A

Facility JPM #: NS07-001

Safety Function: N/A

K/A 2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical records, "no-solo" operation, maintenance of active license status.

Importance: 3.3 / 3.8 CFR: 41.10 / 43.2

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom X Perform X Simulate ________

References: AD-OP-ALL-0107 Maintenance of RO and SRO Licenses

Task Standard: Applicant determines that RP Jones is eligible to work the CRS position on July 1, 2021, while Mike Starnes and Will Fowler are NOT eligible as actively licensed SROs for this shift.

Validation Time: 30 minutes **Time Critical:** Yes ______ No X

Applicant: Time Start: __________
NAME____________________________ Docket #______________ Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____
____
____
____
____
____

Catawba Nuclear Station
Admin. JPM A.1-2S
Sept 2021 NRC Exam

DIRECTION TO APPLICANT:

I will explain the initial conditions and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- You are evaluating the work histories of three Licensed Senior Reactor Operators.
- All three have off-shift assignments at the plant, are current in License Operator Requalification Training, and have had a medical examination in the past 2 years.
- All three operators have worked ONLY the shifts listed for the second quarter of 2021.
- Active/Inactive status and time on shift since April 1, 2021 is as follows for each of the Senior Reactor Operators: (Work History Table Provided)
- Unit Power History
    - Unit 1 has remained in **MODE 1** since 04/01/21
    - Unit 2 entered a refueling outage on 05/04/21 and was in **NO MODE** from 05/15/21 – 05/22/21.

INITIATING CUES:

- Determine if each of the Senior Reactor Operators is eligible to work the CRS position on the 0700 - 1900 shift on July 1, 2021.
- Record your answer below (yes or no). If no, explain why.

Mike Starnes:____

Will Fowler:____

RP Jones:____

EXAMINER NOTE:

After reading cue, provide applicant with a copy of AD-OP-ALL-0107 and the Work History Table provided with this JPM.

Catawba Nuclear Station
Admin. JPM A.1-2S
Sept 2021 NRC Exam

START TIME: _________

STEP 1: Determine the Active/Inactive status of Mike Starnes's SRO License:	**CRITICAL STEP**
STANDARD: From the cue, the applicant reviews the requirements of AD-OP-ALL-0107 and determines that this license is INACTIVE • 04/02/21, 04/03/21, 05/05/21, & 05/06/21 watches met 4 of the 5 required. • 05/21/21 watch did not count because the applicable unit was not in a required mode. • 04/04/21 & 06/17/21 watches were not in required position. **Examiner Note: This step is critical because determining that Mike Starnes is NOT eligible to work, at the CRS position, on July 1, 2021 is necessary to complete the assigned task.** COMMENTS:	Sat __ Unsat __

STEP 2: Determine the Active/Inactive status of Will Fowler's SRO License:	**CRITICAL STEP**
STANDARD: From the cue, the applicant reviews the requirements of AD-OP-ALL-0107 and determines that this license is INACTIVE • 04/01/21, 04/03/21, 04/05/21, & 04/14/21 watches met 4 of the 5 required. • 04/02/21 watch did not count because it did not constitute a full 12 hour shift. • 05/02/21 watch was not in required position. **Examiner Note: This step is critical because determining that Will Fowler is NOT eligible to work, at the CRS position, on July 1, 2021 is necessary to complete the assigned task.** COMMENTS:	Sat __ Unsat __

STEP 3: Determine the Active/Inactive status of RP Jones SRO License:	**CRITICAL STEP**
STANDARD: • From the cue, the applicant reviews the requirements of AD-OP-ALL-0107 and determines that this license is ACTIVE. Although 5 complete watches were not completed within the required positions and unit modes, the five 12 hour proficiency watches are not required to be performed in a quarter where reactivation is accomplished. **Examiner Note: This step is critical because determining that RP Jones is eligible to work, at the CRS position, on July 1, 2021 is necessary to complete the assigned task.** COMMENTS:	Sat __ Unsat __

END TIME: __________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- You are evaluating the work histories of three Licensed Senior Reactor Operators.
- All three have off-shift assignments at the plant, are current in License Operator Requalification Training, and have had a medical examination in the past 2 years.
- All three operators have worked ONLY the shifts listed for the second quarter of 2021.
- Active/Inactive status and time on shift since April 1, 2021 is as follows for each of the Senior Reactor Operators: (Work History Table Provided)
- Unit Power History
    - Unit 1 has remained in **MODE 1** since 04/01/21
    - Unit 2 entered a refueling outage on 05/04/21 and was in **NO MODE** from 05/15/21 – 05/22/21.

INITIATING CUES:

- Determine if each of the Senior Reactor Operators is eligible to work the CRS position on the 0700 - 1900 shift on July 1, 2021.
- Record your answer below (yes or no). If no, explain why.

Mike Starnes:____

Will Fowler:____

RP Jones:____

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

Mike Starnes	**License was active on April 1, 2021.**	
	04/02/21	Worked 0700-1900 shift as CRS.
	04/03/21	Worked 0700-1900 shift as Unit 1 OATC.
	04/04/21	Worked 0700-1900 shift as Unit 1 Supervisor
	05/05/21	Worked 0700-1900 shift as Unit 2 BOP.
	05/06/21	Worked 0700-1900 shift as Unit 2 OATC.
	05/21/21	Worked 1900-0700 shift as Unit 2 BOP.
	06/17/21	Worked 1900-0700 shift as STA.
Will Fowler	**License was active on April 1, 2021.**	
	04/01/21	Worked 0700-1900 shift as CRS.
	04/02/21	Worked 0700-1500 shift as CRS.
	04/03/21	Worked 0700-1900 shift as CRS.
	04/05/21	Worked 0700-1900 shift as CRS.
	04/14/21	Worked 1900-0700 shift as SM.
	05/02/21	Worked 0700-1900 shift as Unit 1 Supervisor
RP Jones	**License was inactive on April 1, 2021.**	
	04/05/21 thru 04/09/21 worked 40 hours under the direction of the CRS and completed all requirements for license reactivation.	
	04/12/21	Worked 0700-1900 shift as Unit 2 BOP.
	04/13/21	Worked 0700-1900 shift as Unit 1 BOP.
	05/16/21	Worked 0700-1900 shift as Unit 2 OATC.
	05/18/21	Worked 1900-0700 shift as Unit 1 BOP.
	05/20/21	Worked 1900-0700 shift as Unit 1 OATC.

Information Use

NUCLEAR OPERATING FLEET

ADMINISTRATIVE PROCEDURE

AD-OP-ALL-0107

MAINTENANCE OF RO AND SRO LICENSES

REVISION 3

Effective Dates:

07/01/2020	07/01/2020	07/01/2020	07/01/2020	07/01/2020
Brunswick	Catawba	Harris (HNP)	McGuire	Oconee
07/01/2020	07/01/2020			
Robinson	NGO			

REVISION SUMMARY
PRR 02321909 **DESCRIPTION**
• Section 3.0.3, sub-bullet: Added guidance for using licensee as an interchangeable term for Licensed Operator. • Section 4.4.1: Revised to incorporate participation in the requalification program. • Section 4.5 and Section 5.7.2: Revised 'Supervisor' to 'Superintendent'. • Section 5.2: Added section and associated guidance for Maintaining Required License Obligations (PRR 02280803 and PRR 02312224). • Section 5.3.1.a: Added guidance for Licensed Operators who will not attend training in accordance with AD-TQ-ALL-0068 (PRR 02280803). • Section 5.5.7 and sub-steps: Deleted guidance for maintaining a Shift Manager proficiency. • Attachment 1, Step 6.a: Updated CNS and MNS NLMS codes (PRR 02300605 and PRR 02315136). • Attachment 2, Step 6.a: Revised Requalification Training Supervisor to Operations Training Superintendent. • Attachment 2, Step 8: Revised to incorporate 'NLMS', and revised RP Staff Representative to Licensed Operator (PRR 02320845). • Attachment 2, Step 12: Revised Ops Training Manager to Ops Training Superintendent (PRR 02270855). • Attachment 2, Step 15.a: Updated CNS and MNS NLMS codes (PRR 02300605 and PRR 02315136). • Attachment 2, Table 8: Revised to delete WGDT room, revised 'AFP Hx to SFP Hx', and revised 'all levels' to 'level 1 and 2'. •

TABLE OF CONTENTS

TABLE OF CONTENTS
(continued)

1.0 PURPOSE

1. This procedure provides guidance for maintaining and reactivating NRC Senior Reactor Operator (SRO), Reactor Operator (RO), and Limited Senior Reactor Operator (LSRO) licenses pursuant to 10 CFR Part 55, Operators' Licenses.

2.0 SCOPE

1. This procedure provides the following instructions:
   - Processing NRC license applications
   - Maintaining required licensed requalification training status
   - Maintaining required Licensed Operator medical status
   - Maintaining required on-shift experience for Licensed Operators
   - Certification of a new or inactive license
   - Removing a Licensed Operator from duty for other than medical or requalification training status
   - Activating an SRO license for fuel handling/core alterations only
2. This procedure applies to NRC licensed personnel at all operating Duke Energy nuclear sites.

3.0 DEFINITIONS

1. **Active Licensed Operator Position(s):** Positions that meet the NRC definition of "actively performing the duties of a Reactor Operator or Senior Reactor Operator". The following are required to be filled by individuals with active licenses, and are the only positions that can be credited for maintaining required on shift experience for Licensed Operator(s).
   - Shift Manager (SRO)
   - Control Room Supervisor (SRO)
   - Operator at the Controls (RO)
   - Balance of Plant Operator (RO)
2. **Duke Energy Medical Information (DEMI):** Database used by Duke Energy Occupational Health (OH) to store employee medical data.

3.0 DEFINITIONS (continued)

3. **Licensed Operator(s):** An individual who has obtained a license issued by the Nuclear Regulatory Commission to perform the function of a Reactor Operator or Senior Reactor Operator as defined in 10 CFR 55.

   - Licensed Operator(s) are an interchangeable term to 'licensee' in 10 CFR 55 and in this procedure.

4. **Licensed Operator Qualification Report (LOQR):** A report used to track specific RO and SRO license attribute status.

5. **Limited Senior Reactor Operator (LSRO):** A SRO limited to fuel handling/core alteration activities only. LSROs are not required to maintain proficiency between refueling outages.

6. **Medical Conditions of Concern:** A new or change in medical condition (physical or mental), illness, or injury that might affect the physical or mental ability of a Licensed Operator to perform required licensed duties.

7. **No Solo:** Operator license restriction that requires another qualified person to be present when the restricted operator is operating or directing the operation of the controls, during all modes of operation including emergency conditions.

8. **Qualified Occupational Health (OH) Personnel:** Physicians and nurses employed or contracted by Duke Energy

4.0 RESPONSIBILITIES

4.1 Operations Management

1. Ensures the status and capability of Licensed Operators to perform licensed duties.

4.2 Assistant Operations Manager - Shift (AOM-Shift) or Designee

1. Ensures Licensed Operators filling Active Licensed Operator Position(s) have the opportunity to perform a minimum of five 12-hour shifts per calendar quarter.

2. Certifies Licensed Operators who are ready to resume the responsibilities of an on-shift Licensed Operator position.

4.3 Shift Managers (SMs)

1. Ensures all Licensed Operators standing watch hold an active license for that position.
2. Communicates operator license status changes to the organizations identified in this procedure.

4.4 Licensed Operators

1. Understands and fulfills NRC license obligations, including participation in the operator requalification program, issued under 10 CFR 55.
2. Understands status in regards to fulfilling Licensed Operator duties.
3. Ensures immediate supervisor and qualified Occupational Health personnel are made aware of any changes in:
   - Prescription medication use
   - Changes in medical status

4.5 Superintendent Nuclear Operations Training

1. Notifies the on duty SM when a Licensed Operator fails to meet the requirements of the requalification program.
2. Ensures Licensed Operators who fail to meet the requirements of the requalification program are disqualified from Nuclear Learning Management System (NLMS) activities per AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS). {7.1.1}

4.6 Regulatory Affairs

1. Generates and submits all regulatory required correspondence per 10 CFR 55 in accordance with the requirements of AD-LS-ALL-0002, Regulatory Correspondence.
2. Coordinates with Occupational Health personnel on reportability determination per 10 CFR 55.23.
3. Coordinates with Operations Management on reportability determination per 10 CFR 50.74(a)(b).

4.7 Corporate Medical Director (CMD)

1. Establishes and oversees the medical components of the OH program.

4.8 Examining Physician (SEP)

1. Reviews test results, performs medical examinations, and performs consultations for nuclear plant sites.

4.9 Qualified Occupational Health Personnel

1. Performs evaluations on Licensed Operator(s) when informed of changes to their medical status.
2. Provides evaluations to Operations Management concerning the ability of Licensed Operators to perform duties while taking prescription or non-prescription medications.
3. Performs biennial medical examinations per ANSI/ANS 3.4/1983.
4. Provides written documentation to Regulatory Affairs for submittal to the NRC as required by ANSI/ANS 3.4/1983.

5.0 INSTRUCTIONS

5.1 <u>Processing NRC License Applications</u>

1. Refer to AD-TQ-ALL-0610, NRC License Application Process, for guidance associated with initial and renewal license applications, waiver requests, and associated NRC correspondence.

2. Expiration of Operator NRC licenses is governed by 10 CFR 55.55, Expiration, which states the following:

   - Each Operator License and Senior Operator License expires six years after the date of issuance, upon termination of employment with the facility licensee, or upon determination by the facility licensee that the licensed individual no longer needs to maintain a license.

   - If a licensee files an application for renewal or an upgrade of an existing license on Form NRC-398 at least 30 days before the expiration of the existing license, it does not expire until disposition of the application for renewal or for an upgraded license has been finally determined by the Commission. Filing by mail will be deemed to be complete at the time the application is deposited in the mail.

5.2 <u>Maintaining Required License Obligations</u>

NOTE
The obligations in this procedure are paraphrased requirements in 10 CFR 55.

1. All Licensed Operators shall meet the following requirements:

   - Neither the license nor any right under the license may be assigned or otherwise transferred.

   - The license is limited to the facility for which it is issued.

   - The license is limited to those controls of the facility specified in the license.

   - The license is subject to, and the licensee shall observe, all applicable rules, regulations, and orders of the Nuclear Regulatory Commission.

   - Maintain on-shift operating experience (i.e., active license) in accordance with Section 5.5.

     ◊ If on-shift operating experience is not maintained, then prior to resumption of functions authorized by the license, refer to Section 5.6 to regain an active license.

5.2 Maintaining Required License Obligations (continued)

- The licensee shall notify the Nuclear Regulatory Commission within 30 days about a conviction for a felony.
- The licensee shall complete a requalification program as described by §55.59.
  - ◊ Compliance of 55.99 is verified by Section 5.3.
- The licensee shall have a biennial medical examination.
  - ◊ Medical requirements shall be maintained in accordance with Section 5.4.
- The licensee shall be Fit-for-Duty and shall **NOT** perform activities authorized by the license while under the influence of alcohol or any prescription, over-the-counter, or illegal substance that could adversely affect the licensees ability to safely and competently perform licensed duties.
  - ◊ The consumption, sale, or use of alcohol or illegal substances is prohibited in the protected area.
- The licensee shall participate in the drug and alcohol testing program established pursuant to 10 CFR Part 26.
- The licensee shall comply with any other conditions that the Commission may impose to protect health or to minimize damage to life or property.

2. If any of the following conditions exist for a Licensed Operator, then terminate the license by processing Attachment 4, Notification Of Change In Operator Status {7.1.2}:

   - Licensed Operator transfers to a new position in which maintaining the license is **NOT** desired.
   - Licensed Operator transfers to a different company.
   - Licensed Operator is no longer employed by the company.

5.3 Maintaining Required Licensed Requalification Training Status

1. All Licensed Operators must comply with the requirements of the AD-TQ-ALL-0068, Licensed Operator Continuing Training Program.

    a. If a Licensed Operator is reassigned to a position that does NOT require a license AND will NOT attend training in accordance with AD-TQ-ALL-0068, Licensed Operator Continuing Training Program, then process Attachment 4, Notification Of Change In Operator Status {7.1.2}, to terminate the Licensed Operator's license.

    b. If a Licensed Operator has failed to meet Licensed Operator qualification requirements, then Operations Training shall disqualify the Licensed Operator from licensed duties in accordance with AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS).

5.4 Medical Reporting Requirements

5.4.1 General

1. The reporting requirements for medical changes contained in this procedure apply to all Active and Inactive Licensed Operators

2. Licensed Operators are required to meet the medical requirements contained in ADMP-SAF-HSF-00091, Occupational Health Programs.

3. The OH Department will initiate notifications for changes in a medical condition that affects the ability of Licensed Operator to perform operator licensed duties in accordance with Attachment 3, Change In Medical Condition Affecting License Status

4. Licensed Operators are required to complete a biennial physical examination per ANSI/ANS 3.4/1983 conducted by the Duke Energy OH facility and the SEP.

    a. The expiration date is two years to the end of the month from the examining physician's medical clearance date.

5. Prior to assuming Licensed Operator duties, Licensed Operators are required to report any physical or mental condition that might impair their ability to perform licensed duties to the License Operator's supervisor and to OH personnel.

    a. Refer to Attachment 5, Common Medical Status Changes/Conditions of Licensed Operators, for examples of changes in medical conditions.

5.4.1 General (continued)

6. If a change in physical or mental condition is discovered, the Licensed Operator is required to share the following information with OH personnel:

   - Specific details of the condition
   - Medications
   - Lab results
   - Treatments
   - Prognosis

7. Notification of a change in physical or mental condition shall be made as soon as the individual has knowledge of the change in condition.

   a. Details of the medical condition are not required to be shared with supervision, just the existence of the condition and how it impacts the ability to perform licensed duties.

   b. Licensed duties shall not be performed until OH personnel have determined that the Licensed Operator is able to perform licensed duties.

8. If a Licensed Operator receives a new prescription, then the Licensed Operator reports the medications to supervision and OH personnel.

   a. If receiving a new prescription, then the Licensed Operator shall ask the prescribing physician if the medication may affect job performance and whether it can be taken while working.

   b. Licensed duties shall not be performed until the Licensed Operator has:

      - Informed Occupational Health personnel
      - Informed the Licensed Operators supervisor
      - OH personnel determines new prescription has no impact on performing licensed duties.

9. Non-prescription medication is not required to be reported to the OH personnel except if side-effects occur (e.g., drowsiness, fatigue, dizziness).

   a. Immediately notify OH personnel if side-effects occur upon taking a non-prescription medication.

5.4.1 General (continued)

10. Supervision shall promptly report any medical condition that results in a restriction being placed on a Licensed Operator by a physician to OH personnel.

11. If a supervisor suspects that a medical condition may have some effect on work (e.g., broken fingers, splints, use of crutches, conditions impacting mobility), then the supervisor shall notify OH personnel even if no restrictions have been placed by a physician.

    a. Most minor illnesses (e.g., flu, virus, infection) are not required to be reported to OH personnel.

    b. OH personnel will initially evaluate the condition presented to them by supervision and notify the CMD or SEP as appropriate.

    c. If required, then the SEP will evaluate and document the condition with a recommendation whether the medical condition should be reported to the NRC.

12. If a change in license restrictions is indicated by a medical condition, then within 30 days, notify the NRC by completing NRC Form 396, Certification of Medical Examination by Facility Licensee.

5.4.2 Medical Examination Process

NOTE
These instructions apply to initial, license renewal, and periodic medical evaluations.

1. The Licensed Operator's supervisor or designee performs the following:

    a. Complete ADMF-SAF-HSF-00004, Facility Operator's Report Form, and ADMF-SAF-HSF-00017, Supervisor's - To Support Nuclear Occupational Medical Evaluations and Surveillances.

        (1) To locate ADMF-SAF-HSF-00004 and ADMF-SAF-HSF-00017, perform the following:

            (a) Go to the DAE and search under the Shortcuts tab for "FileNet P8 Fusion".

            (b) Open FileNet P8 Fusion and select "Enterprise Fusion"

            (c) Under the search fields, select "Document Number" and type in the desired procedure to view.

5.4.2 Medical Examination Process (continued)

b. Submits the completed Facility Operator's Report Form and the Supervisor's Statement forms to OH.

c. Promptly forwards ADMF-SAF-HSF-00013, Occupational Medical History Form, and ADMF-SAF-HSF-00009, Audiological History and Examination Form, to the Licensed Operator or applicant.

2. Licensed Operator or applicant performs the following:

a. Promptly responds to Duke Energy Medical Information (DEMI) notification emails.

b. Promptly completes the Occupational Medical History Form and returns it to OH.

(1) The medical examination consists of testing by the OH nurse and a medical examination performed by the physician.

(2) These are normally scheduled at least 24 hours apart to allow for the return of lab results.

(3) Contact OH to schedule the medical testing and examination.

3. OH performs the following:

a. Receives and reviews the Facility Operator's Report Form and the Supervisor's Statement for each Licensed Operator or applicant.

b. Schedules medical testing and examinations.

c. Performs medical testing and records the data.

d. Provides medical testing and examination data, the Facility Operator's Report Form, and if applicable, a NRC Form 396 to the SEP.

e. Initiates performance of Attachment 3, Change In Medical Condition Affecting License Status.

f. Completes the NRC Form 396 for permanent medical status changes.

g. Ensures a validation of the medical documents and OH database is performed by another member of OH.

h. Incorporates additional medical information provided by the SEP on the completed NRC Form 396 and forwards it to Regulatory Affairs.

i. Documents medical change in Licensed Operator's medical record.

5.4.2 Medical Examination Process (continued)

j. Files the original completed forms, and any associated medical documents in the Licensed Operator's medical record.

4. SEP performs the following:

a. Schedules physician examination component of the medical examination and completes the appropriate documentation.

NOTE

The discovery date on Attachment 3, Change In Medical Condition Affecting License Status is the date used for calculation of the NRC qualification expiration or for calculation of the 30-day NRC notification requirement for permanent medical changes.

b. If there are changes in the Licensed Operator's temporary or permanent medical status, then:

(1) Informs the Licensed Operator at the time of the medical examination.

(2) Notifies OH.

c. If a permanent medical status change is identified, then a required letter with additional medical information shall be prepared by the SEP with assistance from the OH staff to support the NRC Form 396 submittal.

(1) This information is transmitted to Regulatory Affairs in a sealed envelope marked as CONFIDENTIAL.

(2) Regulatory Affairs may open the sealed envelop to validate the information contained in the letter matches the information provided by the physician.

d. Submits completed medical examination documentation to OH Health.

5. Regulatory Affairs performs the following:

a. Ensures the NRC Form 396 is properly completed.

5.4.2 Medical Examination Process (continued)

b. If a permanent medical change was identified, then:

- Obtains Licensed Operator's signature
- Obtains Site VPs signatures
- Submits NRC Form 396 within 30 days of the medical status discovery date
    - ◊ A letter with additional medical information may be prepared by the SEP to support the NRC Form 396 submittal. This information will be provided to the NRC in accordance with 10 CFR 2.390.

c. Forwards a copy of the submitted NRC Form 396 to:

- Site Training
- Occupational Health
- Affected Licensed Operator
- Licensed Operator's Supervisor

5.4.3 Medical Status Changes

NOTE

These instructions apply to medical status changes identified independently of the routine or periodic medical examinations for Licensed Operators.

1. If any change occurs in a Licensed Operator's medical condition, then the operator performs the following:

    a. Refers to Attachment 5, Common Medical Status Changes/Conditions of Licensed Operators.

    b. Prior to performing licensed duties, notifies supervision and OH.

        (1) If the operator is unable to notify OH of the change in medical condition, then Operations Management will notify OH.

5.4.3 Medical Status Changes (continued)

2. OH performs the following upon notification of a potential medical condition change by a Licensed Operator or supervisor:

   a. Obtains specific medical information and documentation pertinent to the medical condition change from the Licensed Operator.

   b. Notifies SEP or CMD of the medical information obtained and follows the physician's instruction.

   c. Notifies the Licensed Operator of the review with the SEP or CMD:

      (1) If the physician needs to examine the Licensed Operator, then schedules an appointment with the physician.

      (2) If the physician does not need to examine the Licensed Operator, then notifies the Licensed Operator of the physician's opinion.

      (3) Documents all actions taken in the Licensed Operator's medical record.

3. SEP or CMD performs the following:

   a. Determines the extent of the evaluation needed to make a decision regarding a potential medical condition change.

      (1) May require contact between the Licensed Operator or the Licensed Operator's personal medical doctor.

   b. If a clinical assessment is not required, then:

      (1) Advises OH.

      (2) Notifies OH personnel of the decision and provide supporting documentation.

   c. If a clinical assessment is required, then determines the Licensed Operator's condition based on available medical records, results of the personal medical doctor's clinical assessment, and any special testing.

      (1) Notifies OH personnel of the decision and provides supporting documentation.

5.4.3 Medical Status Changes (continued)

d. If license restrictions are indicated or determined, then notify OH.

(1) If a temporary medical status change is identified, then additional clinical assessment at a future date will be warranted to determine if or when temporary restrictions may be removed.

(2) If a permanent medical status change is identified, then a required letter with additional medical information shall be prepared by the SEP with assistance from the Occupational Health staff to support the NRC Form 396 submittal.

(a) This information is transmitted to Regulatory Affairs in a sealed envelope marked as CONFIDENTIAL.

(b) Regulatory Affairs may open the sealed envelop to validate the information contained in the letter matches the information provided by the physician.

(3) If OH identifies a permanent medical status change that is not compatible with maintaining a RO or SRO license, then notify the Operations Manager (or designee) and contact Site HR for Medical Accommodations Program consideration.

4. Occupational Health performs the following upon notification of a medical status change by the SEP or CMD:

a. Initiate performance of Attachment 3, Change In Medical Condition Affecting License Status.

b. Completes the NRC Form 396 for permanent medical status changes.

c. Ensures a validation of the medical documents and OH database is performed by another member of OH.

d. Incorporates additional medical information provided by the SEP on the completed NRC Form 396 and forwards it to Regulatory Affairs.

e. Documents medical change in Licensed Operator's medical record.

f. Files the original completed forms, and any associated medical documents in the Licensed Operator's medical record.

5.4.3 Medical Status Changes (continued)

5. Regulatory Affairs performs the following:

   a. Ensures the NRC Form 396 is properly completed.

   b. If a permanent medical change was identified, then:

      - Obtains Licensed Operator's signature
      - Obtains Site VP's signatures
      - Submits NRC Form 396 within 30 days of the medical status discovery date

        ◊ A letter with additional medical information may be prepared by the SEP to support the NRC Form 396 submittal. This information will be provided to the NRC in accordance with 10 CFR 2.390.

   c. Forwards a copy of the submitted NRC Form 396 to:

      - Site Training
      - Occupational Health
      - Affected Licensed Operator
      - Licensed Operator's supervisor.

5.4.4 Medical Restrictions Follow-Up

1. OH performs the following:

   a. Monitors medical reevaluation dates for Licensed Operators with temporary medical restrictions.

   b. Schedules follow-up testing and examinations.

2. SEP performs the following:

   a. Performs the follow-up examinations and tests as necessary.

   b. Completes clinical documentation and submits documentation to OH.

5.4.4 Medical Restrictions Follow-Up (continued)

3. OH performs the following:

   a. Obtains the clinical documents from the SEP and reviews for completeness and accuracy.

   b. If the SEP reports that the temporary medical restriction is no longer valid, then the following actions are taken:

      (1) Initiates performance of Attachment 3, Change In Medical Condition Affecting License Status

      (2) Incorporates additional medical information provided by the SEP on the completed NRC Form 396 and forwards it to Regulatory Affairs.

      (3) Documents medical change in Licensed Operator's medical record.

      (4) Files the original completed forms, and any associated medical documents in the Licensed Operator's medical record.

5.5 <u>Maintaining Required On-Shift Experience (Proficiency) For Licensed Operators</u>

1. Licensed Operators must comply with 10 CFR 55.53(e) to maintain an Active Licensed Operator Position. {7.1.1}

   a. Licensed Operators must perform five 12 hour shifts per calendar quarter to maintain an active license status.

   b. Time as an extra RO or SRO on an Outage Unit or as RO or SRO in the Outage Command Center CANNOT be counted as an Active Licensed Operator Position.

   c. Time during No Mode CANNOT be counted as an Active Licensed Operator Position.

2. Any Licensed Operator that fails to comply with 10 CFR 55.53(e) requirements shall immediately notify their supervisor. {7.1.1}

   a. The Licensed Operator's supervisor shall notify Operations Training to disqualify the Licensed Operator from licensed duties in accordance with AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS).

5.5 Maintaining Required On-Shift Experience (Proficiency) For Licensed Operators (continued)

3. The Licensed Operator shall document performance of standing an Active Licensed Operator Position by one of the following methods:

   - Attachment 1, Operating Experience Maintenance For Active Licensed Operators
   - 'Credit for Standing Watch' eSOMS report

4. If a Licensed SRO stands all of the required proficiency watches in a SRO position, then the RO portion of the license remains active.

5. Individuals who are licensed on two (or more) similar units at a facility are not required to maintain proficiency on each of the similar units (i.e., performing required watches on a single unit will maintain the active licenses on the similar units).

6. If a Licensed SRO stands mostly RO watches, then to maintain the supervisory portion of the SRO License active, the Licensed SRO must stand at least one complete 12 hour shift per calendar quarter as either SM or CRS.

   a. Failure to complete at least one complete watch during a calendar quarter as either a SM or CRS will result in the SRO License becoming inactive.

      (1) The Operator may still stand watch as a RO until the SRO License is reactivated.

7. Two SROs may obtain active hours at the same time in dual unit Control Rooms as long as each SRO is assigned overall responsibility for a specific unit.

   a. SROs assigned to the Control Room for oversight of specific evolutions may not accrue time that can be credited toward their calendar quarter requirement.

8. Four Reactor Operators may obtain active hours at the same time in dual unit Control Rooms as long as each is assigned either the OATC or the BOP position for a specific unit.

   a. Reactor Operators assigned to the Control Room for specific evolutions may not accrue time that can be credited toward their calendar quarter requirement.

5.6 Activation of a New or Inactive License

1. Licensed Operators must ensure compliance with the terms of 10 CFR55.53(e) for activation of a new or inactive license.

2. Prior to performance of Active Licensed Operator Position duties, the Licensed Operator must complete Attachment 2, Activation of New or Inactive License.

   a. Five 12 hour proficiency watches are not required to be performed in a quarter where reactivation or initial activation is accomplished.

   b. Activation hours CANNOT be counted as part of the proficiency watch standing time for the quarter.

> **NOTE**
>
> The requirement detailed in the following step is not specified in 10 CFR 55, but are required by Duke Energy.

