– WRITTEN EXAM SAMPLE PLAN ONLY –

Facility: V.C. SUMMER (UNIT 1) Date of Examination: JUNE 2018

Item	Task Description	Initials a	b*	c**
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model in accordance with ES-401 or ES-401N.	MM	N/A	BAC
	b. Assess whether the outline was systematically and randomly prepared in accordance with Section D.1 of ES-401 or ES-401N and whether all K/A categories are appropriately sampled.	MM	N/A	BAC
	c. Assess whether the outline overemphasizes any systems, evolutions, or generic topics.	MM	N/A	BAC
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	MM	N/A	BAC
2. SIMULATOR	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, technical specifications, and major transients.	N/A		
	b. Assess whether there are enough scenario sets (and spares) to test the projected number and mix of applicants in accordance with the expected crew composition and rotation schedule without compromising exam integrity, and ensure that each applicant can be tested using at least one new or significantly modified scenario, that no scenarios are duplicated from the applicants' audit test(s), and that scenarios will not be repeated on subsequent days.			
	c. To the extent possible, assess whether the outline(s) conforms with the qualitative and quantitative criteria specified on Form ES-301-4 and described in Appendix D and in Section D.5, "Specific Instructions for the 'Simulator Operating Test,'" of ES-301 (including overlap).			
3. WALKTHROUGH	a. Verify that the systems walkthrough outline meets the criteria specified on Form ES-301-2: (1) The outline(s) contains the required number of control room and in-plant tasks distributed among the safety functions as specified on the form. (2) Task repetition from the last two NRC examinations is within the limits specified on the form. (3) No tasks are duplicated from the applicant's audit test(s). (4) The number of new or modified tasks meets or exceeds the minimums specified on the form. (5) The number of alternate-path, low-power, emergency, and radiologically controlled area tasks meets the criteria on the form.			
	b. Verify that the administrative outline meets the criteria specified on Form ES-301-1: (1) The tasks are distributed among the topics as specified on the form. (2) At least one task is new or significantly modified. (3) No more than one task is repeated from the last two NRC licensing examinations.			
	c. Determine whether there are enough different outlines to test the projected number and mix of applicants and ensure that no items are duplicated on subsequent days.			
4. GENERAL	a. Assess whether plant-specific priorities (including probabilistic risk assessment and individual plant examination insights) are covered in the appropriate exam sections.	MM	N/A	BAC
	b. Assess whether the 10 CFR 55.41, 55.43, and 55.45 sampling is appropriate.	MM	N/A	BAC
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	MM	N/A	BAC
	d. Check for duplication and overlap among exam sections and the last two NRC exams.	N/A	N/A	N/A
	e. Check the entire exam for balance of coverage.	MM	N/A	BAC
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	MM	N/A	BAC

	Printed Name/Signature	Date
a. Author	MICHAEL MEEKS / Michael Meeks	04/03/2017
b. Facility Reviewer (*)	N/A	N/A
c. NRC's Chief Examiner (#)	BRUNO CABALLERO / Bruno Caballero	5-2-17
d. NRC Supervisor	Eugene Guthrie / E Guthrie	5/2/17

* Not applicable for NRC-prepared examination outlines.
\# The independent NRC reviewer initials items in column "c"; the chief examiner's concurrence is required.

Facility: VC SUMMER Date of Exam: JUNE 2018

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18	3	3	6
	2	1	2	2				1	2			1	9	2	2	4
	Tier Totals	4	5	5				4	5			4	27	5	5	10
2. Plant Systems	1	3	2	2	2	2	3	3	2	3	3	3	28	3	2	5
	2	1	1	1	1	1	1	1	1	1	1	0	10	mf 1	2	3
	Tier Totals	4	3	3	3	3	4	4	3	4	4	3	38	4	4	8
3. Generic Knowledge and Abilities Categories		1		2		3		4					10	1 / 2	3 / 4	7
		3		2		2		3						2 / 2	1 / 2	

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

ES-401	PWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/SRO)	Form ES-401-2

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		R					(R)EK2.03		
000008 (APE 8) Pressurizer Vapor Space Accident / 3		R					(R) AK2.03		
000009 (EPE 9) Small Break LOCA / 3					R		(R) EA2.39		
000011 (EPE 11) Large Break LOCA / 3			R				(R) EK3.08		
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4						R	(R) AG2.1.31		
000022 (APE 22) Loss of Reactor Coolant Makeup / 2	R						(R) AK1.03		
000025 (APE 25) Loss of Residual Heat Removal System / 4				R			(R) AA1.01		
000026 (APE 26) Loss of Component Cooling Water / 8									
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3					R		(R) AA2.07		
000029 (EPE 29) Anticipated Transient Without Scram / 1									
000038 (EPE 38) Steam Generator Tube Rupture / 3					R S		(R) EA2.15 (S) EA2.08		
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4			R				(R) EK3.1		
000054 (APE 54; CE E06) Loss of Main Feedwater /4					S		(S) AA2.08		
000055 (EPE 55) Station Blackout / 6				R		S	(R) EA1.02 (S) EG2.2.44		
000056 (APE 56) Loss of Offsite Power / 6									
000057 (APE 57) Loss of Vital AC Instrument Bus / 6						R	(R) AG2.4.6		
000058 (APE 58) Loss of DC Power / 6					S	R	(R) AG2.1.20 (S) AA2.02		
000062 (APE 62) Loss of Nuclear Service Water / 4			R				(R) AK3.04		
000065 (APE 65) Loss of Instrument Air / 8									
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6				R		S	(R) AA1.05 (S) AG2.2.25		
(W E04) LOCA Outside Containment / 3		R					(R) EK2.2		
(W E11) Loss of Emergency Coolant Recirculation / 4	R						(R) EK1.2		
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4	R					S	(R) EK1.3 (S) EG2.4.2		
K/A Category Totals: RO	3	3	3	3	3	3	Group Point Total:		18/6
SRO					3	3			

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
007EK2.03	Reactor Trip - Stabilization - Recovery / 1	3.5	3.6	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Reactor trip status panel
008AK2.03	Pressurizer Vapor Space Accident / 3	2.5	2.4	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Controllers and positioners
009EA2.39	Small Break LOCA / 3	4.3	4.7	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Adequate core cooling
011EK3.08	Large Break LOCA / 3	3.9	4.1	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Flowpath for sump recirculation
015AG2.1.31	RCP Malfunctions / 4	4.6	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup.
022AK1.03	Loss of Rx Coolant Makeup / 2	3	3.4	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Relationship between charging flow and PZR level
025AA1.01	Loss of RHR System / 4	3.6	3.7	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	RCS/RHRS cooldown rate
027AA2.07	Pressurizer Pressure Control System Malfunction / 3	3.1	3.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Makeup flow indication
038EA2.15	Steam Gen. Tube Rupture / 3	4.2	4.4	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Pressure at which to maintain RCS during S/G cooldown
055EA1.02	Station Blackout / 6	4.3	4.4	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Manual ED/G start
057AG2.4.6	Loss of Vital AC Inst. Bus / 6	3.7	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge symptom based EOP mitigation strategies.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
058AG2.1.20	Loss of DC Power / 6	4.6	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to execute procedure steps.
062AK3.04	Loss of Nuclear Svc Water / 4	3.5	3.7	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Effect on the nuclear service water discharge flow header of a loss of CCW
077AA1.05	Generator Voltage and Electric Grid Disturbances / 6	3.9	4.0	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Engineered Safety Features
WE04EK2.2	LOCA Outside Containment / 3	3.8	4.0	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.
WE05EK1.3	Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	3.9	4.1	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of Secondary Heat Sink).
WE11EK1.2	Loss of Emergency Coolant Recirc. / 4	3.6	4.1	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Normal, abnormal and emergency operating procedures associated with (Loss of Emergency Coolant Recir).
WE12EK3.1	Steam Line Rupture - Excessive Heat Transfer / 4	3.5	3.9	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure and reactivity changes and operating limitations and reasons for these operating characteristics.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
038EA2.08	Steam Gen. Tube Rupture / 3	3.8	4.4	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Viable alternatives for placing plant in safe condition when condenser is not available
054AA2.08	Loss of Main Feedwater / 4	2.9	3.3	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Steam flow-feed trend recorder
055EG2.2.44	Station Blackout / 6	4.2	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions
058AA2.02	Loss of DC Power / 6	3.3	3.6	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	125V dc bus voltage, low/critical low, alarm
077AG2.2.25	Generator Voltage and Electric Grid Disturbances / 6	3.2	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
we05EG2.4.2	Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	4.5	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/SRO)		Form ES-401-2
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1				R			(R) AA1.02		
000003 (APE 3) Dropped Control Rod / 1									
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7									
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8			R				(R) AK3.03		
000037 (APE 37) Steam Generator Tube Leak / 3					R		(R) AA2.12		
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9									
000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68; BW A06) Control Room Evacuation / 8					R		(R) AA2.02		
000069 (APE 69; W E14) Loss of Containment Integrity / 5					S		(S) EA2.2		
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4						R	(R) EG2.4.21		
000076 (APE 76) High Reactor Coolant Activity / 9					S		(S) AA2.05		
000078 (APE 78*) RCS Leak / 3									
(W E01 & E02) Rediagnosis & SI Termination / 3									
(W E13) Steam Generator Overpressure / 4									
(W E15) Containment Flooding / 5		R					(R) EK2.1		
(W E16) High Containment Radiation /9	R						(R) EK1.3		
(BW A01) Plant Runback / 1									
(BW A02 & A03) Loss of NNI-X/Y/7									
(BW A04) Turbine Trip / 4									
(BW A05) Emergency Diesel Actuation / 6									
(BW A07) Flooding / 8									
(BW E03) Inadequate Subcooling Margin / 4									
(BW E08; W E03) LOCA Cooldown—Depressurization / 4		R				S	(R) EK2.2 (S) EG2.1.20		
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4						S	(S) EG2.1.30		
(BW E13 & E14) EOP Rules and Enclosures									
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4			R				(R) EK3.2		
(CE A16) Excess RCS Leakage / 2									
(CE E09) Functional Recovery									
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
K/A Category Point Totals: RO	1	2	2	1	2	1	Group Point Total:		9/4
SRO					2	2			

RO

T1G2 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
001AA1.02	Continuous Rod Withdrawal / 1	3.6	3.4	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Rod in-out-hold switch
036AK3.03	Fuel Handling Accident / 8	3.7	4.1	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Guidance contained in EOP for fuel handling incident
037AA2.12	Steam Generator Tube Leak / 3	3.3	4.1	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Flow rate of leak
068AA2.02	Control Room Evac. / 8	3.7	4.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Local boric acid flow
WE03EK2.2	LOCA Cooldown - Depress. / 4	3.7	4.0	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.
we06EG2.4.21	Degraded Core Cooling / 4	4.0	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the parameters and logic used to assess the status of safety functions
WE08EK3.2	RCS Overcooling - PTS / 4	3.6	4.0	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Normal, abnormal and emergency operating procedures associated with (Pressurized Thermal Shock).
WE15EK2.1	Containment Flooding / 5	2.8	2.9	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.
WE16EK1.3	High Containment Radiation / 9	3.0	3.3	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	Annunciators and conditions indicating signals, and remedial actions associated with the (High Containment Radiation).

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
076AA2.05	High Reactor Coolant Activity / 9	2.2	2.5	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	CVCS letdown flow rate indication
we03EG2.1.20	LOCA Cooldown - Depress. / 4	4.6	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to execute procedure steps.
we09EG2.1.30	Natural Circ. / 4	4.4	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to locate and operate components, including local controls.
WE14EA2.2	Loss of CTMT Integrity / 5	3.3	3.8	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

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				R								(R) K4.02		
004 (SF1; SF2 CVCS) Chemical and Volume Control										R	S	(R) A4.12 (S) G2.4.47		
005 (SF4P RHR) Residual Heat Removal								R S				(R) A2.02 (S) A2.04		
006 (SF2; SF3 ECCS) Emergency Core Cooling						R						(R) K6.13		
007 (SF5 PRTS) Pressurizer Relief/Quench Tank	R							R				(R) A2.03 (R) K1.01		
008 (SF8 CCW) Component Cooling Water	R										S	(R) K1.01 (S) G2.1.23		
010 (SF3 PZR PCS) Pressurizer Pressure Control						R						(R) K6.01		
012 (SF7 RPS) Reactor Protection					R	R						(R) K5.01 (R) K6.03		
013 (SF2 ESFAS) Engineered Safety Features Actuation				R							R	(R) G2.4.31 (R) K4.11		
022 (SF5 CCS) Containment Cooling											R	(R) G2.2.22		
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray							R	X	R			(R) A1.04 (R) A3.01		
039 (SF4S MSS) Main and Reheat Steam			R						R			(R) A3.02 (R) K3.05		
059 (SF4S MFW) Main Feedwater	R											(R) K1.03		
061 (SF4S AFW) Auxiliary/Emergency Feedwater					R							(R) K5.05		
062 (SF6 ED AC) AC Electrical Distribution										R		(R) A4.04		
063 (SF6 ED DC) DC Electrical Distribution		R								R		(R) A4.01 (R) K2.01		
064 (SF6 EDG) Emergency Diesel Generator		R						S				(R) K2.03 (S) A2.02		
073 (SF7 PRM) Process Radiation Monitoring								S			R	(R) G2.2.44 (S) A2.02		
076 (SF4S SW) Service Water							R					(R) A1.02		
078 (SF8 IAS) Instrument Air									R			(R) A3.01		
103 (SF5 CNT) Containment			R				R					(R) A1.01 (R) K3.03		
053 (SF1; SF4P ICS*) Integrated Control												(NUREG-1122, Rev. 3 only)		
K/A Category Point Totals: RO	3	2	2	2	2	3	3	2	3	3	3	Group Point Total:		28/5
SRO								3			2			

RO

T2G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
003K4.02	Reactor Coolant Pump	2.5	2.7				X								Prevention of cold water accidents or transients
004A4.12	Chemical and Volume Control	3.8	3.3										X		Boration/dilution batch control
005A2.02	Residual Heat Removal	3.5	3.7								X				Pressure transient protection during cold shutdown
006K6.13	Emergency Core Cooling	2.8	3.1						X						Pumps
007A2.03	Pressurizer Relief/Quench Tank	3.6	3.9								X				Overpressurization of the PZR
007K1.01	Pressurizer Relief/Quench Tank	2.9	3.1	X											Containment system
008K1.01	Component Cooling Water	3.1	3.1	X											SWS
010K6.01	Pressurizer Pressure Control	2.7	3.1						X						Pressure detection systems
012K5.01	Reactor Protection	3.3	3.8					X							DNB
012K6.03	Reactor Protection	3.1	3.5						X						Trip logic circuits
013G2.4.31	Engineered Safety Features Actuation	4.2	4.1											X	Knowledge of annunciators alarms, indications or response procedures

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
013K4.11	Engineered Safety Features Actuation	3.2	3.8				✓								Vital power load control
022G2.2.22	Containment Cooling	4.0	4.7											✓	Knowledge of limiting conditions for operations and safety limits.
026A1.04	Containment Spray	3.1	3.3							✓					Containment humidity
026A3.01	Containment Spray	4.3	4.5									✓			Pump starts and correct MOV positioning
039A3.02	Main and Reheat Steam	3.1	3.5									✓			Isolation of the MRSS
039K3.05	Main and Reheat Steam	3.6	3.7			✓									RCS
059K1.03	Main Feedwater	3.1	3.3	✓											S/GS
061K5.05	Auxiliary/Emergency Feedwater	2.7	3.2					✓							Feed line voiding and water hammer
062A4.04	AC Electrical Distribution	2.6	2.7										✓		Local operation of breakers
063A4.01	DC Electrical Distribution	2.8	3.1										✓		Major breakers and control power fuses
063K2.01	DC Electrical Distribution	2.9	3.1		✓										Major DC loads

RO

T2G1 PWR EXAMINATION OUTLINE

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
064K2.03	Emergency Diesel Generator	3.2	3.6	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	Control power
073G2.2.44	Process Radiation Monitoring	4.2	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions
076A1.02	Service Water	2.6	2.6	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Reactor and turbine building closed cooling water temperatures.
078A3.01	Instrument Air	3.1	3.2	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Air pressure
103A1.01	Containment	3.7	4.1	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	Containment pressure, temperature and humidity
103K3.03	Containment	3.7	4.1	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Loss of containment integrity under refueling operations.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
004G2.4.47	Chemical and Volume Control	4.2	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.
005A2.04	Residual Heat Removal	2.9	2.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	RHR valve malfunction
008G2.1.23	Component Cooling Water	4.3	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to perform specific system and integrated plant procedures during all modes of plant operation.
064A2.02	Emergency Diesel Generator	2.7	2.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Load, VARS, pressure on air compressor, speed droop, frequency, voltage, fuel oil level temperatures
073A2.02	Process Radiation Monitoring	2.7	3.2	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Detector failure

ES-401												PWR Examination Outline Plant Systems—Tier 2/Group 2 (RO/SRO)		Form ES-401-2
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive	R											(R) K1.05		
002 (SF2; SF4P RCS) Reactor Coolant					R							(R) K5.14		
011 (SF2 PZR LCS) Pressurizer Level Control											S	(S) G2.2.44		
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation							R					(R) A1.02		
016 (SF7 NNI) Nonnuclear Instrumentation									R			(R) A3.02		
017 (SF7 ITM) In-Core Temperature Monitor														
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control														
029 (SF8 CPS) Containment Purge														
033 (SF8 SFPCS) Spent Fuel Pool Cooling														
034 (SF8 FHS) Fuel-Handling Equipment										R		(R) A4.02		
035 (SF 4P SG) Steam Generator						R						(R) K6.01		
041 (SF4S SDS) Steam Dump/Turbine Bypass Control		R										(R) K2.02		
045 (SF 4S MTG) Main Turbine Generator								R				(R) A2.17		
055 (SF4S CARS) Condenser Air Removal			R									(R) K3.01		
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste				R								(R) K4.01		
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring											S	(S) G2.4.18		
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air														
086 Fire Protection								S				(S) A2.02		
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals: RO	1	1	1	1	1	1	1	1	1	1	0	Group Point Total:		10/3
SRO								1			2			

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
001K1.05	Control Rod Drive	4.5	4.4	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	NIS and RPS
002K5.14	Reactor Coolant	3.8	4.2	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	Consequences of forced circulation loss
015A1.02	Nuclear Instrumentation	3.5	3.6	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	SUR
016A3.02	Non-nuclear Instrumentation	2.9	2.9	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	Relationship between meter readings and actual parameter value
034A4.02	Fuel Handling Equipment	3.5	3.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	☐	Neutron levels
035K6.01	Steam Generator	3.2	3.6	☐	☐	☐	☐	☐	☑	☐	☐	☐	☐	☐	MSIVs
041K2.02	Steam Dump/Turbine Bypass Control	2.8	2.8	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	☐	ICS inverter breakers
045A2.17	Main Turbine Generator	2.7	2.9	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Malfunction of electrohydraulic control
055K3.01	Condenser Air Removal	2.5	2.7	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	☐	Main condenser
068K4.01	Liquid Radwaste	3.4	4.1	☐	☐	☐	☑	☐	☐	☐	☐	☐	☐	☐	Safety and environmental precautions for handling hot, acidic and radioactive liquids

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
011G2.2.44	Pressurizer Level Control	4.2	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions
072G2.4.18	Area Radiation Monitoring	3.3	4.0	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the specific bases for EOPs.
086A2.02	Fire Protection	3.0	3.3	☐	☐	☐	☐	☐	☐	☐	☑	☐	☐	☐	Low FPS header pressure

Facility: VC Summer Unit 1 Date of Exam: June 2018

Category	K/A #	Topic	RO IR	RO #	SRO-only IR	SRO-only #
1. Conduct of Operations	2.1.1	Knowledge of Conduct of Ops Req'ts	3.8			
	2.1.2	" " operator responsibilities.. all modes	4.1			
	2.1.43	Ability to use procedures... effects on reactivity	4.1			
	2.1.15	" " admin req'ts for temp mgmt...			3.4	
	2.1.40	" " refueling admin req'ts			3.9	
	2.1.					
	Subtotal		3		2	
2. Equipment Control	2.2.20	" " process for managing troubleshooting.	3.8			
	2.2.43	" " " to track inop alarms	3.0			
	2.2.23	Ability to track Tech Spec LCOs			4.6	
	2.2.5	" " " for making design/operating Δ's			3.2	
	2.2.					
	2.2.					
	Subtotal		2		2	
3. Radiation Control	2.3.5	Ability to use radiation monitoring systems	2.9			
	2.3.7	" " comply w/ RWP req'ts.. normal/abnormal	3.5			
	2.3.4	Knowledge of rad exposure limits normal/emerg			3.7	
	2.3.					
	2.3.					
	2.3.					
	Subtotal		2		1	
4. Emergency Procedures/Plan	2.4.1	Knowledge of EOP entry conditions & immed action	4.6			
	2.4.43	" " emerg comm systems & techniques	3.2			
	2.4.6	" " symptom based EOP mitigation strat	3.7			
	2.4.26	" " facility protection req'ts.. fire brigade.			3.6	
	2.4.28	" " procedures relating to - sabotage			4.1	
	2.4.					
	Subtotal		3		2	
Tier 3 Point Total			10	10	7	7

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
G2.1.1	Conduct of operations	3.8	4.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of conduct of operations requirements.
G2.1.2	Conduct of operations	4.1	4.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of operator responsibilities during all modes of plant operation.
G2.1.43	Conduct of operations	4.1	4.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to use procedures to determine the effects on reactivity of plant changes
G2.2.20	Equipment Control	2.6	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process for managing troubleshooting activities.
G2.2.43	Equipment Control	3.0	3.3	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process used to track inoperable alarms
G2.3.5	Radiation Control	2.9	2.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to use radiation monitoring systems
G2.3.7	Radiation Control	3.5	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to comply with radiation work permit requirements during normal or abnormal conditions
G2.4.1	Emergency Procedures/Plans	4.6	4.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of EOP entry conditions and immediate action steps.
G2.4.43	Emergency Procedures/Plans	3.2	3.8	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of emergency communications systems and techniques.
G2.4.6	Emergency Procedures/Plans	3.7	4.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge symptom based EOP mitigation strategies.

KA	NAME / SAFETY FUNCTION:	IR RO	IR SRO	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC:
G2.1.15	Conduct of operations	2.7	3.4	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of administrative requirements for temporary management directives such as standing orders, night orders, Operations memos, etc.
G2.1.40	Conduct of operations	2.8	3.9	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of refueling administrative requirements
G2.2.23	Equipment Control	3.1	4.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Ability to track Technical Specification limiting conditions for operations.
G2.2.5	Equipment Control	2.2	3.2	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of the process for making design or operating changes to the facility
G2.3.4	Radiation Control	3.2	3.7	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of radiation exposure limits under normal and emergency conditions
G2.4.26	Emergency Procedures/Plans	3.1	3.6	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage.
G2.4.28	Emergency Procedures/Plans	3.2	4.1	☐	☐	☐	☐	☐	☐	☐	☐	☐	☐	☑	Knowledge of procedures relating to emergency response to sabotage.

Facility: V.C. Summer Unit 1 Date of Examination: 6/4/2018

Examination Level: RO ☒ SRO ☐ Operating Test Number: NRC-ILO-16-01

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations (A1-a)	N R	JPA-102-(R)N18 2018 NRC A1-a (RO) Verification of Operator Watchstanding Certification.
Conduct of Operations (A1-b)	M R	JPA-081E-(R)N18 2018 NRC A1-a (RO) Operational Leak Rate Test without IPCS available .
Equipment Control (A2)	N R	JPA-220-(R)N18 2018 NRC A2 (RO) Determine allowed use of valve extension devices and required documentation.
Radiation Control (A3)	N R	JPA-815-(R)N18 2018 NRC A3 (RO) Locate a component on a survey map and calculate worker stay times.
Emergency Plan (A4)		Not selected for RO.

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

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

JPM SUMMARY STATEMENTS

CONDUCT OF OPERATIONS (A1-a): This is a NEW JPM. The candidate will be given a watchstanding history which will include non-licensed operator and partial licensed watches and will be required to determine the minimum number of hours, and the latest date by which they must be stood, to maintain an active license.

K/A:	2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10cfr55, etc. (RO 3.3)

CONDUCT OF OPERATIONS (A1-b): This is a MODIFICATION of an existing JPM. The candidate will be given plant data and will be required to use a procedure to calculate an RCS leakrate.

K/A:	2.1.20	Ability to interpret and execute procedure steps. (RO 4.6)
VCS Task:	O-002-001-02-01	PERFORM REACTOR COOLANT SYSTEM WATER INVENTORY BALANCE

EQUIPMENT CONTROL (A2): This is a NEW JPM. The candidate will be given a scenario in which a safety-related valve must be operated locally using a valve extension. The candidate must determine the operability status of the valve as result of this operation and the tracking mechanism for recording valve status.

K/A:	2.2.37	Ability to determine operability and/or availability of safety related equipment. (RO 3.6)
VCS Task:	O-119-015-03-01	APPLY TECH SPEC REQUIREMENTS.

RADIATION CONTROL (A3): This is a NEW JPM. The candidate will be given survey maps, a valve location where work will occur, and a dose history for two operators. The candidate will locate the component on the survey map determine which operator(s) can complete work tasks.

K/A:	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions. (RO 3.5)
VCS Task:	O-119-008-04-04	APPLY RADIATION AND CONTAMINATION SAFETY PROCEDURES

EMERGENCY PLAN (A4): Not selected for RO.

Facility: V.C. Summer Unit 1	Date of Examination: 6/4/2018
Examination Level: RO ☐ SRO ☒	Operating Test Number: NRC-ILO-16-01

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations (A1-a)	M R	JPA-066D-(S)N18 2018 A1-a (SRO) Review NIS Power Range Heat Balance
Conduct of Operations (A1-b)	N R	JPA-135-(S)N18 2018 A1-a (SRO) Determine reportability requirements.
Equipment Control (A2)	M R	JPA-040A-(S)N18 2018 A2 (SRO) Review a manual tagout of “A” CCW pump
Radiation Control (A3)	P D R	JPA-190-(S)N18 2018 A3 (SRO) Review a Release Permit prior to approval
Emergency Plan (A4)	N R	JPA-1008-(S)N18 2018 A4 (SRO) Determine Protective Action Recommendations

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

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

JPM SUMMARY STATEMENTS

CONDUCT OF OPERATIONS (A1-a): This is a MODIFICATION of an existing JPM. The candidate will be given a completed STP-102.002 Attachment III heat balance and be required to identify errors and determine if the acceptance criteria are met for the current state of the Power Range instrumentation.

K/A:	2.1.45	Ability to identify and interpret diverse indications to validate the response of another indication. (SRO 4.3)
VCS Task:	O-342-026-03-02	REVIEW RESULTS OF SURVEILLANCE TESTS (SAP-134, GTP-301, AND GTP-302)

CONDUCT OF OPERATIONS (A1-b): This is NEW JPM. The candidate will be given a scenario in which the plant is being shut down. The candidate will be required to reason that the shutdown is due to a Technical Specification action statement and determine the site reportability requirement in accordance with procedure.

K/A:	2.1.18	Ability to make accurate, clear, and concise logs, records, status boards, and reports. (SRO 3.8)
VCS Task:	O-341-013-03-02	REPORT SAFETY LIMIT VIOLATIONS AND REPORTABLE OCCURRENCES PER NL-122

EQUIPMENT CONTROL (A2): This is a MODIFICATION of an existing JPM. The candidate will be given a completed SAP-201, Attachment VIC tagout sheet and will be required to determine the adequacy of the boundary to perform work. The candidate will be required to determine if a mode change can be performed with this tagout in place.

K/A:	2.2.13	Knowledge of tagging and clearance procedures. (SRO 4.3)
VCS Task:	O-115-144-03-02	READ DRAWINGS TO PREPARE AND VERIFY TAGOUT BOUNDARIES

RADIATION CONTROL (A3): This JPM is a BANK JPM used on the 2017 NRC exam. The candidate will be given a partially completed SOP-119, ATTACHMENT VA release work sheet and HPP-709, ATTACHMENT I, Gaseous Waste Release Permit and will determine the conditions that prevent commencing the release.

K/A:	2.3.11	Ability to control radiation releases. (SRO 4.3)
VCS Task:	O-341-012-03-02	APPROVE RADIOACTIVE WASTE DISCHARGE/RELEASE PERMITS (HPP-709 AND HPP-710)

EMERGENCY PLAN (A4): This is a NEW JPM. The candidate will be given conditions in a General Emergency that include a Hostile Action and will determine the required Protective Action Recommendations.

K/A:	2.4.38	Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required. (SRO 4.4)
VCS Task:	O-344-003-03-02	PERFORM REQUIRED NOTIFICATIONS OF ON-SITE AND OFF-SITE PERSONNEL FOR ABNORMAL EVENTS

Facility: V.C.Summer Unit 1 Date of Examination: 6/4/2018

Exam Level: RO ☒ SRO-I ☐ SRO-U ☐ Operating Test No: NRC-ILO-16-01

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

	System / JPM Title	Type Code*	Safety Function
a.	System 006: Emergency Core Cooling System (ECCS) (JPS-001F-N18) Manually actuate Safety Injection and perform EOP-1.0, E-0 Attachment 3 following a Small Break LOCA with failure of both Charging pumps to start.	N,A,EN,L,S	2
b.	APE027: Pressurizer Pressure Control System (PZR PCS) Malfunction (JPS-002F-N18) Stop RCPs per EOP-1.1, ES-0.1 to mitigate a stuck open PZR spray valve following a reactor trip.	M,A,L,S	3
c.	EPE074: Inadequate Core Cooling (JPS-003F-N18) Start RCPs in EOP-14.0, FR-C.1 due to core exit temperature greater than 1200°F. No RCP can be started and all PZR PORVs and Reactor Head vent valves are opened.	D,A,L,S	4P
d.	W/E 13: Steam Generator Overpressure (JPS-004-N18) Respond to Steam Generator Overpressure per EOP-15.1, FR-H.2. Must use SG PORVs since the condenser is not available.	D,L,P,S	4S
e.	System 062: A.C. Electrical Distribution (JPS-005-N18) Restore an ESF bus using transformer XTF-5052 and procedure SOP-304 as referenced by EOP-6.0, ECA-0.0, during a loss of all ESF AC power.	N,L,S	6
f.	System 015: Nuclear Instrumentation System (JPS-006-N18) Respond to power range Instrumentation channel N-44 fail low with rod control in automatic.	D,S	7
g.	System 008: Component Cooling Water System (JPS-007-N18) Swap active CCW loops during normal full power operations.	M,S	8
h	W/E 16: High Containment Radiation (JPS-008F-N18) Align RB HEPA filters using EOP-17.2 following a high containment radiation condition post LOCA.	D,EN,A,L,P,S	9

In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

i	System 001: Control Rod Drive System (JPP-0009F-N18) Locally Trip the Reactor using EOP-13.0, FR-S.1. The trip breakers don't open locally requiring MG set motor breakers to be opened locally.	D,A,E	1
j	APE040; Steam Line Rupture (JPP-010-N18) Locally de-energize and close MS Loop "B" and "C" supplies to the Turbine Driven Emergency Feedwater Pump per EOP-3.0, E-2.	D,E,L,P	4S
k	System 033: Spent Fuel Pool Cooling System (JPP-011-N18) Fill Spent Fuel Pool using Rx M/U water in accordance with SOP-123.	N,R	8

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

* Type Codes	Criteria for RO / SRO-I / SRO-U	RO
(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)	2
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1	7
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1	5
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)	3
(R)CA	≥ 1 / ≥ 1 / ≥ 1	1
(S)imulator		

VC SUMMER NRC RO JPM SUMMARY

a. This is a new JPM. This JPM starts with the plant having just experienced a Small Break LOCA and a failure of Safety Injection to actuate automatically. The Candidate will be told that immediate actions of EOP-1.0, E-0 Reactor Trip or Safety Injection have been completed through step 3 and they are to commence by performing immediate action 4. They will determine that Safety Injection is required and will manually actuate Safety Injection and then will perform EOP-1.0, Attachment 3, SI Equipment Verification. Train "A" Safety Injection will actuate and Train B Safety Injection will not actuate. The Train "A" Charging/SI pump will experience a sheared shaft while the Train B Charging/SI pump will not start due to the Train "B" SI fail to actuate. The Candidate must manually align Train "B" equipment and it will be critical for them to start the Train "B" Charging/SI pump. The JPM is considered alternate path due to the failure of Train "B" equipment to start in response to the Safety Injection signal which will require the operator to take manual action.

K/A000006A2.12: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions requiring actuation of ECCS. (RO 4.5 SRO 4.8)

NUREG 1122 System: 006, Emergency Core Cooling System (ECCS)

VCS Task: O-006-013-01-01, MANUALLY INITIATE SAFETY INJECTION

b. This is a modified bank JPM. The JPM begins with the plant having tripped from 100% power. The Candidate will be told that the crew have transitioned to EOP-1.1, ES-0.1 Reactor Trip Response and have completed steps 1 through 7. The initiating cue to the Candidate will be to perform EOP-1.1 beginning at step 8, verify all Control Rods are fully inserted. Once the Candidate commences the performance of EOP-1.1 PZR spray valve PCV-444C will fail partially open causing PZR pressure to slowly lower. The Candidate will recognize PZR pressure is not stable or trending to 2230 psig at step 11 and will take alternative action beginning with a check of PZR PORVs closed and a check that PZR spray valves are closed. Attempts to close Spray valve PCV-444C will be unsuccessful and the Candidate will stop the “A” RCP and then will stop the “C” RCP. The JPM is considered alternate path due to the failure of the Pressurizer Spray valves to close when the operator attempts to close them which requires the stopping of RCPs.

K/A 027AA1.01: Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs. (RO 4.0 SRO 3.9)

NUREG 1122 APE: 027, Pressurizer Pressure Control System (PZR PCS) Malfunction.

VCS Task: O-000-509-05-02, RECOVER FROM REACTOR TRIP PER EOP-1.1

c. This is a bank JPM. Initial conditions for the JPM are a LOCA with a red path condition on the Core Cooling critical safety function and EOP-14.0, FR-C.1 RESPONSE TO INADEQUATE CORE COOLING in progress. The Candidate will receive the initiating cue to perform Step 22 of EOP-14.0, check if RCPs should be started. The Narrow Range level in the Steam Generators will only support start of the “A” RCP. The “A” RCP will fail to start and the Candidate must apply Alternative Actions and open all PZR PORVs and Reactor Head Vent valves. The JPM is considered alternate path because the available RCP fails to start which requires opening of RCS vents in order to stimulate some core cooling.

K/A 074EA1.05: Ability to operate and monitor the following as they apply to Inadequate Core Cooling: PORV. (RO 4.3 SRO 4.1)

NUREG 1122 EPE: 074, Inadequate Core Cooling.

VCS Task: O-000-088-05-01, RESPONSE TO INADEQUATE CORE COOLING PER SOP-122/EOP-12.0/EOP-2.0/EOP-14.0.

d. This is a revised bank JPM. Initial conditions for the JPM are reactor tripped from 100% power and a yellow path condition exists on the heat sink critical safety function. The Candidate will receive the initiating cue to implement EOP-15.1, FR-H.2 RESPONSE TO STEAM GENERATOR OVERPRESSURE. The candidate uses EOP-15.1 to address high pressure in the “B” SG. The feedwater flow isolation valves to the “B” SG failed to close automatically and will require the operator to close at least one valve to prevent potential SG overfill. The candidate will then proceed to depressurize the “B” Steam Generator using the SG PORV and will then control RCS temperature according to EOP-1.1, ES-0.1 REACTOR TRIP RESPONSE.

K/A W/E13EA1.1: Ability to operate and/or monitor the following as they apply to the (Steam Generator Overpressure): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (RO: 3.1, SRO: 3.3)

K/A W/E13EA2.1: Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (RO 2.9 SRO 3.4)

NUREG 1122 W/E: E13, Steam Generator Overpressure.

VCS Task: O-000-092-05-01, RESPOND TO STEAM GENERATOR OVERPRESSURE

PER EOP-15.1/EOP-12.0.

e. This is a new JPM. The JPM begins with both 7.2 KV ESF Safety Busses de-energized. The Train “B” bus is faulted and the Train “A” Diesel Generator failed to start either automatically or manually. A single offsite power source is available. The Candidate will be directed by the CRS to restore power to the Train “A” ESF bus using the 13.8 KV Parr Hydro line and XTF-5052. The Candidate will be directed to use SOP-304, 115KV/7.2KV OPERATIONS in accordance with the alternative action step 6.a of EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER. The candidate will align the Parr Hydro 13.8 KV line via XTF-5052 to energize the Train “A” ESF bus.

K/A000062A2.05: Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Methods for energizing a dead bus (RO 2.9 SRO 3.3)

NUREG 1122 System: 062, AC Electrical Distribution

VCS Task: O-000-055-05-01, RESPOND TO LOSS OF OFF SITE AND ON SITE POWER

f. This is a revision to a bank JPM. Initial conditions are 75% power all controls in automatic. The Candidate will be directed to respond to any developing plant conditions. Power Range NI channel N-44 will fail low causing auto rod withdrawal. The candidate will place rods in Manual and proceed to remove N-44 from service using AOP-401.10, POWER RANGE CHANNEL FAILURE.

K/A 000015A2.01: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation. (RO 3.5, SRO 3.9)

NUREG 1122 System: 015, Nuclear Instrumentation System.

VCS Task: O-000-034-05-01, RESPOND TO POWER RANGE INSTUMENTATION CHANNEL FAILURE

g. This is a modified bank JPM. The initial conditions for this JPM have the “C” CCW Pump aligned to Train “B” in preparation for swapping the active loop from Train “A” to Train “B” using SOP-118 COMPONENT COOLING WATER. The Candidate will place the “B” CCW Pump in service then perform valve manipulations to remove flow from the “B” RHR Heat Exchanger and the Train “A” non-essential header and to align flow to the Train “B” non-essential header The critical steps are Starting the “B” CCW pump, closing flow to the “B” RHR Heat exchanger and aligning the non-essential header to Train “B” only.

K/A 000008A4.01: Ability to manually operate and/or monitor in the control room: CCW indications and controls (RO: 3.3, SRO: 3.1)

NUREG 1122 System: 008, Component Cooling Water System (CCWS)

VCS Task: O-008-021-01-01 SWITCH COMPONENT COOLING WATER TRAINS PER SOP-117/SOP-118/SOP-501

h. This is a bank JPM that has been revised. The initial conditions for this JPM are a yellow path condition on the Containment Critical safety function due to high radiation levels following a DBA LOCA. The CRS will direct the Candidate to implement EOP-17.2, RESPONSE TO HIGH REACTOR BUILDING RADIATION LEVEL. The Candidate will place all RBCU HEPA filter dampers in the filter position but XDP-110A, RBCU 64A HEPA and XDP-111B, RBCU 65B HEPA Filter Bypass Dampers will remain in the bypassed position. The 64A RBCU fan will be running thus the Candidate will be required to start the 65A RBCU. In addition the 64B RBCU failed to auto start and will require the operator to stop the 64B Normal fan and manually start the 64B slow fan to assure adequate HEPA filtration in service. The JPM is considered alternate path because the HEPA filters fail to properly align which requires the operator to take manual action to align filtration.

K/A W/E16EA1.1: Ability to operate and/or monitor the following as they apply to the (High Containment Radiation): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (RO 3.1 SRO 3.2)

NUREG 1122 W/E: E16, High Containment Radiation.

VCS Task: O-311-018-06-01, RESPONSE TO HIGH RB RADIATION.

i. This is a bank JPM. The initial conditions for this JPM are an ATWS. The CRS will direct the Candidate to implement EOP-13.0, FR-S.1 RESPONSE TO ABNORMAL NUCLEAR POWER GENERATION, Attachment 1-TRIPPING THE REACTOR LOCALLY. The Candidate will attempt to locally open the Reactor Trip and Bypass breakers but they will not open. Next the Candidate will attempt to open Rod Drive MG set motor and generator breakers and they also will not open. Finally the Candidate will open the 480 volt supply breakers to both Rod Drive MG set motors. The JPM is considered alternate path because the action to locally open Reactor Trip breakers fails which requires the Candidate to open the MG set motor supply breakers.

K/A 029EA1.12: Knowledge of the reasons for the following responses as they apply to the ATWS: Actions contained in EOP for ATWS (RO 4.4 SRO 4.7)

K/A 029EA1.12: Ability to operate and monitor the following as they apply to a ATWS: M/G set power supply and reactor trip breakers (RO 4.1 SRO 4.0)

NUREG 1122 EPE: 029, Anticipated Transient Without Scram (ATWS)

VCS Task: O-000-117-05-04 RESPOND TO ABNORMAL NUCLEAR POWER GENERATION

j. This is a bank JPM. This JPM was randomly selected and was last used in the 2016 NRC exam. The Candidate will be directed to de-energize and locally close and the MS Loop "B" and "C" supplies to the turbine driven Emergency Feedwater pump in accordance with EOP-3.0, E-2 FAULTED STEAM GENERATOR ISOLATION.

K/A: 040AA1.10: Ability to operate and/or monitor the following as they apply to the Steam Line Rupture: AFW system (RO: 4.1, SRO: 4.1)

NUREG 1122 APE: 040, Steam Line Rupture.

VCS Task: O-000-169-05-04, LOCALLY ISOLATE A FAULTED STEAM GENERATOR PER EOP-3.0.

k. This is a new JPM. The initial conditions for the JPM are 100% power with Spent Fuel Pool level at 461.1 ft., which is below the Tech Spec 3.7.10 limit in accordance with OAP-106.1, OPERATING ROUNDS. The candidate will be directed to raise Spent Fuel Pool level to 461.8 ft using SOP-123, SPENT FUEL COOLING SYSTEM. The Candidate will simulate lining up the Reactor Makeup Water flowpath to the Spent Fuel Pool and will raise level to the desired value.

K/A: 000033A1.01: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including: Spent Fuel Pool Water Level (RO: 2.7, SRO: 3.3)

NUREG 1122 System: 033, Spent Fuel Pool Cooling System.

VCS Task: O-033-002-01-04, MAKEUP TO THE SPENT FUEL POOL PER SOP-123.

Facility: V.C.Summer Unit 1 Date of Examination: 6/4/2018

Exam Level: RO ☐ SRO-I ☒ SRO-U ☐ Operating Test No: NRC-ILO-16-01

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

	System / JPM Title	Type Code*	Safety Function
a.	System 006: Emergency Core Cooling System (ECCS) (JPS-001F-N18) Manually actuate Safety Injection and perform EOP-1.0, E-0 Attachment 3 following a Small Break LOCA with failure of both Charging pumps to start.	N,A,EN,L,S	2
b.	APE027: Pressurizer Pressure Control System (PZR PCS) Malfunction (JPS-002F-N18) Stop RCPs per EOP-1.1, ES-0.1 to mitigate a stuck open PZR spray valve following a reactor trip.	M,A,L,S	3
c.	EPE074: Inadequate Core Cooling (JPS-003F-N18) Start RCPs in EOP-14.0, FR-C.1 due to core exit temperature greater than 1200°F. No RCP can be started and all PZR PORVs and Reactor Head vent valves are opened.	D,A,L,S	4P
d.	W/E 13: Steam Generator Overpressure (JPS-004-N18) Respond to Steam Generator Overpressure per EOP-15.1, FR-H.2. Must use SG PORVs since the condenser is not available.	D,L,P,S	4S
e.	Not for SRO-I use.	NA	NA
f.	System 015: Nuclear Instrumentation System (JPS-006-N18) Respond to power range Instrumentation channel N-44 fail low with rod control in automatic.	D,S	7
g.	System 008: Component Cooling Water System (JPS-007-N18) Swap active CCW loops during normal full power operations.	M,S	8
h	W/E 16: High Containment Radiation (JPS-008F-N18) Align RB HEPA filters using EOP-17.2 following a high containment radiation condition post LOCA.	D,EN,A,L,P,S	9

In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

	System / JPM Title	Type Code*	Safety Function
i	System 001: Control Rod Drive System (JPP-0009F-N18) Locally Trip the Reactor using EOP-13.0, FR-S.1. The trip breakers don't open locally requiring MG set motor breakers to be opened locally.	D,A,E	1
j	APE040; Steam Line Rupture (JPP-010-N18) Locally de-energize and close MS Loop "B" and "C" supplies to the Turbine Driven Emergency Feedwater Pump per EOP-3.0, E-2.	D,E,L,P	4S
k	System 033: Spent Fuel Pool Cooling System (JPP-011-N18) Fill Spent Fuel Pool using Rx M/U water in accordance with SOP-123.	N,R	8

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

* Type Codes	Criteria for RO / SRO-I / SRO-U	SRO-I
(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)	2
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1	7
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1	4
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)	3
(R)CA	≥ 1 / ≥ 1 / ≥ 1	1
(S)imulator		

VC SUMMER NRC RO JPM SUMMARY

a. This is a new JPM. This JPM starts with the plant having just experienced a Small Break LOCA and a failure of Safety Injection to actuate automatically. The Candidate will be told that immediate actions of EOP-1.0, E-0 Reactor Trip or Safety Injection have been completed through step 3 and they are to commence by performing immediate action 4. They will determine that Safety Injection is required and will manually actuate Safety Injection and then will perform EOP-1.0, Attachment 3, SI Equipment Verification. Train “A” Safety Injection will actuate and Train B Safety Injection will not actuate. The Train “A” Charging/SI pump will experience a sheared shaft while the Train B Charging/SI pump will not start due to the Train “B” SI fail to actuate. The Candidate must manually align Train “B” equipment and it will be critical for them to start the Train “B” Charging/SI pump. The JPM is considered alternate path due to the failure of Train “B” equipment to start in response to the Safety Injection signal which will require the operator to take manual action.

K/A000006A2.12: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions requiring actuation of ECCS. (RO 4.5 SRO 4.8)

NUREG 1122 System: 006, Emergency Core Cooling System (ECCS)

VCS Task: O-006-013-01-01, MANUALLY INITIATE SAFETY INJECTION

b. This is a modified bank JPM. The JPM begins with the plant having tripped from 100% power. The Candidate will be told that the crew have transitioned to EOP-1.1, ES-0.1 Reactor Trip Response and have completed steps 1 through 7. The initiating cue to the Candidate will be to perform EOP-1.1 beginning at step 8, verify all Control Rods are fully inserted. Once the Candidate commences the performance of EOP-1.1 PZR spray valve PCV-444C will fail partially open causing PZR pressure to slowly lower. The Candidate will recognize PZR pressure is not stable or trending to 2230 psig at step 11 and will take alternative action beginning with a check of PZR PORVs closed and a check that PZR spray valves are closed. Attempts to close Spray valve PCV-444C will be unsuccessful and the Candidate will stop the “A” RCP and then will stop the “C” RCP. The JPM is considered alternate path due to the failure of the Pressurizer Spray valves to close when the operator attempts to close them which requires the stopping of RCPs.

K/A 027AA1.01: Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs. (RO 4.0 SRO 3.9)

NUREG 1122 APE: 027, Pressurizer Pressure Control System (PZR PCS) Malfunction.