3. The AOM-Shift completes an evaluation, determining the need for any additional training in the following areas:

   - Ops Management Expectations (e.g., watch standing, communications)
   - Security
   - Radiation Protection
   - Emergency Planning
   - Operation
   - Major Plant Modifications

5.7 Removing a Licensed Operator From Duty for Other Than Medical or Requalification Training Status

1. If a Licensed Operator is removed from licensed duty due to management discretion, then the AOM-Shift ensures the SM and Operations Training is notified.

   a. Operations Training disqualifies the Licensed Operator from licensed duties in accordance with AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS).

2. As necessary, the AOM-Shift and the Superintendent Nuclear Operations Training will develop any needed remediation plan.

5.7 Removing a Licensed Operator From Duty for Other Than Medical or Requalification Training Status (continued)

3. If the removal from license duty is temporary, then the condition is not required to be reported per 10 CFR 55.

4. Upon resolution of the disqualifying issue, the AOM-Shift notifies Operations Training to reinstate the Licensed Operator qualifications to perform licensed duties in accordance with AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS).

5. If a Licensed Operator becomes ineligible to perform licensed duties due to lapses in non-requalification training (e.g., SCBA, Respirator Training), then the Licensed Operator informs the SM.

    a. The SM notifies Operations Training to disqualify the Licensed Operator from licensed duties in accordance with AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS).

    b. Upon resolution of the disqualifying issue, the AOM-Shift notifies Operations Training to reinstate the Licensed Operator qualifications to perform licensed duties in accordance with AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS).

5.8 <u>Activating an SRO License For Fuel Handling/Core Alterations Only</u>

1. Prior to an inactive Licensed Operator independently standing watch as a Fuel Handling/Core Alterations SRO, the actions contained in Attachment 6, Activation Of An SRO License For Fuel Handling/Core Alterations Only must be completed.

5.9 <u>[CNS] Licensed Operators Qualification Report (LOQR)</u>

1. The LOQR is a report updated to provide readily available operator license status.

    a. If there is a conflict between the report and the Licensed Operator's believed qualification status, then the operator shall immediately contact the on-duty SM.

5.9 [CNS] Licensed Operators Qualification Report (LOQR) (continued)

2. At a minimum, the LOQR includes:
   - Licensed Operator's name
   - License status (active or inactive)
   - License expiration date
   - NRC physical expiration date
   - Restrictions placed on an operator's license
3. The LOQR shall be printed at least weekly and be readily available to the on duty Shift Manager for review and updating. {7.1.2}

6.0 RECORDS

6.1 <u>QA Record</u>

1. Completed Attachment 1, Operating Experience Maintenance For Active Licensed Operators as described in the attachment.
2. Completed Attachment 2, Activation of New or Inactive License as described in the attachment.

6.2 <u>Business Record</u>

1. Completed Attachment 3, Change In Medical Condition Affecting License Status as described in the attachment.
2. Completed Attachment 4, Notification Of Change In Operator Status {7.1.2} as described in the attachment.
3. Completed NRC Form 396. Filed by Occupational Health with the individuals medical record.
4. Completed Attachment 6, Activation Of An SRO License For Fuel Handling/Core Alterations Only as described in the attachment.

7.0 REFERENCES

7.1 Commitments

1. ACR 92-797, Reactor Operator With An Inactive Licenses State Stood Brief Watch on BOP
2. CAPR 01898054-11, SRO Standing Watch with a Terminated License

7.2 Procedures

1. AD-DC-ALL-0002, Records Management
2. AD-HU-ALL-0004, Procedure Use and Adherence
3. AD-LS-ALL-0002, Regulatory Correspondence
4. AD-SY-ALL-0460, Managing Fatigue And Work Hour Limits
5. AD-TQ-ALL-0068, Licensed Operator Continuing Training Program
6. AD-TQ-ALL-0610, Nuclear Operator License Application Process
7. AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS)

7.3 Miscellaneous Documents

1. 10CFR50.74, Notification of change in operator or senior operator status
2. 10CFR55.23, Certification
3. 10CFR55.25, Incapacitation because of disability or illness
4. 10CFR55, OPERATORS' LICENSES Subpart C—Medical Requirements and Subpart D - Applications, Part 55.31 and Part 55.33
5. 10CFR55.53, Conditions of licenses
6. ADMF-SAF-HSF-00004, Facility Operator's Report Form
7. ADMF-SAF-HSF-00009, Audiological History and Examination Form
8. ADMF-SAF-HSF-00013, Occupational Health History Form
9. ADMF-SAF-HSF-00017, Supervisor's Statement - To Support Nuclear Occupational Medical Evaluations and Surveillances
10. ADMP-SAF-HSF-00091, Occupational Health Programs

7.3 Miscellaneous Documents (continued)

11. ANSI/ANS-3.4 - 1983, Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants

12. NRC RIS 2007-29, Clarified Guidance for Licensed Operator Watch-Standing Proficiency

13. NUREG 1021, Operator Licensing Examination Standards for Power Reactors, Revision 9, Supplement 1; Section ES-605

14. NUREG 1262, Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operators' Licenses

15. Regulatory Guide 1.134, Rev. 2, Medical Evaluation of Licensed Personnel at Nuclear Power Plants

16. RNP Nuclear Condition Report 224245, Including additional information on NRC Form 396

<< Operating Experience Maintenance For Active Licensed Operators >>

1. Licensed Operator's Name ______________________ Employee Number ______________________

2. Active Licensed Operator Position
   a. Shift Manager (SRO)
   b. Control Room Supervisor (SRO)
   c. Operator at the Controls (RO)
   d. Balance of Plant Operator (RO)

3. Refer to Section 5.5 for the requirements for maintaining an active license.
   a. Refer to Attachment 1 Step 2 and document the watches stood in Table 1.

Table 1, Licensed Operator Watch Standing Log

Licensed Operator Position (a, b, c, or d)	Shift Start Date (Month/Day/Year)	Shift (day/night)

4. I hereby certify that the information set forth above is accurate and complete.

   Licensed Operator Signature _________________________________

5. Completed copies of this attachment shall be retained in accordance with AD-DC-ALL-0002, Records Management.

6. NLMS Code:______________________________ NLMS Entry Completed

ATTACHMENT 1
Page 2 of 2

<< Operating Experience Maintenance For Active Licensed Operators >>

Name:____________________________________ Date:_______________

a. NLMS Codes:

- [BNP] BN TRN LOI0006B ROO
- [CNS] RO: CNROLOQR-NCN
- [CNS] SRO: CNSRLOQR-NCN
- [HNP] HN-OPS-LOI0014H
- [MNS] MCO023
- [ONS] TT4691-N Licensed Operator - Operating Experience Maintenance
- [RNP] RN-OPS-LOC0001R-N

QA Record Retention Rule = 421734 (Life of Plant)

ATTACHMENT 2
Page 1 of 11

<< Activation of New or Inactive License >>

1. Licensed Operator's Name: ____

   Employee Number: ________________ Position: ___________ (RO/CRS/SM)

2. Prior to initiation of the activation process, the Licensed Operator ensures all requirements of AD-SY-ALL-0460, Managing Fatigue And Work Hour Limits, are met, including validating that the individual's status in EmpCenter is "Covered."

3. Date of initiation of the activation process: ____________________

   Required Completion Date: ___________________

   - The required completion date must ensure that all hours required for activation are contained in the same calendar quarter.

4. Pre-job briefing completed.

   AOM-Shift or SM: ________________________________ Date: ______________

NOTE
Attachment 2 Step 5 through Attachment 2 Step 10 can be completed in any order.

5. An evaluation has been performed and additional training needs are identified below: (Section 5.6 Step 3)

   AOM-Shift: _____________________________________ Date: ______________

   ____

   ____

   ____

   ____

6. I hereby certify that all requalification training is up-to-date (N/A for initial licenses).

   a. For SMs and SM reliefs license reactivation, this includes verification that the SM or SM relief has participated in an active simulator evaluation (ASE) or station drill as the SM within the past two years.

   Operations Training Superintendent: ________________ Date: ______________

<< Activation of New or Inactive License >>

7. I hereby certify that medical qualifications are current.

   Occupational Health Representative: __________________ Date: ___________

8. Verify that respirator/SCBA qualifications are current in NLMS.

   Licensed Operator: _________________________________ Date ____________

NOTE
Only one individual is allowed to parallel an active Licensed Operator at a time.

9. Licensed Operator being activated will:

   a. Perform 40 hours in an Active Licensed Operator Position to which the individual will be assigned under the direction of an active Licensed RO or SRO.

      (1) Must include participation in an ongoing and off going turnover, and a shift briefing.

      (2) Up to eight hours of the 40 activation hours may be spent completing the plant familiarization tour.

      (3) All 40 activation hours are required to be completed in the same calendar quarter.

   b. At the beginning of each shift, create a logbook entry stating the watch is being assumed under instruction.

   c. At the end of each shift, obtain an active SRO or RO signature certifying that the Licensed Operator being activated worked under the signatures direction.

   d. Upon completion of under instruction watch, complete Table 2, Licensed Operator Under Instruction Logbook

<< Activation of New or Inactive License >>

e. Complete a plant familiarization tour during the 40 hours or reactivation hours under the direction of an active Licensed SRO. Multiple personnel activating a license may participate in the same plant familiarization tour. The plant familiarization tour shall include the following:

- Up to 8 hours of reactivation time (total 40 hours) should be spent touring the plant.
- Contaminated areas or high radiation areas are not required to be accessed.
- All major equipment in tour areas shall be discussed.
- Plant familiarization tour shall include a review of all AO shift turnover procedures.
- Discuss an plant modification recently installed.
- Document the plant familiarization tour of the site specific areas using the site specific lists at the end of this attachment.
- Attach a copy of the security door printouts for both the licensed individual reactivating and the active Licensed SRO associated with the plant familiarization tour.

Table 2, Licensed Operator Under Instruction Logbook

Date	# of Hours	Under the Direction of	
		Printed Name	Signature
	Oncoming shift turnover		
	Off going shift turnover		

<< Activation of New or Inactive License >>

10. I hereby certify that the plant familiarization tour and 40 hours of activation time in Step 9 are complete and were completed within the same calendar quarter.

    Licensed Operator: ______________________________ Date: ______________

    SM (on duty): __________________________________ Date: ______________

11. I hereby certify that the Training identified in Attachment 2 Step 5 is complete.

    SM (in charge): ________________________________ Date: ______________

12. All licensing requirements met.

    Ops Training Superintendent: _____________________ Date: ______________

13. I hereby certify the Licensed Operator's qualifications and status as current and valid and the required shift functions have been performed and authorize resumption of functions defined by Technical Specifications as follows:

    - Licensed Operator's qualifications are current
    - Licensed Operator's shift functions have been performed
    - Plant tour is complete

    AOM-Shift: ____________________________________ Date: ______________

14. The individual attempting to activate a license notifies the on-duty SM to:

    a. Notify Operations Training to reinstate the Licensed Operator qualifications to perform licensed duties in accordance with AD-OP-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS).

    b. [CNS] Update LOQR active/inactive license status

    AOM-Shift/designee OR SM/designee: ______________ Date: ______________

<< Activation of New or Inactive License >>

15. NLMS Code:________________________________ NLMS Entry Completed

Name:_____________________________________ Date:_____________

a. NLMS Codes:

- [BNP] BN TRN LOI0006B ROO
- [CNS] RO: CNROLOQR-N
- [CNS] SRO: CNSRLOQR-N
- [HNP] HN-OPS-LOI0014H
- [MNS] MCO023
- [ONS] TT4691-N Licensed Operator - Operating Experience Maintenance
- [RNP] RN-OPS-LOC0001R-N

16. The following is retained in accordance with AD-DC-ALL-0002, Records Management.

- Completed copies of this attachment.
- Applicable site completed Plant Familiarization Tour table.
  - ◊ Any Plant Familiarization Tour table not used to complete this attachment can be discarded.
- Copies of the security logs for the time spent on shift for the individual attempting to activate a license and the operator responsible for oversight of the individual attempting to activate a license.
- Copies of the security records for all areas entered during the required plant familiarization tour for the operator attempting to activate a license and the operators conducting the tour.
- Copies of Operations narrative log while standing watch under instruction.

<< Activation of New or Inactive License >>

Table 3, Brunswick Reactivation Plant Familiarization Tour

Date	Area	Active Licensed SRO		Elapsed Time
		Printed Name	Signature	
	Unit 1 Reactor Building			
	Unit 2 Reactor Building			
	Turbine Building			
	DG 4 Day Tanks			
	DG Building			
	CB HVAC			
	Cable Spread			
	Service Water Building			
	Intake Structure			
	AOG			
	Radwaste			
	Transformer Yard			
	MWT - FP Area			
	Security CAS			
	Review of AO Turnover Sheets			
			Total Hours	____________

<< Activation of New or Inactive License >>

Table 4, Catawba Reactivation Plant Familiarization Tour

Date	Area	Active Licensed SRO		Elapsed Time
		Printed Name	Signature	
	Aux Bldg Elevation 594			
	Aux Bldg Elevation 577			
	Aux Bldg Elevation 560			
	Aux Bldg Elevation 543			
	Aux Bldg Elevation 522			
	Unit 1 ETA/ETB Rooms			
	Unit 2 ETA/ETB Rooms			
	Unit 1 Cable Spreading Rm			
	Unit 2 Cable Spreading Rm			
	Unit 1 Spent Fuel Building			
	Unit 2 Spent Fuel Building			
	Unit 1 Vital Battery Area			
	Unit 2 Vital Battery Area			
	Unit 1 CA Pump Room			
	Unit 2 CA Pump Room			
	Unit 1 D/G Rooms			
	Unit 2 D/G Rooms			
	Unit 1 Turbine Operating Deck			
	Unit 2 Turbine Operating Deck			
	Unit 1 Turbine Mezz Level			
	Unit 2 Turbine Mezz Level			
	Unit 1 Turbine Basement			
	Unit 2 Turbine Basement			
	SSF			
	Unit 1 Exterior Doghouse			
	Unit 2 Exterior Doghouse			
	Unit 1 Interior Doghouse			
	Unit 2 Interior Doghouse			
	Review of AO Turnover Sheets			
			Total Hours	________

QA Record Retention Rule = 421734 (Life of Plant)

ATTACHMENT 2
Page 8 of 11

<< Activation of New or Inactive License >>

Table 5, Harris Reactivation Plant Familiarization Tour

Date	Area	Active Licensed SRO		Elapsed Time
		Printed Name	Signature	
	RAB (all levels)			
	Turbine Building			
	Diesel Generator Building			
	Waste Processing Building			
	Water Treatment Facility			
	Fuel Handling Building			
	Diesel Fuel Oil Storage Building			
	ESW Structure			
	ESW Screening Structure			
	Review of AO Turnover Sheets			
			Total Hours	____________

<< Activation of New or Inactive License >>

Table 6, McGuire Reactivation Plant Familiarization Tour

Date	Area	Active Licensed SRO		Elapsed Time
		Printed Name	Signature	
	Aux Bldg Elevation 767			
	Aux Bldg Elevation 750			
	Aux Bldg Elevation 733			
	Aux Bldg Elevation 716			
	Aux Bldg Elevation 695			
	Unit 1 ETA/ETB Rooms			
	Unit 2 ETA/ETB Rooms			
	Unit 1 Cable Spreading Rm			
	Unit 2 Cable Spreading Rm			
	Unit 1 Spent Fuel Building			
	Unit 2 Spent Fuel Building			
	Vital Battery Area			
	Unit 1 CA Pump Room			
	Unit 2 CA Pump Room			
	Unit 1 D/G Rooms			
	Unit 2 D/G Rooms			
	Unit 1 Turbine Operating Deck			
	Unit 2 Turbine Operating Deck			
	Unit 1 Turbine Mezz Level			
	Unit 2 Turbine Mezz Level			
	Unit 1 Turbine Basement			
	Unit 2 Turbine Basement			
	SSF			
	Unit 1 Exterior Doghouse			
	Unit 2 Exterior Doghouse			
	Unit 1 Interior Doghouse			
	Unit 2 Interior Doghouse			
	Review of AO Turnover Sheets			
			Total Hours	________

QA Record Retention Rule = 421734 (Life of Plant)

<< Activation of New or Inactive License >>

Table 7, Oconee Reactivation Plant Familiarization Tour

Date	**Area**	**Active Licensed SRO**		**Elapsed Time**
		Printed Name	Signature	
	CT-4 Blockhouse			
	SSF			
	Turbine Building			
	Auxiliary Building			
	Unit 1 Equipment Room			
	Unit 1 Cable Room			
	Unit 2 Equipment Room			
	Unit 2 Cable Room			
	Unit 3 Equipment Room			
	Unit 3 Cable Room			
	Unit 1 Control Battery Room			
	Unit 2 Control Battery Room			
	Unit 3 Control Battery Room			
	Unit 1 and 2 Spent Fuel Building			
	Unit 3 Spent Fuel Building			
	Keowee Hydro			
	Review of AO Turnover Sheets			
			Total Hours	____________

<< Activation of New or Inactive License >>

Table 8, Robinson Reactivation Plant Familiarization Tour

- Plant tours shall include entry and visual surveillance of each room in the Turbine Building and RCA that is not specifically excluded below or in writing by the SM.
- Areas excluded from the plant familiarization tour:
  - ◊ Containment Vessel
  - ◊ RHR Pit
  - ◊ Office buildings that are not part of the watchstanders normal tour

Date	Area	Active Licensed SRO		Elapsed Time
		Printed Name	Signature	
	CR HVAC			
	4kV Room			
	E1/E2, Battery, Cable Spread Rooms			
	Building 469 - 1st and 2nd floor			
	AFW Pump Room			
	Turbine Building (all levels)			
	Security Building (CAS)			
	SFP Area			
	SFP Hx and Bit Room			
	Auxiliary Building (level 1 and 2)			
	Security Pap Diesel			
	Intake			
	Review of AO Turnover Sheets			
			Total Hours	____________

<< Change In Medical Condition Affecting License Status >>

Operator's Name:________________ Employee Number:__________ Date:_________

NOTE

- This form is to be completed when any change in medical condition affects the employee's ability to perform licensed duties.
- Discovery date is the date and time the Licensed Operator has notified OH of a change in medical condition.
- AD-HU-ALL-0004, Procedure Use and Adherence, Section 5.6, provides guidance on how to appropriately placekeep when performing this attachment.

1. The following is completed by OH:

   a. Record discovery date and time: __________/ __________
   Date Time

   b. Ensure documentation is placed in Licensed Operator's medical file. (Return to work forms or Nurse's documentation).

   c. Identify status of condition (circle one): PERMANENT/TEMPORARY

   d. Notify CMD or SEP, if applicable.

      (1) CMD or SEP Name:________________________

   e. Ensure license duty eligibility is determined by qualified OH health personnel:

   _____ Ineligible

   _____ Eligible with the following temporary restrictions:

   ____

   ____

   _____ Eligible to perform licensed duties

   _____ Eligible with the following permanent restrictions

   ____

   ____

<< Change In Medical Condition Affecting License Status >>

Operator's Name:________________ Employee Number:__________ Date:_________

f. If the Licensed Operator is medically ineligible to perform licensed duties, then OH shall perform the following:

(1) Enter the Licensed Operator medical status change in the Duke Employee Medical Information (DEMI) system.

(a) If DEMI CANNOT be updated by the end of a shift after a Licensed Operator has been determined ineligible to perform licensed duty, then contact the on-duty SM to disqualify the Licensed Operator from licensed duties in accordance with AD-TQ-ALL-0660, Use and Administration of the Nuclear Learning Management System (NLMS).

SM name:___________________________

(2) Verify the Licensed Operator medical status change is reflected in NLMS.

2. OH notifies the affected Licensed Operator of the following:

a. The impact this change in medical status has on license duty eligibility.

(1) If ineligible to perform licensed duties, then provide direct and verbal notification to the Licensed Operator that licensed duties shall NOT be performed.

(a) If direct and verbal notification to the Licensed Operator CANNOT be established, then provide direct and verbal notification to the AOM-Shift that the Licensed Operator shall NOT perform licensed duties.

b. Comply with restrictions and conditions anytime the Licensed Operator performs license duties.

<< Change In Medical Condition Affecting License Status >>

Operator's Name:________________ Employee Number:__________ Date:_________

c. Actions required to address the medical condition may include, but are not limited to the following:

- Additional medical evaluation and treatment
- Obtaining appropriate medications
- Obtaining proper hearing enhancement devices
- Obtaining appropriate corrective lenses, including lenses for SCBA
- Obtaining therapeutic medical device

3. OH performs the following:

a. Contacts the on-duty SM.

(1) Name:______________________________ Date:__________

b. Informs the on-duty SM of the affected Licensed Operator's change in medical status and to notify the AOM-Shift and Licensed Operator's supervisor.

c. [CNS] Informs the on-duty SM to update the LOQR.

4. If the medical condition status is determined to be permanent, then ensure a NTM is generated, documenting that a change in medical status has occurred for the affected Licensed Operator:

a. Record NTM Number:_____________

(1) Include a NTM assignment for Regulatory Affairs to complete required regulatory correspondence.

b. Transmit notification to Regulatory Affairs as follows:

(1) Send email or fax copy of this attachment.

(2) Verify information received by a member of Regulatory Affairs:

(a) Name: _____________________________________

<< Change In Medical Condition Affecting License Status >>

Operator's Name:________________ Employee Number:__________ Date:_________

c. Ensure the following:

- NRC Form 396 is completed.
- Corporate Medical Director or Site Examining Physician notification letter are completed per AD-LS-ALL-0002, Regulatory Correspondence.
- Notification letter is delivered to Regulatory Affairs.

5. OH completes the following:

   a. If permanent medical restrictions are being added, then notify Regulatory Affairs to generate correspondence and notify NRC Region II within 30 days of the date the license restriction was discovered.

   b. If medical restrictions are being removed, then notify Regulatory Affairs to generate correspondence and notify NRC Region II within 30 days of the date of the medical change in condition.

   c. Record NTM number:________________________

6. Place completed copies of this attachment in the Licensed Operator's medical file.

<< Notification Of Change In Operator Status >> {7.1.2}

NOTE

- Review NCR 01898054 {CAPR} before making any changes to this form.

Licensed Operator (name): ______________________________ will no longer perform the duties of a Licensed Senior/Reactor Operator effective as of the License Termination Date listed below.

Employee Number: __________________________

License Termination Date: ____________________

Describe Reason: (Transfer, termination)

____

____

____

____

NOTE

The signatures below indicate that the information stated above is correct.

AOM-Shift/designee Signature	Date
Licensed Operator Signature	Date
Licensed Operator's Manager Signature	Date

<< Notification Of Change In Operator Status >> {7.1.2}

NOTE

The notification of change in operator status must be received by the NRC within 30 days of the effective date per 10CFR50.74.

1. AOM-Shift to ensure the following:

   a. Generate an NTM for operator license termination.

   b. Ensure NTM includes the following:

      - License Termination Date
      - Direction to assign NTM ownership to Regulatory Affairs
      - Direction for Regulatory Affairs to generate all necessary corrective actions for license termination per Attachment 4, Notification Of Change In Operator Status {7.1.2}

   c. Notify Site Operations Training Group to commence actions to deactivate the affected operator's qualifications effective on the License Termination Date.

   d. Route this form to Regulatory Affairs.

______________________	____________	____________
AOM-Shift/designee Signature	Date	NTM Number

2. Regulatory Affairs ensure the following:

   a. Generate License Termination letter.

   b. Assemble verification package.

      (1) A copy of the termination letter shall be distributed to the following individuals for verification of accuracy prior to sending the termination letter to the NRC:

         (a) Licensed Operator

         (b) Licensed Operator's manager

         (c) AOM-Shift

         (d) Operations Manager

<< Notification Of Change In Operator Status >> {7.1.2}

c. Create assignments within the associated license termination NTM as follows:

(1) For each NTM assignment:

- Enable the CAS Lock Due Date function with site Regulatory Affairs group as Due Date Owner.
- Due dates shall not exceed the License Termination Date except as provided.

(2) Create NTM assignments as follows:

- [CNS] Operations to update LOQR to disqualify the Licensed Operator
- Occupational Health staff to update operators file/record that license is terminated.
- Emergency Planning to update operator's EP file/record that license is terminated.
- Operations Training Group to deactivate the operator's qualifications on the License Termination Date (not earlier nor later than License Termination Date).
- Operations Training Group to file completed copy of Attachment 4, Notification Of Change In Operator Status {7.1.2} in operator's training file with Due Date not to exceed two weeks after License Termination Date.
- Other actions and assignments as desired

d. Ensure change in license status is reported to the Nuclear Regulatory Commission per 10CFR 50.74.

e. Route this form to AOM-Shift or deliver to on-duty SM/designee.

____________________________ ____________________
Regulatory Affairs Signature Date

<< Notification Of Change In Operator Status >> {7.1.2}

3. AOM-Shift or on-duty SM ensure the following:

- [CNS] Update LOQR
- Route this form to the Site Occupational Health staff.

______________________________ ____________________
AOM-Shift or SM Signature Date

4. Occupational Health staff ensure the following:

- Update operator's records.
- Route this form to Emergency Planning

______________________________ ____________________
Occupational Health personnel signature Date

5. Emergency Planning ensure the following:

- Update Emergency Planning Records.
- Route form to site Operations Training Group.

______________________________ ____________________
Emergency Planning Staff Signature Date

6. Site Operations Training Group ensure the following:

- Deactivate the affected operator's qualifications as of the effective date of the license termination.
- Completed copies of this attachment shall be retained in accordance with AD-DC-ALL-0002, Records Management.

______________________________ ____________________
Operations Training Signature Date

<< Common Medical Status Changes/Conditions of Licensed Operators >>

NOTE

- Complete details associated with Licensed Operator Medical Examinations is located in ADMP-SAF-HSF-00091, Occupational Health Programs.

1.0 COMMON MEDICAL CONDITIONS REQUIRING OCCUPATIONAL HEALTH AND SUPERVISION NOTIFICATION

- The inability to properly fit or effectively use personal protective equipment (PPE)
- Impairment of the sense of smell
- Loss of the capacity for clear speech
- Hearing becomes impaired
- Vision changes (e.g., visual acuity, peripheral vision, color vision or depth perception)
- Respiratory capacity becomes impaired
- Impairment of the operator's cardiovascular system (e.g., hypertension, myocardial infarction, coronary stent and coronary bypass)
- Development of any type of hernia
- Impairment of muscular-skeletal range of motion or power
- Inability of the skin to tolerate PPE or decontamination procedures
- Changes to the endocrine or metabolic systems, such as diabetes and thyroid disorders, such that the ability to change schedules is affected, or the individual could become incapacitated if meals are delayed
- Impairment of the ability to form blood such as anemia, leukemia, lymphoma and multiple sclerosis
- Impairment of lymphatic function
- Impairment neurological function
- Development of mental, emotional, or behavioral disorders

<< Common Medical Status Changes/Conditions of Licensed Operators >>

1.0 COMMON MEDICAL CONDITIONS REQUIRING OCCUPATIONAL HEALTH AND SUPERVISION NOTIFICATION (continued)

- Abnormal laboratory results
- Chronic fatigue
- Diagnosis of any cancer, including skin cancer
- Diagnosis of sleep apnea
- Use of therapeutic device such as CPAP
- Returning from hospitalizations due to heart attack and major surgery
- Elective surgery that may change or improve an individual's medical condition (e.g., lasik).

2.0 COMMON CHANGES IN MEDICAL STATUS – (NORMALLY REPORTED TO NRC)

- Hypertension/Blood Pressure medication
- Diabetes Mellitus
- Vision change - near/distant
- Myocardial infarction or coronary stents
- Syncopal episode
- Seizure
- Sleep Apnea
- Anxiety
- Depression
- Impaired Hearing Acuity, Hearing Aid
- All cancers including skin cancers
- Drug or alcohol issues

2.0 COMMON CHANGES IN MEDICAL STATUS – (NORMALLY REPORTED TO NRC) (continued)

- Thyroid medication
- Glaucoma
- Impaired Tactile Discrimination
- Chemotherapy
- Pulmonary Embolus or Deep Vein Thrombosis
- Attention Deficit Hyperactivity Disorder (ADHD)
- Multiple Myeloma
- Repair of leg aneurysm
- Sarcoidosis
- Hip or knee replacement (Reported as for information only, unless a permanent restriction is needed)
- Cholesterol medication (Reported as for information only)
- Sleep medications (Reported as for information only)
- Olfactory Deficit
- EpiPen
- Rheumatoid Arthritis
- Asthma (Frequent severe attacks within previous 2 years or need for prolonged or continued use of medication)

<< Common Medical Status Changes/Conditions of Licensed Operators >>

3.0 COMMON MEDICAL CONDITIONS REQUIRING NO SOLO OPERATION

- Insulin requiring diabetes mellitus
- Myocardial infarction
- Coronary stents (No Solo after initial stent insertion, case by case determination)
- Unexplained syncopal episodes
- Kidney transplant
- Narcolepsy
- Olfactory deficit
- Migraine Headache associated with syncope or potential incapacitation
- Treatment with Coumadin (warfarin) (case by case determination)

4.0 MEDICAL CONDITIONS REQUIRING RESTRICTIONS OF LICENSED OPERATORS

- Hypertension requiring medication
- Corrective lens
- Hearing Aid
- Sleep Apnea with CPAP treatment (Therapeutic Medical Device) must have documentation from personal physician stating compliance with machine.
- Myocardial Infarction (No Solo)
- Diabetes Mellitus requiring medication (Use of insulin is No Solo)
- Insulin Pump (Therapeutic Medical Device)
- Glaucoma requiring medication
- Unexplained syncopal episode (No Solo)
- Chemotherapy (Restriction or No Solo)
- Treatment with Coumadin (warfarin) (No Solo)

<< Common Medical Status Changes/Conditions of Licensed Operators >>

4.0 MEDICAL CONDITIONS REQUIRING RESTRICTIONS OF LICENSED OPERATORS (continued)

- Pulmonary Embolus or Deep Vein Thrombosis (case by case determination)
- Olfactory Deficit (No Solo)
- Seizure Disorder (Restriction or No Solo)
- Kidney Transplant (No Solo)
- Migraine Headache associated with syncope or potential incapacitation (No Solo)
- Impaired Tactile Discrimination (No Solo)
- Attention Deficit Disorder treatment with medication
- EpiPen (Case by case determination based on if the operator is at risk for a severe allergic reaction and carries it at all times)

ATTACHMENT 6
Page 1 of 2

<< Activation Of An SRO License For Fuel Handling/Core Alterations Only >>

Operator's Name: ___________________ Employee Number: ___________ Date: ___________

1. Verify refueling training is up-to-date and completed successfully for the Refueling SRO/Reactor Building SRO position(s).