VCS Task: O-000-509-05-02, RECOVER FROM REACTOR TRIP PER EOP-1.1

c. This is a bank JPM. Initial conditions for the JPM are a LOCA with a red path condition on the Core Cooling critical safety function and EOP-14.0, FR-C.1 RESPONSE TO INADEQUATE CORE COOLING in progress. The Candidate will receive the initiating cue to perform Step 22 of EOP-14.0, check if RCPs should be started. The Narrow Range level in the Steam Generators will only support start of the “A” RCP. The “A” RCP will fail to start and the Candidate must apply Alternative Actions and open all PZR PORVs and Reactor Head Vent valves. The JPM is considered alternate path because the available RCP fails to start which requires opening of RCS vents in order to stimulate some core cooling.

K/A 074EA1.05: Ability to operate and monitor the following as they apply to Inadequate Core Cooling: PORV. (RO 4.3 SRO 4.1)

NUREG 1122 EPE: 074, Inadequate Core Cooling.

VCS Task: O-000-088-05-01, RESPONSE TO INADEQUATE CORE COOLING PER SOP-122/EOP-12.0/EOP-2.0/EOP-14.0.

d. This is a revised bank JPM. Initial conditions for the JPM are reactor tripped from 100% power and a yellow path condition exists on the heat sink critical safety function. The Candidate will receive the initiating cue to implement EOP-15.1, FR-H.2 RESPONSE TO STEAM GENERATOR OVERPRESSURE. The candidate uses EOP-15.1 to address high pressure in the “B” SG. The feedwater flow isolation valves to the “B” SG failed to close automatically and will require the operator to close at least one valve to prevent potential SG overfill. The candidate will then proceed to depressurize the “B” Steam Generator using the SG PORV and will then control RCS temperature according to EOP-1.1, ES-0.1 REACTOR TRIP RESPONSE.

K/A W/E13EA1.1: Ability to operate and/or monitor the following as they apply to the (Steam Generator Overpressure): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (RO: 3.1, SRO: 3.3)

K/A W/E13EA2.1: Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (RO 2.9 SRO 3.4)

NUREG 1122 W/E: E13, Steam Generator Overpressure.

VCS Task: O-000-092-05-01, RESPOND TO STEAM GENERATOR OVERPRESSURE PER EOP-15.1/EOP-12.0.

e. Not for SRO-I use.

f. This is a revision to a bank JPM. Initial conditions are 75% power all controls in automatic. The Candidate will be directed to respond to any developing plant conditions. Power Range NI channel N-44 will fail low causing auto rod withdrawal. The candidate will place rods in Manual and proceed to remove N-44 from service using AOP-401.10, POWER RANGE CHANNEL FAILURE.

K/A 000015A2.01: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation. (RO 3.5, SRO 3.9)

NUREG 1122 System: 015, Nuclear Instrumentation System.

VCS Task: O-000-034-05-01, RESPOND TO POWER RANGE INSTUMENTATION CHANNEL FAILURE

g. This is a modified bank JPM. The initial conditions for this JPM have the “C” CCW Pump aligned to Train “B” in preparation for swapping the active loop from Train “A” to Train “B” using SOP-118 COMPONENT COOLING WATER. The Candidate will place the “B” CCW Pump in service then perform valve manipulations to remove flow from the “B” RHR Heat Exchanger and the Train “A” non-essential header and to align flow to the Train “B” non-essential header The critical steps are Starting the “B” CCW pump, closing flow to the “B” RHR Heat exchanger and aligning the non-essential header to Train “B” only.

K/A 000008A4.01: Ability to manually operate and/or monitor in the control room: CCW indications and controls (RO: 3.3, SRO: 3.1)

NUREG 1122 System: 008, Component Cooling Water System (CCWS)

VCS Task: O-008-021-01-01 Switch Component Cooling Water Trains per SOP-117/SOP-118/SOP-501

h. This is a bank JPM that has been revised. The initial conditions for this JPM are a yellow path condition on the Containment Critical safety function due to high radiation levels following a DBA LOCA. The CRS will direct the Candidate to implement EOP-17.2, RESPONSE TO HIGH REACTOR BUILDING RADIATION LEVEL. The Candidate will place all RBCU HEPA filter dampers in the filter position but XDP-110A, RBCU 64A HEPA and XDP-111B, RBCU 65B HEPA Filter Bypass Dampers will remain in the bypassed position. The 64A RBCU fan will be running thus the Candidate will be required to start the 65A RBCU. In addition the 64B RBCU failed to auto start and will require the operator to stop the 64B Normal fan and manually start the 64B slow fan to assure adequate HEPA filtration in service. The JPM is considered alternate path because the HEPA filters fail to properly align which requires the operator to take manual action to align filtration.

K/A W/E16EA1.1: Ability to operate and/or monitor the following as they apply to the (High Containment Radiation): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (RO 3.1 SRO 3.2)

NUREG 1122 W/E: E16, High Containment Radiation.

VCS Task: O-311-018-06-01, RESPONSE TO HIGH RB RADIATION.

i. This is a bank JPM. The initial conditions for this JPM are an ATWS. The CRS will direct the Candidate to implement EOP-13.0, FR-S.1 RESPONSE TO ABNORMAL NUCLEAR POWER GENERATION, Attachment 1-TRIPPING THE REACTOR LOCALLY. The Candidate will attempt to locally open the Reactor Trip and Bypass breakers but they will not open. Next the Candidate will attempt to open Rod Drive MG set motor and generator breakers and they also will not open. Finally the Candidate will open the 480 volt supply breakers to both Rod Drive MG set motors. The JPM is considered alternate path because the action to locally open Reactor Trip breakers fails which requires the Candidate to open the MG set motor supply breakers.

K/A 029EK3.12: Knowledge of the reasons for the following responses as they apply to the ATWS: Actions contained in EOP for ATWS (RO 4.4 SRO 4.7)

K/A 029EA1.12: Ability to operate and monitor the following as they apply to a ATWS: M/G set power supply and reactor trip breakers (RO 4.1 SRO 4.0)

NUREG 1122 EPE: 029, Anticipated Transient Without Scram (ATWS)

VCS Task: O-000-117-05-04 RESPOND TO ABNORMAL NUCLEAR POWER GENERATION

j. This is a bank JPM. This JPM was randomly selected and was last used in the 2016 NRC exam. The Candidate will be directed to de-energize and locally close and the MS Loop "B" and "C" supplies to the turbine driven Emergency Feedwater pump in accordance with EOP-3.0, E-2 FAULTED STEAM GENERATOR ISOLATION.

K/A: 040AA1.10: Ability to operate and/or monitor the following as they apply to the Steam Line Rupture: AFW system (RO: 4.1, SRO: 4.1)

NUREG 1122 APE: 040, Steam Line Rupture.

VCS Task: O-000-169-05-04, LOCALLY ISOLATE A FAULTED STEAM GENERATOR PER EOP-3.0.

k. This is a new JPM. The initial conditions for the JPM are 100% power with Spent Fuel Pool level at 461.1 ft., which is below the Tech Spec 3.7.10 limit in accordance with OAP-106.1, OPERATING ROUNDS. The candidate will be directed to raise Spent Fuel Pool level to 461.8 ft using SOP-123, SPENT FUEL COOLING SYSTEM. The Candidate will simulate lining up the Reactor Makeup Water flowpath to the Spent Fuel Pool and will raise level to the desired value.

K/A: 000033A1.01: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including: Spent Fuel Pool Water Level (RO: 2.7, SRO: 3.3)

NUREG 1122 System: 033, Spent Fuel Pool Cooling System.

VCS Task: O-033-002-01-04, MAKEUP TO THE SPENT FUEL POOL PER SOP-123.

Facility: V.C.Summer Unit 1 Date of Examination: 6/4/2018

Exam Level: RO ☐ SRO-I ☐ SRO-U ☒ Operating Test No: NRC-ILO-16-01

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

	System / JPM Title	Type Code*	Safety Function
a.	Not for SRO-U use.	NA	NA
b.	APE027: Pressurizer Pressure Control System (PZR PCS) Malfunction (JPS-002F-N18) Stop RCPs per EOP-1.1, ES-0.1 to mitigate a stuck open PZR spray valve following a reactor trip.	M,A,L,S	3
c.	Not for SRO-U use.	NA	NA
d.	Not for SRO-U use.	NA	NA
e.	Not for SRO-U use.	NA	NA
f.	System 015: Nuclear Instrumentation System (JPS-006-N18) Respond to power range Instrumentation channel N-44 fail low with rod control in automatic.	D,S	7
g.	Not for SRO-U use.	NA	NA
h	W/E 16: High Containment Radiation (JPS-008F-N18) Align RB HEPA filters using EOP-17.2 following a high containment radiation condition post LOCA.	D,EN,A,L,P,S	9

In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

	System / JPM Title	Type Code*	Safety Function
i	System 001: Control Rod Drive System (JPP-0009F-N18) Locally Trip the Reactor using EOP-13.0, FR-S.1. The trip breakers don't open locally requiring MG set motor breakers to be opened locally.	D,A,E	1
j	Not for SRO-U use.	NA	NA
k	System 033: Spent Fuel Pool Cooling System (JPP-011-N18) Fill Spent Fuel Pool using Rx M/U water in accordance with SOP-123.	N,R	8

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

* Type Codes	Criteria for RO / SRO-I / SRO-U	SRO-U
(A)lternate path	4-6 / 4-6 / 2-3	3
(C)ontrol room		
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4	3
(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	2
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1	2
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)	1
(R)CA	≥ 1 / ≥ 1 / ≥ 1	1
(S)imulator		

VC SUMMER NRC RO JPM SUMMARY

a. Not for SRO-U use.

b. This is a modified bank JPM. The JPM begins with the plant having tripped from 100% power. The Candidate will be told that the crew have transitioned to EOP-1.1, ES-0.1 Reactor Trip Response and have completed steps 1 through 7. The initiating cue to the Candidate will be to perform EOP-1.1 beginning at step 8, verify all Control Rods are fully inserted. Once the Candidate commences the performance of EOP-1.1 PZR spray valve PCV-444C will fail partially open causing PZR pressure to slowly lower. The Candidate will recognize PZR pressure is not stable or trending to 2230 psig at step 11 and will take alternative action beginning with a check of PZR PORVs closed and a check that PZR spray valves are closed. Attempts to close Spray valve PCV-444C will be unsuccessful and the Candidate will stop the “A” RCP and then will stop the “C” RCP. The JPM is considered alternate path due to the failure of the Pressurizer Spray valves to close when the operator attempts to close them which requires the stopping of RCPs.

K/A 027AA1.01: Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs. (RO 4.0 SRO 3.9)

NUREG 1122 APE: 027, Pressurizer Pressure Control System (PZR PCS) Malfunction.

VCS Task: O-000-509-05-02, RECOVER FROM REACTOR TRIP PER EOP-1.1

c. Not for SRO-U use.

d. Not for SRO-U use.

e. Not for SRO-U use.

f. This is a revision to a bank JPM. Initial conditions are 75% power all controls in automatic. The Candidate will be directed to respond to any developing plant conditions. Power Range NI channel N-44 will fail low causing auto rod withdrawal. The candidate will place rods in Manual and proceed to remove N-44 from service using AOP-401.10, POWER RANGE CHANNEL FAILURE.

K/A 000015A2.01: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation. (RO 3.5, SRO 3.9)

NUREG 1122 System: 015, Nuclear Instrumentation System.

VCS Task: O-000-034-05-01, RESPOND TO POWER RANGE INSTUMENTATION CHANNEL FAILURE

g. Not for SRO-U use.

h. This is a bank JPM that has been revised. The initial conditions for this JPM are a yellow path condition on the Containment Critical safety function due to high radiation levels following a DBA LOCA. The CRS will direct the Candidate to implement EOP-17.2, RESPONSE TO HIGH REACTOR BUILDING RADIATION LEVEL. The Candidate will place all RBCU HEPA filter dampers in the filter position but XDP-110A, RBCU 64A HEPA and XDP-111B, RBCU 65B HEPA Filter Bypass Dampers will remain in the bypassed position. The 64A RBCU fan will be running thus the Candidate will be required to start the 65A RBCU. In addition the 64B RBCU failed to auto start and will require the operator to stop the 64B Normal fan and manually start the 64B slow fan to assure adequate HEPA filtration in service. The JPM is considered alternate path because the HEPA filters fail to properly align which requires the operator to take manual action to align filtration.

K/A W/E16EA1.1: Ability to operate and/or monitor the following as they apply to the (High Containment Radiation): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (RO 3.1 SRO 3.2)

NUREG 1122 W/E: E16, High Containment Radiation.

VCS Task: O-311-018-06-01, RESPONSE TO HIGH RB RADIATION.

i. This is a bank JPM. The initial conditions for this JPM are an ATWS. The CRS will direct the Candidate to implement EOP-13.0, FR-S.1 RESPONSE TO ABNORMAL NUCLEAR POWER GENERATION, Attachment 1-TRIPPING THE REACTOR LOCALLY. The Candidate will attempt to locally open the Reactor Trip and Bypass breakers but they will not open. Next the Candidate will attempt to open Rod Drive MG set motor and generator breakers and they also will not open. Finally the Candidate will open the 480 volt supply breakers to both Rod Drive MG set motors. The JPM is considered alternate path because the action to locally open Reactor Trip breakers fails which requires the Candidate to open the MG set motor supply breakers.

K/A 029EK3.12: Knowledge of the reasons for the following responses as they apply to the ATWS: Actions contained in EOP for ATWS (RO 4.4 SRO 4.7)

K/A 029EA1.12: Ability to operate and monitor the following as they apply to a ATWS: M/G set power supply and reactor trip breakers (RO 4.1 SRO 4.0)

NUREG 1122 EPE: 029, Anticipated Transient Without Scram (ATWS)

VCS Task: O-000-117-05-04 RESPOND TO ABNORMAL NUCLEAR POWER GENERATION

j. Not for SRO-U use.

k. This is a new JPM. The initial conditions for the JPM are 100% power with Spent Fuel Pool level at 461.1 ft., which is below the Tech Spec 3.7.10 limit in accordance with OAP-106.1, OPERATING ROUNDS. The candidate will be directed to raise Spent Fuel Pool level to 461.8 ft using SOP-123, SPENT FUEL COOLING SYSTEM. The Candidate will simulate lining up the Reactor Makeup Water flowpath to the Spent Fuel Pool and will raise level to the desired value.

K/A: 000033A1.01: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including: Spent Fuel Pool Water Level (RO: 2.7, SRO: 3.3)

NUREG 1122 System: 033, Spent Fuel Pool Cooling System.

VCS Task: O-033-002-01-04, MAKEUP TO THE SPENT FUEL POOL PER SOP-123.

1. Initial conditions:

   - 100% power.

   Current conditions:

   - EOP-1.1, REACTOR TRIP RECOVERY in progress.
   - The following annunciator windows are **flashing** at the same frequency:
     - XCP-626, 2-3, SG B LVL LO-LO.
     - XCP-626, 4-4, TURB TRIP.
     - XCP-626, 4-5, TURB TRIP & P9 PERMISV.
   - The following annunciator window is **flashing intermittently at a faster rate** than the windows above:
     - XCP-626, 5-3, PZR PRESS LO.

   Which ONE of the following identifies information that is provided by observing **only** the annunciator window indications above?

   Assume that alarms have not been acknowledged.

   A. The Main Turbine was manually tripped.

   B. B SG NR level was less than 35% and is now greater than 35%.

   C.✓ The reactor tripped due to reaching a PZR PRESS LO trip setpoint.

   D. Pressurizer pressure decreased below a trip setpoint and is now greater than the setpoint.

Modification of rps 314 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must determine that the reactor tripped by evaluating annunciators and observing a design feature of the reactor trip first out panel.

A. Plausible. There is a first out for a Turbine trip but the turbine trip actuation originates from reactor trip breakers.

B. Plausible. Annunciator windows on other than the first out panel that are acknowledged and then above alarm setpoint flash to tell the operator that they must be reset. Incorrect because the main turbine trip alarm is flashing at the same rate indicating that the condition has not cleared.

C. CORRECT. The condition that caused the trip will cause a first out that flashes at an intermittent faster rate.

D. Plausible. Annunciators windows will flash at their setpoint and will flash at but will continue to flash until acknowledged and reset.

K/A: 007 EK2.03 Reactor Trip - Stabilization – Recovery /1 - Knowledge of the interrelations between the event and the following: Reactor trip status panel

K/A Match: The KA is matched because the candidate must determine that the reactor tripped by observing a design feature of the reactor trip first out panel.

Selection criteria; **MODIFIED FROM RPS314**

Tier: 1 **Group:** 1
Importance Rating: RO 3.5 SRO 3.6
Technical Reference: **DBD - MAIN CONTROL BOARD (MCB)**
ARP-XCP-626

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-15 04 DESCRIBE the 7300 Process Instrumentation and Solid State Protection System interfaces with the following systems or subsystems:4. Main control board

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

REACTOR PROT SYSTEM 314

Given the following plant conditions:

Initial conditions:

- 100% power.
- Loss of "A" and "B" Feedwater pumps occurred.

Current conditions:

- EOP-1.1, REACTOR TRIP RECOVERY in progress.
- The following annunciator windows are **flashing**:
  - XCP-626, 1-3, SG A LVL LO-LO
  - XCP-626, 3-3, SG C LVL LO-LO
  - XCP-626, 4-4, TURB TRIP
- The following annunciator window is **flashing at a faster rate** than the windows above:
  - XCP-626, 2-3, SG B LVL LO-LO

Which ONE of the following identifies information that is provided by the indications above?

A. The Main Turbine was manually tripped.

B. "B" SG NR level was less than 35% and is now greater than 35%.

C✓ The reactor tripped due to low "B" Steam Generator level.

D. "B" SG level is less than 35% and "A" and "C" SG levels are greater than 35%.

2. Initial Condition:

   - PZR Pressure Transmitter PT-455 failed high.

   Current Condition:

   - PZR Pressure Transmitter PT-444 failed high.

   Which ONE of the choices below completes the following statement?

   1) Which of the Pressurizer PORVs will automatically open due to the failure(s) above?

   2) Will the valve(s) identified in the question above be **closed** when actual Pressurizer pressure decreases to 1950 psig?

   Assume no operator actions.

   A✓ 1) **Only** PCV-444B, PWR RELIEF.
   2) Yes.

   B. 1) **Only** PCV-444B, PWR RELIEF.
   2) No.

   C. 1) PCV-444B, PWR RELIEF **and** PCV-445B, PWR RELIEF.
   2) Yes.

   D. 1) PCV-444B, PWR RELIEF **and** PCV-445B, PWR RELIEF.
   2) No.

Revised without significant modification for 2018 NRC - rj

QUESTION USAGE:
RO-14-01-GOP Makeup Exam
RO-11-01-NRC (**2013-RO NRC**)

REVISION HISTORY:
Rev 1 submitted by RJ - minor adjustments to wording and structure
Ops Review: Danny Rhymer
Approved:
Rev 0 submitted by RJ

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate to assess the effect of a failed control input to the PORV circuits that will create a stuck-open PORV event.

A. CORRECT; The failure of PT-444 will cause only PCV-444B to open. Since only one protection channel is failed, the 2/3 coincidence of P-11 will be satisfied with channels PT-456 and PT-457 and close the valve when pressurizer pressure falls below 1985 psig.

B. The first part is correct. The second part is plausible because two pressurizer pressure channels have failed and P-11 requires a 2/3 coincidence (but not two needed to defeat P-11).

   Incorrect because to PT-444 does not provide input to the function of P-11. Two of the three Pressurizer pressure protection channels (PT-455, 456 and/or 457) must fail for the blocking function to fail.

C. Plausible because for a high failure of PT-445, two PORVs will open. PCV-445B is plausible since it is also entitled as a "B" channel; It may be assumed in error that they will operate together. The second part is correct.

   Incorrect because only PCV-444B will open.

D. Plausible because for a high failure of PT-445, two PORVs will open. PCV-445B is plausible since it is also entitled as a "B" channel; It may be assumed in error that they will operate together. The second part is plausible because two pressurizer pressure channels have failed and P-11 requires a 2/3 coincidence (but not two needed to defeat P-11). PT-444 does not, however, provide input to the P-11 function.

   Incorrect because only PCV-444B will open and P-11 will function as required since two of the three Pressurizer pressure protection channels are still available for the blocking function.

K/A: 008AK2.03 K/A: 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) AK2: AK2: Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: 2.03: Controllers and positioners.

K/A Match: the KA is matched because it requires the candidate to assess the effect of a failed control input to the PORV circuits that will create a stuck-open PORV event.

Selection criteria; **REVISED BANK**

Tier: 1 **Group:** 1
Importance Rating: RO 2.5 SRO 2.4
Technical Reference: **Drawing 1-MS-41-0011 Sh. 11**
Drawing 1-MS-41-0011 Sh. 16

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-3-02 DRAW and LABEL a block diagram of the subsystem, Pressurizer Pressure Control. Include signal flowpaths for the following: 2. Pressure Operated Relief Valves

IC-3-04 DESCRIBE the pressurizer pressure and level control system interfaces with the following systems: 1. Reactor Protection System

Learning Objective:

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

PZR PRESS CNTRL SYS 066

Given the following plant conditions:

Initial Condition:

- PZR Pressure Transmitter PT-455 failed high.

Current Condition:

- PZR Pressure Transmitter PT-444 failed high.

Select ONE of the choices below that completes the following statement:

PORV(s) ____(1)____ will open, and the valve(s) ___(2)____ be closed at an actual pressurizer pressure of 1950 psig.

Assume no operator actions.

	(1)	(2)
A.✓	PCV-444B **only**	will
B.	PCV-444B **only**	will **not**
C.	PCV-445B and PCV-444B	will
D.	PCV-445B and PCV-444B	will **not**

3. Given the following plant conditions:

Time 1200:

- A Small Break LOCA has occurred.
- EOP-2.1, ES-1.2 POST-LOCA COOLDOWN AND DEPRESSURIZATION is in progress.
- An RCS coodown at 70°F/ hour is in progress.
- "A" and "B" RHR Pumps are OFF.
- "A" Charging Pump is running in injection mode.
- "B" Charging Pump has just been stopped.

Time 1205:

- Core exit TCs read 480°F and decreasing.
- LEVEL % LI-459A reads 20% and decreasing.
- WR PRESS PSIG PI-403 reads 825 psig and decreasing.

Which ONE of the choices below completes the following statement in accordance with EOP-2.1?

Core cooling is **_____.**

A. adequate for the evolution in progress and conditions are as expected.

B. adequate; Pump restarts are not required but the cooldown rate should be increased.

C✓ **not** adequate; **Only** "B" Charging pump should be restarted.

D. **not** adequate; "B" Charging pump **and** the RHR pumps should be restarted.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must assess core conditions during a small break LOCA to determine whether core cooling is adequate.

A. Plausible because RCS temperature is decreasing, subcooling is indicated, and one pump is running in injection mode.

   Incorrect because the EOP-2.1 re-initiation criteria is met and core cooling is not adequate.

B. Plausible because RCS temperature is decreasing, subcooling is indicated, and one pump is running in injection mode. The candidate may remember a 52.5°F reference page requirement and surmise that increasing the cooldown rate is required.

   Incorrect because the EOP-2.1 re-initiation criteria is met core cooling is not adequate.

C. CORRECT. For a wide range pressure of 825 psig and core exit temperature of 480°F, saturation temperature is 524°F. This yields a subcooling of 44°F which is less than the required subcooling re-initiation criteria of 52.5°F contained in EOP-2.1. In accordance with the reference page pumps should be restarted as necessary. Since RCS pressure is greater than the 325 psig restart requirement for RHR pumps, only the Charging Pump should be restarted.

D. Plausible. Subcooling is less than the re-initiation criteria of 52.5°F. In accordance with the EOP-2.1 reference page, pumps should be restarted as necessary. RHR pumps can be restarted after stopping them, but only when pressure decreases to less than 325 psig, Only the Charging Pump should be restarted.

   Incorrect because the RHR pumps should not be restarted.

K/A: 009 EA2.39 Small Break LOCA / 3 - Ability to determine and interpret the following as they apply to the event: Adequate core cooling.