   _________________________________ ______________________
   AOM-Shift/designee Date

2. Verify the medical qualification is current for the assigned position(s).

   _________________________________ ______________________
   AOM-Shift/designee Date

3. Record the completion of at least one 12-hour (continuous time period) shift of parallel refueling operation, equipment familiarization, tours, procedure preparation for beginning of fuel movement and shift turnovers required for the assignment/return to active license duties for refueling only in the Refueling SRO/positions. During the 12-hour shift, the inactive SRO shall perform the duties related to fuel handling in parallel with an SRO assigned to the Refueling SRO position that holds an active license. No more than 6 hours can be credited for activities conducted prior to the actual start of fuel movement.

   Date of 12-hour shift Active SRO (initials)

   ______________________ ______

   ______________________ ______

4. I certify the above individual for assignment/return to active Refueling SRO duties for outage _________________ (e.g., 2CR22).

   _________________________________ ______________________
   SM Date

5. AOM-Shift files and retains this form for the duration of the core alteration for which the license was activated.

   _________________________________ ______________________
   AOM-Shift/designee Date

ATTACHMENT 6
Page 2 of 2

<< Activation Of An SRO License For Fuel Handling/Core Alterations Only >>

6. Completed copies of this attachment shall be retained in accordance with AD-DC-ALL-0002, Records Management.

______________________________	______________________
AOM-Shift/designee	Date

JPM A.2S

SRO

Catawba Nuclear Station
Admin. JPM A.2S
Sept 2021 NRC Exam

EVALUATION SHEET

Task: Use Flow Diagrams and Electrical Prints to Determine Work Isolation Boundary

Alternate Path: N/A

Facility JPM #: New

Safety Function: N/A

K/A 2.2.41 Ability to obtain and interpret station electrical and mechanical drawings.

Importance: 3.5 / 3.9 **CFR:** 41.10 / 45.12 / 45.13

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom __X__ Perform __X__ Simulate ________

References: Flow diagram of the CA system (CN 2592-01), 4.16 KV bus one line electrical drawings

Task Standard: Mechanical and electrical isolation boundary determined for work on 2A CA pump per JPM A.2 key.

Validation Time: 30 minutes **Time Critical:** Yes _____ No __X__

Applicant:
NAME____________________________ Docket #______________ Time Start: __________
Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

____

____

____

Catawba Nuclear Station
Admin. JPM A.2S
Sept 2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **The 2A Auxiliary Feedwater (CA) Pump has been shutdown in accordance with OP/2/A/6250/002 (Auxliary Feedwater System) and is to be tagged out for pump casing disassembly and impeller replacement.**

INITIATING CUES:

- **The SM has directed you to use the provided materials to determine the required boundary for isolation for this work.**
- **You are to use the valves closest to the work being performed for isolation to minimize drain and fill time.**
- **Identify components, including the required position, for creation of a Clearance for 2A Auxiliary Feedwater (CA) Pump, including:**
  - **Mechanical isolations**
  - **Electrical isolations**
  - **Applicable Vent and Drain path**
- **Record your answer in the table on the following page.**

EXAMINER NOTE:

After reading cue, provide applicant with a copy of CA flow diagrams (CN 2592-01), and 4.16 KV Switchgear one line diagrams (CN 2702-2.01, CN 2702-2.02).

STEP / STANDARD	**SAT / UNSAT**

START TIME: _______

STEP 1: Determine clearance boundary for the 2A CA Pump boundary. STANDARD: Applicant identifies clearance boundary per table on the next page. Note that only one vent valve or one drain valve plus drain valve 2CA-103 is needed to meet the critical step. **Examiner Note: This step is critical to be able to correctly isolate the 2A Auxiliary Feedwater (CA) Pump for pump casing disassembly and impeller replacement.** COMMENTS:	**CRITICAL STEP** ___ SAT ___ UNSAT

STEP / STANDARD	SAT / UNSAT

Component	Position
2ETA13	Racked Out
2CA-29	Closed
2CA-87	Closed
2CA-25	Closed
2CA-154 (Drain, Can be used)	Open
2CA-103 (Common Drain, Must be used)	Open
2CA-153 (Drain, Can be used)	Open
2CA-101 (Drain, Can be used)	Open
2CA-83 (Vent)	Open

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **The 2A Auxiliary Feedwater (CA) Pump has been shutdown in accordance with OP/2/A/6250/002 (Auxiliary Feedwater System) and is to be tagged out for pump casing disassembly and impeller replacement.**

INITIATING CUES:

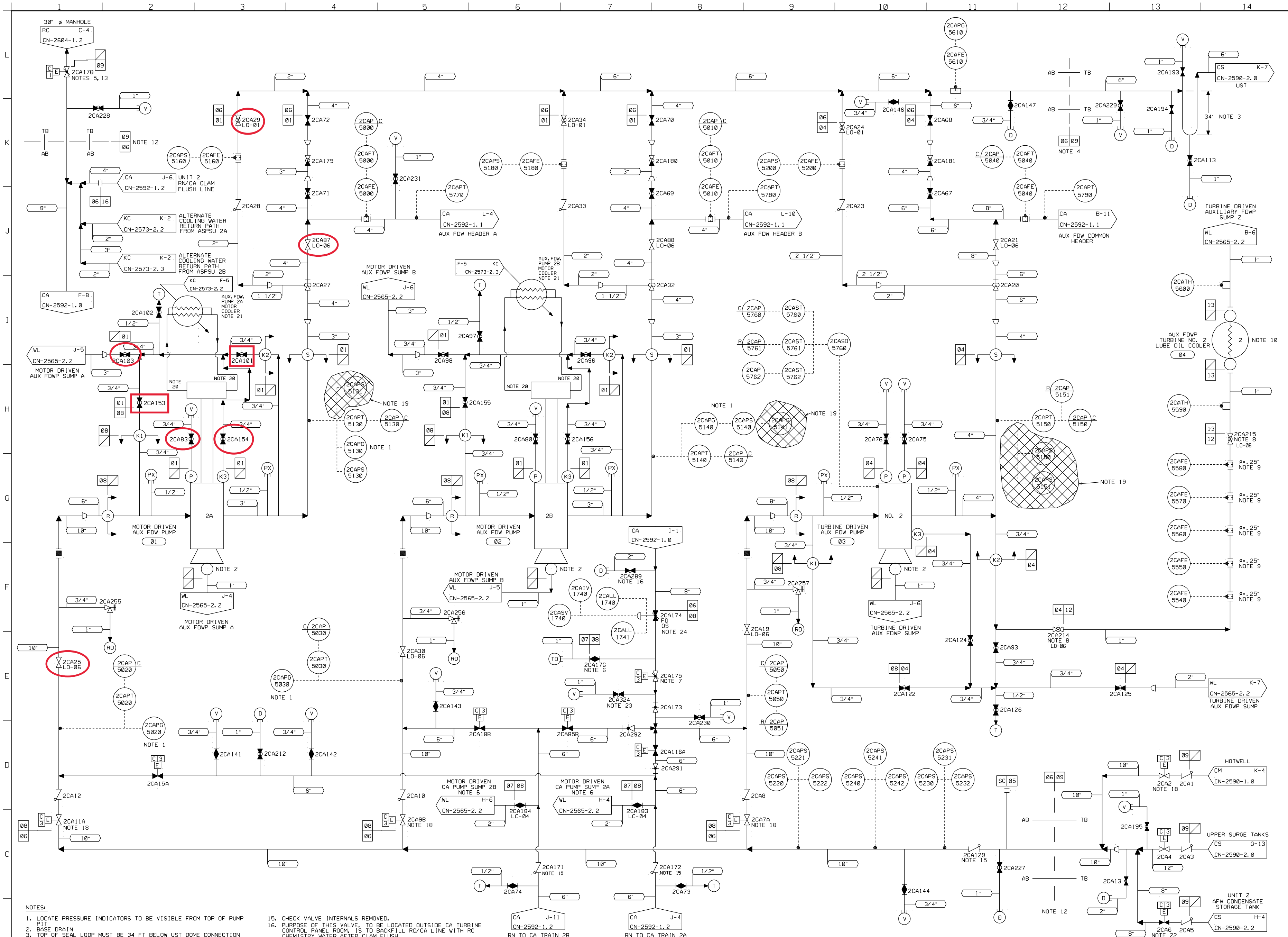
- **The SM has directed you to use the provided materials to determine the required boundary for isolation for this work.**
- **You are to use the valves closest to the work being performed for isolation to minimize drain and fill time.**
- **Identify components, including the required position, for creation of a Clearance for 2A Auxiliary Feedwater (CA) Pump, including:**
  - **Mechanical isolations**
  - **Electrical isolations**
  - **Applicable Vent and Drain path**
- **Record your answer in the table on the following page.**

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

Answer

Component	Position



NOTES:

1. LOCATE PRESSURE INDICATORS TO BE VISIBLE FROM TOP OF PUMP PIT
2. BASE DRAIN
3. TOP OF SEAL LOOP MUST BE 34 FT BELOW UST DOME CONNECTION
4. CLASS F PIPE FROM LAST SUPPORT IN TRENCH BEFORE AUX BLDG PENETRATION
5. LOCATE VALVE 2CA178 AT CONNECTION TO RC SYSTEM. PIPE IN TB TO BE EMBEDDED IN TB BASEMENT FLOOR, VALVE AND RC SYSTEM CONNECTION TO BE PROTECTED BY CONCRETE ENCLOSURE.
6. TEST DRAIN USED FOR FUNCTIONAL TEST OF CHECK VALVE. DISCHARGE TO BE VISIBLE TO VERIFY FLOW
7. POWERED FROM SSF 220/125VDC SYSTEM.
8. KEROTEST VALVES ARE UNACCEPTABLE FOR THIS APPLICATION
9. 2CAFE 5540, 5550, 5560, 5570, 5580 CONSIST OF MACHINED SOCKET WELD PIPE COUPLINGS INSTALLED IN SERIES WITH APPROXIMATELY 8" - 10" SPACING.
10. LUBE OIL (SHELL SIDE) OF HEAT EXCHANGER IS CONNECTED WITH AUX FDW PUMP TURBINE LUBE OIL SYSTEM WHICH IS NOT SHOWN ON ANY FLOW DIAGRAM. CONSULT AUX FDW TURBINE INSTRUCTION MANUAL FOR DETAILS.
11.
12. PIPING IS CLASS F IN TURBINE BLDG. TRENCH.
13. POWERED FROM SSF AC NORMAL AUXILIARY POWER SYSTEM.
14. NO PIPING SHOWN ON THIS FLOW DIAGRAM REQUIRES INSULATION.
15. CHECK VALVE INTERNALS REMOVED.
16. PURPOSE OF THIS VALVE, TO BE LOCATED OUTSIDE CA TURBINE CONTROL PANEL ROOM, IS TO BACKFILL RC/CA LINE WITH RC CHEMISTRY WATER AFTER CLAM FLUSH.
17. FUTURE CONNECTION FOR NSM CN-20525/00
18. MOTOR OPERATOR BREAKER TO BE DISCONNECTED WITH VALVE FULLY OPEN.
19. ABANDONED IN PLACE PER NSM CN-21371/00.
20. VENDOR SUPPLIED PIPING. PROCESS FLUID IS NOT IN THESE LINES.
21. COOLERS SHOW FOR INFORMATION ONLY. REFERENCE KC SYSTEM FLOW DIAGRAMS FOR SPECIFIC INFORMATION.
22. 2CA6 IS CLOSED WITH MOTOR OPERATOR POWER REMOVED (BKR 2MXW F05B OPEN) PER EC 106582 TO PREVENT POSSIBLE VORTEXING IN THE CACST. 2CA6 CAN BE OPEN IN MODES 3 AND 4 DURING PERIODS OF LOW DECAY HEAT (DEFINED AS AFTER REFUELING) WITH BOTH CACSTS ALIGNED AT FULL VOLUME TO THE OUTAGE UNIT (REF. EC 411218). 32
23. 2CA324 SUPPORTS FILLING BETWEEN 2CA174 AND 2CA173 WITH CONDENSATE GRADE WATER. (REF. EC 110084)
24. 2CA174 WILL NOT FAIL OPEN (FOR AT LEAST 60 MINUTES) ON A VI SYSTEM FAILURE UPSTREAM OF 2CACK1740 WHEN 2CASV1740 IS ENERGIZED (REF. CN-2499-CA.16-00 AND EC 110084).

DESIGN PARAMETERS

LINE LISTING	PIPE SPEC	PRESSURE	TEMPERATURE	CLASS	MATERIAL
01	2500.3	1800 PSIA	160°F	C	CS
04	2500.3	2000 PSIA	160°F	C	CS
06	150.4	165 PSIA	160°F	F	CS
07	150.4	50 PSIA	160°F	G	CS
08	150.3	165 PSIA	160°F	C	CS
09	150.4	165 PSIA	160°F	G	CS
12	2501.3	2000 PSIA	160°F	C	SS
13	151.3	165 PSIA	160°F	C	SS
16	151.4	165 PSIA	160°F	F	SS

ABANDONED COMPONENTS RELATED

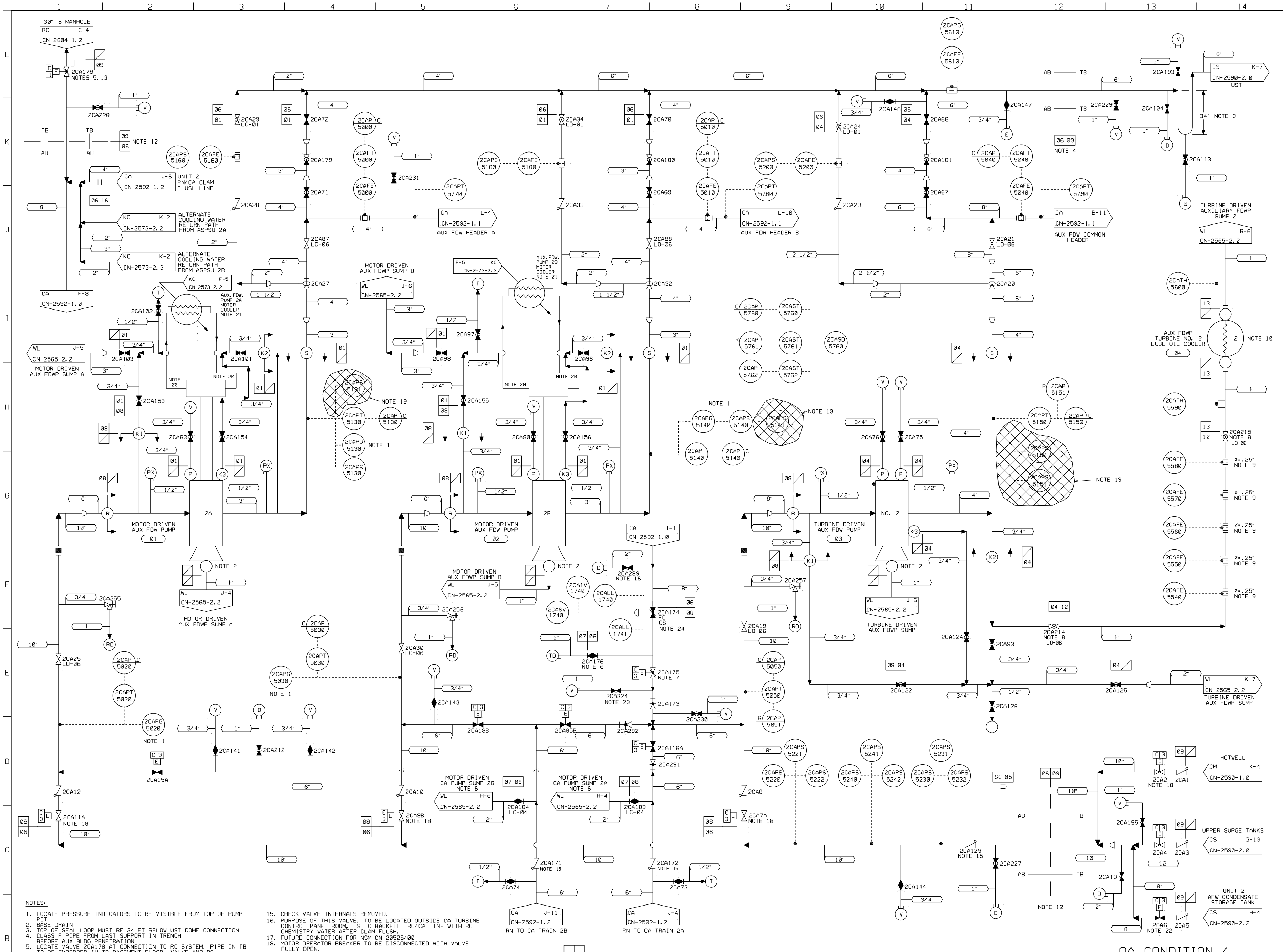
QA CONDITION 4
QA CONDITION 1

DUKE ENERGY
CATAWBA NUCLEAR STATION UNIT 2
FLOW DIAGRAM OF
AUXILIARY FEEDWATER SYSTEM
(CA)

NO.	REVISIONS
32	REV PER EC 411218

SCALE N/A | DWG. NO. CN-2592-01.00 | REV. 32

ERN: CN000THN



NOTES:

1. LOCATE PRESSURE INDICATORS TO BE VISIBLE FROM TOP OF PUMP PIT
2. BASE DRAIN
3. TOP OF SEAL LOOP MUST BE 34 FT BELOW UST DOME CONNECTION
4. CLASS F PIPE FROM LAST SUPPORT IN TRENCH BEFORE AUX BLDG PENETRATION
5. LOCATE VALVE 2CA178 AT CONNECTION TO RC SYSTEM. PIPE IN TB TO BE EMBEDDED IN TB BASEMENT FLOOR, VALVE AND RC SYSTEM CONNECTION TO BE PROTECTED BY CONCRETE ENCLOSURE.
6. TEST DRAIN USED FOR FUNCTIONAL TEST OF CHECK VALVE. DISCHARGE TO BE VISIBLE TO VERIFY FLOW
7. POWERED FROM SSF 220/125VDC SYSTEM.
8. KEROTEST VALVES ARE UNACCEPTABLE FOR THIS APPLICATION
9. 2CAFE 5540, 5550, 5560, 5570, 5580 CONSIST OF MACHINED SOCKET WELD PIPE COUPLINGS INSTALLED IN SERIES WITH APPROXIMATELY 8" - 10" SPACING.
10. LUBE OIL (SHELL SIDE) OF HEAT EXCHANGER IS CONNECTED WITH AUX FDW PUMP TURBINE LUBE OIL SYSTEM WHICH IS NOT SHOWN ON ANY FLOW DIAGRAM. CONSULT AUX FDW TURBINE INSTRUCTION MANUAL FOR DETAILS.
11.
12. PIPING IS CLASS F IN TURBINE BLDG. TRENCH.
13. POWERED FROM SSF AC NORMAL AUXILIARY POWER SYSTEM.
14. NO PIPING SHOWN ON THIS FLOW DIAGRAM REQUIRES INSULATION.
15. CHECK VALVE INTERNALS REMOVED.
16. PURPOSE OF THIS VALVE, TO BE LOCATED OUTSIDE CA TURBINE CONTROL PANEL ROOM, IS TO BACKFILL RC/CA LINE WITH RC CHEMISTRY WATER AFTER CLAM FLUSH.
17. FUTURE CONNECTION FOR NSM CN-20525/00
18. MOTOR OPERATOR BREAKER TO BE DISCONNECTED WITH VALVE FULLY OPEN.
19. ABANDONED IN PLACE PER NSM CN-21371/00.
20. VENDOR SUPPLIED PIPING. PROCESS FLUID IS NOT IN THESE LINES.
21. COOLERS SHOW FOR INFORMATION ONLY. REFERENCE KC SYSTEM FLOW DIAGRAMS FOR SPECIFIC INFORMATION.
22. 2CA6 IS CLOSED WITH MOTOR OPERATOR POWER REMOVED (BKR 2MXW F05B OPEN) PER EC 106582 TO PREVENT POSSIBLE VORTEXING IN THE CACST. 2CA6 CAN BE OPEN IN MODES 3 AND 4 DURING PERIODS OF LOW DECAY HEAT (DEFINED AS AFTER REFUELING) WITH BOTH CACSTS ALIGNED AT FULL VOLUME TO THE OUTAGE UNIT (REF. EC 411218). 32
23. 2CA324 SUPPORTS FILLING BETWEEN 2CA174 AND 2CA173 WITH CONDENSATE GRADE WATER. (REF. EC 110084)
24. 2CA174 WILL NOT FAIL OPEN (FOR AT LEAST 60 MINUTES) ON A VI SYSTEM FAILURE UPSTREAM OF 2CACK1740 WHEN 2CASV1740 IS ENERGIZED (REF. CN-2499-CA.16-00 AND EC 110084).

DESIGN PARAMETERS

LINE LISTING	PIPE SPEC	PRESSURE	TEMPERATURE	CLASS	MATERIAL
01	2500.3	1800 PSIA	160°F	C	CS
04	2500.3	2000 PSIA	160°F	C	CS
06	150.4	165 PSIA	160°F	F	CS
07	150.4	50 PSIA	160°F	G	CS
08	150.3	165 PSIA	160°F	C	CS
09	150.4	165 PSIA	160°F	G	CS
12	2501.3	2000 PSIA	160°F	C	SS
13	151.3	165 PSIA	160°F	C	SS
16	151.4	165 PSIA	160°F	F	SS

ABANDONED COMPONENTS RELATED

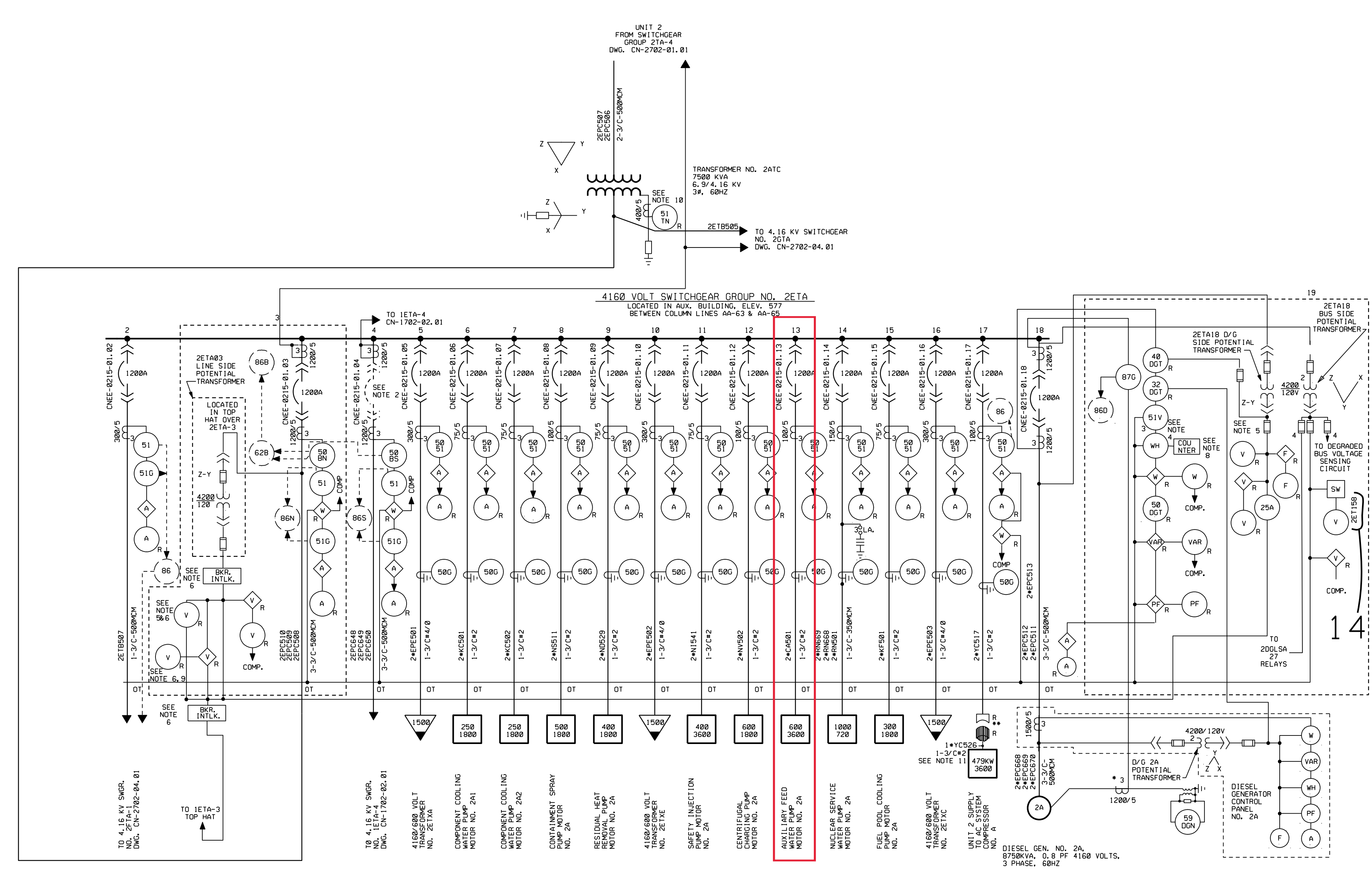
QA CONDITION 4
QA CONDITION 1

DUKE ENERGY
CATAWBA NUCLEAR STATION UNIT 2
FLOW DIAGRAM OF
AUXILIARY FEEDWATER SYSTEM
(CA)

NO.	REVISIONS
32	REV PER EC 411218

SCALE N/A DWG. NO. CN-2592-01.00 REV. 32

ERN: CN000THN



TRAIN A - QA CONDITION 1

NOTES:

1. CABLES ARE SIZED ON THE BASIS OF 5KV, 90C, 90 MILS OF EPR, INTERLOCKED ARMOR SHIELDED, 3 CONDUCTOR COPPER POWER CABLES WITH 0.7 DERATING FACTOR.
2. A BREAKER MAY BE FURNISHED IN 2ETA-4. A KIRK KEY INTERLOCK SCHEME IS PROVIDED WHICH PREVENTS A BREAKER BEING INSERTED ON THE BUS IN 2ETA-4. IF A BREAKER IS INSERTED ON THE BUS IN 2ETA-3, AND VICE-VERSA. THE SCHEME ALSO PREVENTS A BREAKER BEING INSERTED ON THE BUS IN 2ETA-4. IF A BREAKER IS INSERTED ON THE BUS IN 1ETA-4, AND VICE-VERSA. THIS SCHEME PREVENTS, TIEING TWO SOURCES TOGETHER AND TIEING TWO BUSES TO THE, SAME, SOURCE. THE BREAKER IN 2ETA-4 SHOULD BE KEPT IN THE DISCONNECT POSITION WHEN NOT BEING USED.
3. ALL WATT, VAR & VOLT TRANSDUCERS ARE REMOTE. ALL CURRENT TRANSDUCERS ARE LOCAL.
4. 51V RELAYS AND WATTHOUR METER SHOWN IN 2ETA-19 ARE LOCATED IN 2ETA-18.
5. DIRECT READING REMOTE VOLTMETER IS LOCATED IN DIESEL GENERATOR CONTROL PANEL.
6. THE POTENTIAL TRANSFORMER INPUT TO THESE VOLTMETERS ARE INTERLOCKED WITH THE INCOMING BREAKER LOCATION SUCH THAT, IF THE BREAKER IS IN UNIT 3, THE VOLTMETERS WILL REGISTER UNIT 3 INCOMING VOLTAGE; AND IF IN UNIT 4, THE VOLTMETERS WILL REGISTER UNIT 4 INCOMING VOLTAGE.
7. IF A FUTURE VERTICAL IS REQUIRED, IT WILL BE PLACED ADJACENT TO VERTICAL 19 AND THE SWITCHGEAR BUS WILL HAVE TO BE EXTENDED THROUGH VERTICAL 19.
8. DURANT COUNTER CONNECTED THROUGH COMP. CABINET ISOL. COUNTER LOCATED IN 1MC9.
9. REMOTE VOLTMETER LOCATED ON 2MC8.
10. THE 400/5 CURRENT TRANSFORMER WILL BE PLACED IN THE LOW VOLTAGE TERMINAL CHAMBER OF 2ATC AND THE 51TN RELAY IN THE 6.9KV SWGR. CUB. 2TA4
11. ELASTIMOLD CONNECTORS LOCATED IN 1TBOX0345(CN-1751-12.01).

ABBREVIATIONS:

OT - RUN POWER CABLE OUT TOP OF SWITCHGEAR.
R - REMOTE
* - FURNISHED BY DIESEL GENERATOR VENDOR
** - ELASTIMOLD CONNECTORS FURNISHED BY DUKE.
2DGLSA - UNIT 2 DIESEL GENERATOR LOAD SEQUENCER TRAIN A

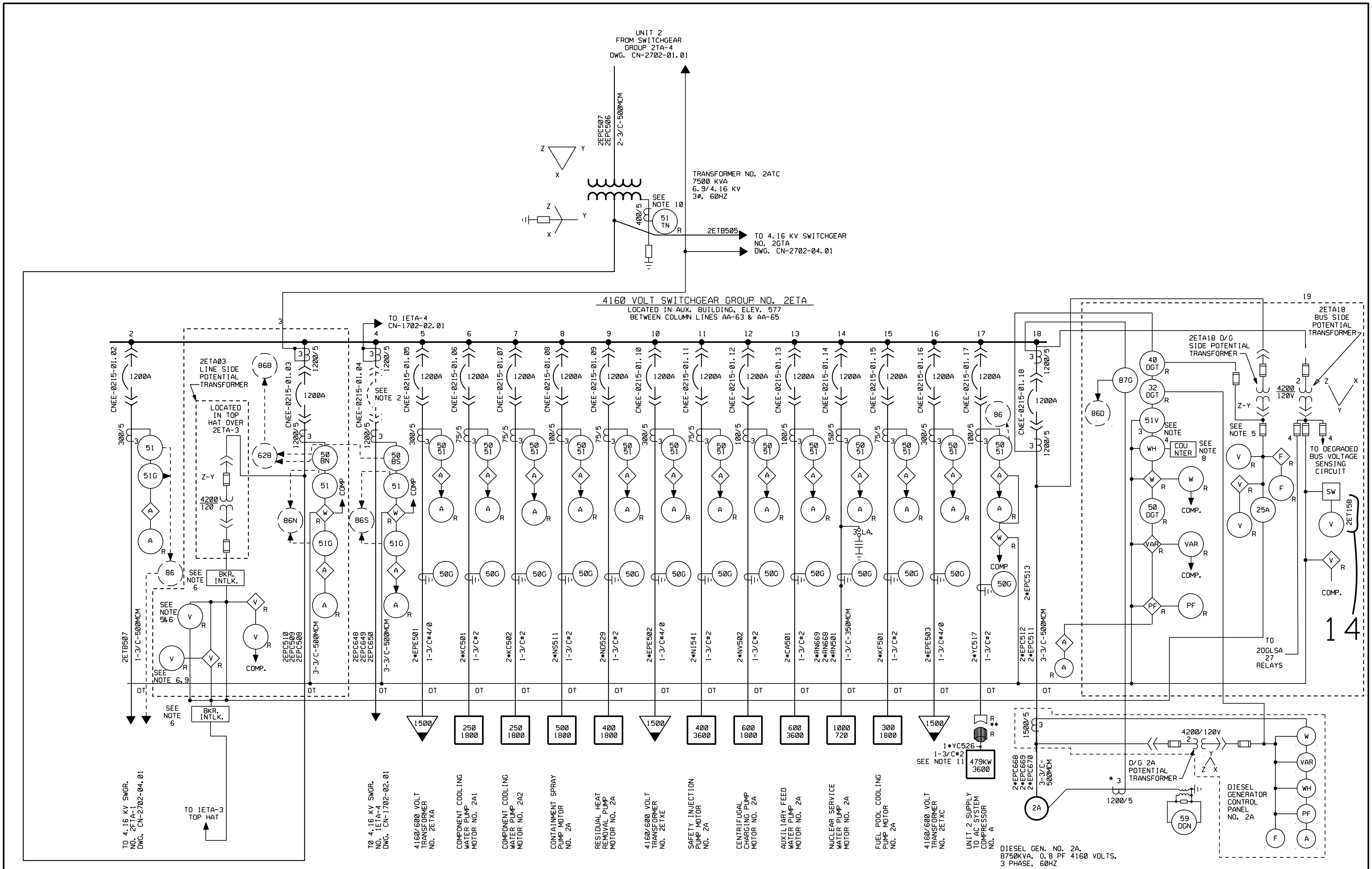
NO.	REVISIONS
14	REVISED PER EC 411851

DUKE ENERGY
CATAWBA NUCLEAR STATION UNIT 2

ONE LINE DIAGRAM
4160 VOLT ESSENTIAL AUXILIARY POWER SYSTEM (EPC)
4160V SWITCHGEAR NO. 2ETA

SCALE N/A | DWG. NO. CN -2702-02.01 | REV. 14

ERN: CN000TW7



NOTES:

1. CABLES ARE SIZED ON THE BASIS OF 5KV, 90C, 90 MILS OF EPR, INTERLOCKED ARMOR SHIELDED, 3 CONDUCTOR COPPER POWER CABLES WITH 0.7 DERATING FACTOR.
2. A BREAKER MAY BE FURNISHED IN 2ETA-4. A KIRK KEY INTERLOCK SCHEME IS PROVIDED WHICH PREVENTS A BREAKER BEING INSERTED ON THE BUS IN 2ETA-4. IF A BREAKER IS INSERTED ON THE BUS IN 2ETA-3, AND VICE-VERSA. THE SCHEME ALSO PREVENTS A BREAKER BEING INSERTED ON THE BUS IN 2ETA-4. IF A BREAKER IS INSERTED ON THE BUS IN 1ETA-4, AND VICE-VERSA. THIS SCHEME PREVENTS, TIEING TWO SOURCES TOGETHER AND TIEING TWO BUSES TO THE, SAME, SOURCE. THE BREAKER IN 2ETA-4 SHOULD BE KEPT IN THE DISCONNECT POSITION WHEN NOT BEING USED.
3. ALL WATT, VAR & VOLT TRANSDUCERS ARE REMOTE. ALL CURRENT TRANSDUCERS ARE LOCAL.
4. 51V RELAYS AND WATTHOUR METER SHOWN IN 2ETA-19 ARE LOCATED IN 2ETA-18.
5. DIRECT READING REMOTE VOLTMETER IS LOCATED IN DIESEL GENERATOR CONTROL PANEL.
6. THE POTENTIAL TRANSFORMER INPUT TO THESE VOLTMETERS ARE INTERLOCKED WITH THE INCOMING BREAKER LOCATION SUCH THAT, IF THE BREAKER IS IN UNIT 3, THE VOLTMETERS WILL REGISTER UNIT 3 INCOMING VOLTAGE, AND IF IN UNIT 4, THE VOLTMETERS WILL REGISTER UNIT 4 INCOMING VOLTAGE.
7. IF A FUTURE VERTICAL IS REQUIRED, IT WILL BE PLACED ADJACENT TO VERTICAL 19 AND THE SWITCHGEAR BUS WILL HAVE TO BE EXTENDED THROUGH VERTICAL 19.
8. DURANT COUNTER CONNECTED THROUGH COMP. CABINET ISOL. COUNTER LOCATED IN 1MC9.
9. REMOTE VOLTMETER LOCATED ON 2MC8.
10. THE 400/5 CURRENT TRANSFORMER WILL BE PLACED IN THE LOW VOLTAGE TERMINAL CHAMBER OF 2ATC AND THE 51TN RELAY IN THE 6.9KV SWGR. CUB. 2TA4
11. ELASTIMOLD CONNECTORS LOCATED IN 1TBOX0345(CN-1751-12.01).

ABBREVIATIONS:

OT - RUN POWER CABLE OUT TOP OF SWITCHGEAR.
R - REMOTE
* - FURNISHED BY DIESEL GENERATOR VENDOR
** - ELASTIMOLD CONNECTORS FURNISHED BY DUKE.
2DGLSA - UNIT 2 DIESEL GENERATOR LOAD SEQUENCER TRAIN A

TRAIN A - QA CONDITION 1

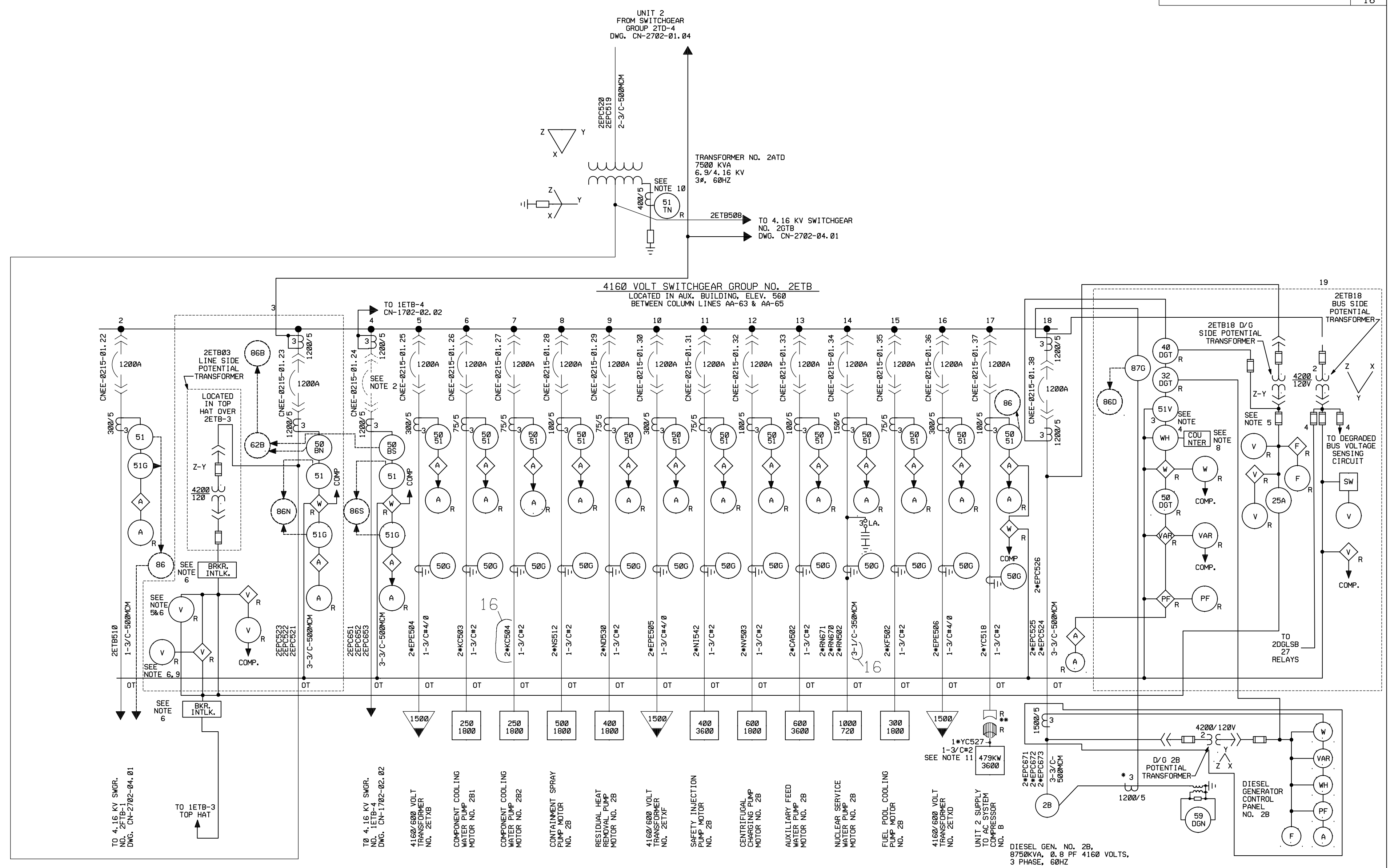
DUKE ENERGY
CATAWBA NUCLEAR STATION UNIT 2

ONE LINE DIAGRAM
4160 VOLT ESSENTIAL AUXILIARY POWER SYSTEM (EPC)
4160V SWITCHGEAR NO. 2ETA

NO.	REVISIONS
14	REVISED PER EC 411851

SCALE N/A | DWG. NO. CN -2702-02.01 | REV. 14

ERN: CN000TW7



NOTES:

1. CABLES ARE SIZED ON THE BASIS OF 5KV, 90C, 90 MILS OF EPR, INTERLOCKED ARMOR SHIELDED, 3 CONDUCTOR COPPER POWER CABLES WITH 0.7 DERATING FACTOR.
2. A BREAKER MAY BE FURNISHED IN 2ETB-4. A KIRK KEY INTERLOCK SCHEME IS PROVIDED WHICH PREVENTS A BREAKER BEING INSERTED ON THE BUS IN 2ETB-4, IF A BREAKER IS INSERTED ON THE BUS IN 2ETB-3 AND VICE-VERSA. THE SCHEME ALSO PREVENTS A BREAKER BEING INSERTED ON THE BUS IN 2ETB-4, IF A BREAKER IS INSERTED ON THE BUS IN 1ETB-4 AND VICE-VERSA. THIS SCHEME PREVENTS TIEING TWO SOURCES TOGETHER AND TIEING TWO BUSES TO THE SAME SOURCE. THE BREAKER IN 2ETB-4 SHOULD BE KEPT IN THE DISCONNECT POSITION WHEN NOT BEING USED.
3. ALL WATT, VAR & VOLT TRANSDUCERS ARE REMOTE. ALL CURRENT TRANSDUCERS ARE LOCAL.
4. 51V RELAYS AND WATTHOUR METER SHOWN IN 2ETB-19 ARE LOCATED IN 2ETB-18.
5. DIRECT READING REMOTE VOLTMETER IS LOCATED IN DIESEL GENERATOR CONTROL PANEL.
6. THE POTENTIAL TRANSFORMER INPUT TO THESE INSTRUMENTS ARE INTERLOCKED WITH THE INCOMING BREAKER LOCATION SUCH THAT, IF THE BREAKER IS IN UNIT 3, THE VOLTMETERS WILL REGISTER UNIT 3 INCOMING VOLTAGE, AND IF IN UNIT 4, THE VOLTMETERS WILL REGISTER UNIT 4 INCOMING VOLTAGE.
7. IF A FUTURE VERTICAL IS REQUIRED, IT WILL BE PLACED ADJACENT TO VERTICAL 19 AND THE SWITCHGEAR BUS WILL HAVE TO BE EXTENDED THROUGH VERTICAL 19.
8. DURANT COUNTER CONNECTED THROUGH COMP. CABINET ISOL. COUNTER LOCATED IN 1MC9.
9. REMOTE VOLTMETER LOCATED ON 2MC8.
10. THE 400/5 CURRENT TRANSFORMER WILL BE PLACED IN THE LOW VOLTAGE TERMINAL CHAMBER OF 2ATD AND THE 51TN RELAY IN THE 6.9KV SWGR. CUB. 2TD4
11. ELASTIMOLD CONNECTORS LOCATED IN 1TBOX0346 (CN-1751-12.01).

ABBREVIATIONS:

- OT - RUN POWER CABLE OUT TOP OF SWITCHGEAR.
- R - REMOTE
- * - FURNISHED BY DIESEL GENERATOR VENDOR
- ** - ELASTIMOLD CONNECTORS IN 1TBOX0346 FURNISHED BY DUKE.
- 2DGLSB - UNIT 2 DIESEL GENERATOR LOAD SEQUENCER TRAIN B

TRAIN B - Q.A. CONDITION 1

NO.	REVISIONS	DRN	DATE	CHKD	DATE	APPR	DATE
16	REV PER C-09-04497, EXEMPT CODE G.5	NSW	8-19-10	TLC	8-20-10	GNK	8-23-10
ORIG.	ORIGINAL DRAWING RETIRED	NSW	8-19-10				

INSPECTED: CIVIL WAIVED — ELEC. -W- — MECH. WAIVED

DUKE ENERGY
CATAWBA NUCLEAR STATION UNIT 2

ONE LINE DIAGRAM
4160 VOLT ESSENTIAL AUXILIARY
POWER SYSTEM (EPC)
4160V SWITCHGEAR NO. 2ETB

DESIGNER HERRINGTON DATE 8-15-77 INSP. WAIVED DATE 8-15-77
DRAWN RA RAMSEY DATE 7-29-77 INSP. WAIVED DATE 8-15-77
CHECKED HW BOOZER DATE 8-15-77 APPR. CE KNEEBURG DATE 8-17-77

DWG. NO. CN-2702-02.02 REV. 16

ERN:CN000TW8

JPM A.3S

SRO

Catawba Nuclear Station
Admin. JPM A.3S
Sept 2021 NRC Exam

EVALUATION SHEET

Task: Review Liquid Waste Release

Alternate Path: N/A

Facility JPM #: WL-002

Safety Function: N/A

K/A G 2.3.6 Ability to approve release permits

Importance: 2.0 / 3.8 **CFR:** 41.10 / 45.12 / 45.13

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom __X__ Perform __X__ Simulate ________

References: OP/0/B/6500/113 (Operations Liquid Waste Release), Enclosure 4.1 (Liquid Waste Release from a Monitor Tank)

Task Standard: The operator will determine that the LWR (Package # 2021056) should not be released due to one, or all, of the following: incorrect EMF-49 Trip 2 setpoint, incorrect RL Flow Interlock setpoint, and incorrect 1WL-124 flowrate setpoint.

Validation Time: 20 minutes **Time Critical:** Yes _____ No __X__

Applicant:
NAME____________________________ Docket #_______________ Time Start: __________
Time Finish: __________

Performance Rating: Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

Catawba Nuclear Station
Admin. JPM A.3S
Sept 2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- Both Units are at 100% power.
- There are currently no liquid waste releases in progress.
- The following equipment is in service:
    - 1A & 1B RC Pumps
    - 0RLP5080 (RL Discharge Total Flow) is operable and indicating 46,400 gpm through A & B RL Discharge Headers
    - 1B RN Pump in normal alignment
    - LWR Integrator (0WLP6160) is operable
- LWR #2021056 has been delivered to the Control Room and was approved by the previous shift's CRS.
    - OP/0/B/6500/113 (Operations Liquid Waste Release), Enclosure 4.1 (Liquid Waste Release from a Monitor Tank) is in progress and has been completed through step 3.14
- The Unit 1 BOP has notified you (per step 3.15) that the LWR is ready to be released.

INITIATING CUE:

Review LWR #2021056 to determine if the release should be initiated. If applicable, list any issues that would prevent release initiation.

Initiate Release **YES** **NO**
circle one

Reason(s) (if any)

____

____

____

Catawba Nuclear Station
Admin. JPM A.3S
Sept 2021 NRC Exam

EXAMINER NOTE:

After reading cue, provide applicant with a copy of OP/0/B/6500/113, Enclosure 4.1, completed through step 3.14 and Student Handout (LWR #2021056).

START TIME: _________

Examiner Note: The operator may review steps performed in Enclosure 4.1. Applicable procedure direction begins at step 3.4. STEP 1: 3.4 CRS performs the following concerning the LWR Permit Report: ☐ Verifies LWR Permit Report refers to correct EMF. ☐ Verifies EMF setpoints correct. ☐ Signs and dates the LWR Permit Report authorizing the release. STANDARD: Operator reviews the LWR permit report and determines that the correct EMF is listed, EMF setpoints are correct, and that the authorization was properly completed. COMMENTS:	Sat __ Unsat __

STEP 2: 3.5 **IF** EMF-49 is functional, perform the following: 3.5.1 Verify EMF-49 (Low Range) is functional per SLC 16.11-2, Table 16.11-2-1 using OP/0/A/6500/080 (EMF Output Modules). 3.5.2 Sign off the "EMF Source Checked & Operable" blank on the LWR Permit Report. STANDARD: Operator reviews the LWR Permit Report and determines that EMF-49 has been source checked and signed off. COMMENTS:	Sat __ Unsat __

STEP 3: 3.5.3 Set EMF-49 (Low Range) trip setpoints to the values listed in "SETPOINT DATA" section on the LWR Permit Report using OP/0/A/6500/080 (EMF Output Modules). 3.5.4 Sign off the "EMF49L Setpoints Set (Low Range)" blank on the LWR Permit Report. STANDARD: Operator reviews the LWR Permit Report and determines that the EMF-49 Trip 2 setpoint has not been set to the value listed. The operator will then determine that the release should not be initiated and list EMF-49 setpoint as one reason why. **Examiner Note: This step may be critical to determine that this release should not be initiated. Proper completion of one of the determination steps (step 3, 9, or 13) meets this requirement.** COMMENTS:	**CRITICAL STEP** Sat __ Unsat __

NOTE: The person performing Steps 3.5.5 and 3.5.6 shall **NOT** be the same as in Step 3.5.4. STEP 4: 3.5.5 Verify the EMF-49 (Low Range) trip setpoints are set as specified in "SETPOINT DATA" section on the LWR Permit Report using OP/0/A/6500/080 (EMF Output Modules). 3.5.6 Sign off the "(I.V.) Independent Verification" blank on the LWR Permit Report. STANDARD: Operator notes that EMF-49 Trip 2 setpoint has not been properly verified and moves on. COMMENTS:	Sat __ Unsat __

NOTE: Monitoring for highest count rate during release using a digital EMF chart recorder does **NOT** require an initial setup. <u>STEP 5:</u> 3.5.7 **IF** using the OAC to obtain the highest count rate during release, monitor OAC point C1E0263 (EMF49L Waste Liquid Discharge). <u>STANDARD</u>: Operator notes this step for future use and moves on. <u>COMMENTS:</u>	Sat __ Unsat __

<u>STEP 6:</u> 3.6 **IF** EMF-49 is nonfunctional, perform the following: <u>STANDARD</u>: Operator determines that this step is not applicable. <u>COMMENTS:</u>	Sat __ Unsat __

STEP 7: 3.7 Complete the following steps on the LWR Permit Report documenting the RN and RL Systems status: 3.7.1 Ensure at least the number of RN and RL pumps assigned to the LWR (listed on page 2 of LWR Permit) are operating. 3.7.2 Sign off the "Ensure number of RN Pumps Operating is at least #.##" blank on the LWR Permit Report where #.## is the number of RN Pumps listed on page 2 of LWR Permit. 3.7.3 Sign off the "Ensure number of RL Pumps Operating is at least #.##" blank on the LWR Permit Report where #.## is the number of RL Pumps listed on page 2 of LWR Permit. STANDARD: Operator determines that the proper number of RN and RL pumps are operating and properly documented. COMMENTS:	Sat __ Unsat __

STEP 8: 3.8 **IF** 0RLP5080 (RL Discharge Total Flow) is operable, perform the following: 3.8.1 Enter the RL flow rate on the LWR Permit Report as read on 0RLP5080 (RL Discharge Total Flow) (located on 1MC9). 3.8.2 Sign off the "RL Flowrate _______ gpm" blank on the LWR Permit Report. STANDARD: Operator determines that RL flowrate is properly recorded. COMMENTS:	Sat __ Unsat __

NOTE: If RL flow drops below this setpoint, it will automatically close 1WL-124 (Waste Monit Tnk Pmps Disch) and terminate the release.	**CRITICAL STEP**
STEP 9: 3.8.3 Set the flow interlock on 0RLP5080 (RL Discharge Total Flow) to the value specified in the "SPECIAL INSTRUCTIONS FOR RELEASE" section of the LWR Permit Report. 3.8.4 Record the RL Flow interlock setpoint on the LWR Permit Report. 3.8.5 Circle the RL header or headers used for this release. 3.8.6 Sign off the "RL Flow Interlock Set @ _______ gpm for Appropriate Headers (A and/or B)" blank on the LWR Permit Report STANDARD: Operator determines the RL Flow Interlock has been set for the incorrect value. The operator will determine that this is an additional reason for which the release should not be initiated. **Examiner Note: This step may be critical to determine that this release should not be initiated. Proper completion of one of the determination steps (step 3, 9, or 13) meets this requirement.** COMMENTS:	Sat __ Unsat __

STEP 10: 3.9 **IF** 0RLP5080 (RL Discharge Total Flow) is inoperable, perform the following: STANDARD: Operator determines that this step is not applicable. COMMENTS:	Sat __ Unsat __

STEP 11: 3.10 **IF** the LWR Integrator (Liquid Rad Waste Disch, 0WLP6160) is operable, perform the following: 3.10.1 Reset the LWR Integrator (located on 1MC11). 3.10.2 Sign off the "Reset LWR Integrator" blank on the LWR Permit Report. STANDARD: Operator determines the LWR Integrator has been reset. COMMENTS:	Sat __ Unsat __

STEP 12: 3.11 **IF** the LWR Integrator (Liquid Rad Waste Disch, 0WLP6160) is inoperable, perform the following: STANDARD: Operator determines that this step is not applicable. COMMENTS:	Sat __ Unsat __

STEP 13: 3.12 Set 1WL-124 (Waste Monit Tnk Pmps Disch) (located on 1MC11) to the "Recommended Release Rate (gpm)" specified on the LWR Permit Report. 3.13 Enter the "Recommended Release Rate (gpm)" specified on the LWR Permit Report. STANDARD: Operator determines the 1WL-124 has NOT been set to the recommended release rate. The operator will determine that this is an additional reason for which the release should not be initiated. **Examiner Note: This step may be critical to determine that this release should not be initiated. Proper completion of one of the determination steps (step 3, 9, or 13) meets this requirement.** COMMENTS:	**CRITICAL STEP** Sat __ Unsat __

END TIME: __________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- Both Units are at 100% power.
- There are currently no liquid waste releases in progress.
- The following equipment is in service:
  - 1A & 1B RC Pumps
  - 0RLP5080 (RL Discharge Total Flow) is operable and indicating 46,400 gpm through A & B RL Discharge Headers
  - 1B RN Pump in normal alignment
  - LWR Integrator (0WLP6160) is operable
- LWR #2021056 has been delivered to the Control Room and was approved by the previous shift's CRS.
  - OP/0/B/6500/113 (Operations Liquid Waste Release), Enclosure 4.1 (Liquid Waste Release form a Monitor Tank) is in progress and has been completed through step 3.14
- The Unit 1 BOP has notified you (per step 3.15) that the LWR is ready to be released.

INITIATING CUE:

Review LWR #2021056 to determine if the release should be initiated. If applicable, list any issues that would prevent release initiation.

Initiate Release **YES** **NO**

circle one

Reason(s) (if any)

____

____

____

<table>
<tr><td rowspan="2">Duke Energy
Catawba Nuclear Station

Operations Liquid Waste Release</td><td>Procedure No.
OP/0/B/6500/113</td></tr>
<tr><td>Revision No.
010</td></tr>
<tr><td>Continuous Use</td><td>Electronic Reference No.
CP0095WS</td></tr>
</table>

Operations Liquid Waste Release

1. Purpose

To aid the operator in the correct methods of performing steps in Radwaste procedure OP/0/B/6500/015 (Discharging a Monitor Tank to the Environment) and Radiation Protection procedure HP/0/B/1004/004 (Radioactive Liquid Waste Release). Also to aid the operator as to limits and results expected while these procedures are being performed.

2. Limits and Precautions

2.1 Ensure that RN is discharging through at least one RL header.

2.2 Ensure that RN is **NOT** discharging to SNSWP.

2.3 If the pre-set radiation levels are exceeded on EMF-49 or the dilution flow rate drops below the setpoint for 0RLP5080 (RL Discharge Total Flow), 1WL-124 (Waste Monit Tnk Pmps Disch) will trip closed.

2.4 Releases that are interrupted by EMF-49 "HI-RAD" trips may be initiated up to a maximum of three times, including original initiation, without re-sampling per HP/0/B/1004/004 (Radioactive Liquid Waste Release).

2.5 Turbine Building Sump releases are secured if the pre-set levels are exceeded on 1/2EMF-31.

3. Procedure

Refer to Section 4 (Enclosures)

4. Enclosures

4.1 Liquid Waste Release from a Monitor Tank

4.2 Discharging a Contaminated Turbine Building Sump to Holdup Pond

1. Limits and Precautions

1.1 Ensure that RN is discharging through at least one RL header.

1.2 Ensure that RN is **NOT** discharging to SNSWP.

1.3 If the pre-set radiation levels are exceeded on EMF-49 or the dilution flow rate drops below the setpoint for 0RLP5080 (RL Discharge Total Flow), 1WL-124 (Waste Monit Tnk Pmps Disch) will trip closed.

1.4 Releases that are interrupted by EMF-49 "HI-RAD" trips may be initiated up to a maximum of three times, including original initiation, without re-sampling per HP/0/B/1004/004 (Radioactive Liquid Waste Release).

2. Initial Conditions

CC___ 2.1 Verify Radwaste has initiated OP/0/B/6500/015 (Discharging a Monitor Tank to the Environment).

CC___ 2.2 Verify LWR (Liquid Waste Release) Permit Report has been delivered to the CRS.

3. Procedure

_____ 3.1 **IF AT ANY TIME** within 60 seconds after initiating the release with EMF-49 functional, 1RAD-1, F/4 "EMF-49 LIQUID WASTE DISCH LOSS OF FLOW" alarm **CANNOT** be cleared, the release shall be secured and EMF-49 declared nonfunctional.

_____ 3.2 **IF AT ANY TIME** during a release, a Site Assembly occurs, secure release and have Radwaste ensure the following valves are locked closed:

- 1WL-949 (RMT Outlet To Waste Monitor Tank Disch Radiation Monitor EMF-49)
- 1WL-113 (Waste Monitor Tank A Pump Disch To Radiation Monitor 1EMF-49)
- 1WL-187 (Waste Monitor Tank B Pump Disch To Radiation Monitor)

_____ 3.3 **IF AT ANY TIME** during this release the OAC is out of service, initiate a 15 minute increased surveillance per OMP 2-31 (Control Room Instrumentation Status) to verify:

☐ 3.3.1 The following RN valves remain OPEN:

- 1RN-57A (Station RN Disch To RL Sys)
- 1RN-843B (Station RN Disch To RL Sys)
- 1RN-54A (Station RN Disch Hdr X-Over)
- 1RN-53B (Station RN Discharge Hdr X-Over)
- 1RN-1A (RN P/H Pit A Isol From Lake)
- 1RN-2B (RN P/H Pit A Isol From Lake)
- 1RN-5A (RN P/H Pit B Isol From Lake)
- 1RN-6B (RN P/H Pit B Isol From Lake)

☐ 3.3.2 The following valves remain CLOSED:

- 1RN-3A (RN P/H Pit A Isol From SNSWP)
- 1RN-4B (RN P/H Pit B Isol From SNSWP)
- 1RN-63A (RN Hdr A Return To SNSWP)
- 1RN-58B (RN Hdr B Return To SNSWP)

_____ 3.3.3 **IF** any of the above RN Valve(s) are found out of position, secure the release immediately.

NOTE:

1. The following steps are to be completed, the information is to be entered and the appropriate steps (designated by "0") signed off on the LWR Permit Report.
2. Verifying EMF setpoints correct requires:
   - Verification that Trip 1 is higher than background
   - Verification that Trip 2 is higher than Trip 1
3. Steps 3.4 - 3.16 provide instructions for completing the "COMPLETE PRIOR TO RELEASE" section of the LWR Permit Report.

BB
CRS 3.4 CRS performs the following concerning the LWR Permit Report:

☒ Verifies LWR Permit Report refers to correct EMF.

☒ Verifies EMF setpoints correct.

☒ Signs and dates the LWR Permit Report authorizing the release.

CC 3.5 **IF** EMF-49 is functional, perform the following:

CC 3.5.1 Verify EMF-49 (Low Range) is functional per SLC 16.11-2, Table 16.11-2-1 using OP/0/A/6500/080 (EMF Output Modules).

CC 3.5.2 Sign off the "EMF Source Checked & Operable" blank on the LWR Permit Report.

CC 3.5.3 Set EMF-49 (Low Range) trip setpoints to the values listed in "SETPOINT DATA" section on the LWR Permit Report using OP/0/A/6500/080 (EMF Output Modules).

CC 3.5.4 Sign off the "EMF49L Setpoints Set (Low Range)" blank on the LWR Permit Report.

NOTE: The person performing Steps 3.5.5 and 3.5.6 shall **NOT** be the same as in Step 3.5.4.

DD 3.5.5 Verify the EMF-49 (Low Range) trip setpoints are set as specified in "SETPOINT DATA" section on the LWR Permit Report using OP/0/A/6500/080 (EMF Output Modules).

DD 3.5.6 Sign off the "(I.V.) Independent Verification" blank on the LWR Permit Report.

NOTE: Monitoring for highest count rate during release using a digital EMF chart recorder does **NOT** require an initial setup.

CC 3.5.7 **IF** using the OAC to obtain the highest count rate during release, monitor OAC point C1E0263 (EMF49L Waste Liquid Discharge).