K/A Match: The KA is matched because the candidate must assess core conditions during a small break LOCA to determine whether core cooling is adequate.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 4.3 SRO 4.7
Technical Reference: **STEAM TABLES**
EOP-2.1

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.1 04. STATE the bases or reasons for each action contained in EOP-2.1 This should include, but not be limited to, the following: 8. Determination of appropriate CHG/SI pump alignment

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

4. Given the following plant conditions:

    - A Large break LOCA occurred one hour ago.
    - "A" RHR pump failed to start.
    - EOP-2.2, ES-1.3 TRANSFER TO COLD LEG RECIRCULATION is in progress.
    - Operators are taking the manual actions required to complete the long-term recirculation lineup of ECCS systems.
    - Operators just **closed** MVG-8109A(B)(C), CHG PP A(B)(C).

    Which ONE of the choices below completes the following statement?

    The reason why this valve is closed, in accordance with EOP-2.2, is to ______

    A. prevent backflow through an idle loop.

    B.✓ isolate a path from the RHR sump to outside of containment.

    C. prepare systems for the transfer to hot leg-recirculation.

    D. provide the maximum charging pump discharge pressure.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the reason for closing a fluid path while transferring to cold leg recirculation.

A. Plausible because there is a caution in EOP-2.2 stating that 8706 A(B) should not be opened from a non-running RHR pump and "A" RHR pump not running is given in the stem.

Incorrect because the prevention of backflow is not the reason 8109A, B and C are closed.

B. CORRECT. 8109A, B and C are closed in EOP-2.2 to isolate a path from the containment sump to outside of the containment.

C. Plausible because one valve on the Charging Pump discharge, 8801A is closed in preparation for hot leg recirculation.

Incorrect because closing of 8109A, B and C is not performed to prepare for hot leg recirculation.

D. Plausible because closing recirculation valves increases discharge pressure flow and there is no need for recirculation flow since the charging pumps are pumping to a low pressure RCS during a large break LOCA.

Incorrect because increasing discharge flow and pressure is not the reason that 8109A, B and C are closed.

K/A: 011 EK3.08 Large Break LOCA/3 - Knowledge of the reasons for the following responses as they apply to the event: Flowpath for sump recirculation

K/A Match: the KA is matched because it requires the candidate to recall the reason for closing a fluid path while transferring to cold leg recirculation.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.9 SRO 4.1
Technical Reference: **ERG BASES - ES-1.3**
EOP-2.2

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.2 04. STATE the bases or reasons for each action contained in EOP-2.2. This should include, but not be limited to, the following:

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments (2011 NRC Exam):

Facility Response:

Comments;

5. Given the following plant conditions:

   - 100% power.
   - Operators are evaluating to determine the cause of the following alarm:

     XCP-619, 2-3, RCP C #2 STG SL PRESS HI alarm.

   Which ONE of the following identifies an **abnormal** component position that could cause XCP-619, 2-3 to alarm?

   A. HCV-186, INJ FLOW fully **open**.

   B.✓ PVT-8141C, C CBO ISOL fully **closed**.

   C. FCV-122, CHG FLOW fully **closed**.

   D. MVT-8112, SEAL WTR RTN ISOL fully **closed**.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev 0 submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must identify a valve that is out of position at 100% power that is causing an alarm indicating an RCP malfunction.

A. Plausible because HCV-186 regulates the injection flow from charging pump discharge to the seals but fully opening this valve does not cause XCP-619, 2-3 to annunciate.

B. CORRECT. Closing the path for controlled bleed off will cause XCP-619, 2-3.

C. Plausible because closing FCV-122 will cause pressure on the seal injection line to increase raising pressure on the #1 seal but will not cause XCP-619, 2-3 to annunciate.

D. Plausible because MVT-8812 is in series with PVT-8141C but will not cause XCP-619, 2-3 to annunciate since there is a relief that allows CBO to flow to the PRT.

K/A: 015 AG2.1.31 RCPMalfunctions/4 - Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup.

K/A Match: The KA is matched because the candidate to identify a valve that is out of position at 100% power that is causing an alarm associated with an RCP malfunction.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 4.6 SRO 4.3
Technical Reference: **ARP-XCP-619, 2-3**
DRAWING 302-673

Proposed references to be provided to candidates during examination: None

Learning Objective: AB-4 021. LIST the condition or setpoint associated with the following system limitations: 5. Controlled Bleedoff Flow and Temperature 6. Individual Seal Flow

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(3)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

6. Initial conditions:

   - 3% power and stable with a plant startup in progress.
   - "B" Charging Pump is in service.
   - BU GRP 1 Pressurizer heaters are energized.

   Current conditions.

   - "B" Charging pump has tripped.
   - XCP-616, 1-3, BLCK HTRS ISOL LTDN PZR LCS LO is in alarm.

   Which ONE of the choices below completes the following statement?

   The **highest** Pressurizer level at which XCP-616, 1-3 came into alarm is __(1)__;

   Operators __(2)__ manually energize BU GRP 2 heaters from the main control boards.

   A. 1) 21%;
      2) can

   B. 1) 21%.
      2) cannot

   C. 1) 17%.
      2) can

   D.✓ 1) 17%.
      2) cannot

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine that pressurizer heaters cannot be energized due to a lowering pressurizer level after a loss of charging flow.

A. The first part is plausible XCP-616, 1-5 alarms at 5% below programmed level. The Pressurizer level program is 25% to 60% from 0% to 100% power respectively; thus the programmed level at 3% power is (3% x (60%- 25%)) + 25% = 26%. 5% below that level is 21%. The second part is plausible because there is no interlock at that level that would prevent energization to heaters.

   Incorrect because XCP-616, 1-3 did not alarm at 21%

B. The first part is plausible. See explanation for A. The second part is plausible because XCP-616, 1-3 alarms at a 17% pressurizer level at which there is an interlock preventing energization of heaters.

   Incorrect because XCP-616, 1-3 did not alarm at 21%.

C. The first part is correct. XCP-616, 1-3 alarms at a 17% pressurizer level. The second part is plausible because there are other alarms such as XCP-616, 2-3 which alarm at a level which still allow energization of heaters.

   Incorrect because pressurizer heaters cannot be energized at 17% pressurizer level.

D. CORRECT. XCP-616, 1-3 alarms at a 17% pressurizer level and an interlock at that level prevents heaters from being energized.

K/A: 022 AK1.03 Loss of Rx Coolant Makeup / 2 - Knowledge of the operational implications of the following concepts as they apply to the event: Relationship between charging flow and PZR level

K/A Match: The KA is matched because the candidate must determine that pressurizer heaters cannot be energized due to a lowering pressurizer level after a loss of charging flow.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.0 SRO 3.4
Technical Reference: **DBD - SETPOINT BASES**

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-3 016. DE SCRIBE the following pressurizer pressure and level control system interlocks. Specify purpose and setpoints: 2. Pressurizer Low Level Heater Cutout

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

7. Initial conditions:

   - The plant was in Mode 4 with a plant cooldown in progress.
   - A loss of "A" Train RHR cooling occurred.
   - The RCS began to heatup.
   - The crew entered AOP-115.3, LOSS OF RHR WITH THE RCS INTACT.

   Current conditions:

   - "A" RHR pump has been restored to operation.
   - Operators have exited AOP-115.3.
   - RCS temperature is 305°F and decreasing.
   - FCV-605A, A BYP, is 40% open in MANUAL.
   - HCV-603A, A OUTLET is partially open.
   - STP-103.001, REACTOR COOLANT SYSTEM AND PRESSURIZER HEATUP/COOLDOWN SURVEILLANCE has been implemented.
   - The CRS had directed the NROATC to increase the RCS cooldown rate using HCV-603A.

   Which ONE of the choices below answers both of the following questions?

   1) What is direction of HCV-603A valve movement required to increase the cooldown in accordance with AOP-115.3?

   2) What is the **largest** cooldown rate allowed by T.S. 3.4.9.1 PRESSURE / TEMPERATURE LIMITS - REACTOR COOLANT SYSTEM for the current RCS temperature?

   A✓ 1) HCV-603A will be **opened**.
   2) 100°F/hour.

   B. 1) HCV-603A will be **opened**.
   2) 50°F/hour.

   C. 1) HCV-603A will be **closed**.
   2) 100°F/hour.

   D. 1) HCV-603A will be **closed**.
   2) 50°F/hour.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine the operation of an RHR heat exchanger outlet valve to adjust a cooldown rate.

A. CORRECT. HCV-603A is opened to increase the cooldown rate and the maximum cooldown rate allowed in accordance with T.S. 3.4.9.1 is 100°F/hr.

B. The first part is correct. HCV-603A is opened to increase the cooldown rate. The second part is plausible because GOP-6 specifies a cooldown rate of 50°F/hr at low temperatures.

   Incorrect because the cooldown rate allowed in accordance with T.S. 3.4.9.1 is not 50°F/hr.

C. The first part is plausible. FCV-605A, A BYP is closed to increase the cooldown rate. The second part is correct. the maximum cooldown rate allowed in accordance with T.S. 3.4.9.1 is 100°F/hr.

   Incorrect because HCV-603A is not closed to increase the cooldown rate.

D. The first part is plausible. FCV-605A, A BYP is closed to increase the cooldown rate. The second part is plausible because GOP-6 specifies a cooldown rate of 50°F/hr at low temperatures.

   Incorrect because HCV-603A is not closed to increase the cooldown rate and the cooldown rate allowed in accordance with T.S. 3.4.9.1 is not 50°F/hr

K/A: 025 AA1.01 Loss of RHRSystem/4 - Ability to operate and / or monitor the following as they apply to the event: RCS/RHRS cooldown rate.

K/A Match: The KA is matched because the candidate must determine the operation of an RHR heat exchanger bypass valve to adjust a cooldown rate.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.6 SRO 3.7
Technical Reference: **AOP-115.3**
T.S. 3.4.9.1
GOP-6

Proposed references to be provided to candidates during examination: None

Learning Objective: AOP-115.3 06. RELATE any systems'/components' indication operation, or malfunction to its effect on AOP-115.3.

Question Cognitive Level: **Memory or Fundamental Knowledge** ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

8. Given the following plant conditions:

   - 100% power.
   - Leakage via a Pressurizer PORV valves seat is occurring.
   - All Pressurizer PORVs indicated closed on the main control board.
   - Pressurizer level is 60% and stable.
   - VCT level is decreasing at 20%/ hour.

   Which ONE of the following choices completes both of the following statements?

   The leakrate for this event is in a range of __(1)__ gpm.

   Operators __(2)__ determine which of the Pressurizer PORVs is leaking by using control board temperature meters.

   Assume no changes to equipment alignments.

   A. 1) 1 to 2
      2) can

   B. 1) 1 to 2
      2) cannot

   C. 1) 4 to 6
      2) can

   D.✓ 1) 4 to 6
      2) cannot

NOTE TO EXAMINER:
THE QUESTION ORIGINALLY SUBMITTED AS PART OF THE SAMPLE FOR THIS KA HAS BEEN COMPLETELY REPLACED.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must interpret the rate of makeup required for a leaking PORV using the rate of decrease in the volume control tank.

A. The first part is plausible. If the volume per % level of the RCDT is used in error, the range of leakage would be 1-2 gpm. 3.13 gallons per % of RCDT level -> 20%/hr x 1hr/60 min x 3.13 gals/% = 1.04 gpm. The second part is plausible because Pressurizer code safeties have individual temperature sensors on the discharge manifold that display on the main control board.

   Incorrect because the range of leakage is not 1-2 percent and control board instrumentation cannot be used to determine the leaking PORV.

B. The first part is plausible. If the volume per % level of the RCDT is used in error, the range of leakage would be 1-2 gpm. The second part is correct. Control board temperature instrumentation cannot be used to determine the leaking PORV.

   Incorrect because the range of leakage is not 1-2 percent.

C. The first part is correct. Using 14 gallons per % of VCT level -> 20%/hr x 1hr/60 min x 14 gals/% = 4.66 gpm. The second part is plausible because Pressurizer code safeties have individual temperature sensors on the discharge manifold that display on the main control board.

   Incorrect control board instrumentation cannot be used to determine the leaking PORV.

D. CORRECT. Using 14 gallons per % of VCT level -> 20%/hr x 1hr/60 min x 14 gals/% = 4.66 gpm. PORVs do not have individual temperature sensors and therefore which PORV is leaking cannot be determined using temperature meters.

K/A: 027 AA2.07 Pressurizer Pressure Control System Malfunction /3 - Ability to determine and interpret the following as the apply to the event: Makeup flow indication

K/A Match: The KA is matched because it requires the candidate to interpret the rate of makeup required for a leaking PORV using the rate of decrease in the volume control tank.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.1 SRO 3.1
Technical Reference: **CURVE BOOK - VI-31**
CURVE BOOK - VI-22
SOP-101, Section V.A

Proposed references to be provided to candidates during examination: None

Learning Objective: AB-5 09. DESCRIBE the following Reactor Makeup System Interlocks. Specify purpose and setpoints: 3. Reactor makeup control system

Question Cognitive Level: **Memory or Fundamental Knowledge** ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

9. Given the following plant conditions:

   - 100% power initially.
   - A steam generator tube rupture has occurred on "C" Steam Generator (SG).
   - Actions of EOP-4.0, E-3 STEAM GENERATOR TUBE RUPTURE are complete.
   - EOP-4.1C, ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP is in progress.
   - An RCS cooldown is in progress.
   - "C" SG narrow range level is 86% and decreasing.
   - Pressurizer level is 19% and decreasing.

   Which ONE of the choices below identifies the method that is used for the **current** condition to stabilize level in "C" Steam Generator in accordance with EOP-4.1C?

   A. Establish EFW flow to "C" Steam Generator.

   B. Energize Pressurizer heaters.

   C. Close steam dumps.

   D.✓ Increase Charging flow.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine that an increase in charging flow is required to maintain a stable RCS and SG pressure during a post-sgtr cooldown.

A. Plausible because an increase in EFW can raise ruptured SG level and operators are given direction in EOP-4.1C on how to control ruptured SG level. Additionally, after the cooldown, efw flow is controlled to cycle level between 40 and 82%,

   Incorrect because the current conditions do not require increasing EFW in accordance with EOP-4.1C.

B. Plausible because energizing pressurizer heaters is a method used to counteract a decreasing ruptured SG level in EOP-4.1C.

   Incorrect because pressurizer heaters would not be used at a pressurizer level of 19%.

C. Plausible because closing steam dumps stop the cooldown and mitigate the RCS pressure decrease which would help counteract the decrease in ruptured SG level.

   Incorrect because stopping the cooldown is not a requirement for a lowering SG level for the conditions given.

D. CORRECT. Using the table in EOP-4.1C step 9, with pressurizer level less than 22% and a decreasing ruptured SG level, an increase in charging is required.

K/A: 038 EA2.15 Steam Gen. Tube Rupture / 3 - Ability to determine and interpret the following as the apply to the event: Pressure at which to maintain RCS during SIG cooldown.

K/A Match: The KA is matched because the candidate must determine that an increase in charging flow is required to maintain a stable RCS and SG pressure during a post-sgtr cooldown.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 4.2 SRO 4.4
Technical Reference: **EOP-4.1C**

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-4.1C 04. STATE the bases or reasons for each action contained in EOP-4.1C This should include, but not be limited to, the following: 7. Use of table provided to minimize primary-to-secondary leakage.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

10. Given the following plant conditions:

- Mode 3.
- Buses 1DA and 1DB are **deenergized**.
- The EDGs failed to start automatically or manually from the main control boards.
- XCP-636, 6-1, DG A ENG START FAIL is in alarm.
- XCP-637, 6-1, DG B ENG START FAIL is in alarm.
- The crew entered EOP-6.0 ECA 0.0 LOSS OF ALL ESF AC POWER.

Which ONE of the choices below completes the following statements?

EOP-6.0 Attachment 1, LOCAL STARTING AND LOADING DIESEL GENERATOR A(B) will be implemented __(1)__ loads on buses 1DA and 1DB are placed in PULL TO LK NON-A.

The maximum KW loading for **continuous** operation of EDGs allowed in EOP-6.0 is __(2)__ .

A✓ 1) after
2) 4250 KW.

B. 1) after
2) 5100 KW.

C. 1) before
2) 4250 KW.

D. 1) before
2) 5100 KW.

NOTE TO EXAMINER:
THIS QUESTION WAS SUBMITTED FOR SAMPLE AND WAS REVISED TO ADDRESS COMMENTS.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall whether EDGs are started before or after bus load controls are placed in pull to lock and recall procedure limits for continuous diesel operation.

A. CORRECT. The manual start of a diesel is attempted after loads on 1DA and 1DB are placed in pull to lock on main control boards while in EOP-6.0. step 6. The max continuous load limit for the EDGs is 4250 KW.

B. The first part is correct. The manual start of a diesel is attempted after loads on 1DA and 1DB are placed in pull to lock on main control boards while in EOP-6.0. step 6. The second part is plausible because 5100 KW is the maximum load limit for 30 minutes as contained in EOP-6.0.

   Incorrect because the 5100 KW is not the continuous load limit contained in EOP-6.0.

C. The first part is plausible because operator will attempt to locally start diesel in step 6 of EOP-6.0. The second part is correct. The max continuous load limit for the EDGs is 4250 KW.

   Incorrect because operators will not attempt to locally start EDGs prior to placing ESF loads on buses in pull to lock.

D. The first part is plausible because operator will attempt to locally start diesel in step 6 of EOP-6.0. The second part is plausible because 5100 KW is the maximum load limit for 30 minutes as contained in EOP-6.0.

   Incorrect because operators will not attempt to locally start EDGs prior to placing ESF loads on buses in pull to lock and because 5100 KW is not the continuous load limit contained in EOP-6.0.

K/A: 055 EA1.02 Station Blackout/6 - Ability to operate and / or monitor the following as they apply to the event: Manual ED/G start

K/A Match: the KA is matched because it requires the candidate to recall the maximum kilowatt load for diesels while in EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER and to recall load limits for the diesel.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 4.3 SRO 4.4
Technical Reference: **EOP-6.0, LOSS OF ALL ESF AC POWER.**

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-6.0 06. RELATE each CAUTION and NOTE in Emergency Operating Procedure EOP-6.0 to its corresponding step by identifying the bases or reason for the CAUTION or NOTE. This should include, but not be limited to, the following:

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

11. Given the following plant conditions:

Time 1100:

- 100% power initially.
- "A" Train Work Week.
- APN-5901 has deenergized due to a static switch malfunction that **cannot** be immediately corrected.

Time 1101:

- A main steam line break has occurred.
- RB pressure is 13 psig and increasing.
- RCS pressure is 1400 psig and decreasing.
- EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION has been entered.

Which ONE of the following identifies an item that will be performed by the Balance of Plant Operator using directions provided in ATTACHMENT 3, SI EQUIPMENT VERIFICATION?

A. The Safety Injection will be backed up using SI Actuation switches.

B. One of the Service Water Booster Pumps will be placed in PULL TO LOCK NON A.

C✓ One of the Charging Pumps will be manually started.

D. One of the RB Spray Pumps will be manually started.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 Submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must identify an ESF pump that must be started while in EOP-1.0 (E-0) after a loss of a vital ac bus that deenergizes an ESF sequencer.

A. Plausible because Safety Injection is manually actuated as part of immediate actions in Step 5 if the automatic actuation occurs.

   Incorrect. The SI actuation switch is not specified for use in Attachment 3.

B. Plausible because the Attachment 3 contains a requirement to place an idle SW booster pump in pull to lock if the associated 3107 valve is not shut.

   Incorrect because the 3107 valve is an air operated valve that will automatically close if the SW booster pump is off.

C. Correct. The “A” ESFLS is de-energized because APN-5901(“A” ESFLS power supply) and therefore automatic loading of 1DA will not occur.

D. Plausible because the RB spray pump is an ESF load and the candidate may assume in error that an auto start will not occur due to the deenergized sequencer.

   Incorrect. The RB spray pump will start automatically as required.

K/A: 057 AG2.4.6 Loss of Vital AC Inst. Bus / 6 - Knowledge symptom based EOP mitigation strategies.

K/A Match: The KA is matched because the candidate must identify an ESF pump that must be started while in EOP-1.0 (E-0) after a loss of a vital ac bus.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.7 SRO 4.7
Technical Reference: **EOP-1.0**
EOP-1.0, ATT 3

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-1.0 08. RELATE any systems/components operation, indication, or malfunction to its effect on EOP-1.0.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

12. Given the following plant conditions:

- 100% power.
- An open circuit has occurred that has removed DC power from the control circuit for LCV-459, LTDN LINE ISOL.
- DC power to other components has **not** been interrupted.

Which ONE of the following identifies an action that will be procedurally required to stabilize plant conditions at the current power level?

A✓ Throttle open HCV-137, XS, LTDN HX.

B. Open LCV-460, LTDN LINE ISOL.

C. Fully close HCV-186, INJ FLOW.

D. Adjust FCV-122, CHG FLOW to 70 gpm.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the valve operation for excess letdown after normal letdown is lost due to a loss of dc power.

A. CORRECT. If DC is lost to LCV-459, it will close and be incapable of being opened from the main control boards. In order to stabilize pressurizer level, Excess Letdown using HCV-137, XS LTDN HX will be placed in service.

B. Plausible because LCV-460, LTDN LINE ISOL performs a function similar to LCV-459 in the same line and other letdown valves such as the orifice isolation valves are in parallel. The candidate may assume that the valves are in parallel and that opening this valve could restore letdown.

   Incorrect because LCV-460 is in series with LCV-459.

C. Plausible because seal injection flow will be reduced when placing excess letdown in service.

   Incorrect because HCV-186 will not be fully closed.

D. Plausible because the procedure for restoring normal letdown to service contains a step that requires adjustment of FCV-122 to 70 gpm.

   Incorrect because that adjustment would not allow normal letdown to be placed in service and cause conditions to stabilize.

K/A: 058 AG2. 1.20 Loss of DC Power / 6 - Ability to execute procedure steps.

K/A Match: The KA is matched because the candidate must recall the valve operation for excess letdown after normal letdown is lost due to a loss of dc power.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 4.6 SRO 4.6
Technical Reference: **AOP-102.1**

Proposed references to be provided to candidates during examination: None

Learning Objective: AOP-102.1 03. Given a set of plant conditions, DETERMINE the appropriate plant response and operator actions in accordance with AOP-102.1.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

13. Given the following initial plant conditions:

Time 0700:

- 100% power.
- "C" Service Water Pump is tagged out of service.

Time 0705:

- Both "A" and "B" Service Water pumps have tripped.
- AOP-117.1, LOSS OF SERVICE WATER has been entered.

Which ONE of the following describes an action that must be performed if restoration of a Service Water loop does **not** occur in accordance with AOP-117.1?

A. Trip RCPs when time 0715 is reached.

B. Reduce Main Turbine load to zero within 1 hour.

C.✓ Perform Attachment 3, Charging Pump Temperature monitoring.

D. Open valves to supply Industrial Cooling to RBCUs.

NOTE TO EXAMINER:
THIS QUESTION WAS SUBMITTED FOR SAMPLE AND WAS REVISED TO ADDRESS COMMENTS.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate Knowledge of the reasons for the following responses as they apply to the event: Effect on the nuclear service water discharge flow header of a loss of CCW

A. Plausible because AOP-117.1 satisfies an entry condition for AOP-118.1, LOSS OF COMPONENT COOLING WATER and in that procedure if CCW is lost to the motor bearing coolers for 10 minutes, an RCP trip would be required.

   Incorrect because a loss of CCW to the motor bearing coolers has not occurred and the 10 minute RCP trip requirement is not in effect.

B. Plausible because a loss of stator cooling would require a rapid power reduction and stator cooling is supplied by closed cycle cooling which supplies component cooling similar to service water and component cooling water for various Turbine Building loads.