N/A CC 3.6 **IF** EMF-49 is nonfunctional, perform the following:

_____ 3.6.1 Notify Radiation Protection to take action per HP/0/B/1004/004 (Radioactive Liquid Waste Release).
Person notified ____

_____ 3.6.2 N/A the following steps on the LWR Permit Report:

- "EMF Source Checked"
- "EMF49L Setpoints Set (Low Range)"
- "(I.V.) Independent Verification"

3.7 Complete the following steps on the LWR Permit Report documenting the RN and RL Systems status:

CC 3.7.1 Ensure at least the number of RN and RL pumps assigned to the LWR (listed on page 2 of LWR Permit) are operating.

CC 3.7.2 Sign off the "Ensure number of RN Pumps Operating is at least #.##" blank on the LWR Permit Report where #.## is the number of RN Pumps listed on page 2 of LWR Permit.

CC 3.7.3 Sign off the "Ensure number of RL Pumps Operating is at least #.##" blank on the LWR Permit Report where #.## is the number of RL Pumps listed on page 2 of LWR Permit.

CC 3.8 **IF** 0RLP5080 (RL Discharge Total Flow) is operable, perform the following:

CC 3.8.1 Enter the RL flow rate on the LWR Permit Report as read on 0RLP5080 (RL Discharge Total Flow) (located on 1MC9).

CC 3.8.2 Sign off the "RL Flowrate _______ gpm" blank on the LWR Permit Report.

NOTE: If RL flow drops below this setpoint, it will automatically close 1WL-124 (Waste Monit Tnk Pmps Disch) and terminate the release.

CC 3.8.3 Set the flow interlock on 0RLP5080 (RL Discharge Total Flow) to the value specified in the "SPECIAL INSTRUCTIONS FOR RELEASE" section of the LWR Permit Report.

CC 3.8.4 Record the RL Flow interlock setpoint on the LWR Permit Report.

CC 3.8.5 Circle the RL header or headers used for this release.

CC 3.8.6 Sign off the "RL Flow Interlock Set @ _______ gpm for Appropriate Headers (A and/or B)" blank on the LWR Permit Report.

N/A CC 3.9 **IF** 0RLP5080 (RL Discharge Total Flow) is inoperable, perform the following:

_____ 3.9.1 Estimate the flow rate every 4 hours per PT/0/A/4250/011 (RL Temperature and Discharge Flow Determination).

_____ 3.9.2 Record estimated flow rate on the LWR Permit Report and attach copies of all enclosures used to the LWR Permit Report.

_____ 3.9.3 Sign off the "RL Flowrate _______ gpm" blank on the LWR Permit Report.

CAUTION:	Setting the RL flow interlock at "0" overrides the interlock on 1WL-124 (Waste Monit Tnk Pmps Disch). RL flow rate shall be closely monitored and the release stopped if the flow drops below the RL Flow interlock Setting specified on the "SPECIAL INSTRUCTIONS FOR RELEASE" Section of the LWR Permit Report.

_____ 3.9.4 Set the RL flow interlock on "0".

_____ 3.9.5 Record the RL Flow interlock setpoint on the LWR Permit Report.

_____ 3.9.6 Note on the LWR Permit Report that the RL Flow interlock is inoperable.

_____ 3.9.7 Circle the RL header or headers used for this release.

_____ 3.9.8 Sign the "RL Flow Interlock Set @ _______ gpm for Appropriate Headers (A and/or B)" blank on the LWR Permit Report.

CC 3.10 **IF** the LWR Integrator (Liquid Rad Waste Disch, 0WLP6160) is operable, perform the following:

CC 3.10.1 Reset the LWR Integrator (located on 1MC11).

CC 3.10.2 Sign off the "Reset LWR Integrator" blank on the LWR Permit Report.

N/A CC 3.11 **IF** the LWR Integrator (Liquid Rad Waste Disch, 0WLP6160) is inoperable, perform the following:

_____ 3.11.1 Notify Radwaste that Alternate Flow Instruments, Data and Data Sheets per OP/0/B/6500/015 (Discharging a Monitor Tank to the Environment) must be used to determine volume released.
Person notified ____

_____ 3.11.2 N/A the "Reset LWR Integrator" blank on the LWR Permit Report.

CC 3.12 Set 1WL-124 (Waste Monit Tnk Pmps Disch) (located on 1MC11) to the "Recommended Release Rate (gpm)" specified on the LWR Permit Report.

CC 3.13 Enter the "Recommended Release Rate (gpm)" specified on the LWR Permit Report.

CC 3.14 Sign off the "1WL124 Flow Set @ _____ gpm" blank on the LWR Permit Report.

_____ 3.15 Notify the CRS that LWR is ready to be released.

_____ 3.16 **<u>IF AT ANY TIME</u>** an automatic closure on low RL flow **<u>OR</u>** high radiation level occurs, complete the following to reopen 1WL-124 (Waste Monit Tnk Pmps Disch):

_____ 3.16.1 Reset 1WL-124 (Waste Monit Tnk Pmps Disch) to "0".

_____ 3.16.2 Set 1WL-124 (Waste Monit Tnk Pmps Disch) (located on 1MC11) to the "Recommended Release Rate (gpm)" specified on the LWR Permit Report.

NOTE:	The Liquid Waste Release is now ready to be initiated per OP/0/B/6500/015 (Discharging a Monitor Tank to the Environment). Subsequent steps in this procedure are to be completed after the release is terminated.

_____ 3.17 Close 1WL-124 (Waste Monit Tnk Pmps Disch).

NOTE:	The LWR integrator reading shall be recorded prior to flushing.

_____ 3.18 **<u>IF</u>** the LWR Integrator (Liquid Rad Waste Disch, 0WLP6160) is operable, perform the following:

_____ 3.18.1 Record the volume released on the LWR Permit Report.
Volume released = Integrator reading X 10.

_____ 3.18.2 Sign off the "Volume Released ______ gal" blank on the LWR Permit Report.

_____ 3.18.3 Reset the LWR Integrator.

_____ 3.18.4 Sign off the "Reset LWR Integrator" blank on the LWR Permit Report.

_____ 3.19 **<u>IF</u>** the LWR Integrator (Liquid Rad Waste Disch, 0WLP6160) is inoperable, perform the following:

_____ 3.19.1 Record the volume released as calculated from Alternate Flow Instruments, Data and Data Sheets per OP/0/B/6500/015 (Discharging a Monitor Tank to the Environment).

_____ 3.19.2 Sign off the "Volume Released ________ gal" blank on the LWR Permit Report.

_____ 3.19.3 N/A the "Reset LWR Integrator" blank on the LWR Permit Report.

_____ 3.20 **IF** EMF-49 was functional during the release, perform the following:

_____ 3.20.1 Record the EMF reading after flushing on the LWR Permit Report.

_____ 3.20.2 Sign off the "EMF Reading after Flush ________ cpm" blank on the LWR Permit Report.

3.20.3 Reset the EMF-49 (low range) setpoints as follows:

3.20.3.1 Calculate EMF-49 setpoints as follows:

_____ A. Record the un-rounded setpoint values:

- Trip 2 = 3 x (EMF Reading after Flush) = ________
- Trip 1 = Trip 2 x .70 = ________

NOTE:	The EMF setpoints are rounded to 3 significant digits (using standard mathematical rounding) before entry on EMF. For example 2342 cpm = 2.34E+03 and 1635 cpm = 1.64E+03.

_____ B. Round the setpoints calculated in Step 3.20.3.1A to 3 significant digits in scientific notation:

- Trip 2 = _______ cpm
- Trip 1 = _______ cpm

_____ 3.20.3.2 Set EMF-49 (low range) Trip 1 and Trip 2 setpoints to the values determined in Step 3.20.3.1B per OP/0/A/6500/080 (EMF Output Modules).

_____ 3.20.3.3 Enter Trip 1 and Trip 2 setpoints on the LWR Permit Report.

_____ 3.20.4 Sign off the "Reset EMF49L Setpoints
EMF Reading (cpm) after Flush"
____________ cpm X 3 = _________ (Trip2)
Trip 2 X .70 = ___________(Trip 1)" blank on the LWR Permit Report.

3.20.5 Record the highest EMF-49 reading during the release on the LWR Permit Report from one of the following:

☐ 3.20.5.1 Highest reading from OAC point C1E0263.

OR

3.20.5.2 Perform the following on the applicable digital EMF chart recorder:

_____ A. Depress the "Historical" icon. (Located at bottom of screen, left of keyboard icon)

_____ B. Select "Memory".

_____ C. Select "Start of History". (Binocular icon)

_____ D. Select "Search by Time".

_____ E. Enter the start date and start time.

_____ F. Select "Search".

NOTE:	Time intervals per inch can be changed by depressing the "+" or "-" buttons.

_____ G. While viewing the digital values on the left side of the screen, scroll across the trend by depressing the ">" or ">>" buttons and obtain the highest reading.

_____ H. Depress the "Historical" icon to exit history.

_____ 3.20.6 Sign off the "Highest EMF Reading During Release:_______ cpm" blank on the LWR Permit Report.

NOTE:	The person performing Step 3.20.7 shall **NOT** be the same individual who originally performed the associated actions in Steps 3.20.3.1 and 3.20.3.2.

_____ 3.20.7 Independently verify trip setpoints are reset as described in Steps 3.20.3.1 and 3.20.3.2 using OP/0/A/6500/080 (EMF Output Modules).
IV

_____ 3.21 **IF** EMF-49 was nonfunctional during the release, N/A the following steps on the LWR Permit Report:

- "EMF Reading after Flush _______ cpm"
- "Reset EMF49L Setpoints
  EMF Reading (cpm) after Flush
  ____________ cpm X 3 = _________ (trip 2)
  Trip 2 X .70 = ___________(trip 1)"
- "Highest EMF Reading During Release:_______ cpm"

_____ 3.22 Set the RL flow interlock to "0".

_____ 3.23 Sign off the "RL Flow Interlock Set @ ZERO (0)".

_____ 3.24 Verify all blanks on the LWR Permit Report are properly filled out and signed or N/A'd as appropriate.

_____ 3.25 Sign and date the "COMPLETION OF RELEASE ACKNOWLEDGED:" "OPS SRO" blank on the LWR Permit Report.
SRO

_____ 3.26 Place the completed LWR Permit Report in the completed release box.

3.27 Do **NOT** file this enclosure in the Control Copy of this procedure.

LIQUID WASTE RELEASE PERMIT REPORT

LWR Number: 2021056

=== RL/RN PUMP DATA ===
RL pumps assigned to release.................................... 1.00
RN pumps assigned to release.................................... 1.00
Minimum RL flow interlock setpoint for radionuclides (gpm)..... 2.76E+04

=== RECOMMENDED RELEASE RATE ==
Allowable release rate (gpm).................................... 3.73E+02
Recommended release rate (gpm).................................. 1.00E+02
Release rate margin (%)... 272.74

=== SETPOINT DATA ===
EMF49L in Service .... Yes
EMF49L Background (cpm)... 3.91E+02

Cs-137 Equivalence (uCi/ml)..................................... 2.11E-06
Expected CPM.. 1.13E+03
50 % of Expected CPM.. 5.64E+02
Trip 1 setpoint (cpm)... 2.43E+04
Trip 2 setpoint (cpm)... 3.24E+04

=== SPECIAL INSTRUCTIONS FOR RELEASE ==================================
* RL flow interlock must be greater than or equal to 2.76E+04 gpm *
0 EMF 49 FUNCTIONAL

Performed by: [signature] Date: Today

Verified by: [signature] Date: Today

LWR Number: 2021056
Release ID: 2 Waste Monitor Tank "B"
Release Mode: 2 Batch

CRS AUTHORIZING RELEASE *Control Room Supervisor* DATE/TIME *Today / 0500*

COMPLETE PRIOR TO RELEASE:

(O) *CC* EMF49L Source Checked & Operable

(O) *CC* EMF49L Setpoints Set (Low Range)

TRIP 1 *2.43 e4* cpm TRIP 2 *2.43 e6* cpm

(O) *DD* (I.V.) Independent Verification

(O) *CC* Ensure number of RL Pumps Operating is at least 1.00

(O) *CC* Ensure number of RN Pumps Operating is at least 1.00

(O) *CC* RL Flowrate *46,400* gpm

(O) *CC* RL Flowrate Interlock set @ *2,760* gpm For Appropriate Headers (A, B, or A&B)

(O) *CC* Reset LWR Integrator

(O) *CC* 1WL124 Flowrate set @ *1000* gpm

(O) ____ Date/Time Release Started ____/____

(O) ____ 0WLP6160 channel check

____/____
(OPS Contact)

(O) ____ Date/Time Release Secured ____/____

(O) ____ Date/Time First Restart ____/____

(O) ____ Date/Time Second Restart ____/____

COMPLETE FOLLOWING RELEASE:

(O) ____ Date/Time First Trip ____/____

(O) ____ Date/Time Second Trip ____/____

(O) ____ Date/Time Release Secured ____/____

(O) ____ Tank Level ____ %

(O) ____ Flush per Procedure

(O) ____ Flush secured

(O) ____ EMF Reading after Flush:

____ cpm

(O) ____ Volume Released ____ gal

(O) ____ Reset LWR Integrator

(O) ____ Reset EMF49L Setpoints

EMF Reading (cpm) After Flush:

____ cpm X 3 = ____ cpm (Trip2)

Trip 2 X 0.7 = ____ cpm (Trip1)

(O) ____ Highest EMF Reading During Release: ____ cpm

(O) ____ RL Flowrate Interlock Set @ Zero (0)

COMPLETION OF RELEASE ACKNOWLEDGED:

CRS ____ DATE/TIME ____/____

RP SHIFT REVIEW ____ DATE/TIME ____/____

Ensure all signoffs are legible. Print name where indicated on next page.

JPM A.4S

SRO

Catawba Nuclear Station
Admin. JPM A.4S
Sept 2021 NRC Exam

EVALUATION SHEET

Task: Classify an Event and fill out the Emergency Notification Form

Alternate Path: N/A

Facility JPM #: NEW

Safety Function: N/A

K/A 2.4.40 Knowledge of SRO responsibilities in emergency plan implementation.

Importance: 2.7 / 4.5 **CFR:** 41.10 / 43.5 / 45.11

Preferred Evaluation Location: **Preferred Evaluation Method:**

Simulator ______ Classroom __X__ Perform __X__ Simulate ______

References: AD-EP-ALL-0101 (Emergency Classification)
CSD-EP-CNS-0101-02 (EAL Wallcharts)
AD-EP-ALL-0304 (State and County Notifications)

Task Standard: Using AD-EP-ALL-0101 (Emergency Classification) CSD-EP-CNS-0101-02 (EAL Wallcharts), applicant classifies the event as an Alert (SA1.1) in ≤ 15 minutes, and then completes the Emergency Notification Form in ≤ 15 minutes.

Validation Time: 30 minutes **Time Critical:** Yes __X__ No ______

Applicant:
NAME____________________________ Docket #_______________

Time Start: ___________
Time Finish: ___________

Time Critical 1 (<15 min):
Time Start: ___________
Time Finish: ___________

Time Critical 2 (<15 min):
Time Start: ___________
Time Finish: ___________

Performance Rating:

Performance Time _____

SAT ____ UNSAT ____

Examiner: ______________________________ ______________________________/________
NAME SIGNATURE DATE

==

COMMENTS

____

____

____

Catawba Nuclear Station
Admin. JPM A.4S
Sept 2021 NRC Exam

READ TO APPLICANT

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Both Units are at 100% RTP.**
- **A seismic event has been felt within the protected area.**
- **The following events have occurred:**
  - **Annunciator 1AD-4, B/8 OBE EXCEEDED is received in the control room.**
  - **A Loss of Offsite Power (LOOP) occurs on Unit 2.**
  - **2B D/G fails to start.**
  - **The rounds AO reports the 1A NI (Safety Injection) Pump discharge piping is cracked with water leaking out at 125 drops per minute.**

INITIATING CUES:

- **You are the Emergency Coordinator.**
- **Classify this event per AD-EP-ALL-0101 (Emergency Classification).**
- **Emergency Coordinator Judgment is NOT to be used when making this classification.**
- **Fill out the Emergency Notification Form per AD-EP-ALL-0304 (State and County Notifications).**
- **This JPM is time critical.**

EXAMINER NOTE:

After reading cue, provide applicant with a copy of AD-EP-ALL-0101 (Emergency Classification) CSD-EP-CNS-0101-02 (EAL Wallcharts) and AD-EP-ALL-0304 (State and County Notifications)

STEP / STANDARD	SAT / UNSAT

START TIME: _______

<table>
<tr><th>STEP / STANDARD</th><th>SAT / UNSAT</th></tr>
<tr><td>

<u>Time Critical 1 Start:</u> ____________

<u>STEP 1:</u> Classify the Event:

Unit 2 is in an Alert (SA1.1) due to the loss of all but one AC power source for > 15 minutes. Note 1 states “The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.”

Unit 1 is in an Unusual Event (HU2.1) due to the OBE exceeded alarm being received in the control room. Alert (SA9.1) is not applicable because only one train of a required SAFETY SYSTEM needed for the current operating mode is affected.

Since Unit 2 is in a higher classification than Unit 1, the emergency classification for the site will be an Alert (SA1.1).

<u>STANDARD</u>:

Applicant determines from CSD-EP-CNS-0101-02 (EAL Wallcharts), that Catawba is in an Alert SA1.1.

<u>Time Critical 1 Finish:</u> ____________

Examiner Note: This step is critical to determine the proper Emergency Action Level prior to notification of the States and Counties. This time critical must be complete in $\leq$ 15 minutes.

<u>COMMENTS:</u>

</td><td>

CRITICAL STEP

___ SAT

___ UNSAT

</td></tr>
</table>

STEP / STANDARD	SAT / UNSAT
Time Critical 2 Start: ____________ STEP 2: Fill out Emergency Notification Form: STANDARD: Applicant properly fills out the emergency notification form within 15 minutes. **Time Critical 2 Finish:** ____________ **Examiner Note: This step is critical to ensure timely and accurate notification of the States and Counties. This time critical must be complete in ≤ 15 minutes.** COMMENTS: **END OF TASK**	**CRITICAL STEP** ___ SAT ___ UNSAT

STOP TIME ________

APPLICANT CUE SHEET

(RETURN TO EXAMINER UPON COMPLETION OF TASK)

DIRECTION TO APPLICANT:

I will explain the initial conditions, and state the task to be performed. All control room steps shall be performed for this JPM, including any required communications. I will provide initiating cues and reports on other actions when directed by you. Ensure you indicate to me when you understand your assigned task. To indicate that you have completed your assigned task return the handout sheet I provided you.

INITIAL CONDITIONS:

- **Both Units are at 100% RTP.**
- **A seismic event has been felt within the protected area.**
- **The following events have occurred:**
  - **Annunciator 1AD-4, B/8 OBE EXCEEDED is received in the control room.**
  - **A Loss of Offsite Power (LOOP) occurs on Unit 2.**
  - **2B D/G fails to start.**
  - **The rounds AO reports the 1A NI (Safety Injection) Pump discharge piping is cracked with water leaking out at 125 drops per minute.**

INITIATING CUES:

- **You are the Emergency Coordinator.**
- **Classify this event per AD-EP-ALL-0101 (Emergency Classification).**
- **Emergency Coordinator Judgment is NOT to be used when making this classification.**
- **Fill out the Emergency Notification Form per AD-EP-ALL-0304 (State and County Notifications).**
- **This JPM is time critical.**

Declared EAL: _______________

Declaration Time: ______________

Items marked with a * indicate critical tasks

JPM A.4 KEY

NUCLEAR POWER PLANT EMERGENCY NOTIFICATION FORM

MESSAGE # 1 **Confirmation Phone #:** SIM:(803) 831-3807 **AUTHENTICATION CODE #:** ________

Lines 1 – 6 are required for INITIAL Notifications

1. EVENT: ■ DRILL ☐ ACTUAL DECLARATION ☐ TERMINATION (ONLY Lines 1, 2, & 4 required)

2. AFFECTED SITE:

Catawba

3.* EMERGENCY CLASSIFICATION

☐ UNUSUAL EVENT ■ ALERT ☐ SITE AREA EMERGENCY ☐ GENERAL EMERGENCY

4.* EAL # SA1.1 * **Declaration Date:** ___/___/___ **Time:** ________

Termination Date: ___/___/___ **Time:** ________(mark "N/A" for EAL # & Description)

EAL DESCRIPTION: Loss of all but one AC power source to essential buses for 15 minutes or longer.

5.* RELEASE TO THE ENVIRONMENT (caused by the emergency): ■ NONE ☐ IS OCCURRING ☐ HAS OCCURRED

6.* PROTECTIVE ACTION RECOMMENDATIONS:

■ NONE

☐ EVACUATE: ________

☐ SHELTER: ________

☐ CONSIDER THE USE OF KI (POTASSIUM IODIDE) IN ACCORDANCE WITH ORO PLANS AND POLICIES

☐ OTHER: ________

*Lines 7-11 are **NOT** required for INITIAL notifications. Lines 7-11 may be provided separately for follow-up notifications.*

7. PROGNOSIS: Upgrade in classification or PAR change is likely before the next follow-up notification ☐ **Yes** ☐ **No**

8. SITE UNIT(S) STATUS:

AFFECTED UNIT

☐ YES Unit 1 - ________% Power Shutdown: Date ___/___/___Time ________

☐ YES Unit 2 - ________% Power Shutdown: Date ___/___/___Time ________

9. METEOROLOGICAL DATA:

Wind direction from: ________ degrees Wind Speed: ______ mph Precipitation: ______ inches

Stability Class: ☐ A ☐ B ☐ C ☐ D ☐ E ☐ F ☐ G

*Lines 10 - 11 are completed for follow-up notifications, **IF** Line 5 IS OCCURRING or HAS OCCURRED is selected*

10. AIRBORNE RELEASE CHARACTERIZATION: ☐ GROUND ☐ MIXED ☐ ELEVATED

MAGNITUDE UNITS: ☐ Ci ☐ Ci/sec ☐ µCi/sec

Noble Gases: ________ Iodines: ________ Particulates: ________

11. DOSE PROJECTION: Projection period: ________ Hours Estimated Release Duration ______Hours

Performed: Date ___/___/___ Time: ________	**DISTANCE**	**TEDE (mrem)**	**Thyroid CDE (mrem)**
	Site Boundary		
	2 Miles		
	5 Miles		
	10 Miles		

12. REMARKS (As Applicable): ________

13.* APPROVED BY: Operator Name TITLE: Emergency Coordinator Date XX/ XX / XX Time ________

14. NOTIFIED BY: ________ Date ___/___/___ Time ________

15. RECEIVED BY (ORO use only): ________ Date ___/___/___ Time ________

JPM A.4 KEY

GOVERNMENT AGENCIES NOTIFIED

Record the name, date, time, and agencies notified as applicable.

1. ______________________________ York County WP/EOC
(name)
______________________________ 9-1-803/329-1110
(date) (time)

2. ______________________________ Mecklenburg County WP/EOC
(name) 9-704/336-2441 (WP)
______________________________ 9-704/432-4120 (EOC)
(date) (time)

3. ______________________________ Gaston County WP/EOC
(name)
______________________________ 9-704/866-3300
(date) (time)

4. ______________________________ North Carolina EOC/WP
(name) 9-1-919/733-3300 (Primary)
______________________________ 9-1-800/858-0368 (Alt.)
(date) (time)

5. ______________________________ North Carolina Alt. WP
(name) 9-1-828/466-5500
______________________________ 9-1-828/466-5501
(date) (time)

6. ______________________________ North Carolina Alt. EOC
(name) 9-1-919/733-3300 (Primary)
______________________________ 9-1-800-858-0368 (Alt.)
(date) (time)

7. ______________________________ South Carolina WP
(name) 9-1-803/737-8500 (Primary)
______________________________ 9-1-800/811-8045 (Alt.)
(date) (time)

8. ______________________________ South Carolina Alt. WP
(name)
______________________________ 9-1-803/896-9621
(date) (time)

9. ______________________________ South Carolina EOC
(name) 9-1-803/737-8500 (Primary)
______________________________ 9-1-803-737-8724 (Alt.)
(date) (time)

Information Use

NUCLEAR OPERATING FLEET

ADMINISTRATIVE PROCEDURE

AD-EP-ALL-0101

EMERGENCY CLASSIFICATION

REVISION 2

Effective Dates:

07/29/2020	07/29/2020	07/29/2020	07/29/2020	07/29/2020
Brunswick	Catawba	Harris (HNP)	McGuire	Oconee
07/29/2020	07/29/2020			
Robinson	NGO			

REVISION SUMMARY
PRR 2196707 **DESCRIPTION**

- Section 4.1 Step 1: Changed 'Evaluating, classifying and declaring' to 'Evaluates, classifies and declares' to align with AD-DC-ALL-0202.
- Section 4.1 Step 2: New responsibility, 'Enters AD-EP-ALL-0111, Control Room Activation of the ERO'. PRR 2305765
- Section 4.2: Added '(TSC)'.
- Section 4.2 Step 1: Changed 'Evaluating, classifying and declaring' to 'Evaluates, classifies and declares' to align with AD-DC-ALL-0202.
- Section 4.2 Step 2: New responsibility, 'Enters AD-EP-ALL-0105, Activation and Operation of the Technical Support Center'.
- Section 5.1 Step 1 Old CAUTION: Deleted CAUTION.
- Section 5.1 Step 1 NOTE: Moved information from old CAUTION to NOTE. Added second bullet, 'The primary tool for determining the Emergency Classification Level (ECL) is the Emergency Classification Wallchart' (PRR 2229095).
- Section 5.1 Old Steps 1 through 3: Relocated to new Attachment 1, EAL Wallchart Guidance. PRR 2196707
- Section 5.1 Step 1: New step, 'The Emergency Classification Wallchart user may (but is not required to) consult the EAL Technical Basis Document in order to obtain additional information concerning the Initial Conditions (ICs) and EALs under classification consideration.' PRR 2229095
- Section 5.1 Step 2: New step, 'Determine emergency classification per EAL Wallchart.' PRR 2196707
- Section 5.1 Step 2.a: New step, 'If needed, then refer to Attachment 1, EAL Wallchart Guidance.' PRR 2196707
- Section 5.1 Step 3.c Bullet: Changed '[time]' to '[current time]'. PRR 2295310
- Section 5.1 Old Steps 6 through 8: Deleted Old Step 6 and relocated Old Steps 7 and 8 to new Attachment 1, EAL Wallchart Guidance. PRR 2196707
- Section 6.0 Old Step 1: Deleted 'All checklists, logs and forms completed as the result of implementing this procedure shall be collected at the end of the event and provided to the Site Emergency Preparedness Manager'; renumbered remaining step.
- Section 7.2 Old Steps 2, 3, 5, 6, 7, and 8: Deleted [BNP] 0PEP-02.1, [BNP] 0PEP-02.2.1 **(PRR 2280927)**, [RNP] Emergency Action Level Matrix1, [RNP] Emergency Action Level Matrix2, [RNP] Emergency Action Level Matrix3, and [RNP] EPCLA-04 **(PRR 2303130)**; renumbered remaining references.
- Section 7.3 Old Miscellaneous Documents 2 through 8: Deleted [HNP EP-EAL **(PRR 2310065)**, [CNS] EP-EAL-EALMATRIX **(PRR 2304471)**, [ONS] EP-EAL-EALMATRIX **(PRR 2305242)**, [MNS] EP - EAL-WALLCHART, [CNS] EPA D **(PRR 2304471)**, [ONS] EPA SECTION D **(PRR 2305242)**, and [HNP] FAD-HNP-EP-EPEAL MATRIX **(PRR 2310065)**.
- Section 7.3 Miscellaneous Document 2 through Section 7.3 Miscellaneous Document 13, Section 7.3 Miscellaneous Document 16: Added [BNP] CSD-EP-BNP-0101-01, [BNP] CSD-EP-BNP-0101-02 **(PRR 2280927)**, [CNS] CSD-EP-CNS-0101-01, [CNS] CSD-EP-CNS-0101-02 **(PRR 2304471)**, [HNP] CSD-EP-HNP-0101-01, [HNP] CSD-EP-HNP-0101-02 **(PRR 2310065)**, [MNS] CSD-EP-MNS-0101-01, [MNS] CSD-EP-MNS-0101-02 **(PRR 2298233)**, [ONS] CSD-EP-ONS-0101-01, [ONS] CSD-EP-ONS-0101-02 **(PRR 2305242)**, [RNP] CSD-EP-RNP-0101-01, [RNP] CSD-EP-RNP-0101-02 **(PRR 2303130)**, and RIS 2007-02 **(PRR 2213104)**.
- Attachment 1: New attachment, EAL Wallchart Guidance; renumbered remaining attachment. PRR 2196707

TABLE OF CONTENTS

1.0 PURPOSE

1. This procedure provides instruction for the evaluation, classification and declaration of an emergency at a Duke Energy nuclear site.

2. This procedure provides guidance for event termination and entry into Recovery.

2.0 SCOPE

1. This procedure applies to all Duke Energy nuclear operating sites and is continually used to assess events and conditions at the site in order to classify and declare emergencies.

3.0 DEFINITIONS

None

4.0 RESPONSIBILITIES

4.1 Shift Manager

1. Evaluates, classifies and declares emergencies prior to TSC Activation.

2. Enters AD-EP-ALL-0111, Control Room Activation of the ERO.

4.2 Emergency Coordinator (TSC)

1. Evaluates, classifies and declares emergencies after activation of the TSC.

2. Enters AD-EP-ALL-0105, Activation and Operation of the Technical Support Center.

5.0 INSTRUCTIONS

5.1 <u>Assessment, Classification and Declaration of Events</u>

NOTE

- The highest emergency classification for which an Emergency Action Level (EAL) is exceeded shall be declared.
- The primary tool for determining the Emergency Classification Level (ECL) is the Emergency Classification Wallchart.

1. The Emergency Classification Wallchart user **may** (but is not required to) **consult** the EAL Technical Basis Document in order to obtain additional information concerning the Initial Conditions (ICs) and EALs under classification consideration.

2. **Determine** emergency classification per EAL Wallchart.

   a. **<u>IF</u>** needed,
   <u>THEN</u> refer to Attachment 1, EAL Wallchart Guidance.

3. **Declare** the event using the "Update" method as follows:

   a. After completing all required verifications and determining an EAL applies, the EC shall **perform** an update as follows:

   - "Update. I intend to declare a(n) ________ [GE, SAE, Alert, UE] for EAL _______. Are there any challenges to this declaration?"

   b. **<u>IF</u>** there are challenges,
   <u>THEN</u> make corrections.

   c. **<u>IF</u>** there are no challenges or challenges have been resolved,
   <u>THEN</u> announce in the same update:

   - "Update, at _______ [current time] a(n) _________ [GE, SAE, Alert, UE] has been declared for EAL __________. Possible upgrades include __________. End of update."

5.2 Terminating from a Declared Emergency

NOTE

The decision to terminate an event is **NOT** time dependent. There is no regulatory 15 minute notification requirement for exiting a declared emergency.

1. **Complete** Attachment 2, Event Termination Checklist, to determine if termination conditions are met.

2. **IF** conditions do **NOT** allow event termination,
THEN continue monitoring events.

3. **WHEN** conditions allow for event termination,
THEN perform AD-EP-ALL-0110, Recovery, to complete termination of the event.

6.0 RECORDS

1. All logs, forms and records completed as result of implementing this procedure during an actual declared event shall be retained as permanent plant records.

   a. Nuclear Generation Record Retention Rule number 421734, Life of Plant (LOP), Record Type Code NUC-LIC-003, Licensing Life of Plant Records.

7.0 REFERENCES

7.1 Commitments

None

7.2 Procedures

1. [BNP] 0ERP, Radiological Emergency Response Plan (ERP)

2. AD-EP-ALL-0110, Recovery

3. [RNP] PLP-007, Robinson Emergency Plan

4. [HNP] PLP-201, Emergency Plan

7.3 Miscellaneous Documents

1. [CNS] Catawba Emergency Plan
2. [BNP] CSD-EP-BNP-0101-01, EAL Technical Basis Document
3. [BNP] CSD-EP-BNP-0101-02, EAL Wallchart (Both Hot and Cold)
4. [CNS] CSD-EP-CNS-0101-01, EAL Technical Basis Document
5. [CNS] CSD-EP-CNS-0101-02, EAL Wallcharts
6. [HNP] CSD-EP-HNP-0101-01, EAL Technical Basis Document
7. [HNP] CSD-EP-HNP-0101-02, EAL Wallchart (Both Hot and Cold)
8. [MNS] CSD-EP-MNS-0101-01, EAL Technical Basis Document
9. [MNS] CSD-EP-MNS-0101-02, EAL Wallchart (Both Hot and Cold)
10. [ONS] CSD-EP-ONS-0101-01, EAL Technical Basis Document
11. [ONS] CSD-EP-ONS-0101-02, EAL Wallchart (Both Hot and Cold)
12. [RNP] CSD-EP-RNP-0101-01, EAL Technical Basis Document
13. [RNP] CSD-EP-RNP-0101-02, EAL Wallchart (Both Hot and Cold)
14. [MNS] McGuire Emergency Plan
15. [ONS] Oconee Emergency Plan
16. RIS 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events

<< EAL Wallchart Guidance >>

1. **Evaluate** the “All Conditions” EAL Wallchart.

   - **Read** the EAL Wallchart from left to right and top to bottom.
   - **Read** the EAL Category.
   - **Read** the EAL subcategory.
   - **Read** the Initiating Condition.
   - **Read** the Mode Applicability bar.
   - **Read** the category number criterion.
   - **Read** any applicable notes or tables.
   - **Determine** EAL classification threshold applicability

2. **IF** the Reactor Coolant System temperature is greater than [BNP] 212°F **OR** [CNS, HNP, MNS, ONS, RNP] 200°F, **THEN evaluate** the “Hot Conditions” EAL Wallchart.

   - **Read** the EAL Wallchart from left to right and top to bottom.
   - **Read** the EAL Category.
   - **Read** the EAL subcategory.
   - **Read** the Initiating Condition.
   - **Read** the Mode Applicability bar.
   - **Read** the category number criterion.
   - **Read** any applicable notes or tables.
   - **Determine** EAL classification threshold applicability.

<< EAL Wallchart Guidance >>

3. **IF** the Reactor Coolant System temperature is less than or equal to [BNP] 212°F **OR** [CNS, HNP, MNS, ONS, RNP] 200°F, **THEN evaluate** the "Cold Conditions" EAL Wallchart.

   - **Read** the EAL Wallchart from left to right and top to bottom.
   - **Read** the EAL Category.
   - **Read** the EAL subcategory.
   - **Read** the Initiating Condition.
   - **Read** the Mode Applicability bar.
   - **Read** the category number criterion.
   - **Read** any applicable notes or tables.
   - **Determine** EAL classification threshold applicability.

4. **Identify** the highest applicable Emergency Classification Level (ECL).

5. **Declare** the event using the "Update" method as follows:

   a. After completing all required verifications and determining an EAL applies, the EC shall **perform** an update as follows:

      - "Update. I intend to declare a(n) ________ [GE, SAE, Alert, UE] for EAL _______. Are there any challenges to this declaration?"

   b. **IF** there are challenges, **THEN make** corrections.

   c. **IF** there are no challenges or challenges have been resolved, **THEN announce** in the same update:

      - "Update, at _______ [current time] a(n) _________ [GE, SAE, Alert, UE] has been declared for EAL __________. Possible upgrades include __________. End of update."

<< EAL Wallchart Guidance >>

6. **IF** the classification level is below a General Emergency,
   THEN continue to monitor conditions for possible ECL upgrade.

   a. **Return** to the applicable ERO position checklist.

7. **IF** an EAL threshold has **NOT** been met or exceeded,
   THEN perform the following:

   a. **Continue** to monitor conditions for potential changes to ECL.

   b. **Return** to the applicable ERO position checklist.

ATTACHMENT 2
Page 1 of 2

<< Event Termination Checklist >>

Site: ____________________ Event: ______________________________________

		True	False	N/A
1.	Conditions no longer meet an Emergency Action Level and it appears unlikely that conditions will deteriorate.	☐	☐	☐

List any EAL(s) which is/are still exceeded and a justification as to why a state of emergency is no longer applicable:

____

____

____

____

____

____

- If Unusual Event level Emergency Classification Levels (ECLs), then go to the comments/approval section below.
- If all other ECLs, then continue with following questions.

		True	False	N/A
2.	Plant releases of radioactive materials to the environment are under control (within Technical Specifications) or have ceased and the potential for a radioactive release is acceptably low.	☐	☐	☐
3.	The radioactive plume has dissipated and plume tracking is no longer required. The only environmental assessment activities in progress are those necessary to determine the extent of deposition resulting from passage of the plume.	☐	☐	☐
4.	In-plant radiation levels are stable or decreasing, and acceptable given the plant conditions.	☐	☐	☐
5.	The reactor is in a stable operating condition or shutdown condition with long-term core cooling available.	☐	☐	☐

<< Event Termination Checklist >>

		True	False	N/A
6.	The integrity of containment is within Technical Specifications limits for the current plant mode.	☐	☐	☐
7.	The operability and integrity of radioactive waste systems, decontamination facilities, power supplies, electrical equipment and plant instrumentation including radiation monitoring equipment is acceptable.	☐	☐	☐
8.	Any fire, flood, earthquake or similar emergency condition no longer exists.	☐	☐	☐
9.	Any security issues have been resolved.	☐	☐	☐
10.	Any onsite medical issues have been resolved.	☐	☐	☐
11.	Offsite conditions do not unreasonably limit access of outside support to the site and qualified personnel and support services are available.	☐	☐	☐
12.	Discussions have been held with Federal, State and County agencies and agreement has been reached and coordination established to terminate the emergency.	☐	☐	☐
13.	All required state, local and NRC notifications for event termination have been prepared.	☐	☐	☐

It is not necessary that all responses listed above be 'TRUE'; however, all items must be considered prior to event termination from Alert or higher classification or entry into Recovery. For example, it is possible that some conditions remain which exceed an EAL threshold following a severe accident, but entry into Recovery is appropriate. Additionally, other significant items not included on this list may warrant consideration (such as severe weather).

Comments:

Approved by: ______________________________ Date/Time: _______________________

SM, EC or EOF Director in C&C of the event

DUKE ENERGY

Information Use

NUCLEAR OPERATING FLEET

ADMINISTRATIVE PROCEDURE

AD-EP-ALL-0304

STATE AND COUNTY NOTIFICATIONS

REVISION 3

Effective Dates:

08/24/2020	11/16/2020	11/16/2020	11/16/2020	08/24/2020
Brunswick	Catawba	Harris (HNP)	McGuire	Oconee
11/16/2020	11/16/2020			
Robinson	NGO			

REVISION SUMMARY

PRR 2322562
DESCRIPTION

- Old Section 5.2: Deleted 'Documentation'. PRR 2322562
- Section 6.0 Step 1: Changed from 'All logs, forms, and records completed as the result of implementing this procedure during the actual declared event shall be retained as permanent plant records' to 'Provide all checklists, forms, and other documentation generated by this procedure for retention in the Drill, Exercise, or actual event record package, as necessary'. PRR 2322562
- Section 6.0 Step 1.a: Changed 'Nuclear Generation Record Retention Rule Number 421734, Life of Plant (LOP), Record Type Code NUC-LIC-003, Life of Plant Records' to 'In case of an actual event, retain documentation generated by this procedure in the event record package in accordance with AD-EP-ALL-1000, Conduct of Emergency Preparedness'. PRR 2322562
- Section 6.0 Step 1.b: New step, 'In case of a Drill or Exercise, retain documentation generated by this procedure in the Drill record package in accordance with AD-EP-ALL-0803, Evaluation and Critique of Drills and Exercises'. PRR 2322562
- Section 7.2: Added AD-EP-ALL-0803 and AD-EP-ALL-1000; renumbered remaining references.
- Attachment 1: Changed 'Sample' to 'Emergency Notification Form' and placed parentheses around ENF.
- Attachment 2 Section 1.3 Step 2: Deleted 'from AD-EP-ALL-0304, State and County Notifications,'. Moved Attachment 1, Emergency Notification Form (ENF), to new Step 2.a and added 'should be used as a blank ENF' **(PRR 2305089)**. **Attachment 2 Section 1.3 Step 3 NOTE:** Added bullet concerning peer check of the ENF and the means for peer check. PRR 2304167, 2304590
- Attachment 2 Section 1.4 Old Step 1.b: Deleted 'Verify the 'Recipient Name' list is correct'; renumbered remaining steps. PRR 2299450
- Attachment 2 Section 1.4 Step 2: Changed from 'If manually faxing the ENF, then transmit the fax according to the site specific fax machine's user guide to send the ENF' to 'If **NOT** using WebEOC, then communicate information on the ENF verbally while making contact for transmission in Attachment 3, ENF Transmission'. PRR 2329139
- Attachment 4 Section 1.1 Step 3: Changed 'is located in' to 'may be found in the'. Deleted 'Procedure file cabinet' **(PRR 2311970)** and added 'or on hard copies stocked within each sites' TSC and the EOF'.
- Attachment 5: For Lines 10 & 11 added bullet 'Projection period is not required when performing a URI Rapid Dose Assessment' **(PRR 2163335)**. Changed 'Fax or Email' to 'Transmit'; changed 'notification' to 'ENF'; changed 'If completing a pre-printed or blank ENF, then use fax machine to send fax to the appropriate recipients' to 'If using a paper ENF, then transmit the ENF verbally as specified in Attachment 8, ENF Transmissions'. PRR 2329139
- Attachment 6: Under Step 5 changed 'Pre-printed ENF and Blank ENF' to 'Paper ENF'. Moved Old Step 7 to Step 8 and added EOF checkboxes. PRR 2231992
- Attachment 7 Changes **(PRR 2318460)**: On Pages 2of 8 through 8 of 8 added 'The first page of this attachment is applicable to all sites. If any of the statements below are true, then a Release is occurring' prior to Step 1. Numbered items on these pages. Under BNP Step 2 added 'with no operational radiation monitor at the release point'. On Pages 3 of 8 through 8 of 8 Step 3 added 'is' before 'occurring' and deleted 'steam line' and 'with known release path to environment (e.g., stuck open steam line valve, steam line break)'. Under CNS added new Step 3.c; under HNP added new Step 3.b; under MNS in Step 3 changed 'Primary and Secondary' to 'Primary to Secondary' and added new Step 3.c; under ONS added new Step 3.b; under RNP added new Step 3.b.
- Attachment 8 Changes: Arranged ENF Transmissions in alphabetical order (BNP, CNS, HNP, MNS, ONS, RNP) versus the current order (MNS, CNS, ONS, BNP, HNP, RNP). **PRR 2325215** Indented scripted text. **(PRR 2316401).** Under Step 1 of each site's transmission, added 'Please standby' after 'A/an Unusual Event / Alert / Site Area Emergency / General Emergency has been declared'. PRR 2317710

TABLE OF CONTENTS

1.0 PURPOSE

1. This procedure describes the instructions for Initial and Follow-up Notifications to state and county response organizations in the event of a declared emergency at a Duke Energy nuclear site.

2.0 SCOPE

1. This procedure applies to all operating Duke Energy nuclear sites when an emergency is declared in accordance with site Emergency Plans.

3.0 DEFINITIONS

1. **Authentication Code List:** A controlled list of numbers and corresponding words provided by the state(s) to authenticate communications between various parties.
2. **DEMNET:** The primary communication method used by the licensee to communicate emergency information to offsite response organizations.
3. **Emergency Notification Form (ENF):** The document prepared by the licensee to communicate Initial and Follow-up Notifications to the offsite response organizations.
4. **Emergency Release:** An unplanned, quantifiable airborne radiological release to the environment attributed to the emergency event.
5. **Follow-up Notifications:** Periodic notifications to provide updated information to offsite response organizations following an Initial Notification.
6. **Initial Notification:** The first notification made to offsite response organizations upon declaration of any emergency classification, upgrade in classification (Alert, Site Area Emergency, or General Emergency), or change in Protective Action Recommendations (PARs).
7. **Termination Notification:** The last notification sent to offsite response organizations communicating termination of the emergency.
8. **WebEOC:** An electronic emergency response communication system used to provide information within the Duke Energy emergency response facilities and to offsite response organizations.

4.0 RESPONSIBILITIES

4.1 <u>Shift Manager, Emergency Coordinator, or EOF Director</u>

1. Ensures required notifications are made when in Command and Control.

4.2 <u>TSC Dose Assessor/Radiation Assessment Manager</u>

1. Determines release levels and provides accurate and timely dose projections.

4.3 <u>Emergency Coordinator/Accident Assessment Manager</u>

1. Determines PARs.

4.4 <u>Offsite Communicators</u>

1. Complete and transmit ENFs as outlined in their position-specific checklists using the guidance provided in this procedure.

5.0 INSTRUCTIONS

5.1 General Instructions

1. Follow instructions in the position-specific checklists and procedures provided for the Control Room, TSC and EOF to make required notifications to State and County agencies.

2. Use guidance provided in the following attachments, as appropriate:

   - Attachment 1, Emergency Notification Form (ENF)
   - Attachment 2, ENF Completion
   - Attachment 3, ENF Transmission
   - Attachment 4, Authentication Guideline
   - Attachment 5, ENF Quick Reference
   - Attachment 6, Offsite Communications Turnover Checklist
   - Attachment 7, Determining Radiological Release Status
   - Attachment 8, ENF Transmissions

6.0 RECORDS

1. Provide all checklists, forms and other documentation generated by this procedure for retention in the Drill, Exercise, or actual event record package, as necessary.

   a. In case of an actual event, retain documentation generated by this procedure in the event record package in accordance with AD-EP-ALL-1000, Conduct of Emergency Preparedness, if required.

   b. In case of a Drill or Exercise, retain documentation generated by this procedure in the Drill record package in accordance with AD-EP-ALL-0803, Evaluation and Critique of Drills and Exercises, if required.

7.0 REFERENCES

7.1 Commitments

None

7.2 Procedures

1. [BNP] 0ERP, Radiological Emergency Response Plan (ERP)
2. AD-EP-ALL-0104, ERO Common Guideline and Forms
3. AD-EP-ALL-0406, Duke Emergency Management Network (DEMNET)
4. AD-EP-ALL-0803, Evaluation and Critique of Drills and Exercises
5. AD-EP-ALL-1000, Conduct of Emergency Preparedness
6. [RNP] PLP-007, Robinson Emergency Plan
7. [HNP] PLP-201, Emergency Plan

7.3 Miscellaneous Documents

1. 66 FR 5427, Consideration of Potassium Iodide in Emergency Plans
2. [CNS] Catawba Emergency Plan
3. CSD-EP-ALL-0104-01, Emergency Telephone Directory
4. [MNS] McGuire Emergency Plan
5. NRC Regulatory Issue Summary (RIS) 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events
6. [ONS] Oconee Emergency Plan

ATTACHMENT 1
Page 1 of 1

<< Emergency Notification Form (ENF) >>

NUCLEAR POWER PLANT EMERGENCY NOTIFICATION FORM

MESSAGE # ________ **Confirmation Phone #:** _________________ **AUTHENTICATION CODE #:** ___________

Lines 1 – 6 are required for INITIAL Notifications

1. **EVENT:** ☐ DRILL ☐ ACTUAL DECLARATION ☐ TERMINATION (ONLY Lines 1, 2, & 4 required)
2. **AFFECTED SITE:**
   ☐ BRUNSWICK ☐ CATAWBA ☐ HARRIS ☐ MCGUIRE ☐ OCONEE ☐ ROBINSON
3. **EMERGENCY CLASSIFICATION**
   ☐ UNUSUAL EVENT ☐ ALERT ☐ SITE AREA EMERGENCY ☐ GENERAL EMERGENCY
4. **EAL #**______________ **Declaration Date**: ____/____/____ **Time:** ________
   Termination Date: ____/____/____**Time:** ________(mark "N/A" for EAL # & Description)
   EAL DESCRIPTION:____
   ____
5. **RELEASE TO THE ENVIRONMENT** (caused by the emergency): ☐ NONE ☐ IS OCCURRING ☐ HAS OCCURRED
6. **PROTECTIVE ACTION RECOMMENDATIONS**:
   ☐ NONE
   ☐ EVACUATE: ____
   ☐ SHELTER: ____
   ☐ CONSIDER THE USE OF KI (POTASSIUM IODIDE) IN ACCORDANCE WITH ORO PLANS AND POLICIES
   ☐ OTHER:____

*Lines 7-11 are **NOT** required for INITIAL notifications. Lines 7-11 may be provided separately for follow-up notifications.*

7. **PROGNOSIS: Upgrade in classification or PAR change is likely before the next follow-up notification** ☐ **Yes** ☐ **No**
8. **SITE UNIT(S) STATUS:**
   AFFECTED UNIT
   ☐ YES Unit 1 - _________% Power Shutdown: Date ___/_____/____Time _________
   ☐ YES Unit 2 - _________% Power Shutdown: Date ___/_____/____Time _________
   ☐ YES Unit 3 - _________% Power Shutdown: Date ___/_____/____Time _________
   ☐ YES Unit 4 - _________% Power Shutdown: Date ___/_____/____Time _________
9. **METEOROLOGICAL DATA:**
   Wind direction from: ________ degrees Wind Speed: _______ mph Precipitation: _______ inches
   Stability Class: ☐ A ☐ B ☐ C ☐ D ☐ E ☐ F ☐ G

*Lines 10 - 11 are completed for follow-up notifications, **IF** Line 5 IS OCCURRING or HAS OCCURRED is selected*

10. **AIRBORNE RELEASE CHARACTERIZATION:** ☐ GROUND ☐ MIXED ☐ ELEVATED
    MAGNITUDE UNITS: ☐ Ci ☐ Ci/sec ☐ µCi/sec
    Noble Gases: ____________ Iodines: ____________ Particulates: ____________
11. **DOSE PROJECTION:** Projection period: __________ Hours Estimated Release Duration ________Hours

Performed: Date ___/_____/____ Time: _________	**DISTANCE**	**TEDE (mrem)**	**Thyroid CDE (mrem)**
	Site Boundary		
	2 Miles		
	5 Miles		
	10 Miles		

12. **REMARKS (As Applicable):**____
    ____
    ____
13. **APPROVED BY:** ________________________TITLE: __________________Date ___/_____/____Time __________
14. **NOTIFIED BY:** ____Date ___/_____/____Time __________
15. **RECEIVED BY** (ORO use only): ______________________________________Date ___/_____/____Time __________

<< ENF Completion >>

1.0 INSTRUCTIONS

1.1 Initial Notifications

1. Make the Initial Notification to the state(s) and counties within 15 minutes of the event declaration time or a change in Protective Action Recommendation (PAR) (evacuation and shelter only) using the information contained in the ENF. The ENF is preferred to be approved to make notifications, however it is not required to make notifications of a declared event.

2. The Initial Notification for a General Emergency classification must include PARs.

3. If a higher Emergency Classification Level (ECL) is declared before the notification begins for the lesser ECL, then perform the following: [7.3.5]

   a. If possible, then update the ENF to reflect the higher ECL and complete the notification within 15 minutes of the lesser ECL.

   b. If it is not possible to update the ENF within 15 minutes of the lesser classification, then add a Line 12 remark that explains a change in classification is forthcoming **AND** continue notification for lesser ECL to meet the 15 minute requirement.

      (1) Complete an Initial ENF for the higher ECL **AND** perform the notification within 15 minutes of declaration of the higher ECL.

<< ENF Completion >>

1.2 Follow-up Notifications

NOTES

- Examples of significant changes requiring a Follow-up Notification include: Dose Assessment Data characterizing a release or addition of KI, change in release status, evacuation or relocation of site personnel, fires onsite, chemical spills, explosions, MERT activation or injured personnel transported offsite, and any event that would cause or require offsite agency response.
- Follow-up Notifications are to occur within approximately 60 minutes from the first contact of the previous notification until a new time period is agreed upon by all offsite agencies.
- A Termination Notification is a type of Follow-up Notification. Termination Notifications must occur within 60 minutes from the time the last notification was made.
- The purpose for timely turnover from the Shift Manager/EC to the TSC/EC is to relieve the Control Room crew from burdens and distractions that might delay plant stabilization and recovery.
  - ◊ The transfer of Command and Control is a priority over ENF follow-up notifications since the regulatory requirement for timely ERO activation (i.e., within 75 minutes of classification) takes precedence over follow-up.
  - ◊ Do not allow follow-up notification to delay Command and Control turnover, which is required to be completed prior to ERO facility activation.
- It is acceptable to perform an ENF follow-up early, event oversight permitting.
  - ◊ The Shift Manager can request the Offsite Communicator to contact the Offsite Response Organizations to alert the individuals that turnover is in progress, provide a status of the event, and inform the ORO the follow-up form will be sent imminently upon turnover completion.

1. If a significant change to plant conditions occurs, then perform a Follow-up Notification as soon as possible, with the expectation of completion of notification within 30 minutes.

2. An Initial General Emergency Notification should be followed by another Follow-up Notification to include Dose assessment and Meteorological data as soon as possible, with the expectation of completion of notification within 30 minutes.

3. Follow-up Notifications continue to be made throughout the event within approximately 60 minutes from the first contact of the previous notification in Attachment 8, Section 2, or as agreed upon by all offsite agencies receiving the ENF.

<< ENF Completion >>

1.2 Follow-up Notifications (continued)

a. Documentation (log name of officials agreeing to new schedule) shall be maintained for any agreed-upon schedule change.

4. If a Follow-up Notification is due and an upgrade to a higher classification is declared, then do not complete the Follow-up ENF and complete the Initial Notification for the higher classification.

1.3 ENF Completion

1. If WebEOC is available, then access the WebEOC ENF.

2. If WebEOC is **NOT** available, then obtain a pre-printed or blank ENF.

a. Attachment 1, Emergency Notification Form (ENF), should be used as a blank ENF.

NOTE

- Attachment 5, Page 2, lists the positions responsible for providing information to complete the ENF.
- Only one ENF can be open at a time for the same event (e.g., opening an ENF in the TSC and then opening in the EOF will not allow the EOF to finalize the ENF).
- Peer check of the ENF is a best practice. Means for peer check include, but are not limited to, the following:
  - ◊ Control Room - Review by another person in the Control Room
  - ◊ TSC or EOF - Using a video projector for all TSC or EOF personnel review
  - ◊ EOF - EOF Offsite Communicator sharing their computer screen with the TSC Offsite Communicator for cross-facility review

3. Complete the ENF per Attachment 5, ENF Quick Reference.

4. Select the 'Approve' button on the WebEOC ENF to automatically open the WebEOC Emergency Notification Management panel with the recipient name list auto-populated.

<< ENF Completion >>

1.4 Sending the ENF

1. If using WebEOC, then perform the following:

   a. Access the Emergency Notification Management panel for the applicable ENF.

   b. Select the 'Send ENF' button.

   c. Select 'OK'.

      (1) The 'Emergency Notification Management' panel will indicate it is sending the messages.

   d. When 'Completed Sending Messages' appears, then select 'OK'.

2. If **NOT** using WebEOC, then communicate information on the ENF verbally while making contact for transmission in Attachment 3, ENF Transmission.

<< ENF Transmission >>

1.0 INSTRUCTIONS

1.1 General Information

NOTE

- Blast dial is a reverse conference call that calls recipients rather than the recipients calling in.
- Once initiated, the recipients will be called.
- Once the recipient answers, recipient will join the call automatically.
- Notification can then be made.

1. Duke Emergency Management Network (DEMNET) is the primary communication device.
   a. Commercial telephone (blast dial) is the first back-up.
   b. Commercial Telephone line (Individual Line) is the second back-up.
   c. Satellite Phone is the third back-up.
2. Information regarding blast dial and back-up phone numbers is located in CSD-EP-ALL-0104-01, Emergency Telephone Directory.
3. DEMNET instructions are contained in AD-EP-ALL-0406, Duke Emergency Management Network (DEMNET).
4. The NRC requires that ALL state and county agencies be notified within 15 minutes of emergency declaration. Attachment 3, ENF Transmission, Step 7 meets the 15 minute notification time requirement.

1.2 Communicating the ENF using DEMNET Ethernet Phone Group Call

1. Verify that there is not another DEMNET call in progress that another Notify call would override by either method below:
   - Verifying DEMNET icon status
   - Contacting the Control Room, TSC, or EOF via landline

<< ENF Transmission >>

1.2 Communicating the ENF using DEMNET Ethernet Phone Group Call (continued)

2. Initiate a "Notify" call to the offsite agencies as follows:

   a. Verify the appropriate screen for the affected nuclear site has been selected.

   b. Select the orange oval group "Notify" button.

   c. When the prompt appears on screen asking to connect the call, then select 'Yes'.

      (1) Verify as the call is being connected that the 'Call in Progress' screen is displayed.

   d. Press **AND** hold the push-to-talk (PTT) button whenever it is desired to talk to the agencies.

NOTE

Attachment 8, ENF Transmissions, is used to perform and document communications with the offsite agencies for the appropriate site

3. When agencies start to answer the call, then state the plant name and the declaration of the emergency using Attachment 8, Section 1.

4. Note the time of the first contact on Attachment 8, Section 2. This is the start time for the next Follow-Up Notification.

5. Repeat Attachment 3 Section 1.2 Step 3 until it is believed that all agencies have answered the call.

NOTE

Message authentication can be requested any time; however, authentication is only required if message transmittal is other than via DEMNET.

6. Refer to Attachment 4, Authentication Guideline as needed.

7. Once it is believed that all agencies have answered the notification call, state the reason for the notification using Attachment 8, Section 3.

8. Conduct and document a roll call using Attachment 8, Section 3 to verify all required agencies are on the line by stating each State or County agency and allowing time for a reply.

<< ENF Transmission >>

1.2 Communicating the ENF using DEMNET Ethernet Phone Group Call (continued)

9. If an offsite agency does not answer, then contact the missing agency by any other means (i.e. commercial phone, cell phone, satellite phone, radio) while continuing with the DEMNET notification if possible:

   a. If needed, then request another communicator contact the agency using other means (i.e. commercial phone, cell phone, satellite phone, radio) to complete the notification.

10. Record the time that all agencies were notified on Attachment 8, Section 4.

11. Verify each agency has received the ENF using Attachment 8, Section 5, and record the names of the person receiving the notification.

    a. If the form was transmitted verbally, then there is no need to verify that the agency/agencies received the ENF by other means.

12. Determine if there are any agency questions and then conclude the message using Attachment 8, Section 6.

    a. If a question applies to information on the ENF, then provide the information to the requesting agency.

    b. If a question requires follow up, then document the questions and state that you will follow up with the requesting agency after the notification is complete.

13. Depress the "Hang up" button to end the DEMNET Call.

14. If using the WebEOC ENF, then perform the following:

    a. Open the control panel.

    b. Select 'EN Form' from the WebEOC control panel.

    c. Select 'View' button in the Notification Management column for the applicable message.

    d. Record recipient names in the Government Agencies Notified 'Received By' field and enter times and dates.

    e. Select the 'Update' Button.

<< ENF Transmission >>

1.2 Communicating the ENF using DEMNET Ethernet Phone Group Call (continued)

f. Enter the Time and Date that the first agency responded into the Notification Time and Date fields.

g. Verify or record the name of the Off Site Communicator making the notification call in the 'Notified By' field.

h. Select the 'Save' button to auto populate the ENF with the Notification Time and Date on Line 14.

15. If not using WebEOC, then:

a. If desired, then record offsite agency recipient names on the back of the ENF.

b. Document the notification time and date on Line 14 of the approved original ENF.

16. If an agency question requires follow up, then perform the following:

a. Document the questions in the Communicator's position log.

b. Document the name, agency, and contact information of the individual making the request.

c. Inform the individual you will contact them regarding the question.

d. Obtain the answer to the question from the appropriate ERO member and request the SM, Emergency Coordinator, or EOF Director to approve release of the information to the off-site agency.

e. Document the answer provided by the Shift Manager, Emergency Coordinator, or EOF Director in the Communicator's position log.

f. Contact requesting agency.

g. Provide the answer to the requesting agency.

h. Document the time the answer was provided to the requesting agency in the Communicator's position log.

<< Authentication Guideline >>

1.0 INSTRUCTIONS

1.1 General Information

1. When using the DEMNET phone, then authentication is typically **NOT** required unless requested by an off-site agency.

2. The Authentication Code List is a controlled list of numbers and corresponding words provided by the state(s) to authenticate communications between the various parties.

   a. Authentication provides assurance to the receiver that the information is valid.

   b. Authentication may be performed anytime the receiver wishes to assure the information received from the transmitter is valid.

   c. The receiver provides a number from the Authentication Code List.

   d. The transmitter provides the word that corresponds to the number provided by the receiver.

3. The Authentication Code List may be found in the WebEOC Emergency Notification Management panel using 'Get Authentication Code' button or on hard copies stocked within each sites' TSC and the EOF.

1.2 Responding to a Request for Authentication

1. If using WebEOC, then perform the following:

   a. Access the Emergency Notification Management panel for the appropriate message (ENF).

   b. If Authentication is requested, then perform the following:

      (1) Request the state or county representative to provide a number from the Authentication Code List.

      (2) Enter the number provided by the Agency into the AUTHENTICATION # field.

      (3) Select 'Get Authentication Code' (the Code Word(s) will appear).

      (4) Provide code word(s) to the requestor.