   Incorrect because stator cooling is not cooled by service water and a rapid power reduction would not be required due to a loss of that system.

C. CORRECT. Local temperature monitoring of charging pumps must be performed because service water cooling has been lost to the CCW heat exchangers.

D. Plausible because the RBCUs can be supplied by either Service Water or Industrial Cooling water.

   Incorrect because Industrial Cooling Water is normally aligned.

K/A: 062 AK3.04 Loss of Nuclear Svc Water / 4 - Knowledge of the reasons for the following responses as they apply to the event: Effect on the nuclear service water discharge flow header of a loss of CCW.

K/A Match: The KA is matched because it requires the candidate to determine that charging pumps temperatures must be monitored due to a loss of SW cooling to CCW.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.5 SRO 3.7
Technical Reference: **AOP-117.1 LOSS OF SERVICE WATER**
AOP-118.1
SOP-117
XCP-632, 3-3

Proposed references to be provided to candidates during examination: NONE

Learning Objective: AOP-117.1 03. Given a set of plant conditions, DETERMINE the appropriate plant response and operator actions in accordance with AOP-117.1.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

14. Given the following plant conditions:

- 100% power.
- A grid disturbance was reported by the System Controller.
- AOP-301.1, RESPONSE TO ELECTRICAL GRID ISSUES, in progress.
- Grid voltage is lowering.
- The BOP is monitoring voltages on the main control boards.
- The following indications have been present for **5 seconds**.
  - Bus 1DA at 6500 vac.
  - Bus 1DB at 6600 vac.
  - Bus 1C at 6500 vac.

Which ONE of the choices describes automatic action(s) based on system voltage that should have occurred, or will occur, based on the conditions above?

A.✓ **Only** "A" EDG should have started.

B. "A" **and** "B" EDG should have started.

C. **Only** "A" EDG will start after 5 more seconds have passed.

D. "A" EDG should have started **and** "C" RCP should have tripped.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 Submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the the candidate must identify which diesel should have started after a grid disturbance causing a lowered voltage.

A. CORRECT. The "A" EDG will start at a 1DA bus voltage at 91.34% of 7200 volts for 3 seconds - > 6576 volts. 1DA is indicated at 6500 vac. No other condition for EDG start exists in question.

B. Plausible because both 1DA and 1DB voltages are shown significantly reduced.

   Incorrect because 1DB has not reached a setpoint that would start "B" EDG.

C. Plausible because The EDG will start at bus voltage at 91.34% of 7200 volts after a time delay of 3 seconds. In addition, the diesel is designed to come to speed and voltage and be ready for load in 10 seconds. The candidate may assume that the time delay is 10 seconds.

   Incorrect because the "A" EDG is already started.

D. Plausible because the EDGs will start at bus voltage at 91.34% of 7200 volts for 3 seconds - > 6576 volts. In addition, an RCP will automatically trip at a 1C voltage setpoint of 4830 vac.

   Incorrect because 1C voltage is not at the setpoint for a trip of "C" RCP.

K/A: 077 AA1 .05 Generator Voltage and Electric Grid Disturbances / 6 - Ability to operate and / or monitor the following as they apply to the event: Engineered Safety Features

K/A Match: The KA is matched because the candidate must identify which diesel should have started after a grid disturbance causing a lowered voltage.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.9 SRO 4.0
Technical Reference: **FSAR PAGE 8.3-8**
DRAWING E-206-005

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.1 06. APPLY EOP-2.1 by predicting a discrete path through EOP-2.1 given a set of plant conditions

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(8)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

15. Initial conditions:

- A LOCA outside the RB has occurred.
- All ESF equipment performed as designed.
- The crew is performing actions as directed by EOP-2.5, LOCA OUTSIDE CONTAINMENT.
- The **first** closure of a motor-operated valve required by EOP-2.5 had just been performed in an attempt to stop the LOCA.
- The crew is evaluating RCS pressure to determine if the action was successful.

Which ONE of the choices identifies the motor-operated valve that operators closed above?

A.✓ MVG-8888A, RHR LP A TO COLD LEGS.

B. MVT-8100 SEAL WTR RTN ISOL.

C. MVG-8801A(B), HI HEAD TO COLD LEG INJ.

D. MVG-8889, RHR LP A&B TO HOT LEGS.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must recall an isolation of a low head injection line that is performed in procedure EOP-2.5, LOCA OUTSIDE CONTAINMENT.

A. CORRECT; MVG-8888A is the first MOV that must be manually closed. All other MOVs contained in EOP-2.5 prior to that closure are verifications of an automatic action that should have occurred.

B. Plausible because MVT-8100 is verified closed in EOP-2.5.

   Incorrect because MVT-8100 is not the first valve actually closed and there is no procedural verification of RCS conditions after that verification.

C. Plausible because low head injection lines are the first isolations and 8801 is a isolation in a high head injection line.

   Incorrect because MVG 8801A and B are not closed in EOP-2.5..

D. Plausible because MVG-8889 is a low head injection line.

   Incorrect because MVG-8889 is not closed in EOP-2.5..

K/A: WE04 EK2.2 LOCA Outside Containment / 3 - Knowledge of the interrelations between the event and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.

K/A Match: K/A is met because candidate must recall an isolation of a low head injection line that is performed in procedure EOP-2.5, LOCA OUTSIDE CONTAINMENT.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.8 SRO 4.0
Technical Reference: **EOP-2.5, LOCA OUTSIDE CONTAINMENT**
DBD-SAFETY INJECTION SYSTEM PAGE 2.3-6

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.5 06. APPLY EOP-2.5 by predicting a discrete path through EOP-2.5 given a set of plant conditions

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

16. Given the following plant conditions:

Time 1200:

- 100% power initially.
- A turbine and reactor trip occurred.
- All EFW was subsequently lost.
- The crew entered EOP-15.0, FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK.

Time 1215.

- Feed to steam generators was recovered via a Feedwater Pump.
- All Narrow Range steam generator levels are 30% and increasing.
- Operators exited EOP-15.0 to procedure and step in effect.

Time 1217:

- XCP-626, RB PRESS HI-1 SI is in alarm.
- All Narrow Range steam generator levels are 33% and decreasing.

Which ONE of the choices below completes the following statements?

Steam generator feed __(1)__ been lost and re-entry into EOP-15.0 __(2)__

Assume no other RED paths are present on the CSFSTs.

A. 1) has
2) will be required if NR range Steam Generator levels decrease an additional 7%.

B.✓ 1) has
2) is required at the current values of NR Steam Generator levels.

C. 1) has **not**
2) will be required if NR range Steam Generator levels decrease an additional 7%.

D. 1) has **not**
2) is required at the current values of NR Steam Generator levels.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must identify adverse containment conditions based on an annunciator and determine that entry into a loss of heat sink procedure is required

A. The first part is correct because an SI will cause main feedwater pumps to trip. The second part is plausible because under normal containment conditions, re-entry would not be required until SG levels decrease to 26% and current levels are at 33%.

   Incorrect because with XCP-626, 6-1 alarm in, adverse containment is present and reentry into EOP-15.0 is required at SG levels below 41%.

B. An SI will cause main feedwater pump to trip and with XCP-626, 6-1 alarm in at 3.6 psig RB pressure, adverse containment is present and reentry into EOP-15.0 is required at SG levels below 41%. Current levels are at 33%.

C. Plausible because the candidate may think that because a previous defeat of feed water isolation was performed that the Feedwater pumps are now prevented from tripping. The second part is plausible because under normal containment conditions, initial entry in EOP-15.0 is required with no EFW flow and SG levels less than 26%. Current levels are at 33%.

   Incorrect because main feed has been lost due to the SI signal and because conditions to re-enter EOP-15.0 are met.

D. Plausible because the candidate may think that because a previous defeat of feed water isolation was performed that the Feedwater pumps are now prevented from tripping. The second part is plausible because entry into EOP-15.0 is required at SG levels below 41% when adverse containment is present. Current levels are at 33%..

   Incorrect because main feed has been lost due to the SI signal.

K/A: WE05 EK1 .3 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4 - Knowledge of the operational implications of the following concepts as they apply to the event: Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of secondary Heat Sink).

K/A Match: The KA is matched because the candidate must identify adverse containment conditions based on an annunciator and determine that entry into a loss of heat sink procedure is required.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.9 SRO 4.1
Technical Reference: **EOP-12.0 HEAT SINK STATUS TREE**
ARP-XCP-626, 6-1

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-15.0 02. STATE the entry conditions of EOP-12.0 1. Symptoms 2. Transitions from other procedures.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

17. Given the following plant conditions:

- A Large Break Loss of Coolant Accident has occurred.
- Neither RHR pump is available.
- The crew is implementing EOP-2.4 LOSS OF EMERGENCY COOLANT RECIRCULATION.
- RWST level is 20% and lowering.
- Both spray pumps are taking suction from the RWST.
- Operators are preparing to stop RB spray pumps.

Which ONE of the choices below completes the following statement?

RB spray pumps are stopped in this plant condition in order to ________(1)________ and MVG-3003A(B), SPRAY HDR ISOL LOOP A(B) will be closed to ensure that the ____________(2)_____________.

A. 1) protect the pumps from a loss of suction
   2) containment is isolated.

B. 1) protect the pumps from a loss of suction
   2) RWST does not drain to the RB sump.

C.✓ 1) conserve RWST inventory
   2) containment is isolated.

D. 1) conserve RWST inventory
   2) RWST does not drain to the RB sump.

Proposed for use on 2018 NRC - rj

QUESTION USAGE:
ILO 13-01 Audit (2015)

REVISION HISTORY:
Rev. 0 Submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must recall the reasons for stopping ESF pumps and performing isolations during a loss of emergency cooling recirculation.

A. Plausible; RB Spray pumps are stopped to protect the pumps if RWST level drops less than 10% due to potential loss of suction and the second half of the question stating containment isolation as the basis is correct. The second part is right per basis for EOP-2.4 (WOG ECA-1.1 step 6, in NOTES).

   Incorrect because the pump is not required to be stopped due to a loss of suction.

B. Plausible; RB Spray pumps are stopped to protect the pumps if RWST level drops less than 10% due to potential loss of suction and other valves such as the RHR suction supplies are shut to prevent draining the RWST.

   Incorrect because the pump is not required to be stopped due to a loss of suction and preventing a drain of the RWST is not the reason for closure.

C. CORRECT; The RB spray pumps will be stopped in this condition to conserve RWST inventory and their discharge valves are shut to ensure containment isolation. The second part is right per basis for EOP-2.4 (WOG ECA-1.1 step 6, in NOTES). Answer is not obvious due to RB Spray being semi-closed outside of Containment; release path is through RB Spray piping back to the Refueling Water Storage Tank (RWST) then out RWST vent to the Aux Building Charcoal Exhaust (drawings 302-651 and 912-125 C1).

D. Plausible because The RB spray pumps will be stopped in this condition to conserve RWST inventory and other valves such as the RHR suction supplies are shut to prevent draining the RWST.

   Incorrect because prevent a drain of the RWST is not the reason for closure.

K/A: WE11 EK1 .2 Loss of Emergency Coolant Recirc. / 4 - Knowledge of the operational implications of the following concepts as they apply to the event: Normal, abnormal and emergency operating procedures -associated with (Loss of Emergency Coolant Recirc).

K/A Match: the K/A is met because the candidate must recall the reasons for stopping ESF pumps and performing isolations during a loss of emergency cooling recirculation.

Selection criteria; **BANK**

Tier: 1 **Group:** 1
Importance Rating: RO 3.6 SRO 4.1
Technical Reference: **EOP-2.4**
EOP-2.4 (WOG ECA-1.1) basis

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.4 03. STATE the major action categories of EOP-2.4

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification:

NRC Form ES-401-9 Comments (2011 NRC Exam):

Facility Response:

comments;

18. Given the following plant conditions:

Time 1400:

- 100% power.
- A Main steam line break has occurred in the Reactor Building.
- All MSIVs failed to close.

Time 1440:

- Actions of EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION have been completed.
- EOP-3.1 ECA-2.1 UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS is in progress.
- RCS pressure is 1400 psig and increasing.
- Operators are taking actions to terminate SI.

Which ONE of the choices below completes the following statements?

The **highest** value for a T_{COLD} temperature at which operators **must** immediately leave EOP-3.1 is __(1)__ in accordance with EOP-12.0, MONITORING OF CRITICAL SAFETY FUNCTIONS.

The purpose for performing a Function Restoration Procedure for the T_{COLD} value identified in question (1) is to ensure that __(2)__.

Assume all other Critical Safety Functions are GREEN.

A. 1) 250°F
2) the RCS cooldown does not cause a loss of shutdown margin.

B.✓ 1) 250°F
2) thermal stresses do not cause a growth in a reactor vessel flaw.

C. 1) 280°F
2) the RCS cooldown does not cause a loss of shutdown margin.

D. 1) 280°F
2) thermal stresses do not cause a growth in a reactor vessel flaw.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the temperature at which excessive thermal stresses could occur and that a procedure for excessive cooldown must be used after a steam line break.

A. The first part is correct. 250°F a T_{COLD} temperature at which operators must immediately leave EOP-3.1 due to an ORANGE path on INTEGRITY. The second part is plausible EOP-13.0, which is a procedure for ATWS makes considerations for faulted SGs as are presented in the stem.

Incorrect EOP-16.0 is not entered because of a concern for loss of shutdown margin.

B. CORRECT. 250°F a T_{COLD} temperature at which operators must immediately leave EOP-3.1 to go to EOP-16.0 due to an ORANGE path on INTEGRITY. The procedure addresses a potential thermal stresses that can cause a growth in a reactor vessel flaw.

C. Plausible because 280°F is the threshold for a yellow path procedure entry. The second part is plausible EOP-13.0, which is a procedure for ATWS makes considerations for faulted SGs as are presented in the stem.

Incorrect because the transition for ORANGE path integrity is not at 280°F and EOP-16.0 is not entered because of a concern for loss of shutdown margin.

D. Plausible because 280°F is the threshold for a yellow path procedure entry. The second part is plausible because the procedure addresses a potential thermal stresses that can cause a growth in a reactor vessel flaw.

Incorrect because the transition for ORANGE path integrity is not at 280°F.

K/A: WE12 EK3.1 Steam Line Rupture - Excessive Heat Transfer / 4 - Knowledge of the reasons for the following responses as they apply to the event: Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure and reactivity changes and operating limitations and reasons for these operating characteristics.

K/A Match: the KA is matched because the candidate must recall the temperature at which excessive thermal stresses could occur and that a procedure for excessive cooldown must be used after a steam line break.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.5 SRO 3.9
Technical Reference: **EOP-12.0 INTEGRITY STATUS TREE**
ERG Background document for EOP-16.0(FR-P.1)
ERG BACKGROUND FOR EOP-13.0(FR-S.1)

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-12.0 03. STATE the bases or reasons for each action contained in EOP-12.0 as well as each of the decision blocks of each Status Tree in EOP-12.0, Attachments 1 through 6. This should include, but not be limited to, the following:

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 43(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments (2011 NRC Exam):

Facility Response:

Comments;

19. Initial conditions:

- Mode 1.
- Control Bank "D" rods began stepping out in AUTO.

Current conditions:

- Power is 50% stable.
- The Immediate Actions of AOP-403.3, CONTINUOUS ROD MOTION are **complete**.
- T_{AVG} is 2.5 °F above $T_{REF.}$
- Control Bank D Rods are 12 steps above their pre-incident position.

Which ONE of the following is the **first** method that will be used for reducing T_{AVG} in accordance with AOP-403.3?

A✓ Adjust control rods with ROD CONTRL BANK SEL in MAN.

B. Adjust control rods with ROD CONTRL BANK SEL in CBD.

C. Perform a boration using SOP-106, REACTOR MAKEUP WATER SYSTEM.

D. Perform a boration while referring to AOP-106.1, EMERGENCY BORATION.

Revised without significant modification for 2018 NRC - rj

QUESTION USAGE:
RO-14-01-AOP-2 Exam
RO-05-01-PRACTICE-AUDIT-1 (2006-RO PRACTICE AUDIT)
RO-11-01-AOP-2 Exam

REVISION HISTORY:
Rev. 2 submitted by RJ - restructured question
Reviewed by: Danny Rhymer
Approved:
Rev. 1 Submitted by MRB 7/1/12 Removed "Rod motion stopped when the Bank Selector Switch was placed in MANUAL." from stem to prevent overlap issues with other questions. Changed choices so that grammatically match stem. Changed stem to remove teaching of 1°F.

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the method for controlling temperature with rods after a continuous rod withdrawal.

A. CORRECT. AOP-403.3 requires operators to adjust TAVG with control rods. Control rods were placed in manual in step 2.

B. Plausible because Control Bank D rods can be moved with the rods in individual bank select to restore rods to the previous position this will also adjust temperature.

   Incorrect because rods would remain in manual during the adjustment of temperature.

C. Plausible because this is a method used to adjust temperature during normal evolutions such as power changes and is also a method used in AOP-403.5 during a misaligned rod event and rods are misaligned when compared with their original positions.

   Incorrect because a boration is not required.

D. Plausible because AOP-106.1, EMERGENCY BORATION is used in AOP-403.5 for stuck or misaligned rods and rods are in manual and are misaligned when compared with their original positions.

   Incorrect because a boration is not required.

K/A: 001 AA1.02 Continuous Rod Withdrawal/1 - Ability to operate and / or monitor the following as they apply to the event: Rod in-out-hold switch

K/A Match: The KA is matched because the candidate must recall the method for controlling temperature with rods after a continuous rod withdrawal.

Selection criteria; **REVISED BANK**

Tier: 1 **Group:** 2
Importance Rating: RO 3.6 SRO 3.4
Technical Reference: **AOP-403.3**
AOP-403.5

Proposed references to be provided to candidates during examination: None

Learning Objective: AOP-403.3 04. GIVEN a set of plant conditions, DETERMINE the following: a. If control rod motion is caused by an event other than a valid rod motion signal. b. Plant response and appropriate operator actions in accordance with AOP-403.3.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

AOPS 337

Given the following plant conditions:

- Power was at 50%.
- Control Bank “D” rods began stepping out in Automatic.
- The operating crew is in AOP-403.3, Continuous Control Rod Motion.
- T_{avg} is 2.5 °F above $T_{ref.}$
- Rods are 12 steps above their pre-incident position.

Which ONE of the following is the method for reducing Tavg in accordance with step 5 of AOP-403.3?

A.✓ Manually insert control rods.

B. Manually raise turbine load.

C. Initiate normal boration.

D. Initiate emergency boration.

20. Given the following plant conditions:

- Core re-load was in progress.
- The Refueling SRO reported the following from the RB:
  - A fuel assembly appeared to be bound momentarily.
  - The fuel assembly was withdrawn approximately one foot
  - Bubbles were observed coming from the mast.
- The crew has entered AOP-123.3, POTENTIAL FUEL ASSEMBLY DAMAGE WHILE HANDLING FUEL.
- The Fuel Transfer Cart is on its way to the Fuel Handling Building.

Which ONE of the choices below completes the following statements?

In accordance with AOP-123.3, the Fuel Transfer Cart will be moved to the __(1)__.

This action will __(2)__.

A.✓ 1) Fuel Handling Building;
2) allow closure of the Fuel Transfer Tube Valve.

B. 1) Fuel Handling Building;
2) allow installation of the Fuel Transfer Tube blind flange.

C. 1) Reactor Building;
2) allow closure of the Fuel Transfer Tube valve.

D. 1) Reactor Building;
2) provide the upender as a safe location for the damaged fuel assembly.

Modified aops734 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 Submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question the candidate must recall the reason for moving the fuel transfer cart to the Fuel Handling Building during a refueling accident.

A. CORRECT. The Fuel Transfer Cart will be transferred back to the Fuel Handling Building to allow for closure of the Fuel Transfer Tube Valve.

B. The first part is correct. The Fuel Transfer Cart will be transferred back to the Fuel Handling Building to allow for closure of the Fuel Transfer Tube Valve. The second part is plausible because the Fuel Transfer tube blind flange would provide isolation equivalent to closing the fuel transfer tube and cannot be installed unless the fuel transfer cart has been withdrawn to the Fuel Handling Building.

   Incorrect because the blind flange is not the method used for isolation.

B. Plausible because moving the cart is done to enable closure of the fuel transfer tube in AOP-123.3.

   Incorrect because the cart is not moved to the Reactor Building.

C. Plausible because operators are directed to place any assemblies in transit in a safe location and the unpender is one such location and additionally, moving an assembly to the Fuel Handling building for safety would require moving the cart to the RB if not already there.

   Incorrect because the cart is not moved to the Reactor Building.

K/A: 036 AK3.03 Fuel Handling Accident! 8 - Knowledge of the reasons for the following responses as they apply to the event: Guidance contained in EOP for fuel handling incident

K/A Match: The KA is matched because the candidate must recall the reason for moving the fuel transfer cart to the Fuel Handling Building during a refueling accident.

Selection criteria; **MODIFIED FROM AOPS734**

Tier: 1 **Group:** 2
Importance Rating: RO 3.7 SRO 4.1
Technical Reference: **AOP-123.3**
OAP-108.4

Proposed references to be provided to candidates during examination: None

Learning Objective: AOP-123.3 05. STATE/IDENTIFY the bases for the overall mitigating strategies and the bases for the steps, notes, or cautions (as applicable), for AOP-123.3.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

AOPS 734

Given the following plant conditions:

- Core re-load is in progress.
- A fuel assembly was dropped.
- RM-A4, RB PURGE EXH PARTICULATE (IODINE)(GAS) ATMOS MONITOR, is in alarm.
- The Fuel Transfer Cart is in the RB.
- The crew has entered AOP-123.3, POTENTIAL FUEL ASSEMBLY DAMAGE WHILE HANDLING FUEL.

Which ONE of the choices below completes the following statements in accordance with AOP-123.3?

The fuel transfer cart will ___(1)___ and the RB Coordinator ___(2)___ need to initiate Containment Closure.

A. 1) remain in the RB
   2) will **not**

B. 1) remain in the RB
   2) will

C. 1) be transferred back to the Fuel Handling Building
   2) will **not**

D✓ 1) be transferred back to the Fuel Handling Building
   2) will

21. Given the following plant conditions:

- 100% power.
- A steam generator tube leak is in progress.
- Operators entered AOP-112.2, STEAM GENERATOR TUBE LEAK NOT REQUIRING SI.
- Automatic systems have stabilized plant conditions without operator action.
- Operators have determined that the leakrate is stable.

Which ONE of the choices below answers both of the following questions?

Operators can use a computer calculation based on radiation levels measured by __(1)__ to determine the need for plant shutdown.

The lowest **stable** leakrate at which a plant shutdown is required is __(2)__ gallons per day.

A. 1) RM-G19A, B,C Main Steamline Radiation Monitors
   2) 150

B. 1) RM-G19A, B,C Main Steamline Radiation Monitors
   2) 75

C. 1) RM-A9 Main Condenser Exhaust.
   2) 150

D✓ 1) RM-A9 Main Condenser Exhaust.
   2) 75

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the instrumentation required to determine a steam generator tube leak rate.

A. The first part is plausible the RM-A19s are a monitor used to determine entry in AOP-112.2 and monitor radiation in the secondary. The second part is plausible because 150 gpm is a threshold for determining the rate of shutdown in AOP-112.2.

   Incorrect because the leakrate calculation is not based on the RM-G19s and 150 gpd is not the minimum leakrate at which plant shutdown is required,

B. The first part is plausible the RM-A19s are a monitor used to determine entry in AOP-112.2 and monitor radiation in the secondary. The second part is correct. 75 gpd is the minimum stable leakrate at which a plant shutdown is required.

   Incorrect because the leakrate calculation is not based on the RM-G19s.

C. The first part is correct. A computer calculation based on RM-A9 is used in AOP-112.2 to determine leakrate. The second part is plausible because 150 gpd is a threshold for determining the rate of shutdown in AOP-112.2.