      (5) Select "OK" in the Pop up window to make the window disappear.

<< Authentication Guideline >>

1.2 Responding to a Request for Authentication (continued)

(6) Select 'Save' to populate the ENF.

c. If Authentication is **NOT** requested, then perform the following:

(1) Enter N/A into the AUTHENTICATION # field.

(2) Select 'Save' to auto-populate the ENF.

2. If not using WebEOC, then perform the following:

a. If Authentication is requested, then request state or county representative to provide a number from the Authentication Code List.

(1) Provide code word(s) corresponding to number from the Authentication Code List.

(2) Document the number in the AUTHENTICATION # field located at the top of the ENF.

b. If Authentication is **NOT** requested, then enter N/A in the AUTHENTICATION # field located at the top of the ENF.

1.3 Receiving a Call

1. If receiving a call from an off-site agency and the identity of call is **NOT** known, then perform the following:

a. Tell the caller you'd like to authenticate the call.

b. Provide a number from Authentication Code List to caller.

c. Obtain code word(s) corresponding with number on the Authentication Code List from the caller.

d. If caller has questions pertaining to the event in progress, then perform Attachment 3 Section 1.2 Step 16.

ATTACHMENT 5
Page 1 of 2

<< ENF Quick Reference >>

* Performance Indicator Accuracy Measure

Above line 1	Select 'Initial' or 'Follow-up'. Verify or record **Message Number.** • Only lines 1-6 and line 13 are required for an Initial Notification. • Only lines 1-6, 7-11 and 13 are required for a Follow-up Notification
Line 1*	**Event** - Select or ensure appropriate block for Drill, Actual Declaration, or Termination. • Only lines 1, 2, 4 and 13 are required for a Termination Message.
Line 2*	Verify, record, or select correct site. Verify, record, or select appropriate **Confirmation Phone#.**
Line 3*	**Emergency Classification** - Select or ensure correct classification.
Line 4*	• Verify, record, or select correct Emergency Action Level **(EAL) #** and corresponding EAL Description**.** • If termination, then verify, select or record "N/A for EAL# and EAL Description. • Select or enter **Declaration** or **Termination Date and Time**.
Line 5*	**Release to the environment -** Select or verify appropriate block for None, Is Occurring**,** or Has Occurred. (Refer to Attachment 7, Determining Radiological Release Status, for additional guidance)
Line 6*	**Protective Action Recommendations** • If Unusual Event, Alert, or Site Area Emergency, then verify, select or mark None. • If General Emergency, then select or mark 'Evacuate' and/or 'Shelter' as appropriate. Verify, select or record appropriate zones. If circumstance warrant, then select or mark 'KI' or 'Other'. • Once an evacuation PAR has been issued for a zone, it shall **NOT** be rescinded until recovery.
Line 7	**Prognosis** - If it is likely a higher emergency classification or a change in PARS will be required before the next follow-up, then select or mark 'Yes'. Otherwise mark 'No'.
Line 8	**Site Unit(s) status** • Select or verify 'Yes' for the unit(s) affected with the highest classification or units with the same classification caused by the same event. **Complete for all Units** • If Unit(s) is (are) Shutdown, then record 0% power **AND** Shutdown Time and Date. • If Unit(s) is (are) **NOT** Shutdown, then record % reactor power only.
Line 9	**Meteorological Data –** Verify, record or import Meteorological data including wind speed, direction, precipitation, and stability class.
Lines 10 & 11	Airborne Release Characterization and Dose Projection - Record or import radiological information. Lines 10 and 11 are only completed if Line 5 has 'Is Occurring' or Has Occurred' selected. • Only BNP Vent Stack is an Elevated Release. • All releases other than BNP Vent Stack are Ground Releases. • Projection period is not required when performing a URI Rapid Dose Assessment.
Line 12	**Remarks -** Record any additional information using short narratives without acronyms.
Line 13	**Approved By -** Enter or record approvers name, title, and date and time.
Line 14	**Notified By -** If known, then enter the name of the person who will be notifying the State/Counties, OR if unknown, then leave blank and it will be filled out when the notification is complete.
Line 15	**Received By – This field is only used by off-site agencies.** This field will not be present on WebEOC.
Validate	• If using WebEOC, then select 'Validate' to identify issues for resolution. • If completing a pre-printed or blank ENF, then review all data to identify and resolve issues.
Approve	• If using WebEOC, then obtain approval and select 'Approve'. • If completing a pre-printed or blank ENF, then obtain approval by having approver sign the ENF
Transmit	If using WebEOC, after the form is approved, then the screen will advance to the notification management screen. Ensure correct recipients are specified and select **Send ENF.** If using a paper ENF, then transmit the ENF verbally as specified in Attachment 8, ENF Transmissions.
Record Notification	Record the **notification date, time, and notified by, and authentication** (if performed) information either in WebEOC or manually.

<< ENF Quick Reference >>

ENF Completion Responsibilities

Above Line 1	Off-Site Communicator
Line 1	Off-Site Communicator
Line 2	Off-Site Communicator
Line 3	Shift Manager / Operations Manager / Accident Assessment Manager
Line 4	Shift Manager / Operations Manager / Accident Assessment Manager
Line 5	Shift Manager / Radiation Protection Manager / Radiological Assessment Manager
Line 6	Shift Manager / Radiation Protection Manager / Radiological Assessment Manager
Line 7	Shift Manager / Operations Manager / Accident Assessment Manager
Line 8	Shift Manager / Operations Manager / Accident Assessment Manager
Line 9	Shift Manager / Radiation Protection Manager / Radiological Assessment Manager
Line 10	Shift Manager / Radiation Protection Manager / Radiological Assessment Manager
Line 11	Shift Manager / Radiation Protection Manager / Radiological Assessment Manager
Line 12	Anyone
Line 13	Shift Manager / Emergency Coordinator / EOF Director
Line 14	Off-Site Communicator

<< Offsite Communications Turnover Checklist >>

1. Affected Site(s):

Brunswick Nuclear Plant (BNP)	
Catawba Nuclear Station (CNS)	
Harris Nuclear Plant (HNP)	
McGuire Nuclear Station (MNS)	
Oconee Nuclear Station (ONS)	
Robinson Nuclear Plant (RNP)	

2. Obtain the most recent notification forms.

3. Emergency Classification (check):

   ☐ UE ☐ Alert ☐ Site Area Emergency ☐ General Emergency

   Emergency Declaration (time): ___________

4. Last Emergency Notification Form Message #: ______
   Notification (time): ________

5. Using: ☐ WebEOC, ☐ Paper ENF

6. Next Message Due at (time): ________

7. Communications Problems or Offsite agencies activated:

   ____

   ____

   ____

8. Alternate Facility Activated - TSC: ☐ Yes ☐ No OSC: ☐ Yes ☐ No

   EOF: ☐ Yes ☐ No

9. Site Assembly Status: N/A ☐ In progress ☐ Completed ☐

   Time Site Assembly Initiated: _______

   Number of persons unaccounted for: ________

<< Offsite Communications Turnover Checklist >>

10. Site Evacuation: ☐ Yes ☐ No Time Evacuation Initiated: _______

Approximate Number of persons being evacuated: ________

Site Evacuation Location (indicate relocation area):

BNP	**Yes**	**No**
Technical Training Center (TTC)		
Home		

CNS	**Yes**	**No**
Plant Allen (Belmont, NC)		
York Operations Center (York, SC)		
Home		

HNP	**Yes**	**No**
Administration Building		
Home		

MNS	**Yes**	**No**
TTTC (Bldg. 7403)		
Cowans Ford Dam Service Bay		
Mt. Holly Training Center		
McGuire Office Complex (MOC) Auditorium (Bldg. 7422)		
Home		

ONS	**Yes**	**No**
Daniel High School		
Keowee Elementary School		
Home		

RNP	**Yes**	**No**
Unit 2 Administration Building		
Building 110 (Next to Lake Robinson)		
Home		

ATTACHMENT 6
Page 3 of 3

<< Offsite Communications Turnover Checklist >>

11. Other Pertinent Information (e.g., fires or explosions onsite, MERT activation, injured personnel transported offsite, chemical spills.)

    ____

    ____

    ____

12. Turnover Completed by: ______________________________

    At (date/time): ________________

<< Determining Radiological Release Status >>

Use the following guidance in determining how to report radiological release status on ENF.

A release in progress must be:

- Quantifiable
- Airborne
- Attributed to the declared emergency event

An increase on a radiation monitor is defined as a validated reading resulting from plant conditions (e.g., not attributable to instrument drift or electronic noise) above pre-emergency levels.

The following are release designations as listed on the Emergency Notification Form (ENF):

None - No release of quantifiable airborne radioactivity attributed to the emergency event.

Has Occurred - Any quantifiable airborne radioactivity released to the environment attributed to the emergency event, but has stopped.

Is Occurring - Any quantifiable airborne radioactivity release to the environment attributed to the emergency event, and is currently in progress as defined by the following criteria.

<< Determining Radiological Release Status >>

BNP

The first page of this attachment is applicable to all sites. If any of the statements below are true, then a Release is occurring.

1. Any gaseous radiation monitor below shows increase in activity:
   a. Main Stack: D12-RM-23S
   b. Reactor Building Vent Noble Gas: CAC-AQH-1264-3
   c. Turbine Building Vent Rad: D12-RM-23
2. Any drywell radiation monitor below showing an increase in activity **AND** a known leak path to environment exists with no operational radiation monitor at the release point:
   a. Drywell High Rad Monitor 30 ft elevation: D-22-RM-4195
   b. Drywell High Rad Monitor 57 ft elevation: D-22-RM-4196
   c. Drywell High Rad Monitor 23 ft elevation: D-22-RM-4197
   d. Drywell High Rad Monitor 57 ft elevation: D-22-RM-4198
3. A known unmonitored release path exists **AND** a radioactive source exists (RCS or fuel damage).
4. Field Monitoring Team results of airborne radioactivity (other than naturally occurring) detected by survey or sampling.

<< Determining Radiological Release Status >>

CNS

The first page of this attachment is applicable to all sites. If any of the statements below are true, then a Release is occurring.

1. Any gaseous radiation monitor below that shows an increase in activity:
   a. Unit Vent Low/High: 1/2 EMF-36 L/H
   b. Unit Vent Extended: 1/2 EMF-54
2. Any containment radiation monitor below that shows an increase in activity **AND** known leak path to environment exists:
   a. Containment High Range: 1/2 EMF-53 A/B
   b. Containment Gas Low/High: 1/2 EMF-39 L/H
3. Any radiation monitor below that shows an increase in activity **AND** Primary to Secondary leakage is occurring:
   a. Unit 1 Steam Line: 1 EMF-26/27/28/29
   b. Unit 2 Steam Line: 2 EMF-10/11/12/13
   c. Condenser Steam Air Ejector: 1/2 EMF-33 (not for use in dose assessment)
4. A known unmonitored release path exists **AND** a radioactive source exists (RCS or fuel damage).
5. Field Monitoring Team results of airborne radioactivity (other than naturally occurring) detected by survey or sampling.

<< Determining Radiological Release Status >>

HNP

The first page of this attachment is applicable to all sites. If any of the statements below are true, then a Release is occurring.

1. Any gaseous radiation monitor below that shows an increase in activity:
   a. Plant Vent: RM-21AV-3509-1SA
   b. Turbine Building: RM-1TV-3536-1
   c. Waste Processing Building Vent 5: RM-1WV-3546-1
   d. Waste Processing Building Vent 5A: RM-1WV-3547-1
2. Any containment radiation monitor below that shows an increase in activity **AND** known leak path to environment exists:
   a. Containment High Range: RM-01CR-3589SA
   b. Containment High Range: RM-01CR-3590SB
   c. Containment Gas: REM-01LT-3502ASA
3. Any radiation monitor below that shows an increase in activity **AND** Primary to Secondary leakage is occurring:
   a. Main Steam Line A/B/C: RM-01MS-3591/3592/3593
   b. Turbine Building Vent Stack Wide Range Gas Monitor: RM-01TV-3536-1

<< Determining Radiological Release Status >>

HNP (continued)

4. Damage to irradiated fuel in conjunction with a valid high alarm on any Containment or Fuel Handling Building radiation monitor listed below:

   a. Containment Ventilation Isolation: RM-1CR-3561 A/B/C/D

   b. Spent Fuel Pool SW, SE, SW: RM-1FR-3564A-SA

   c. Spent Fuel Pool SW, SE, SE: RM-1FR-3564B-SB

   d. Spent Fuel Pool SW, SE, SW: RM-1FR-3565A-SA

   e. Spent Fuel Pool SW, SE, SE: RM-1FR-3565B-SB

   f. Spent Fuel Pool NE, NW, NE: RM-1FR-3566A-SA

   g. Spent Fuel Pool NW, NE, NW: RM-1FR-3566B-SB

   h. Spent Fuel Pool NW, NE, NW: RM-1FR-3567A-SA

   i. Spent Fuel Pool NE, NW, NE: RM-1FR-3567B-SB

5. A known unmonitored release path exists **AND** a radioactive source exists (RCS or fuel damage).

6. Field Monitoring Team results of airborne radioactivity (other than naturally occurring) detected by survey or sampling.

<< Determining Radiological Release Status >>

MNS

The first page of this attachment is applicable to all sites. If any of the statements below are true, then a Release is occurring.

1. Any gaseous radiation monitor below that shows an increase in activity:
   a. Unit Vent Low/High/High-High: 1/2 EMF-36 L/H/HH
2. Any Containment radiation monitor below that shows an increase in activity **AND** known leak path to environment exists:
   a. Containment High Range: 1/2 EMF-51 A/B
   b. Containment Gas Low/High: 1/2 EMF-39 L/H
3. Any radiation monitor below that shows an increase in activity **AND** Primary to Secondary leakage is occurring:
   a. Unit 1 Steam Line: 1 EMF-24/25/26/27
   b. Unit 2 Steam Line: 2 EMF-10/11/12/13
   c. Condenser Steam Air Ejector: 1 & 2 EMF-33
4. A known unmonitored release path exists **AND** a radioactive source exists (RCS or fuel damage).
5. Field Monitoring Team results of airborne radioactivity (other than naturally occurring) detected by survey or sampling.

<< Determining Radiological Release Status >>

ONS

The first page of this attachment is applicable to all sites. If any of the statements below are true, then a Release is occurring.

1. Any gaseous radiation monitor below that shows an increase in activity:
   a. Unit Vent Low: 1/2/3 RIA-45
   b. Unit Vent High: 1/2/3 RIA-46
   c. Unit Vent High Gross Gamma: 1/2/3 RIA-56
2. Any Containment radiation monitor below that shows an increase in activity **AND** known leak path to environment exists:
   a. Containment High Range: 1/2/3 RIA-57
   b. Containment High Range: 1/2/3 RIA-58
   c. Containment Gas Low: 1/2/3 RIA-49
   d. Containment Gas High: 1/2/3 RIA-49A
3. Any radiation monitor below that shows an increase in activity **AND** Primary to Secondary leakage is occurring:
   a. Steam Line: 1/2/3 RIA-16/17
   b. Air Ejector Off Gas: 1/2/3 RIA-40
4. A known unmonitored release path exists **AND** a radioactive source exists (RCS or fuel damage).
5. Field Monitoring Team results of airborne radioactivity (other than naturally occurring) detected by survey or sampling.

<< Determining Radiological Release Status >>

RNP

The first page of this attachment is applicable to all sites. If any of the statements below are true, then a Release is occurring.

1. Any gaseous radiation monitor below that shows an increase in activity:
   a. Plant Vent Low/Mid/High: R-14 C/D/E
   b. Fuel Handling Building Exhaust: R-20
   c. Fuel Handling Building Exhaust High: R-30
2. Any Containment radiation monitor below that shows an increase in activity **AND** known leak path to environment exists:
   a. Containment High Range: R-32 A/B
   b. Containment Gas: R-12
3. Any radiation monitor below that shows an increase in activity **AND** Primary to Secondary leakage is occurring:
   a. Steam Line: R-31 A/B/C
   b. Condenser Air Ejector Gas: R-15
4. A known unmonitored release path exists **AND** a radioactive source exists (RCS or fuel damage).
5. Field Monitoring Team results of airborne radioactivity (other than naturally occurring) detected by survey or sampling.

<< ENF Transmissions >>

1.	"This is the **Brunswick Nuclear Plant** A/an **Unusual Event / Alert / Site Area Emergency / General Emergency** has been declared. Please standby." **OR** "We have terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**. Please standby."		ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE
2.	Record time of first agency on the phone. Repeat the above until all agencies are on-line.		Time: _________	Time: _________	Time: _________
3.	"The Brunswick Nuclear Plant has declared a/an Unusual Event / Alert / Site Area Emergency / General Emergency." If Initial or Follow-up of GENERAL EMERGENCY, then state: "We recommend the following protective actions:" (read ENF LINE 6) "I'm now going to conduct a roll call." (verify each agency is on the line) **OR** "The Brunswick Nuclear Plant has Terminated from the Unusual Event / Alert / Site Area Emergency / General Emergency." "I need to confirm required agencies are on-line."	Brunswick County	Verified ☐	Verified ☐	Verified ☐
		New Hanover County	Verified ☐	Verified ☐	Verified ☐
		North Carolina	Verified ☐	Verified ☐	Verified ☐
		Coast Guard	Verified ☐	Verified ☐	Verified ☐
4.	Record time all required agencies are notified.		Time: _________	Time: _________	Time: _________
5.	"A copy of message # _____ has been sent to you. When I call your agency, please state whether you have received the message and also state your name." Verify receipt of message and RECORD names. → If any agency did not receive the ENF **OR** you cannot transmit the ENF, then request the agency to obtain a blank ENF and transmit the message verbally line by line	Brunswick County			
		New Hanover County			
		North Carolina			
		Coast Guard			
6.	"Are there any questions?" Record questions per Attachment 3 Section 1.2 Step 16 "This concludes the notification, my name is ______________________________. Duke Energy clear."				Page ___ of ___

<< ENF Transmissions >>

1.	"This is the **Catawba Nuclear Station** A/an **Unusual Event / Alert / Site Area Emergency / General Emergency** has been declared. Please standby." **OR** "We have terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**. Please standby."		ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE
2.	Record time of first agency on the phone. Repeat the above until all agencies are on-line.		Time: _________	Time: _________	Time: _________
3.	"The Catawba Nuclear Station has declared a/an **Unusual Event / Alert / Site Area Emergency / General Emergency**." If Initial or Follow-up of GENERAL EMERGENCY, then state: "We recommend the following protective actions:" (read ENF LINE 6) "I'm now going to conduct a roll call." (verify each agency is on the line) **OR** "The Catawba Nuclear Station has Terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**." "I need to confirm required agencies are on-line."	York County	Verified ☐	Verified ☐	Verified ☐
		Mecklenburg Co.	Verified ☐	Verified ☐	Verified ☐
		Gaston County	Verified ☐	Verified ☐	Verified ☐
		North Carolina	Verified ☐	Verified ☐	Verified ☐
		South Carolina	Verified ☐	Verified ☐	Verified ☐
4.	Record time all required agencies are notified.		Time: _________	Time: _________	Time: _________
5.	"A copy of message # _____ has been sent to you. When I call your agency, please state whether you have received the message and also state your name." Verify receipt of message and RECORD names. → If any agency did not receive the ENF **OR** you cannot transmit the ENF, then request the agency to obtain a blank ENF and transmit the message verbally line by line	York County			
		Mecklenburg Co.			
		Gaston County			
		North Carolina			
		South Carolina			
6.	"Are there any questions?" Record questions per Attachment 3 Section 1.2 Step 16 "This conclues the notification, my name is ______________________________. Duke Energy clear."				Page ___ of ___

<< ENF Transmissions >>

1.	"This is the **Harris Nuclear Plant** A/an **Unusual Event / Alert / Site Area Emergency / General Emergency** has been declared. Please standby." **OR** "We have terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**. Please standby."		ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE
2.	Record time of first agency on the phone. Repeat the above until all agencies are on-line.		Time: _________	Time: _________	Time: _________
3.	"The Harris Nuclear Plant has declared a/an Unusual Event / Alert / Site Area Emergency / General Emergency." If Initial or Follow-up of GENERAL EMERGENCY, then state: "We recommend the following protective actions:" (read ENF LINE 6) "I'm now going to conduct a roll call." (verify each agency is on the line) **OR** "The Harris Nuclear Plant has Terminated from the Unusual Event / Alert / Site Area Emergency / General Emergency." "I need to confirm required agencies are on-line."	Wake County	Verified ☐	Verified ☐	Verified ☐
		Chatham County	Verified ☐	Verified ☐	Verified ☐
		Harnett County	Verified ☐	Verified ☐	Verified ☐
		Lee County	Verified ☐	Verified ☐	Verified ☐
		North Carolina	Verified ☐	Verified ☐	Verified ☐
4.	Record time all required agencies are notified.		Time: _________	Time: _________	Time: _________
5.	"A copy of message # _____ has been sent to you. When I call your agency, please state whether you have received the message and also state your name." Verify receipt of message and RECORD names. → If any agency did not receive the ENF **OR** you cannot transmit the ENF, then request the agency to obtain a blank ENF and transmit the message verbally line by line	Wake County			
		Chatham County			
		Harnett County			
		Lee County			
		North Carolina			
6.	"Are there any questions?" Record questions per Attachment 3 Section 1.2 Step 16 "This concludes the notification, my name is ______________________________. Duke Energy clear."				Page ___ of ___

<< ENF Transmissions >>

1.	"This is the **McGuire Nuclear Station** A/an **Unusual Event / Alert / Site Area Emergency / General Emergency** has been declared. Please standby." **OR** "We have terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**. Please standby."		ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE
2.	Record time of first agency on the phone. Repeat the above until all agencies are on-line.		Time: _________	Time: _________	Time: _________
3.	"The McGuire Nuclear Station has declared a/an **Unusual Event / Alert / Site Area Emergency / General Emergency**." If Initial or Follow-up of GENERAL EMERGENCY, then state: "We recommend the following protective actions:" (read ENF LINE 6) "I'm now going to conduct a roll call." (verify each agency is on the line) **OR** "The McGuire Nuclear Station has Terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**." "I need to confirm required agencies are on-line."	Gaston County	Verified ☐	Verified ☐	Verified ☐
		Lincoln County	Verified ☐	Verified ☐	Verified ☐
		Iredell County	Verified ☐	Verified ☐	Verified ☐
		Mecklenburg Co.	Verified ☐	Verified ☐	Verified ☐
		Catawba County	Verified ☐	Verified ☐	Verified ☐
		Cabarrus County	Verified ☐	Verified ☐	Verified ☐
		North Carolina	Verified ☐	Verified ☐	Verified ☐
4.	Record time all required agencies are notified.		Time: _________	Time: _________	Time: _________
5.	"A copy of message # _____ has been sent to you. When I call your agency, please state whether you have received the message and also state your name." Verify receipt of message and RECORD names. → If any agency did not receive the ENF **OR** you cannot transmit the ENF, then request the agency to obtain a blank ENF and transmit the message verbally line by line.	Gaston County			
		Lincoln County			
		Iredell County			
		Mecklenburg Co.			
		Catawba County			
		Cabarrus County			
		North Carolina			
6.	"Are there any questions?" Record questions per Attachment 3 Section 1.2 Step 16 "This concludes the notification, my name is ____. Duke Energy clear."				Page ___ of ___

<< ENF Transmissions >>

1.	"This is the **Oconee Nuclear Station** A/an **Unusual Event / Alert / Site Area Emergency / General Emergency** has been declared. Please standby." **OR** "We have terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**. Please standby."		ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE
2.	Record time of first agency on the phone. Repeat the above until all agencies are on-line.		Time: _________	Time: _________	Time: _________
3.	"The Oconee Nuclear Station has declared a/an **Unusual Event / Alert / Site Area Emergency / General Emergency**." If Initial or Follow-up of GENERAL EMERGENCY, then state: "We recommend the following protective actions:" (read ENF LINE 6) "I'm now going to conduct a roll call." (verify each agency is on the line) **OR** "The Oconee Nuclear Station has Terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**." "I need to confirm required agencies are on-line."	Oconee County	Verified ☐	Verified ☐	Verified ☐
		Pickens County	Verified ☐	Verified ☐	Verified ☐
		South Carolina	Verified ☐	Verified ☐	Verified ☐
4.	Record time all required agencies are notified.		Time: _________	Time: _________	Time: _________
5.	"A copy of message # _____ has been sent to you. When I call your agency, please state whether you have received the message and also state your name." Verify receipt of message and RECORD names. → If any agency did not receive the ENF **OR** you cannot transmit the ENF, then request the agency to obtain a blank ENF and transmit the message verbally line by line	Oconee County			
		Pickens County			
		South Carolina			
6.	"Are there any questions?" Record questions per Attachment 3 Section 1.2 Step 16 "This concludes the notification, my name is ________________________________. Duke Energy clear."				

<< ENF Transmissions >>

1.	"This is the **Robinson Nuclear Plant** A/an **Unusual Event / Alert / Site Area Emergency / General Emergency** has been declared. Please standby." **OR** "We have terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**. Please standby."		ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE	ENF # ____ Initial / Follow-up UE / AL / SAE / GE
2.	Record time of first agency on the phone. Repeat the above until all agencies are on-line.		Time: _________	Time: _________	Time: _________
3.	"The Robinson Nuclear Plant has declared a/an **Unusual Event / Alert / Site Area Emergency / General Emergency**." If Initial or Follow-up of GENERAL EMERGENCY, then state: "We recommend the following protective actions:" (read ENF LINE 6) "I'm now going to conduct a roll call." (verify each agency is on the line) **OR** "The Robinson Nuclear Plant has Terminated from the **Unusual Event / Alert / Site Area Emergency / General Emergency**." "I need to confirm required agencies are on-line."	Darlington County	Verified ☐	Verified ☐	Verified ☐
		Chesterfield County	Verified ☐	Verified ☐	Verified ☐
		Lee County	Verified ☐	Verified ☐	Verified ☐
		South Carolina	Verified ☐	Verified ☐	Verified ☐
4.	Record time all required agencies are notified.		Time: _________	Time: _________	Time: _________
5.	"A copy of message # _____ has been sent to you. When I call your agency, please state whether you have received the message and also state your name." Verify receipt of message and RECORD names. → If any agency did not receive the ENF **OR** you cannot transmit the ENF, then request the agency to obtain a blank ENF and transmit the message verbally line by line	Darlington County			
		Chesterfield County			
		Lee County			
		South Carolina			
6.	"Are there any questions?" Record questions per Attachment 3 Section 1.2 Step 16 "This concludes the notification, my name is ________________________________. Duke Energy clear."				Page ___ of ___

NUCLEAR POWER PLANT EMERGENCY NOTIFICATION FORM

MESSAGE # ________ **Confirmation Phone #:** SIM:(803) 831-3807 **AUTHENTICATION CODE #:** ___________

Lines 1 – 6 are required for INITIAL Notifications

1. EVENT: ☒ DRILL ☐ ACTUAL DECLARATION ☐ TERMINATION (ONLY Lines 1, 2, & 4 required)

2. AFFECTED SITE:

Catawba

3. EMERGENCY CLASSIFICATION

☐ UNUSUAL EVENT ☒ ALERT ☐ SITE AREA EMERGENCY ☐ GENERAL EMERGENCY

4. EAL # SA1.1 **Declaration Date**: ____/____/____ **Time:** ________

Termination Date: ____/____/____ **Time:** ________(mark "N/A" for EAL # & Description)

EAL DESCRIPTION: Loss of all but one AC power source to essential buses for 15 minutes or longer.

5. RELEASE TO THE ENVIRONMENT (caused by the emergency): ☒ NONE ☐ IS OCCURRING ☐ HAS OCCURRED

6. PROTECTIVE ACTION RECOMMENDATIONS:

☒ NONE
☐ EVACUATE: ____________________
☐ SHELTER: ____________________
☐ CONSIDER THE USE OF KI (POTASSIUM IODIDE) IN ACCORDANCE WITH ORO PLANS AND POLICIES
☐ OTHER: ____________________

*Lines 7-11 are **NOT** required for INITIAL notifications. Lines 7-11 may be provided separately for follow-up notifications.*

7. PROGNOSIS: Upgrade in classification or PAR change is likely before the next follow-up notification ☐ **Yes** ☐ **No**

8. SITE UNIT(S) STATUS:

AFFECTED UNIT

☐ YES Unit 1 - _________% Power Shutdown: Date ___/_____/____Time _________
☐ YES Unit 2 - _________% Power Shutdown: Date ___/_____/____Time _________

9. METEOROLOGICAL DATA:

Wind direction from: ________ degrees Wind Speed: _______ mph Precipitation: _______ inches
Stability Class: ☐ A ☐ B ☐ C ☐ D ☐ E ☐ F ☐ G

*Lines 10 - 11 are completed for follow-up notifications, **IF** Line 5 IS OCCURRING or HAS OCCURRED is selected*

10. AIRBORNE RELEASE CHARACTERIZATION: ☐ GROUND ☐ MIXED ☐ ELEVATED

MAGNITUDE UNITS: ☐ Ci ☐ Ci/sec ☐ µCi/sec
Noble Gases: ____________ Iodines: ____________ Particulates: ____________

11. DOSE PROJECTION: Projection period: __________ Hours Estimated Release Duration ________Hours

Performed: Date ___/_____/____ Time: _________	**DISTANCE**	**TEDE (mrem)**	**Thyroid CDE (mrem)**
	Site Boundary		
	2 Miles		
	5 Miles		
	10 Miles		

12. REMARKS (As Applicable):____________________

13. APPROVED BY: ____________________ TITLE: Emergency Coordinator Date ___/_____/____Time __________

14. NOTIFIED BY: ____________________ Date ___/_____/____Time __________

15. RECEIVED BY (ORO use only)**:** ____________________ Date ___/_____/____Time __________

GOVERNMENT AGENCIES NOTIFIED

Record the name, date, time, and agencies notified as applicable.