   Incorrect because 150 gpm is not the minimum leakrate at which plant shutdown is required,

D. CORRECT. A computer calculation based on RM-A9 is used in AOP-112.2 to determine leakrate. 75 gpd is the minimum stable leakrate at which a plant shutdown is required.

K/A: 037 AA2. 12 Steam Generator Tube Leak / 3 - Ability to determine and interpret the following as the apply to the event: Flow rate of leak

K/A Match: The KA is matched because the candidate must recall the instrumentation required to determine a steam generator tube leak rate.

Selection criteria; NEW

Tier: 1 **Group:** 2
Importance Rating: RO 3.3 SRO 4.1
Technical Reference: **AOP-112.2**

Proposed references to be provided to candidates during examination: None

Learning Objective: AOP-112.2 04. Given a set of plant conditions, DETERMINE the following for AOP-112.2: a. Requirements for a power reduction or unit shutdown based on primary to secondary leakage

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10) and (11)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

22. Given the following plant conditions:

- Operators manually tripped the Reactor from the Main Control Board before evacuating the Control Room.
- Two (2) control rods did **not** fully insert on the Reactor trip.
- AOP-600.1, CONTROL ROOM EVACUATION is in progress.
- Operators are referring to AOP-106.1 EMERGENCY BORATION and aligning for boration in Step 4, "Emergency Borate by gravity drain of the Boric Acid Tank."

Which ONE of the choices below completes the following statements?

Based on the given conditions, AOP-600.1 requires operators to borate ___(1)___ .

The Operators are cautioned in AOP-106.1 to monitor Charging pump amps and flow carefully while performing the gravity drain alignment because ___(2)___ must be closed in that lineup.

A.✓ 1) a fixed amount of 2500 gallons.
2) LCV-115C(E), VCT OUTLET ISOL valves

B. 1) a fixed amount of 2500 gallons.
2) XVD08329–CS, CHARGING/SI PUMPS SUCTION HEADER VALVE and XVD08331–CS, CHARGING/SI PUMPS SUCT HDR ISOL VALVE

C. 1) an amount determined by a Shutdown Margin calculation based on temperature.
2) LCV-115C(E), VCT OUTLET ISOL valves

D. 1) an amount determined by a Shutdown Margin calculation based on temperature.
2) XVD08329–CS, CHARGING/SI PUMPS SUCTION HEADER VALVE and XVD08331–CS, CHARGING/SI PUMPS SUCT HDR ISOL VALVE

Proposed for use on 2018 NRC - rj

QUESTION USAGE
ILO 13-01 NRC (2015 Q23)

The following is as entered by the NRC.
Answer Choice Analysis

A. CORRECT. Per AOP-600.1, the given boration amount of 2500 gallons corresponds to that required if only two control rods have not fully inserted following a reactor trip and operators are cautioned to monitor charging pump amps during closure of LCV-115C and E.

B. The first part is correct. Per AOP-600.1, the given boration amount of 2500 gallons corresponds to that required if only two control rods have not fully inserted following a reactor trip and . The second half answer proposes valves on the suction of the charging pumps that are opened during the alignment to gravity drain and would not cause a loss of suction source.

Incorrect because XVD08329–CS, CHARGING/SI PUMPS SUCTION HEADER VALVE and XVD08331–CS, CHARGING/SI PUMPS SUCT HDR ISOL VALVE are opened rather than closed and will not cause a loss of suction.

C. The first half is plausible because other procedures, such as GOP-8, PLANT SHUTDOWN FROM HOT STANDBY TO COLD SHUTDOWN WITH THE CONTROL ROOM INACCESSIBLE MODE 3 TO MODE 5 will require use of STP-134.001 and will calculate boration based on temperature. The second part of the answer is correct; Operators are cautioned to monitor charging pump amps during closure of LCV-115C and E

Incorrect because AOP-600.1 requires a boration of 2500 gallons.

D. INCORRECT. The first half answer is plausible but incorrect; GOP-8, PLANT SHUTDOWN FROM HOT STANDBY TO COLD SHUTDOWN WITH THE CONTROL ROOM INACCESSIBLE MODE 3 TO MODE 5 will require use of STP-134.001 and will calculate boration based on temperature. The second part of the answer is plausible because

Incorrect because AOP-600.1 requires a boration of 2500 gallons and because XVD08329–CS, CHARGING/SI PUMPS SUCTION HEADER VALVE and XVD08331–CS, CHARGING/SI PUMPS SUCT HDR ISOL VALVE are opened rather than closed and will not cause a loss of suction.

Supporting References
1. AOP-600.1, Revision 2, Page 9
2. GOP-8, Revision 6, Page 4

References Provided to candidate
None.

K/A: 068 AA2.02 Control Room Evac. / B - Ability to determine and interpret the following as the apply to the event: Local boric acid flow

K/A Match: The KA is matched because the candidate must recall the amount of boric acid that must be injected due to two stuck control rods during a control room evacuation.

Selection criteria; **BANK**

Tier: 1 **Group:** 2
Importance Rating: RO 3.7 SRO 4.2
Technical Reference: **AOP-600.1**
AOP-106.1

Proposed references to be provided to candidates during examination: None

Learning Objective: AOP-600.1 03. Given a set of plant conditions, DETERMINE the plant response and appropriate operator actions in accordance with AOP-600.1

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

23. Initial conditions:

- A small break LOCA occurred.
- Operators turned off all RCPs during the performance of EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION.

Current conditions:

- The crew is performing actions contained in EOP-2.1, ES-1.2 POST- LOCA COOLDOWN AND DEPRESSURIZATION.
- "A" and "B" Charging pumps are running in the injection mode.
- RCS pressure is 800 psig and stable.
- RCS subcooling is 75°F and increasing.
- Pressurizer level is 30% and stable.
- The conditions required to start and run RCPs in accordance with SOP-101, REACTOR COOLANT SYSTEM are present.

Which ONE of the following describes the operation of RCP(s) during the implementation of EOP-2.1?

A. RCPs will be left OFF.

B✓ **Only** "A" RCP will be restarted.

C. **Only** "B" and "C" RCPs will be restarted.

D. "A", "B" and "C" RCPs will be restarted.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the method for operating reactor coolant pumps during a post loca cooldown and depressurization.

A. Plausible because EOP-2.1 can be performed without RCPs running and the pumps were manually shut off by operators.

   Incorrect because "A" RCP will be started.

B. CORRECT. EOP-2.1 will start RCP "A".

C. Plausible because starting two pumps would cause greater RCS circulation and would provide acceptable spray flow.

   Incorrect because "A" RCP will be started.

D. Plausible three RCPs running is the normal configuration at power.

   Incorrect because only "A" RCP will be started.

K/A: WE03 EK2.2 LOCA Cooldown - Depress. / 4 - Knowledge of the interrelations between the event and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems and relations between the proper operation of these systems to the operation of the facility.

K/A Match: The KA is matched because the candidate must recall the method for operating reactor coolant pumps during a post loca cooldown and depressurization.

Selection criteria; NEW

Tier: 1 **Group:** 2
Importance Rating: RO 3.7 SRO 4.0
Technical Reference: **EOP-2.1**
EOP-1.0

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.1 06. APPLY EOP-2.1 by predicting a discrete path through EOP-2.1 given a set of plant conditions

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

24. Given the following plant conditions:

- A loss of coolant accident has occurred one hour ago.
- EOP-2.0, E-1 LOSS OF REACTOR OR SECONDARY COOLANT is in progress.
- RB Spray pumps did not start automatically or manually.
- RB pressure is 31 psig and decreasing.
- NR RVLIS is 32% and decreasing.
- Core Exit TCs are 670°F and increasing.
- RCS pressure is 31 psig and decreasing.
- "A" and "B" RCS Loop T_{COLD} Indications at 350°F and decreasing.
- All RCPs are off.
- Power is in the source range at 10^2 cps and increasing.

Based **only** on the conditions above, which ONE of the choices below is a complete list of **all** the Critical Safety Functions that are **either in a RED or ORANGE** status,

A. Subcriticality and Integrity.

B. Subcriticality and Core Cooling.

C.✓ Containment and Core Cooling.

D. Containment and Integrity.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must assess parameters to determine the status of core cooling and other critical safety functions.

A. Plausible because source range counts are increasing and because TCOLDs are have decreased to 350°F in one hour which is in excess of 100°F/hr. Exceeding this cooldown rate is a question asked on the integrity status tree.

Incorrect because neither Subcriticality or Integrity CSFs are in a RED or ORANGE status.

B. Plausible because source range counts are increasing and because Core Cooling is in an ORANGE status due to NR RVLIS level <34%.

Incorrect because the Subcriticality CSF is not in a RED or ORANGE status.

C. CORRECT. Containment is in ORANGE status because RB pressure is greater that 12 psig with no spray flow and Core Cooling is in an ORANGE status due to NR RVLIS level < 34%.

D. Plausible because Containment is in ORANGE status because RB pressure is greater that 12 psig with no spray flow and because TCOLDs are have decreased to 350°F in one hour which is in excess of 100°F/hr. Exceeding this cooldown rate is a question asked on the integrity status tree.

Incorrect because the Integrity CSF is not in a RED or ORANGE status.

K/A: WE06 EG2.4.21 Degraded Core Cooling /4 - Knowledge of the parameters and logic used to assess the status of safety functions

K/A Match: The KA is matched because the candidate must assess parameters to determine the status of core cooling and other critical safety functions.

Selection criteria; NEW

Tier: 1 **Group:** 2
Importance Rating: RO 4.0 SRO 4.6
Technical Reference: **EOP-12.0, MONITORING OF CRITICAL SAFETY FUNCTIONS**

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-12.0 02. STATE the entry conditions of EOP-12.0 1. Symptoms 2. Transitions from other procedures.

Question Cognitive Level: **Memory or Fundamental Knowledge __X__**
Comprehension or Analysis _____

10 CFR Part 55 Content: 43(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

25. Given the following plant conditions:

- 100% power initially.
- A small break LOCA occurred.
- EOP-16.0 RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK was implemented due to Critical Safety Function status.
- Safety injection has been terminated.
- All RCPs have been stopped.
- The crew is performing step 12, **"Verify SI flow is NOT required".**
- The following indications are present on the main control board:
    - Pressurizer level is 0%.
    - NR RVLIS is 50% and stable.
    - WR RVLIS is 15% and stable.
    - Subcooling is 70°F.

Which ONE of the following identifies whether SI will be re-initiated in accordance with EOP-16.0 and, if so, the reason for that determination?

A. Will **not** be re-initiated.

B.✓ Will be initiated as required due to a low NR RVLIS indication.

C. Will be initiated as required due to a low WR RVLIS indication.

D. Will be initiated as required due to a low Pressurizer level indication.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the procedural requirement for re-initiating SI during a PTS event.

A. Plausible because subcooling is 70°F which is greater than the typical adverse containment 67.5°F requirement for re-initiation of SI in other procedures.

   Incorrect because SI will be re-initiated due to a low narrow range RVLIS level.

B. CORRECT. 50% NR RVLIS with no RCPs running requires reinitiation of SI in accordance with the reference page of EOP-16.0.

C. Plausible because WR RVLIS would be used for re-initiation if RCPs were running and 15% is lower than any value of WR RVLIS for this determination.

D. Plausible because Pressurizer level is 0% and the typical re-initiation criterion is 10%.

   Incorrect because Pressurizer level is not used for re-initation criteria in EOP-16.0.

K/A: WE08 EK3.2 RCS Overcooling - PTS /4 - Knowledge of the reasons for the following responses as they apply to the event: Normal, abnormal and emergency operating procedures associated with (Pressurized Thermal Shock).

K/A Match: The KA is matched because the candidate must determine the procedural requirement for re-initiating SI during a PTS event.

Selection criteria; NEW

Tier: 1 **Group:** 2
Importance Rating: RO 3.6 SRO 4.0
Technical Reference: **EOP-16.0**

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-16.0 04. STATE the bases or reasons for each action contained in EOP-16.0. This should include, but not be limited to, the following: 3. Conditions requiring SI reinitiation

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

26. Given the following conditions:

- A large break LOCA has occurred.

NOTE:

- **<u>Refer to the circled instrumentation in the reference provided</u>**.

Which ONE of the choices below answers both of the following questions:

1) Which instrument set can be used to determine whether an ORANGE status exists on a Critical Safety Function?

2) What is the concern if that ORANGE status is present?

REFERENCE PROVIDED

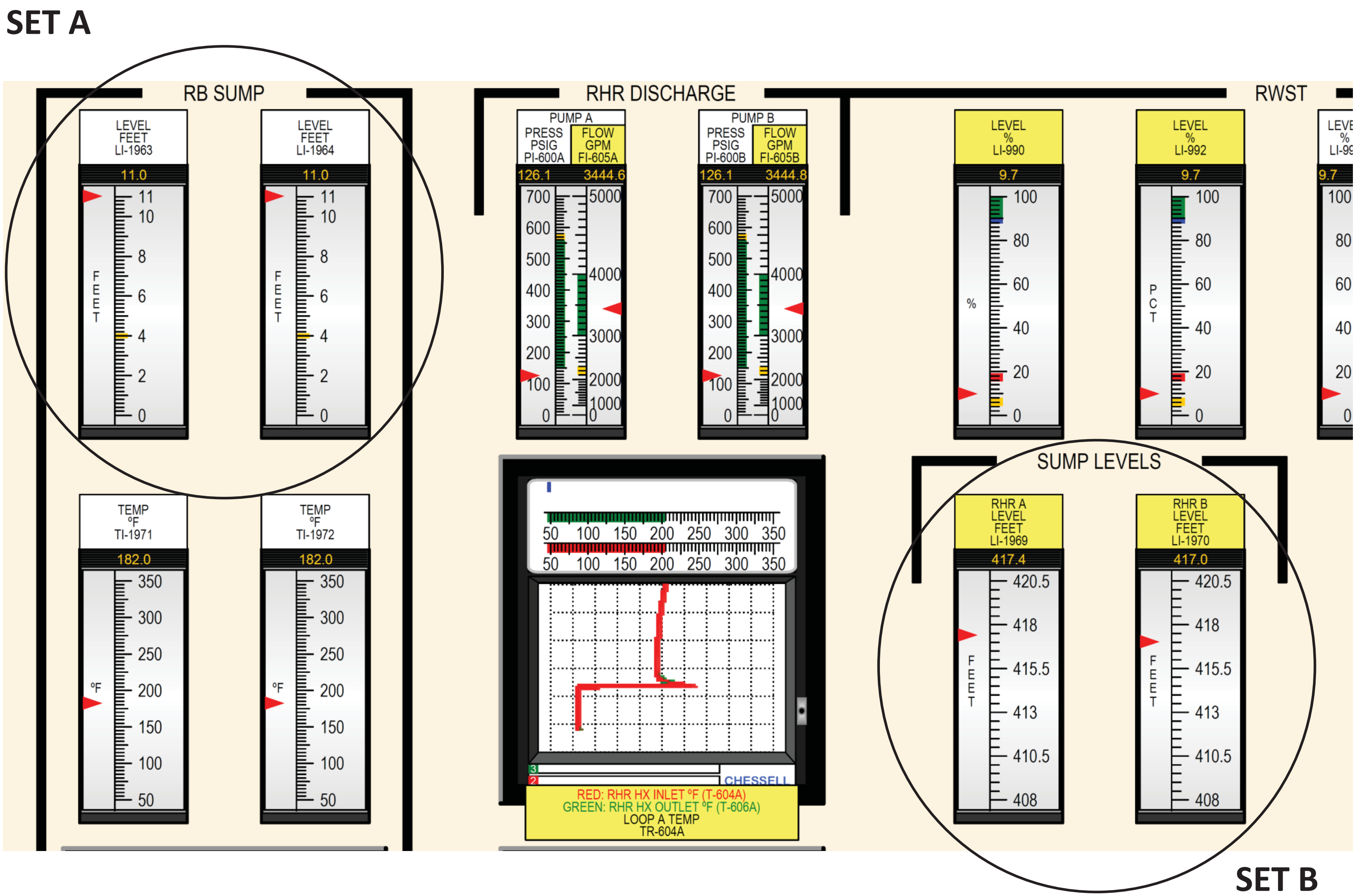
A. 1) Set A.
   2) Equipment in the RB that is needed for long-term recovery may become submerged.

B. 1) Set A.
   2) The level required for long-term recirculation cooling of the RCS may not be met.

C.✓ 1) Set B.
   2) Equipment in the RB that is needed for long-term recovery may become submerged.

D. 1) Set B.
   2) The level required for long-term recirculation cooling of the RCS may not be met.

Q #26


SET A
RB SUMP
LEVEL FEET LI-1963
11.0
LEVEL FEET LI-1964
11.0
TEMP °F TI-1971
182.0
TEMP °F TI-1972
182.0
RHR DISCHARGE
PUMP A
PRESS PSIG PI-600A
126.1
FLOW GPM FI-605A
3444.6
PUMP B
PRESS PSIG PI-600B
126.1
FLOW GPM FI-605B
3444.8
RWST
LEVEL % LI-990
9.7
LEVEL % LI-992
9.7
SUMP LEVELS
RHR A LEVEL FEET LI-1969
417.4
RHR B LEVEL FEET LI-1970
417.0
CHESSELL
RED: RHR HX INLET °F (T-604A)
GREEN: RHR HX OUTLET °F (T-606A)
LOOP A TEMP
TR-604A
SET B

PROVIDE GRAPHIC OF MAIN CONTROL BOARD PANEL XCP6106.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the instrumentation used to determine an orange path due to containment flooding and the concern due to that status.

A. The first part is plausible. A level indication for a sump inside the RB is used to determine an ORANGE path for the Containment CSF due to flooding. The second part is correct. The concern is submergence of required equipment.

   Incorrect because the RB Sump level indications are not an input to the Containment CSF Status Tree.

B. The first part is plausible. A level indication for a sump inside the RB is used to determine an ORANGE path for the Containment CSF due to flooding. The second part is plausible because an RHR sump level is used to determine if sufficient sump level exists to supply suction to RHR during recirculation.

   Incorrect because the RB Sump level indications are not an input to the Containment CSF Status Tree and the concern for an ORANGE status is not recirculation suction.

C. CORRECT. The RHR sump level indications are used to determine an ORANGE path for the Containment CSF due to flooding. The concern is submergence of required equipment.

D. The first part is correct. The RHR sump level indications are used to determine an ORANGE path for the Containment CSF due to flooding. The second part is plausible because an RHR sump level is used to determine if sufficient sump level exists to supply suction to RHR during recirculation.

   Incorrect because the concern for an ORANGE status is not recirculation suction

K/A: WE15 EK2.1 Containment Flooding / 5 - Knowledge of the interrelations between the event and the following: Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features.

K/A Match: The KA is matched because the candidate must recall the instrumentation used to determine an orange path due to containment flooding and the concern due to that status.

Selection criteria; NEW

Tier: 1 **Group:** 2
Importance Rating: RO 2.8 SRO 2.9
Technical Reference: **EOP-12.0, CONTAINMENT CSFST**
ERG BASES - F0.5

Proposed references to be provided to candidates during examination:

GRAPHIC OF MAIN CONTROL BOARD PANEL XCP6106.

Learning Objective: EOP-12.0 02. STATE the entry conditions of EOP-12.0 1. Symptoms 2. Transitions from other procedures.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

27. Initial conditions:

- A LOCA has occurred.
- EOP-2.0, E-1 LOSS OF REACTOR OR SECONDARY COOLANT.
- The BOP is controlling Steam Generator (SG) levels.
- NR SG levels are 20% and increasing.

Current conditions:

- The following annunciator have come into alarm:
  - XCP-642, 3-1, RB AREA RM-G18 HI RAD
  - XCP-645, 4-5, RB AREA RM-G7 HI RAD
- RM-G7 and RM-G18 read 1200 R/hr and increasing.
- IPCS is **not** available, and engineering is **not** evaluating integrated dose.
- Containment pressure is 2.8 psig and increasing.
- Total EFW flow is 375 gpm.
- NR SG levels are 35% and stable.

Which ONE of the choices below complete the following statements in accordance with OAP-103.4, EOP/AOP USER'S GUIDE?

In the **current conditions**, adverse containment setpoint values ___(1)___ required to be used in the Emergency Operating Procedures (EOPs).

In the **current conditions**, the BOP ___(2)___ required to raise EFW flow.

A✓ 1) are
2) is

B. 1) are
2) is **not**

C. 1) are **not**
2) is

D. 1) are **not**
2) is **not**

Modified eops 2 2 for 2018 NRC - rj

QUESTION USAGE

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the applicant must assess radiation monitor indications to determine adverse containment and determine the effect on a critical safety function.

A. CORRECT. With IPCS not in service, adverse containment values must be used when containment radiation exceeds 1000 R/hr. In the current condition, the BOP must raise EFW flow to bring flow rate greater than the 450 gpm required by the Heat Sink CSF.

B. The first part is correct. With IPCS not in service, adverse containment values must be used when containment radiation exceeds 1000 R/hr. The second part is plausible because SG levels are greater than 26% and the candidate may not recall the adverse value of 41%.

   Incorrect because EFW flow must be raised.

C. The first part is plausible because RB pressure is less than the 3.6 value for adverse containment. The second part correct, but also plausible in combination with the first part because EOPs give a high value for SG level control of 60%.

   Incorrect because adverse containment is in effect due to radiation.

D. The first part is plausible because RB pressure is less than the 3.6 value for adverse containment. The second part is plausible because SG levels are greater than the non-adverse minimum value of 26%.

   Incorrect because adverse containment is in effect due to radiation and because EFW flow must be raised.

K/A: WE16 EK1.3 High Containment Radiation / 9 - Knowledge of the operational implications of the following concepts as they apply to the event: Annunciators and conditions indicating signals, and remedial actions associated with the (High Containment Radiation).

K/A Match: The KA is matched because the candidate must assess radiation monitor indications to determine adverse containment and determine the effect on a critical safety function.

Selection criteria; **MODIFIED FROM EOPS 2 2**

Tier: 1 **Group:** 2
Importance Rating: RO 3.0 SRO 3.3
Technical Reference: **OAP-103.4, EOP/FSP/AOP USER'S GUIDE**
EOP-2.0

Proposed references to be provided to candidates during examination: None

Learning Objective: OAP-103.4 06. DEFINE "Adverse Containment" and EXPLAIN its implications.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

EOPS 2 002

Given the following plant conditions:

Time 03:45

- A LOCA has occurred.

Time 0400:

- Containment radiation level is 1200 R/hr and increasing slowly.
- Containment pressure is 2.8 psig and increasing.
- IPCS is **not** available, and engineering can **not** determine integrated dose.

Time 0600:

- Containment radiation level is 900 R/hr and decreasing slowly.
- Containment pressure peaked at 12 psig, and is now 2.8 psig and decreasing.
- IPCS is **not** available, and engineering can **not** determine integrated dose.

Which ONE of the choices below complete the following statements in accordance with OAP-103.4, EOP/AOP USER'S GUIDE?

At 0400, adverse containment setpoint values ___(1)___ required to be used in the Emergency Operating Procedures (EOPs) for designated parameters.

At 0600, adverse containment setpoint values ___(2)___ required to be used in the EOPs for designated parameters.

A.✓ 1) are
2) are

B. 1) are
2) are **not**

C. 1) are **not**
2) are

D. 1) are **not**
2) are **not**

28. Initial condition:

- 90% power initially.

Current condition:

- A failure of a distance relay in the electrical system occurred.
- XCB 8902, MAIN XFMR FEED tripped open.
- Power range Channels N-41 through N-44 are 91% and increasing at 5%/second.

Which ONE of the choices below completes the following statements?

Power is increasing because of reactivity inserted by an __(1)__.

The reactor will trip due to a __(2)__ trip.

A. 1) increase in RCP frequency.
   2) Overpower ΔT

B.✓ 1) increase in RCP frequency.
   2) Power Range Neutron Flux High Positive rate

C. 1) actuation of steam dumps.
   2) Overpower ΔT

D. 1) actuation of steam dumps.
   2) Power Range Neutron Flux High Positive rate

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine that an increase in RCP frequency will increase power and cause a power positive rate trip.

A. The first part is correct. Power is increasing because of reactivity inserted by an increase in RCP frequency. The second part is plausible because the overpower ΔT protective function protects against a high linear heat generation in conjunction with a power increase.

   Incorrect because the Overpower ΔT trip will not occur.

B. CORRECT. Power is increasing because of reactivity inserted by an increase in RCP frequency. When this event occurred at V.C. Summer, a Power Range Neutron flux High Positive rate trip occurred. The stem includes a rate in excess of the setpoint of 5% power range increase in 2 seconds.

C. The first part is plausible steam dumps are expected to actuate for a loss of load and the candidate could conclude that an oversteaming event will occur.

   Incorrect because the Power Range Neutron flux High Positive Rate trip will occur.

D. The first part is plausible.

   Incorrect because

K/A: 003 K4.02 Reactor Coolant Pump - Knowledge of system design feature(s) and or interlock(s) which provide for the following: Prevention of cold water accidents or transients

K/A Match: The KA is matched because the candidate must determine that an increase in RCP frequency will increase power and cause a power positive rate trip.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 2.5 SRO 2.7
Technical Reference: **EOP-1.0**
GS-1, SERVICE POWER SYSTEM

Proposed references to be provided to candidates during examination: None

Learning Objective: GS-1 021. Using selected operating experiences related to this course, DESCRIBE their applicability to your job, their significance to plant operations, and which of the Human Performance tools could have been used to prevent or mitigate the event.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(5) and (7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

29. Given the following plant conditions:

- 100% power.
- VCT level is 45% and decreasing.
- **No** automatic or manual makeup to the VCT is in progress.

Which ONE of the choices below completes the following statements?

Without operator action, an automatic makeup to the VCT is expected to start when VCT level decreases to __(1)__.

VCT level transmitter __(2)__ provides the signal to start the automatic makeup initiation.

A. 1) 40%.
   2) LT-112.

B. 1) 40%.
   2) LT-115.

C. 1) 20%.
   2) LT-112.

D✓ 1) 20%.
   2) LT-115.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the instrument and setpoint at which a makeup to the VCT occurs.

A. The first part is plausible because the auto makeup stops at 40%. The second part is plausible because it monitors VCT level and performs alarm and interlock functions.

   Incorrect because the auto makeup does not start at 40% and LT-112 does not provide the input for this function.

B. The first part is plausible because the auto makeup stops at 40%. The second part is correct.

   Incorrect because the auto makeup does not start at 40%.

C. The first part is correct. The automatic makeup begins at 20%.The second part is plausible because it monitors VCT level and performs alarm and interlock functions.

   Incorrect because LT-112 does not provide the input for this function.

D. CORRECT. The automatic makeup begins at 20% as detected by LT-115.

K/A: 004 A4.12 Chemical and Volume Control - Ability to manually operate and/or monitor in the control room: Boration/dilution batch control

K/A Match: The KA is matched because the candidate must recall the instrument and setpoint at which a makeup to the VCT occurs.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 3.8 SRO 3.3
Technical Reference: **SOP-106**
1MS-51-032, Sh. 23

Proposed references to be provided to candidates during examination: None

Learning Objective: AB-5 09. DESCRIBE the following Reactor Makeup System Interlocks. Specify purpose and setpoints: 3. Reactor makeup control system

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

30. Given the following plant conditions:

- GOP-6, PLANT SHUTDOWN FROM HOT STANDBY TO COLD SHUTDOWN MODE 3 TO MODE 5 is in progress.
- RCS temperature is 350°F and lowering.
- Operators will use "A" RHR train to complete the RCS cooldown to Mode 5.
- The following valves are currently **closed**:

  MVG-8701A, RCS LP A TO PUMP A.
  MVG-8701B, RCS LP C TO PUMP B.
  MVG-8702A, RCS LP A TO PUMP A.
  MVG-8702B, RCS LP C TO PUMP B.

Which ONE of the choices below completes the following statements?

The **highest** RCS temperature at which Cold Overpressure Protection is required to be placed in service is ___(1)___ in accordance with T.S. 3.4.9.3, REACTOR COOLANT SYSTEM - OVERPRESSURE PROTECTION SYSTEMS.

When RHR is placed in service, guidance in GOP-6 will ensure that T.S. 3.4.9.3 is met by requiring operators to open a **minimum** of ___(2)___ .

A. 1) 200°F.
   2) MVG-8701A and MVG-8702A.

B. 1) 200°F.
   2) MVG-8701A, MVG-8702A, MVG-8701B and MVG-8702B.

C. 1) 300°F.
   2) MVG-8701A and MVG-8702A.

D✓ 1) 300°F.
   2) MVG-8701A, MVG-8702A, MVG-8701B and MVG-8702B.

Modified rhr172 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the alignments of RHR required for putting cold overpressure protection in service.

A. The first part is plausible because cold overpressure protection is placed in service at a reduced temperature for brittle fracture concerns. The second part is plausible because opening one set of valves places one relief in service which provides the required relief capacity for protection.

   Incorrect because cold overpressure protection is required above 200°F and opening MVG-8701A and MVG-8702A will not place the minimum equipment in service in accordance with T.S. 3.4.9.3.

B. The first part is plausible because cold overpressure protection is placed in service at a reduced temperature for brittle fracture concerns. The second part is correct.

   Incorrect because cold overpressure protection is required above 200°F.

C. The first part is correct. The second part is plausible because opening one set of valves places one relief in service which provides the required relief capacity for protection.

   Incorrect because opening MVG-8701A and MVG-8702A will not place the minimum equipment in service in accordance with T.S. 3.4.9.3.

D. CORRECT. The highest temperature for placing cold overpressure protection in service per T.S> 3.4.9.3 is 300°F. Opening MVG-8701A, MVG-8702A, MVG-8701B and MVG-8702B will place 2 reliefs in service as required by the specification..

K/A: 005 A2.02 Residual Heat Removal - Ability to (a) predict the impacts of the following on the system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown

K/A Match: The KA is matched because the candidate must recall the alignments of RHR required for putting cold overpressure protection in service.

Selection criteria; **MODIFIED FROM RHR172**

Tier: 2 **Group:** 1
Importance Rating: RO 3.5 SRO 3.7
Technical Reference: **T.S. 3.4.9.3**
GOP-6

Proposed references to be provided to candidates during examination: None

Learning Objective: SB-4 015 Given a limiting condition for operation and a mode, DEFINE the requirements to satisfy the LCO, the actions if required within one hour or less, and describe the bases for the LCO.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

RHR SYSTEM 172

Given the following plant conditions:

- Plant cooldown is in progress in accordance with GOP-6, PLANT SHUTDOWN FROM HOT STANDBY TO COLD SHUTDOWN MODE 3 TO MODE 5.
- RCS temperature is 330°F and lowering.

Which ONE of the choices below completes the following statements?

The **highest** RCS temperature at which Cold Overpressure Protection is required to be placed in service is ___(1)___ in accordance with 3.4.9.3, REACTOR COOLANT SYSTEM - OVERPRESSURE PROTECTION SYSTEMS.

When XVR08708A-RH, RH INLET HEADER A RELIEF VALVE is placed in service, it will lift at a setpoint of ___(2)___ .

A. 2) 200°F.
   1) 450 psig

B. 1) 200°F.
   2) 600 psig

C. ✓ 2) 300°F.
   1) 450 psig

D. 1) 300°F.
   2) 600 psig

31. Initial conditions:

- 100% power initially.
- A small break loss of coolant accident occurred.
- "C" Charging pump is tagged **out of service**.
- RCS pressure is 500 psig and stable.
- The following pumps could **not** be started automatically or manually.

  "A" RHR pump
  "A" Charging pump
- Operators have **completed** alignment of long-term recirculation using EOP-2.2, ES-1.3 TRANSFER TO COLD LEG RECIRCULATION.

Current condition:

- The NROATC reports that "B" RHR Pump has **tripped**.
- RWST level is 9%.
- RCS pressure is 500 psig and stable.
- Operators are attempting to restore ESF pumps to operation.

Which ONE of the choices below answers both of the following questions?

1) Is there currently pumped flow to the core?

2) What is a **minimum** "A" Train pump restoration that would allow operators to **increase** pumped flow to the core with alignments established in EOP-2.2 ?

A. 1) Yes.
2) "A" Charging pump **only**.

B. 1) Yes.
2) **Either** "A" RHR pump or "A" Charging Pump.

C. 1) No.
2) "A" RHR Pump **only**.

D✓ 1) No.
2) **Both** "A" RHR pump **and** "A" Charging pump.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 Submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must determine the effect of a loss of ESF pumps on core cooling and the restoration that is required.

A. The first part is plausible because before alignments of EOP-2,2 are completed, the Charging pumps continue to draw from the RWST so the candidate my think that B pump is still pumping. The second part is plausible because with RCS pressure at 500 psig, only a Charging pump provides the necessary head to provide flow.

   Incorrect because there is not pumped flow to the core for the given conditions.

B. The first part is plausible because before alignments of EOP-2,2 are completed, the Charging pumps continue to draw from the RWST so the candidate my think that B pump is still pumping. The second part is plausible because the candidate may assume that either pump provides the necessary head to provide flow.

   Incorrect because there is not pumped flow to the core for the given conditions.

C. The first part is correct. The second part is plausible because the candidate may assume that starting "A" RHR pump will provide the required head pressure and is the minimum restoration required.

   Incorrect because starting only "A" RHR pump will not provide additional flow.

D. CORRECT. With "A" and "B" RHR pumps recirculation has been lost. In order to increase flow to the core from "A" train, both "A" RHR pump to provide suction from the RHR sump and "A" Charging pump to overcome RCS pressure are required.

K/A: 006 K6.13 Emergency Core Cooling — - Knowledge of the of the effect of a loss or malfunction on the following will have on the system: Pumps

K/A Match: the K/A is met because the candidate must determine the effect of a loss of ESF pumps on core cooling and the restoration that is required.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 2.8 SRO 3.1
Technical Reference: **EOP-2.2**
EOP-2.1
AB-10 EMERGENCY CORE COOLING SYSTEM

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.2 07. RELATE any systems/components operation, indication, or malfunction to its effect on EOP-2.2

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(8)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

32. Initial conditions:

- 100% power initially.
- A loss of all electrical load occurred.
- Pressurizer PORVs lifted during the transient.

Current conditions:

- XCP-616, 4-4 PRT LVL LO/TEMP/LVL/PRESS HI is in alarm.
- PRT parameters are as follows:

Level	84% and stable.
Pressure	7 psig and stable.
Temperature	114°F and stable.

- The CRS has directed the NROATC to restore PRT conditions as necessary to clear the XCP-616, 4-4.

Consider the following three actions:

(1) Drain the PRT.
(2) Vent the PRT.
(3) Recirculate the PRT contents through the RCDT heat exchanger.

Which ONE of the choices below identifies the actions that the NROATC **must** take to clear XCP-616, 4-4 in accordance with SOP-101, REACTOR COOLANT SYSTEM?

Assume that ambient heat loss will not occur.

A. (1) and (2) **only**.

B.✓ (1) and (3) **only**.

C. (2) and (3) **only**.

D. (1), (2) and (3).

Modified rcs96 for 2018 NRC - rj

REVISION HISTORY:

REVISION HISTORY:

Rev. 0 submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:

In order to answer this question correctly, the candidate must recall PRT alarm setpoints and design features and operation of systems that can drain and cool the Pressurizer Relief tank contents.

A. Plausible because (1) is correct and (2) would be correct for a pressure greater than 8 psig.

   Incorrect because pressure is not in excess of the alarm setpoint.

B. CORRECT. The PRT must be drained to reduce level less than 83% and must be recirculated and cooled to reduce temperature less than 113°F.

C. Plausible because venting would be correct if pressure were greater than 8 psig and because (3) is correct.

   Incorrect because pressure is not in excess of the alarm setpoint.

D. Plausible because (1) and (3) are correct and because venting would be correct at a higher pressure.

   Incorrect because pressure is not in excess of the alarm setpoint.

K/A: 007 A2.03 Pressurizer Relief/Quench Tank - Ability to (a) predict the impacts of the following on the system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overpressurization of the PZR

K/A Match: the KA is matched because it requires the candidate to recall the design features and operation of systems that cool and control Pressurizer Relief tank contents after the lift of a PORV.

Selection criteria; **MODIFIED FROM RCS96**

Tier: 2 **Group:** 1
Importance Rating: RO 3.6 SRO 3.9
Technical Reference: **SOP-101, REACTOR COOLANT SYSTEM**
XCP-616 4-4, PRT LVL LO/TEMP/LVL/PRESS HI

Proposed references to be provided to candidates during examination: None

Learning Objective: AB-2-30 For the following annunciators: XCP-616 4-1 PRT LVL LO/TEMP/LVL/PRESS HI a. STATE the setpoint. b. DESCRIBE the associated automatic actions. c. STATE the associated automatic actions. d. DESCRIBE the operator guidance contained in the alarm response procedure.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(4)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

RCS 096

Given the following plant conditions:

- 100% power.
- XCP-616, 4-4, PRT LVL LO/TEMP/LVL/PRESS HI.
- PRT conditions are as follows:

Temperature	115°F.
Level	66%.
Pressure	7 psig.

Which ONE of the following describes the parameter causing the alarm and the method that can be used to clear the alarm?

A.✓ Temperature is high;

Cool the PRT with the RCDT heat exchanger using Component Cooling Water to cool the heat exchanger.

B. Temperature is high;

Cool the PRT with the RCDT heat exchanger using Service Water to cool the heat exchanger.

C. Pressure is high;

Drain the PRT to the Recycle Holdup tank.

D. Pressure is high;

Vent the PRT to the Waste Gas System.

33. Given the following plant conditions:

- 100% power initially.
- "A" and "B" RB Spray pumps are **inoperable**.
- A Pressurizer Code Safety fully opened.
- A reactor trip and safety injection occurred automatically.
- EOP-1.0 E-0, REACTOR TRIP OR SAFETY INJECTION is in progress.
- PRT pressure is 70 psig and increasing.
- RB pressure is 0 psig and stable.

Which ONE of the choices completes the following statement?

Under the above conditions, a device that protects the PRT from overpressure relieves at __(1)__;

As a result of this relief, Adverse Containment bracketed values will be **first** be used in the EOPs due to an increase in __(2)__ .

Assume that the Pressurizer Code Safety valve will not reseat.

A. 1) 75 psig;
 2) RB pressure.

B. 1) 75 psig;
 2) RB radiation levels.

C✓ 1) 90 psig;
 2) RB pressure.

D. 1) 90 psig;
 2) RB radiation levels.

Modified rcs43 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must assess the effect of a PRT rupture disk relief on containment conditions.

A. The first part is plausible the rupture discs relieve at a differential pressure of 90 psig. The candidate may think that they relieve at an absolute pressure of 90 psia which is 75 psig. The second part is correct.

   Incorrect because the PRT rupture discs would not relieve at 75 psig.

B. The first part is plausible the rupture discs relieve at a differential pressure of 90 psig. The candidate may think that they relieve at an absolute pressure of 90 psia which is 75 psig. The second part is plausible because if the PRT relieves this will expel reactor coolant to the RB atmosphere and could increase radiation level.

   Note the following: The setpoint of 2000 R/hr indicates failed fuel in accordance with EAL tables. At that level indicated by RM-G7 or RM-G18, it would take at least 50 hours to reach 100,000 of integrated dose to require adverse containment values. Since the stem does not provide indication of failed fuel, it would take considerably longer for the conditions given.

   Incorrect because the PRT rupture discs would not relieve at 75 psig and because the setpoint of 100,000 accumulated rads would not be reached for 50 hours or more.

C. CORRECT. The rupture discs relieve at a differential pressure of 90 psi and with 0 psig RB pressure indicated this is 90 psig. Adverse containment would first be reached due to RB pressure.

D. The first part is correct. The second part is plausible because if the PRT relieves this will expel reactor coolant to the RB atmosphere and could increase radiation level.

   Incorrect because the setpoint of 100,000 accumulated rads would not be reached for 50 hours or more.

K/A: 007 K1 .01 Pressurizer Relief/Quench Tank - Knowledge of the physical connections and/or cause-effect relationships between the system and the following: Containment system

K/A Match: The KA is matched because the candidate must assess the effect of a PRT rupture disk relief on containment conditions.

Selection criteria; **MODIFIED FROM RCS43**

Tier: 2 **Group:** 1
Importance Rating: RO 2.9 SRO 3.1
Technical Reference: **XCP-616, 4-4**
DBD - REACTOR COOLING SYSTEM

Proposed references to be provided to candidates during examination: None

Learning Objective: MD-3 1151DESCRIBE factors that affect the reliability and potential failure of instrumentation associated with each key parameter, including the following: 1. Probable failure modes 2. Response and degree of accuracy when exposed to an accident environment, including magnitude and direction of inaccuracies

Question Cognitive Level: **Memory or Fundamental Knowledge** _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(3)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

RCS 043

A small break LOCA has occurred. Pressurizer PORVs are being used to reduce RCS pressure per EOP-2.1, "Post-LOCA Cooldown and Depressurization."

- Containment pressure is 14 psig.

Which ONE of the following represents the maximum pressure that could be reached inside the Pressurizer Relief Tank (PRT) before the PRT rupture disc ruptures?

A. 91 psig

B. 100 psig

C.✓ 114 psig

D. 128 psig

34. Initial conditions:

- 100% power.
- "B" CCW train is the active loop.
- "B" Service Water Pump has just tripped.

Which ONE of the choices below completes the following statements in accordance with SOP-117, SERVICE WATER SYSTEM?

In the current condition, the "C" SWP Train "B" supply breaker is racked __(1)__ .

A flow path from "A" SWP to the "B" CCW Heat exchanger __(2)__.

A. 1) up
   2) is available without additional operator action required.

B. 1) up
   2) is isolated by at least one closed manually-operated valve.

C. 1) down
   2) is available without additional operator action required.

D✓ 1) down
   2) is isolated by at least one closed manually-operated valve.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall

A. The first part is plausible because "A" and "B" Service Water pumps are normally racked up and because in the CCW system "C" CCW pump is racked up and in standby. The second part is plausible because systems such as EFW and RHR discharge path are normally cross-connected at 100% power.

   Incorrect because "C" SWP breakers are normally racked down and because the Service Water trains are maintained split at 100% power.

B. The first part is plausible because "A" and "B" Service Water pumps are normally racked up and because in the CCW system "C" CCW pump is racked up and in standby. The second part is correct.

   Incorrect because "C" SWP breakers are normally racked down.

C. The first part is correct. The second part is plausible because systems such as EFW and RHR discharge path are normally cross-connected at 100% power.

   Incorrect because the Service Water trains are maintained split at 100% power.

D. CORRECT. Either 3118A or B is closed to maintain the Service Water trains split.

K/A: 008 K1.01 Component Cooling Water - Knowledge of the physical connections and/or cause-effect relationships between the system and the following: SWS

K/A Match: The KA is matched because the candidate must recall the alignments of SW and determine whether a path of Service Water to a CCW heat exchanger exists after a loss of a Service Water pump.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 3.1 SRO 3.1
Technical Reference: **SOP-117**

Proposed references to be provided to candidates during examination: None

Learning Objective: IB-1 07. DESCRIBE the normal operation of the following Service Water System Component. Include component types and applicable setpoints: 8. Service Water Pump

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(8)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

35. Given the following plant conditions:

- 100% power, stable.
- Pressurizer Pressure Control was in AUTO.
- Pressurizer pressure was 2235 psig, stable.
- PT-444, PZR Pressure Transmitter **instantly** failed to an output corresponding to 2290 psig.

Which ONE of the choices below completes the following statement?

Immediately after the PT-444 failure, the Master Pressure Controller will demand the spray valves to be __(1)__ .

As time continues, the spray valves will __(2)__.

ASSUME NO OPERATOR ACTIONS

A. 1) fully closed
2) remain **fully closed**.

B. 1) 60% open
2) **close** to less than 60% open as integral action takes effect.

C. 1) 60% open
2) **open** more than 60% open as integral action takes effect.

D. 1) 100% open
2) remain **fully open**.

Revised przrpresscntlsys68 to create this version for 2018 NRC - rj

QUESTION USAGE (as przrpresscon68)
ILO 13-01 NRC (2015)

REVISION HISTORY:
Rev. 1 submitted by RJ - changed pressure values and answers
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine the effect of a loss of a pressure transmitter on the pressure control system.

A. The first part and second parts are incorrect but plausible because this would be the position if PT-445 were failing high. (PT-445 has no impact on spray valves).

   Incorrect because PT-444 will demand the spray valves to open for the PT-444 failure.

B. The first part is correct. The second part is plausible because this is the normal characteristic of the controller as the spray valves open to reduce pressure.

   INCORRECT because integral action of the controller will cause the spray valve to close.

C. The first part is correct. Since the PZR spray valves start to open at an error of 25 psi, and are fully open at a 75 psi error, and the Spray Valve Controller output varies proportionally, the spray valves should be open about 60% [(2290-2260 psig)/50 psi x 100% = 60%]. As the integral controller detects continued deviation above setpoint due to the failure, the controller output will rise to attempt to bring system pressure back to setpoint.

D. The first part is plausible because two PORVs would open and remain open for a high failure of PT-445. In addition, the valve would be 100% open due to integral action if no action is taken to take control of pressure.

   Incorrect because the valve demand will be 60% after the failure.

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

K/A Match: The KA is matched because the candidate must determine the effect of a loss of a pressure transmitter on the pressure control system.

Selection criteria; **MODIFIED FROM PZRPRESSCNTRLSYS68**

Tier: 2 **Group:** 1
Importance Rating: RO 2.7 SRO 3.1
Technical Reference: **DBD - SETPOINT BASES DOCUMENT**
DBD - REACTOR COOLING SYSTEM

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-3 022. DE SCRIBE the response of the PZR PCS and LCS to the following failures: 3. PT-444 Fail High/Fail Low

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

Given the following plant conditions:

- 100% power, stable.
- Pressurizer Pressure Control was in AUTO.
- Pressurizer pressure was 2235 psig, stable.
- PT-444, PZR Pressure Transmitter **instantly** failed to an output corresponding to 2280 psig.

Which ONE of the choices below completes the following statement?

Immediately after the PT-444 failure, the Master Pressure Controller will demand the spray valves to be ____(1)____ .

As time continues, the spray valves will _______(2)_________.

ASSUME NO OPERATOR ACTIONS

A. 1) fully closed
   2) remain **fully closed**.

B✓ 1) 40% open
   2) **open** more than 40% open as integral action takes effect.

C. 1) 40% open
   2) **close** to less than 40% open as integral action takes effect.

D. 1) 60% open
   2) **open** more than 60% open as integral action takes effect.

36. Initial conditions:

- 100% power.
- Loop A, B and C OT∆T setpoints all read 123%.

Current condition:

- An operator has opened PCV-444D, PZR SPRAY.

Which ONE of the following choices complete the following statement?

Opening PCV-444D at 100% power causes core conditions to become __(1)__ a Departure from Nucleate Boiling (DNB)?

10 seconds after opening PCV-444D, OT∆T setpoints will be __(2)__ 123%.

Assume that the plant does not trip.

Assume no operator actions

A. 1) farther from
2) above

B. 1) farther from
2) at

C. 1) closer to
2) at

D✓ 1) closer to
2) below

Modified rps49 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine the effect of a depressurization on an RPS setpoint that protects against Departure from Nucleate Boiling.

A. The first part is plausible if the candidate believes that reducing pressure improves core conditions. This is also plausible since an overpressure condition can have an adverse affect due to the suppression of nucleate boiling at full power. A rising setpoint in the second part would indicate an improving condition.