1. ______ York County WP/EOC
(name)
______ 9-1-803/329-1110
(date) (time)

2. ______ Mecklenburg County WP/EOC
(name) 9-704/336-2441 (WP)
______ 9-704/432-4120 (EOC)
(date) (time)

3. ______ Gaston County WP/EOC
(name)
______ 9-704/866-3300
(date) (time)

4. ______ North Carolina EOC/WP
(name) 9-1-919/733-3300 (Primary)
______ 9-1-800/858-0368 (Alt.)
(date) (time)

5. ______ North Carolina Alt. WP
(name) 9-1-828/466-5500
______ 9-1-828/466-5501
(date) (time)

6. ______ North Carolina Alt. EOC
(name) 9-1-919/733-3300 (Primary)
______ 9-1-800-858-0368 (Alt.)
(date) (time)

7. ______ South Carolina WP
(name) 9-1-803/737-8500 (Primary)
______ 9-1-800/811-8045 (Alt.)
(date) (time)

8. ______ South Carolina Alt. WP
(name)
______ 9-1-803/896-9621
(date) (time)

9. ______ South Carolina EOC
(name) 9-1-803/737-8500 (Primary)
______ 9-1-803-737-8724 (Alt.)
(date) (time)

EAL WALLCHARTS

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

R – Abnorm. Rad Levels / Rad Effluent

1 – Rad Effluent

General Emergency – Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RG1.1
Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RG1.2
Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 4)

RG1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Site Area Emergency – Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RS1.1
Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RS1.2
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 4)

RS1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Alert – Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA1.1
Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

RA1.2
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Notes 4)

RA1.3
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)

RA1.4
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

Unusual Event – Release of gaseous or liquid radioactivity greater than 2 times the SLC/TS limits for 60 minutes or longer
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU1.1
Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

RU1.2
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x SLC/TS limits for ≥ 60 min. (Notes 1, 2)

2 – Irradiated Fuel Event

General Emergency – Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RG2.1
Spent fuel pool level cannot be restored to at least 14.5 ft. (Level 3) on 1(2)KFP5780 or 1(2)NVP8790 for ≥ 60 min. (Note 1)

Site Area Emergency – Spent fuel pool level at the top of the fuel racks
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RS2.1
Lowering of spent fuel pool level to 14.5 ft. (Level 3) on 1(2)KFP5780 or 1(2)NVP8790

Alert – Significant lowering of water level above, or damage to, irradiated fuel
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity
AND
A Trip 2 radiation alarm on **any** of the following radiation monitor indications:
- 1EMF15 (2EMF4) Spent Fuel Building Refueling Bridge
- 1EMF17 (2EMF2) Reactor Building Refueling Bridge
- 1EMF42 (2EMF42) Spent Fuel Pool Ventilation
- 1EMF39L (2EMF39L) Containment Noble Gas

RA2.3
Lowering of spent fuel pool level to 24.5 ft. (Level 2) on 1(2)KFP5780 or 1(2)NVP8790

Unusual Event – Unplanned loss of water level above irradiated fuel
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU2.1
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication
AND
UNPLANNED rise in corresponding area radiation levels as indicated by **any** of the following radiation monitors:
- 1EMF15 (2EMF4) Spent Fuel Building Refueling Bridge
- 1EMF17 (2EMF2) Reactor Building Refueling Bridge

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent Noble Gas Low	1/2EMF36L	--------	--------	4.18E+6 cpm	5.75E+3 cpm
Gaseous	Unit Vent Noble Gas High	1/2EMF36H	2.21E+4 cpm	2.22E+3 cpm	2.42E+2 cpm	--------
Liquid	Liquid Waste Effluent Line	0EMF49L	--------	--------	--------	4.50E+6 cpm
Liquid	Monitor Tank Discharge	0EMF57L	--------	--------	--------	4.97E+5 cpm

3 – Area Rad Levels

General Emergency – (none listed)

Site Area Emergency – (none listed)

Table R-2 Safe Operation & Shutdown Rooms/Areas

Bldg. Elevation	Unit 1 Room/Area	Unit 2 Room/Area	Mode
Auxiliary 577'	Rm 478 (1EMXA)	Rm 469 (2EMXA)	4
	Rm 496 (1ETA)	Rm 486 (2ETA)	4
	Rm 496 (1EMXS)	Rm 486 (2EMXS)	4
	AB-577', JJ-57 (1MXK)	AB-577', JJ-57 (2MXK)	4
Auxiliary 560'	Rm 330 (1EMXJ)	Rm 320 (2EMXJ)	4
	Rm 372 (1ETB)	Rm 362 (2ETB)	4
	Rm 372 (1EMXD)	Rm 362 (2EMXD)	4

Alert – Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA3.1
Dose rates > 15 mR/hr in **EITHER** of the following areas:
Control Room (EMF12)
OR
Central Alarm Station (by survey)

RA3.2
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Unusual Event – None

H – Hazards

1 – Security

General Emergency – None

Site Area Emergency – HOSTILE ACTION within the PROTECTED AREA
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

Alert – HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Unusual Event – Confirmed SECURITY CONDITION or threat
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU1.1
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision

HU1.2
Notification of a credible security threat directed at the site

HU1.3
A validated notification from the NRC providing information of an aircraft threat

2 – Seismic Event

Alert – None

[Refer to EAL CA6.1 OR SA9.1 for escalation due to seismic event]

Unusual Event – Seismic event greater than OBE level
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU2.1
Seismic event > OBE as indicated by OBE EXCEEDED alarm on 1AD-4, B/8

3 – Natural or Tech. Hazard

Alert – None

[Refer to EAL CA6.1 OR SA9.1 for escalation due to natural or technological hazard]

Unusual Event – Hazardous event
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU3.1
A tornado strike within the PROTECTED AREA

HU3.2
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

HU3.3
Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

NOTES

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Note 9: In the absence of reliable NCS temperature indication caused by the loss of decay heat removal capability, classification should be based on time to boil data when in Mode 5 and 6.

Note 10: If the loss of containment cooling threshold is exceeded due to loss of both trains of VX-CARF, this EAL only applies if at least one train of VX-CARF is not operating, per design, after the 10 minute actuation delay for greater than or equal to 15 minutes.

Note 11: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 12: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

4 – Fire

General Emergency – None

Site Area Emergency – None

Alert – **[Refer to EAL CA6.1 OR SA9.1 for escalation due to FIRE]**

Table H-1 Fire Areas
- Reactor Building (Containment)
- Auxiliary Building
- Diesel Generator Rooms
- RN Pump House
- Dog Houses
- Standby Shutdown Facility (SSF)

Unusual Event – FIRE potentially degrading the level of safety of the plant
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU4.1
A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within any Table H-1 area

HU4.2
Receipt of a single fire alarm (i.e., no other indications of a FIRE)
AND
The fire alarm is indicating a FIRE within any Table H-1 area
AND
The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1)

HU4.3
A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

CNS

5 – Hazardous Gases

Table H-2 Safe Operation & Shutdown Rooms/Areas

Bldg. Elevation	Unit 1 Room/Area	Unit 2 Room/Area	Mode
Auxiliary 577'	Rm 478 (1EMXA)	Rm 469 (2EMXA)	4
	Rm 496 (1ETA)	Rm 486 (2ETA)	4
	Rm 496 (1EMXS)	Rm 486 (2EMXS)	4
	AB-577', JJ-57 (1MXK)	AB-577', JJ-57 (2MXK)	4
Auxiliary 560'	Rm 330 (1EMXJ)	Rm 320 (2EMXJ)	4
	Rm 372 (1ETB)	Rm 362 (2ETB)	4
	Rm 372 (1EMXD)	Rm 362 (2EMXD)	4

Alert – Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown
Modes: 4

HA5.1
Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas
AND
Entry into the room or area is prohibited or IMPEDED (Note 5)

Unusual Event – None

6 – Control Room Evacuation

General Emergency – None

Site Area Emergency – Inability to control a key safety function from outside the Control Room
Modes: 1 | 2 | 3 | 4 | 5 | 6

HS6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panels or Standby Shutdown Facility
AND
Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):
- Reactivity (Modes 1, 2 and 3 **only**)
- Core Cooling
- NCS heat removal

Alert – Control Room evacuation resulting in transfer of plant control to alternate locations
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panels or Standby Shutdown Facility

Unusual Event – None

7 – EC Judgment

General Emergency – Other conditions exist which in the judgment of the Site Emergency Coordinator warrant declaration of a General Emergency
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HG7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

Site Area Emergency – Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a Site Area Emergency
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

Alert – Other conditions exist that in the judgment of the Site Emergency Coordinator warrant declaration of an Alert
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA7.1
Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Unusual Event – Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a UE
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

E – ISFSI

General Emergency – None

Table E-1 ISFSI Dose Limits

NAC UMS	NAC MAGNASTOR
• 100 mrem/hr (neutron + gamma) on the side	• 240 mrem/hr gamma on the vertical surfaces
• 100 mrem/hr (neutron + gamma) on the top	• 10 mrem/hr neutron on the vertical surfaces
• 200 mrem/hr (neutron + gamma) at the air inlets or outlets	• 900 mrem/hr (neutron + gamma) on the top

Unusual Event – Damage to a loaded cask CONFINEMENT BOUNDARY
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

EU1.1
Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 dose limit

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | DEF Defueled

DUKE ENERGY
Catawba Nuclear Station
CSD-EP-CNS-0101-02 Rev 000

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

C – Cold SD/ Refuel System Malfunct.

1 – NCS Level

General Emergency – Loss of NCS inventory affecting fuel clad integrity with containment challenged
Modes: 5 | 6

CG1.1
NCS level **cannot** be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **any** of the following:
- UNPLANNED increase in **any** Table C-6 sump or tank level due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage
- Reactor Building Refueling Bridge Monitor 1EMF17 (2EMF2) reading > 9,000 mR/hr
- Erratic Source Range or Gamma Metric Monitor indication

AND
Any Containment Challenge indication, Table C-1

Site Area Emergency – Loss of NCS inventory affecting core decay heat removal capability
Modes: 5 | 6

CS1.1
NCS level **cannot** be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **any** of the following:
- UNPLANNED increase in **any** Table C-6 sump or tank level due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage
- Reactor Building Refueling Bridge Monitor 1EMF17 (2EMF2) reading > 9,000 mR/hr
- Erratic Source Range or Gamma Metric Monitor indication

Alert – Loss of NCS inventory
Modes: 5 | 6

CA1.1
UNPLANNED loss of NCS inventory as indicated by NCS water level < 6.5% (wide range)

CA1.2
NCS water level **cannot** be monitored for ≥ 15 min. (Note 1)
AND EITHER
- UNPLANNED increase in **any** Table C-6 sump or tank level due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage

Unusual Event – UNPLANNED loss of NCS inventory for 15 minutes or longer
Modes: 5 | 6

CU1.1
UNPLANNED loss of NCS inventory results in NCS water level less than a required lower limit for ≥ 15 min. (Note 1)

CU1.2
NCS water level **cannot** be monitored
AND EITHER
- UNPLANNED increase in **any** Table C-6 sump or tank level due to a loss of NCS inventory
- Visual observation of UNISOLABLE NCS leakage

2 – Loss of Essential AC Power

General Emergency – None

Site Area Emergency – None

Alert – Loss of **all** offsite and **all** onsite AC power to essential buses for 15 minutes or longer
Modes: 5 | 6 | DEF

CA2.1
Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1)

Unusual Event – Loss of **all but one** AC power source to essential buses for 15 minutes or longer
Modes: 5 | 6 | DEF

CU2.1
AC power capability, Table C-2, to essential 4160V buses 1(2)ETA and 1(2)ETB reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

3 – NCS Temp

General Emergency – None

Site Area Emergency – None

Alert – Inability to maintain plant in cold shutdown
Modes: 5 | 6

CA3.1
UNPLANNED increase in NCS temperature to > 200°F for > Table C-3 duration (Notes 1, 9)
OR
UNPLANNED NCS pressure increase > 10 psig due to a loss of NCS cooling (this does **not** apply during water-solid plant conditions)

Unusual Event – UNPLANNED increase in NCS temperature
Modes: 5 | 6

CU3.1
UNPLANNED increase in NCS temperature to > 200°F due to loss of decay heat removal capability

CU3.2
Loss of **all** NCS temperature and NCS level indication for ≥ 15 min. (Note 1)

4 – Loss of Vital DC Power

General Emergency – None

Site Area Emergency – None

Alert – None

Unusual Event – CU4 Loss of Vital DC power for 15 minutes or longer
Modes: 5 | 6

CU4.1
< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC buses for ≥ 15 min. (Note 1)

5 – Loss of Comm

General Emergency – None

Site Area Emergency – None

Alert – None

Unusual Event – Loss of **all** onsite or offsite communications capabilities
Modes: 5 | 6 | DEF

CU5.1
Loss of **all** Table C-4 onsite communication methods
OR
Loss of **all** Table C-4 ORO communication methods
OR
Loss of **all** Table C-4 NRC communication methods

6 – Hazardous Event Affecting Safety Systems

General Emergency – None

Site Area Emergency – None

Alert – Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode
Modes: 5 | 6

CA6.1
The occurrence of **any** Table C-5 hazardous event
AND
- Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER of the following:
- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode.
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Note 11, 12)

Unusual Event – None

Table C-1 Containment Challenge Indications
- CONTAINMENT CLOSURE **not** established (Note 6)
- Containment hydrogen concentration > 6%
- UNPLANNED rise in containment pressure

Table C-2 AC Power Sources

Offsite
- ATC (Train A)
- SATA (Train A) (if already aligned)
- ATD (Train B)
- SATB (Train B) (if already aligned)

Onsite
- D/G A (Train A)
- D/G B (Train B)

Table C-3 NCS Heat-up Duration Thresholds

NCS Status	Containment Closure Status	Heat-up Duration
Intact (but **not** reduced inventory)	N/A	60 min.*
Not intact **OR** At reduced inventory	established	20 min.*
	not established	0 min.

* If an NCS heat removal system is in operation within this time frame and NCS temperature is being reduced, the EAL is **not** applicable

Table C-4 Communication Methods

System	Onsite	ORO	NRC
Public Address	X		
Internal Telephones	X		
Onsite Radios	X		
DEMNET		X	
Commercial Telephones		X	X
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telecommunications System (ETS)		X	X

Table C-5 Hazardous Events
- Seismic event (earthquake)
- Internal or external FLOODING event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Shift Manager

Table C-6 Sumps/Tanks
- Containment Floor & Equipment Sump
- Incore Sump (alarm)
- ND/NS sump
- NCDT
- PRT

NOTES

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Note 9: In the absence of reliable NCS temperature indication caused by the loss of decay heat removal capability, classification should be based on time to boil data when in Mode 5 and 6.

Note 10: If the loss of containment cooling threshold is exceeded due to loss of both trains of VX-CARF, this EAL only applies if at least one train of VX-CARF is not operating, per design, after the 10 minute actuation delay for greater than or equal to 15 minutes.

Note 11: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 12: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

Date & Time of Shutdown
______ Date ______ Time

CNS

EAL Identifier
XXX.X
- Category (R, H, E, C, S, F)
- Emergency classification (G, S, A, U)
- Sequential number within subcategory/classification
- Subcategory number (1 if no subcategory)

MODES 5, 6 & Defueled

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

R — Abnorm. Rad Levels / Rad Effluent

1 Rad Effluent

General Emergency — Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RG1.1
Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RG1.2
Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RG1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Site Area Emergency — Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RS1.1
Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RS1.2
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RS1.3
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Alert — Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA1.1
Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

RA1.2
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RA1.3
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)

RA1.4
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

Unusual Event — Release of gaseous or liquid radioactivity greater than 2 times the SLC/TS limits for 60 minutes or longer
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU1.1
Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

RU1.2
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x SLC/TS limits for ≥ 60 min. (Notes 1, 2)

2 Irradiated Fuel Event

General Emergency — Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RG2.1
Spent fuel pool level cannot be restored to at least 14.5 ft. (Level 3) on 1(2)KFP5780 or 1(2)NVP8790 for ≥ 60 min. (Note 1)

Site Area Emergency — Spent fuel pool level at the top of the fuel racks
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RS2.1
Lowering of spent fuel pool level to 14.5 ft. (Level 3) on 1(2)KFP5780 or 1(2)NVP8790

Alert — Significant lowering of water level above, or damage to, irradiated fuel
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA2.1
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2
Damage to irradiated fuel resulting in a release of radioactivity
AND
A Trip 2 radiation alarm on **any** of the following radiation monitor indications:
- 1EMF15 (2EMF4) Spent Fuel Building Refueling Bridge
- 1EMF17 (2EMF2) Reactor Building Refueling Bridge
- 1EMF42 (2EMF42) Spent Fuel Pool Ventilation
- 1EMF39L (2EMF39L) Containment Noble Gas

RA2.3
Lowering of spent fuel pool level to 24.5 ft. (Level 2) on 1(2)KFP5780 or 1(2)NVP8790

Unusual Event — Unplanned loss of water level above irradiated fuel
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RU2.1
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication
AND
UNPLANNED rise in corresponding area radiation levels as indicated by **any** of the following radiation monitors:
- 1EMF15 (2EMF4) Spent Fuel Building Refueling Bridge
- 1EMF17 (2EMF2) Reactor Building Refueling Bridge

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent Noble Gas Low	1/2EMF36L	--------	--------	4.18E+6 cpm	5.75E+3 cpm
Gaseous	Unit Vent Noble Gas High	1/2EMF36H	2.21E+4 cpm	2.22E+3 cpm	2.42E+2 cpm	--------
Liquid	Liquid Waste Effluent Line	0EMF49L	--------	--------	--------	4.50E+6 cpm
Liquid	Monitor Tank Discharge	0EMF57L	--------	--------	--------	4.97E+5 cpm

3 Area Rad Levels

Table R-2 Safe Operation & Shutdown Rooms/Areas

Bldg. Elevation	Unit 1 Room/Area	Unit 2 Room/Area	Mode
Auxiliary 577'	Rm 478 (1EMXA)	Rm 469 (2EMXA)	4
	Rm 496 (1ETA)	Rm 486 (2ETA)	4
	Rm 496 (1EMXS)	Rm 486 (2EMXS)	4
	AB-577', JJ-57 (1MXK)	AB-577', JJ-57 (2MXK)	4
Auxiliary 560'	Rm 330 (1EMXJ)	Rm 320 (2EMXJ)	4
	Rm 372 (1ETB)	Rm 362 (2ETB)	4
	Rm 372 (1EMXD)	Rm 362 (2EMXD)	4

Alert — Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

RA3.1
Dose rates > 15 mR/hr in **EITHER** of the following areas:
Control Room (EMF12)
OR
Central Alarm Station (by survey)

RA3.2
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Unusual Event — None

H — Hazards

1 Security

General Emergency — None

Site Area Emergency — HOSTILE ACTION within the PROTECTED AREA
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS1.1
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

Alert — HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA1.1
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision

HA1.2
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Unusual Event — Confirmed SECURITY CONDITION or threat
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU1.1
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision

HU1.2
Notification of a credible security threat directed at the site

HU1.3
A validated notification from the NRC providing information of an aircraft threat

2 Seismic Event

Alert — None
[Refer to EAL CA6.1 OR SA9.1 for escalation due to seismic event]

Unusual Event — Seismic event greater than OBE level
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU2.1
Seismic event > OBE as indicated by OBE EXCEEDED alarm on 1AD-4, B/8

3 Natural or Tech. Hazard

Alert — None
[Refer to EAL CA6.1 OR SA9.1 for escalation due to natural or technological hazard]

Unusual Event — Hazardous event
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU3.1
A tornado strike within the PROTECTED AREA

HU3.2
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

HU3.3
Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

NOTES

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Note 9: In the absence of reliable NCS temperature indication caused by the loss of decay heat removal capability, classification should be based on time to boil data when in Mode 5 and 6.

Note 10: If the loss of containment cooling threshold is exceeded due to loss of both trains of VX-CARF, this EAL only applies if at least one train of VX-CARF is not operating, per design, after the 10 minute actuation delay for greater than or equal to 15 minutes.

Note 11: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 12: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

4 Fire

General Emergency — None

Site Area Emergency — None

Alert — [Refer to EAL CA6.1 OR SA9.1 for escalation due to FIRE]

Table H-1 Fire Areas
- Reactor Building (Containment)
- Auxiliary Building
- Diesel Generator Rooms
- RN Pump House
- Dog Houses
- Standby Shutdown Facility (SSF)

Unusual Event — FIRE potentially degrading the level of safety of the plant
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU4.1
A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within any Table H-1 area

HU4.2
Receipt of a single fire alarm (i.e., no other indications of a FIRE)
AND
The fire alarm is indicating a FIRE within any Table H-1 area
AND
The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1)

HU4.3
A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

Table H-2 Safe Operation & Shutdown Rooms/Areas

Bldg. Elevation	Unit 1 Room/Area	Unit 2 Room/Area	Mode
Auxiliary 577'	Rm 478 (1EMXA)	Rm 469 (2EMXA)	4
	Rm 496 (1ETA)	Rm 486 (2ETA)	4
	Rm 496 (1EMXS)	Rm 486 (2EMXS)	4
	AB-577', JJ-57 (1MXK)	AB-577', JJ-57 (2MXK)	4
Auxiliary 560'	Rm 330 (1EMXJ)	Rm 320 (2EMXJ)	4
	Rm 372 (1ETB)	Rm 362 (2ETB)	4
	Rm 372 (1EMXD)	Rm 362 (2EMXD)	4

5 Hazardous Gases

General Emergency — None

Site Area Emergency — None

Alert — Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown
Modes: 4

HA5.1
Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas
AND
Entry into the room or area is prohibited or IMPEDED (Note 5)

Unusual Event — None

6 Control Room Evacuation

CNS

Site Area Emergency — Inability to control a key safety function from outside the Control Room
Modes: 1 | 2 | 3 | 4 | 5 | 6

HS6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panels or Standby Shutdown Facility
AND
Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):
- Reactivity (Modes 1, 2 and 3 **only**)
- Core Cooling
- NCS heat removal

Alert — Control Room evacuation resulting in transfer of plant control to alternate locations
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA6.1
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panels or Standby Shutdown Facility

Unusual Event — None

7 EC Judgment

General Emergency — Other conditions exist which in the judgment of the Site Emergency Coordinator warrant declaration of a General Emergency
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HG7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area

Site Area Emergency — Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a Site Area Emergency
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HS7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

Alert — Other conditions exist that in the judgment of the Site Emergency Coordinator warrant declaration of an Alert
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HA7.1
Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Unusual Event — Other conditions existing that in the judgment of the Site Emergency Coordinator warrant declaration of a UE
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

HU7.1
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.

E — ISFSI

General Emergency — None

Table E-1 ISFSI Dose Limits

NAC UMS	NAC MAGNASTOR
• 100 mrem/hr (neutron + gamma) on the side	• 240 mrem/hr gamma on the vertical surfaces
• 100 mrem/hr (neutron + gamma) on the top	• 10 mrem/hr neutron on the vertical surfaces
• 200 mrem/hr (neutron + gamma) at the air inlets or outlets	• 900 mrem/hr (neutron + gamma) on the top

Unusual Event — Damage to a loaded cask CONFINEMENT BOUNDARY
Modes: 1 | 2 | 3 | 4 | 5 | 6 | DEF

EU1.1
Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > **any** Table E-1 dose limit

Modes: 1 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refuel | DEF Defueled

DUKE ENERGY

Catawba Nuclear Station

CSD-EP-CNS-0101-02 Rev 000

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

S — System Malfunct.

1 Loss of Essential AC Power

General Emergency — Prolonged loss of **all** offsite and **all** onsite AC power to essential buses
Modes: 1 | 2 | 3 | 4

SG1.1
Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB
AND
SSF fails to supply NC pump seal injection
OR
CA supply to SGs
AND EITHER:
- Restoration of at least one essential bus in < 4 hours is **not** likely (Note 1)
- Core Cooling RED PATH conditions met

Site Area Emergency — Loss of **all** offsite and **all** onsite AC power to essential buses for 15 minutes or longer
Modes: 1 | 2 | 3 | 4

SS1.1
Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1)

Alert — Loss of **all but one** AC power source to essential buses for 15 minutes or longer
Modes: 1 | 2 | 3 | 4

SA1.1
AC power capability, Table S-1, to essential 4160V buses 1(2)ETA and 1(2)ETB reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

Unusual Event — Loss of **all** offsite AC power capability to essential buses for 15 minutes or longer
Modes: 1 | 2 | 3 | 4

SU1.1
Loss of **all** offsite AC power capability, Table S-1, to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min. (Note 1)

Table S-1 AC Power Sources

Offsite
- ATC (Train A)
- SATA (Train A) (if already aligned)
- ATD (Train B)
- SATB (Train B) (if already aligned)

Onsite
- D/G A (Train A)
- D/G B (Train B)

2 Loss of Vital DC Power

General Emergency — Loss of **all** essential AC and vital DC power sources for 15 minutes or longer
Modes: 1 | 2 | 3 | 4

SG1.2
Loss of **all** offsite and **all** onsite AC power capability to essential 4160V buses 1(2)ETA and 1(2)ETB for ≥ 15 min.
AND
Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses EDA, EDC, EDB and EDD for ≥ 15 min.
(Note 1)

Site Area Emergency — Loss of all vital DC power for 15 minutes or longer
Modes: 1 | 2 | 3 | 4

SS2.1
Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses EDA, EDC, EDB and EDD for ≥ 15 min. (Note 1)

Alert — None

Unusual Event — None

3 Loss of CR Indications

General Emergency — None

Site Area Emergency — None

Table S-3 Significant Transients
- Reactor trip
- Runback > 25% thermal power
- Electrical load rejection > 25% electrical load
- Safety injection actuation

Alert — UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress
Modes: 1 | 2 | 3 | 4

SA3.1
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)
AND
Any significant transient is in progress, Table S-3

Unusual Event — UNPLANNED loss of Control Room indications for 15 minutes or longer
Modes: 1 | 2 | 3 | 4

SU3.1
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

4 NCS Activity

General Emergency — None

Table S-2 Safety System Parameters
- Reactor power
- NCS level
- NCS pressure
- In-core T/C temperature
- Level in at least one S/G
- Auxiliary or emergency feed flow in at least one S/G

Unusual Event — NCS activity greater than Technical Specification allowable limits
Modes: 1 | 2 | 3 | 4

SU4.1
NCS activity > Technical Specification 3.4.16 limits or Facility Operating License limits (151/159), whichever is more restrictive

5 NCS Leakage

General Emergency — None

Site Area Emergency — None

Unusual Event — NCS leakage for 15 minutes or longer
Modes: 1 | 2 | 3 | 4

SU5.1
NCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.
OR
NCS identified leakage > 25 gpm for ≥ 15 min.
OR
Leakage from the NCS to a location outside containment > 25 gpm for ≥ 15 min.
(Note 1)

6 RPS Failure

General Emergency — None

Site Area Emergency — Inability to shut down the reactor causing a challenge to core cooling or NCS heat removal
Modes: 1

SS6.1
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
All actions to shut down the reactor are not successful as indicated by reactor power ≥ 5%
AND EITHER:
- Core Cooling RED PATH conditions met
- Heat Sink RED PATH conditions met

Alert — Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor
Modes: 1

SA6.1
An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%
AND
Manual trip actions taken at the reactor control console (manual reactor trip switches or turbine manual trip) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)

Unusual Event — Automatic or manual trip fails to shut down the reactor
Modes: 1

SU6.1
An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RPS setpoint is exceeded
AND
A subsequent automatic trip or manual trip action taken at the reactor control console (manual reactor trip switches or turbine manual trip) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

SU6.2
A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated
AND
A subsequent automatic trip or manual trip action taken at the reactor control console (manual reactor trip switches or turbine manual trip) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

NOTES

Note 1: The Emergency Coordinator should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies.

Note 9: In the absence of reliable NCS temperature indication caused by the loss of decay heat removal capability, classification should be based on time to boil data when in Mode 5 and 6.

Note 10: If the loss of containment cooling threshold is exceeded due to loss of both trains of VX-CARF, this EAL only applies if at least one train of VX-CARF is not operating, per design, after the 10 minute actuation delay for greater than or equal to 15 minutes.

Note 11: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 12: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

7 Loss of Comm

Table S-4 Communication Methods

System	Onsite	ORO	NRC
Public Address	X		
Internal Telephones	X		
Onsite Radios	X		
DEMNET		X	
Commercial Telephones		X	X
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telecommunications System (ETS)		X	X

Unusual Event — Loss of **all** onsite or offsite communications capabilities
Modes: 1 | 2 | 3 | 4

SU7.1
Loss of **all** Table S-4 onsite communication methods
OR
Loss of **all** Table S-4 ORO communication methods
OR
Loss of **all** Table S-4 NRC communication methods

8 CMT Failure

Unusual Event — Failure to isolate containment or loss of containment pressure control
Modes: 1 | 2 | 3 | 4

SU8.1
EITHER:
Any penetration is **not** isolated within 15 min. of a VALID containment isolation signal (Note 1)
OR
Containment pressure > 3 psig with < one full train of containment cooling operating per design for > 15 min. (Notes 1, 10)

9 Hazardous Event Affecting Safety Systems

Table S-5 Hazardous Events
- Seismic event (earthquake)
- Internal or external FLOODING event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Shift Manager

Alert — SA9 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode
Modes: 1 | 2 | 3 | 4

SA9.1
The occurrence of **any** Table S-5 hazardous event
AND
- Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER of the following:
- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode.
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode

(Note 11, 12)

CNS

Table F-2 Containment Radiation - R/hr (EMF53A/B)

Time After S/D	FC Loss	NCS Loss	CMT Potential Loss
0 - 1	550	8.8	5500
1 - 2	400	8.4	4000
2 - 8	160	7.0	1600
>8	100	6.2	1000

F — Fission Product Barriers

FG1.1 Modes: 1 | 2 | 3 | 4
Loss of **any** two barriers
AND
Loss or potential loss of third barrier (Table F-1)

FS1.1 Modes: 1 | 2 | 3 | 4
Loss or potential loss of **any** two barriers (Table F-1)

FA1.1 Modes: 1 | 2 | 3 | 4
Any loss or **any** potential loss of either Fuel Clad or NCS (Table F-1)

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		Reactor Coolant System (NCS) Barrier		Containment (CMT) Barrier	
	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
A. NCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER**: • UNISOLABLE NCS leakage • SG tube RUPTURE	1. CSFST Integrity-RED PATH conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B. Inadequate Heat Removal	1. CSFST Core Cooling-RED PATH conditions met	1. CSFST Core Cooling-ORANGE PATH conditions met 2. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED PATH conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED PATH conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C. CMT Radiation / NCS Activity	1. EMF53A/B > Table F-2 column "FC Loss" 2. Dose equivalent I-131 coolant activity > 300 µCi/gm	None	1. EMF53A/B > Table F-2 column "NCS Loss"	None	None	1. EMF53A/B > Table F-2 column "CMT Potential Loss"
D. CMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of NCS leakage outside of containment	1. CSFST Containment-RED PATH conditions met 2. Containment hydrogen concentration > 6% 3. Containment pressure > 3 psig with < one full train of containment cooling operating per design for > 15 min. (Notes 1, 10)
E. EC Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the NCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

EAL Identifier

XXX.X

- Category (R, H, E, C, S, F)
- Emergency classification (G, S, A, U)
- Subcategory number (1 if no subcategory)
- Sequential number within subcategory/classification

MODES 1, 2, 3 & 4