   Incorrect because the OTΔT setpoints will lower and core conditions will not be farther from DNB.

B. The first part is plausible if the candidate believes that reducing pressure improves core conditions. This is also plausible since an overpressure condition can have an adverse affect due to the suppression of nucleate boiling at full power. The second part is plausible because OPΔT setpoints do not change for a change in pressure.

   Incorrect because the OTΔT setpoints will lower and core conditions will not be farther from DNB.

C. The first part is correct. The second part is plausible because OPΔT setpoints do not change for a change in pressure.

   Incorrect because the OTΔT setpoints will lower.

D. CORRECT. Opening PCV-444D cause pressure to lower which cause core conditions to be closer to DNB. 10 seconds afterwards the OTΔT setpoint will lower the normal state of 123% due to the lowering input of pressurizer pressure to the associated calculation.

K/A: 012 Reactor Protection K5.01 Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB

K/A Match: the K/A is met because the candidate must determine the effect of a depressurization on an RPS setpoint that protects against Departure from Nucleate Boiling.

Selection criteria; **MODIFIED FROM RPS49**

Tier: 2 **Group:** 1
Importance Rating: RO 3.3 SRO 3.8
Technical Reference: **T.S. TABLE 2.2-1**

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-9 010. IDENTIFY the Reactor Protection and Safeguards Actuation System interfaces with the following systems and/or subsystem: 3. Pressurizer

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

REACTOR PROT SYSTEM 049

Given the following plant conditions:

- 100% power.
- PCV-444D, PZR SPRAY, sticks open.

Which ONE of the following describes:

1) The Reactor Protection System Trip that protects against Departure from Nucleate Boiling?

2) How the above failure will affect the associated setpoint within the first TEN (10) seconds after the failure?

Assume NO Operator Action

A. 1) Overpower Delta T (OPΔT),

2) Setpoint will increase.

B. 1) Overpower Delta T (OPΔT),

2) Setpoint will decrease.

C. 1) Overtemperature Delta T (OTΔT),

2) Setpoint will increase.

D.✓ 1) Overtemperature Delta T (OTΔT),

2) Setpoint will decrease.

37. Given the following plant conditions:

    - 100% power.

    Which ONE of the choices below answers both of the following question?

    1) Which permissive bistable is required to operate properly to ensure that a Reactor trip occurs for a loss of **one** Reactor Coolant Pump.

    2) What is the normal setpoint for this bistable?

    A. 1) P-7.
       2) 10% power.

    B. 1) P-8.
       2) 10% power.

    C. 1) P-7.
       2) 38% power.

    D✓ 1) P-8.
       2) 38% power.

Revised version of rps193 for 2018 NRC - rj

QUESTION USAGE (AS RPS193):
RO-14-01 TAA Exam
RO-11-01-NRC (**2013-RO NRC**)

REVISION HISTORY:
Rev 1. submitted by RJ - adjusted power level and stem wording
Reviewed by: Danny Rhymer
Approved:

Rev. 0 Submitted by RJ based on RPS-58 in response to NRC comments during review for 2013 NRC.

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must recall the permissive that must function properly to enable trip logic at the correct condition.

A. Plausible, P-7 would block the anticipatory 2-loop loss of flow trips on UV and UF that are present. 10% is the correct setpoint for P-7.

Incorrect because the single loop loss of flow > P-8 would still operate since it is not blocked by P-7.

B. Plausible, P-8 prevents a reactor trip on a loss of flow on one loop and 10 is the normal setpoint for P-7.

Incorrect because the setpoint for P-8 is 38% power.

C. Plausible, P-7 would block the anticipatory 2-loop loss of flow trips on UV and UF that are present. 38% is the correct setpoint for P-8.

Incorrect because the single loop loss of flow > P-8 would still operate since it is not blocked by P-7.

D. CORRECT; P-8 would prevent a reactor trip on loss of flow in one loop and the correct setpoint is 38% power.

K/A: 012 K6.03 Reactor Protection - Knowledge of the of the effect of a loss or malfunction on the following will have on the system: Trip logic circuits

K/A Match: K/A is met because the candidate must recall the permissive that must function properly to enable trip logic at the correct condition.

Selection criteria; **REVISED BANK**

Tier: 2 **Group:** 1
Importance Rating: RO 3.1 SRO 3.5
Technical Reference: **SOP-401**

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-9-17 DESCRIBE the following permissives associated with the reactor protection system, include function, setpoint and coincidence:4. P-8, Loss of Flow

Question Cognitive Level: **Memory or Fundamental Knowledge** __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

REACTOR PROT SYSTEM 193

Given the following plant conditions:

- 100% power.
- Flow in the RCS loop "A" drops to 75%.

Which ONE of the following permissive bistables, if in the wrong condition for current plant status, would **prevent** an automatic reactor trip and the normal setpoint for the this bistable?

A. P-7; 10% power.

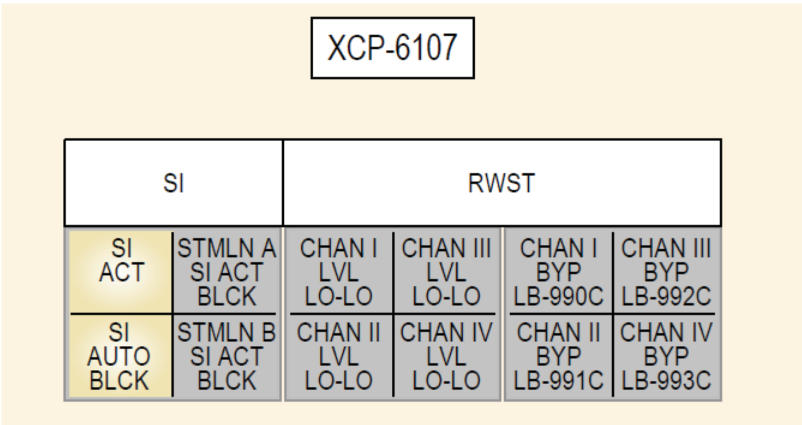
B. P-8 10% power.

C. P-7; 38% power.

D.✓ P-8; 38% power.

38. Given the following plant conditions:

- 100% power initially.
- A spurious safety injection occurred.
- The RO is performing actions to reset SI.
- After performing SI reset, the following indications exist:



- With the indications above in effect, operators **stopped "A" Charging pump**.

Which ONE of the choices below answers **both** of the following questions?

1) Were operators successful in resetting both trains of SI?
2) Under the conditions above, will manual operation of SI ACTUATION switches cause "A" Charging pump to restart?

A. 1) Yes.
   2) Yes.

B. 1) Yes.
   2) No.

C. 1) No.
   2) Yes.

D. 1) No.
   2) No.

Modified eops85 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must assess indications to determine the status of SI actuation circuitry.

A. The first part is plausible because the two lights for SI ACT and SI BLCK have different states of dime and bright depending on whether no, one or both trains are reset. The second part is plausible because a manual actuation of SI will occur even after SI reset.

   Incorrect because both trains of SI have not been reset and a manual actuation can occur after SI reset.

B. The first part is plausible because the two lights for SI ACT and SI BLCK have different states of dime and bright depending on whether no, one or both trains are reset. The second part is plausible because auto SI cannot occur after SI reset until RX trip breakers have been cycled.

   Incorrect because both trains of SI have not been reset.

C. CORRECT. The SI ACT light will extinguish when both trains of SI have been reset. Since "A" Charging pump has been stopped, "A" train of SI was reset. A manual actuation of Train "A" will occur.

D. The first part is correct. The second part is plausible because auto SI cannot occur after SI reset until RX trip breakers have been cycled.

   Incorrect because a manual SI actuation will start "A" Charging pump.

K/A: 013 G2.4.31 Engineered Safety Features Actuation - Knowledge of annunciators alarms, indications or response procedures

K/A Match: The KA is matched because the candidate must assess indications to determine the status of SI actuation circuitry.

Selection criteria; **MODIFIED FROM EOPS85**

Tier: 2 **Group:** 1
Importance Rating: RO 4.2 SRO 4.1
Technical Reference: **SOP-112**
MS-41-011-0008

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-9 024. DESCRIBE the component operation associated with each position for the following switches/controls of the safety injection system: 1. Safety Injection Train A/B Reset Switches

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

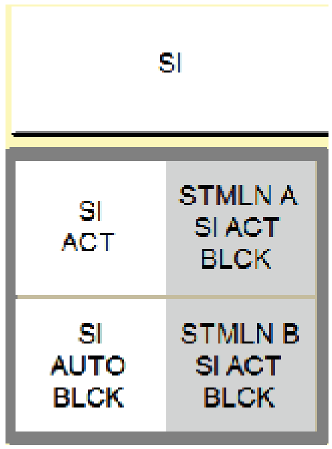
*****************ORIGINAL FOR Q38***********

Mods and revs

EOPS 805

Given the following plant conditions:

- A reactor trip and safety injection have occurred.
- The RO is performing actions to reset SI.
- After performing SI reset, the following indications exist:
  - SI ACT Monitor light is ILLUMINATED
  - SI AUTO BLCK Monitor light is ILLUMINATED



Which ONE of the choices below answers **both** of the following:

1) What is the current status of SI Actuation reset, and
2) Is it possible to activate SI components from the SI switches after SI has been reset?

A. 1) Both trains of SI are RESET.

2) SI will reinitiate from the MCB SI switches.

B. 1) Both trains of SI are RESET.

2) SI will **not** reinitiate from the MCB SI switches.

C.✓ 1) Both trains of SI are **not** RESET.

2) SI will reinitiate from the MCB SI switches.

D. 1) Both trains of SI are **not** RESET.

2) SI will **not** reinitiate from the MCB SI switches.

39. Given the following plant conditions:

- 100% power.
- ALL Offsite Power (115 KV and 230 KV) is lost.
- The crew is performing SOP-306, Section V. A and B. OPERATION OF DIESEL GENERATOR A and B AFTER AN AUTOMATIC START AND LOAD.
- The plant is initiating a natural circulation cooldown in accordance with site procedures.

Which ONE of the choices below answers both of the following questions?

1) Which pressurizer heaters will operators use to maintain RCS pressure?

2) What **minimum** action(s) must be take to make those heaters available?

A✓ 1) **Only** Backup Group heaters will be used;
2) ESF LOADING SEQ RESETS must be taken to NON-ESF LCKOUTS **only**.

B. 1) **Only** Backup Group heaters will be used;
2) ESF LOADING SEQ RESETS must be taken to NON-ESF LCKOUTS **and** AUTO-START BLOCKS.

C. 1) Backup **and** Control Group heaters will be used;
2) ESF LOADING SEQ RESETS must be taken to NON-ESF LCKOUTS **only**.

D. 1) Backup **and** Control Group heaters will be used;
2) ESF LOADING SEQ RESETS must be taken to NON-ESF LCKOUTS **and** AUTO-START BLOCKS.

Revised without significant modification for 2018 NRC - rj

QUESTION USAGE:
RO-11-01-AUDIT W/U Exam
RO-10-01-NRC **(2011-RO NRC)**

REVISION HISTORY:
Rev 3. submitted by RJ - restructured question and wording
Reviewed by: Danny Rhymer
Approved:

Rev. 0 as a modifiPZR PRESS CNTRL SYS 48
Rev.1 (wdb 9/27/11) swapped APN numbers in B. and D. per validator comment.
Rev. 2 Replaced second half of question with what is required by operators to get heaters to function per the SOP.

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must understand the operation of interlocks required to re-enable pressurizer heaters after an SI actuation.

A. CORRECT. The PZR Control Heaters are powered from APN-4106 which is powered from 7.2 KV Bus 1C, and the PZR Backup Heaters are powered from APN-4104 and 4105, which are powered from 7.2 KV ESF Bus 1DA and 1DB, respectively. Only the Backup Heaters are available. When a Blackout occurs on an ESF Bus the ESFLS actuation several Auxiliary Trips and Lockouts including the trip of the feeder breakers to the PZR BU Heaters (APN 4104 and 4105). The ESFLS reset switches reset the automatic lockout of non-ESF loads.

B. Plausible because the first half is correct and EOP-1.0 directs that the reset switch be taken to both non-esf lckouts and auto-start blocks.

   Incorrect because the switch only has to be taken to NON-ESF LCKOUTS to be able to control the BU Heaters.

C. Plausible because which heaters come off ESF power could be misunderstood and the second part is correct.

   Incorrect because the control group heaters are powered by BOP power and so will not have power in this case.

D. Plausible because which heaters come off ESF power could be misunderstood and EOP-1.0 directs that the reset switch be taken to both non-esf lckouts and auto-start blocks.

   Incorrect. B/U heaters are available (not control group) AND the switch only has to be taken to NON-ESF LCKOUTS to be able to control the BU Heaters.

K/A: 013 K4.11 Engineered Safety Features Actuation - Knowledge of system design feature(s) and or interlock(s) which provide for the following: Vital power load control

K/A Match: K/A is met because the candidate must understand the operation of interlocks required to re-enable pressurizer heaters after an SI actuation.

Selection criteria; **REVISED BANK**

Tier: 2 **Group:** 1
Importance Rating: RO 3.2 SRO 3.8
Technical Reference: **SOP-101**
DRAWING 203-203

Proposed references to be provided to candidates during examination: None

Learning Objective: AB-2 016. IDENTIFY the power supplies for the following: 3. Pressurizer Heaters.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

PZR PRESS CNTRL SYS 054

Given the following plant conditions:

- 100% power.
- ALL Offsite Power (115 KV and 230 KV) is lost.
- Both DG "A" and "B" start and restore power to their respective ESF Bus.
- The crew is performing SOP-306, Section V. A and B. OPERATION OF DIESEL GENERATOR A and B AFTER AN AUTOMATIC START AND LOAD.
- The plant is initiating a natural circulation cooldown in accordance with site procedures.

Which ONE of the following describes which pressurizer heaters operators will use to maintain subcooled conditions in the RCS and the **minimum** required action(s) to make those heaters available?

A.✓ Only the BU Heaters will be used;

ESF LOADING SEQ RESETS must be taken to NON-ESF LCKOUTS **only**.

B. Only the BU Heaters will be used;

ESF LOADING SEQ RESETS must be taken to NON-ESF LCKOUTS **and** AUTO-START BLOCKS.

C. Only the Control Heaters will be used;

ESF LOADING SEQ RESETS must be taken to NON-ESF LCKOUTS **only**.

D. Only the Control Heaters will be used;

ESF LOADING SEQ RESETS must be taken to NON-ESF LCKOUTS **and** AUTO-START BLOCKS.

40. Given the following plant conditions:

- Mode 3.
- Reactor Building Cooling Units are OPERABLE in the applicable speeds as follows:

Fan	**Slow speed OPERABLE?**	**Fast speed OPERABLE?**
XFN0064A	NO	YES
XFN0065A	NO	YES
XFN0064B	YES	NO
XFN0065B	YES	YES

NOTE the following fan names:

XFN0064A-AH, REACTOR BLDG COOLING UNIT 1A EMERG FAN
XFN0065A-AH, REACTOR BLDG COOLING UNIT 2A EMERG FAN
XFN0064B-AH, REACTOR BLDG COOLING UNIT 1B EMERG FAN
XFN0065B-AH, REACTOR BLDG COOLING UNIT 2B EMERG FAN

Which ONE of the choices below answer both of the following questions in accordance with T.S. 3.6.2.3, REACTOR BUILDING COOLING SYSTEM?

1) What is the **lowest** plant MODE of operation in which T.S. 3.6.2.3 applies?
2) Is an action statement contained in T.S. 3.6.2.3 L.C.O in effect for the current conditions?

A. 1) 3
2) No.

B. 1) 3
2) Yes.

C. 1) 4
2) No.

D.✓ 1) 4
2) Yes.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the applicability for an LCO for containment cooling and determine if the LCO is met.

A. The first part is plausible because some LCO such as T.S. 3.5.2 for ECCS subsystems are only applicable in Mode 1, 2 and 3. The second part is plausible because there are two slow speed fans operable.

   Incorrect because the two fans are on the same train and do not meet the LCO. An action statement is in effect for one train inoperable and Mode 3 is not the lowest mode of applicability.

B. The first part is plausible because some LCO such as T.S. 3.5.2 for ECCS subsystems are only applicable in Mode 1, 2 and 3. The second part is correct.

   Incorrect because Mode 3 is not the lowest mode of applicability.

C. The first part is correct. The second part is plausible because there are two slow speed fans operable

   Incorrect because the two fans are on the same train and do not meet the LCO. An action statement is in effect for one train inoperable.

D. CORRECT. Mode 4 is the lowest mode of applicability for T.S. 3.6.2.3 and it specifies that two separate trains must be operable. The two fans indicated are on the same train and do not meet the LCO. An action statement is in effect for one train inoperable.

K/A: 022 G2.2.22 Containment Cooling - Knowledge of limiting conditions for operations and safety limits.

K/A Match: The KA is matched because the candidate must recall the applicability for an LCO for containment cooling and determine if the LCO is met.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 4.0 SRO 4.7
Technical Reference: **T.S. 3.6.2.3**

Proposed references to be provided to candidates during examination: None

Learning Objective: SB-4 015 Given a limiting condition for operation and a mode, DEFINE the requirements to satisfy the LCO, the actions if required within one hour or less, and describe the bases for the LCO.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

41. Given the following plant conditions:

- 100% power initially.
- A large break LOCA has occurred.
- Operators entered EOP-2.2 ES-1.3 TRANSFER TO COLD LEG RECIRCULATION and are reading **step 1**.
- RWST is at 17% and decreasing.
- Both RB Spray Pumps are running.
- LI-1969, RHR A LEVEL FEET and LI-1970, RHR B LEVEL FEET read 411 feet and increasing.

Which ONE of the choices below completes the following statements?

The RB spray system __(1)__ capable of performing its design function in the current condition because __(2)__.

A. 1) is
2) valves have aligned automatically to continue RB Spray flow.

B. 1) is
2) the contents of the RWST remains available until the RB Spray pump suctions are aligned manually to the RHR sumps.

C.✓ 1) is **not**
2) the RHR sump levels do not provide the required suction head for RB Spray pumps.

D. 1) is **not**
2) the RB Spray pumps must be stopped to protect the pumps from a loss of suction from the RWST.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine if the containment sump level is sufficient to provide operation of containment spray during recirculation.

A. Plausible aligned automatically to the RHR sump to continue RB Spray flow on recirculation at 18% RWST level.

   Incorrect because RHR sump level is not sufficient to continue recirculation spray flow and EOP-2.4 cautions that pumps must be stopped.

B. Plausible because some alignments such as RHR discharge to Charging Pump suction occur manually.

   Incorrect because RB Spray was automatically aligned and is not capable of providing spray flow in the given conditions.

C. CORRECT. RHR sump level is not sufficient to continue recirculation spray flow and EOP-2.4 cautions that spray pumps must be stopped.

D. Plausible because the EOP-2.4 provide steps to stop pumps taking suction from the RWST at low levels.

   Incorrect because low RWST level is not the reason RB spray pumps will be stopped.

K/A: 026 A1.03 Containment Spray - Ability to predict and/or monitor changes in parameters associated with operating the system controls including: Containment sump level.

K/A Match: The KA is matched because the candidate must determine if the containment sump level is sufficient to provide operation of containment spray during recirculation.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 3.5 SRO 3.5
Technical Reference: **EOP-2.2**
OAP-103.2 EOP FSP SETPOINT DOCUMENT

Proposed references to be provided to candidates during examination: None

Learning Objective: AB- 8 04. DESCRIBE the Reactor Building Spray System Interfaces with the following systems and/or subsystems: 2. RB Recirculation Sump

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(8)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

42. Given the following plant conditions:

- 100% power initially.
- The following events occurred at the indicated times:

  0730 Reactor trip.
  0740 Safety Injection occurs.
  0750 RB Pressure is 7 psig.
  0800 RB Pressure is 12.1 psig.

Which ONE of the choices below completes the following statements?

MVG-3002A and 3002B, NAOH TO SPRAY PUMP A(B) SUCT **first** opened at __(1)__.

Reactor Building Spray Pumps were **first** running at time __(2)__.

A. 1) 0740.
   2) 0750.

B.✔ 1) 0740.
   2) 0800.

C. 1) 0750.
   2) 0750.

D. 1) 0750.
   2) 0800.

Modified contspray77 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the applicant must determine the auto operation of a containment spray MOVs and pumps when conditions were met.

A. The first part is correct. The second part is plausible because Containment HI-2 actuated at 6.35 psig RB pressure at time 0750 so the candidate may think that this is when spray actuates.

   Incorrect because RB spray pumps did not start on HI-2 at 0750.

B. CORRECT. MVG-3002A and B open upon phase A actuation which occurred when SI actuated at 0740.

C. The first part is plausible. Containment HI-2 occurs at 6.35 psig so the candidate may assume that all spray related features occur at that pressure.

   Incorrect because 3002A and B did not open and RB spray pumps did not start on HI-2 at 0750.

D. The first part is plausible because Containment HI-2 occurs at 6.35 psig so the candidate may assume 3002A and B open at that pressure. The second part is correct.

   Incorrect because 3002A and B opened before 0750.

K/A: 026 A3.01 Containment Spray - Ability to monitor automatic operation of the system, including: Pump starts and correct MOV positioning

K/A Match: The KA is matched because the candidate must determine the auto operation of a containment spray MOVs and pumps when conditions were met.

Selection criteria; **MODIFIED FROM CONTSPRAY77**

Tier: 2 **Group:** 1
Importance Rating: RO 4.3 SRO 4.5
Technical Reference: **SOP-112**
ARP-XCP-612, 3-1

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-9 027. STATE the conditions that actuate a containment isolation signal, including setpoint and coincidence.

AB-8 019. DESCRIBE the following actuation signals associated with the Reactor Building Spray System. Include purpose and setpoints: 1. RB Spray Pump Start

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

CONT SPRAY SYSTEM 077

Given the following plant conditions:

The plant was in Mode 1.

0735 Reactor trip.
0738 Reactor Building Pressure is 2.3 psig.
0741 Reactor Building Pressure is 3.5 psig.
0747 Reactor Building Pressure is 12.1 psig.
0748 Bus 1DA loses power. "A" EDG operates as designed.
0749 Reactor Building Pressure is 14.9 psig and rising.

Which ONE of the following describes the operation of the Reactor Building spray system for this event?

A. At 0738, MVG-3003A and 3003B SPRAY HDR ISOL LOOP A(B) are open.

B. At 0741, MVG-3002A and 3002B, NAOH TO SPRAY PUMP A(B) SUCT are open.

C.✓ At 0747, Reactor Building Spray Pump "A" is running.

D. At 0749, Reactor Building Spray Pump "A" is not running.

43. Given the following plant conditions:

Time 0500:

- Mode 3, Plant startup in progress.
- Selected valves, controls and indications are as follows:

Valve	Control switch	Indicated position
PVM-2801A, MS Isolation	AUTO	OPEN
PVM-2869A, MS Isolation Bypass	AUTO	OPEN
PVT-2879A,B, LINE DRN	AUTO	OPEN

Time 0531:

- A main steam line break has occurred to the RB.
- RB pressure is 5.5 psig and increasing.
- Steam generator pressures are 600 psig and decreasing.
- Valves, controls and indications are as follows:

Valve	Control switch	Indicated position
PVM-2801A, MS Isolation	AUTO	CLOSED
PVM-2869A, MS Isolation Bypass	AUTO	OPEN
PVT-2879A,B, LINE DRN	AUTO	OPEN

Given the conditions above, which ONE of the following correctly describes whether the three system valves above reflect their proper positions?

A. All valve positions are appropriate for the stated conditions.

B. PVM-2801A should have remained open but all other positions are correct.

C✓ PVM-2869A should have closed automatically but all other positions are correct.

D. All the valves should have closed automatically.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine if main steam line isolations occurred as required after an auto main steam line isolation.

A. Plausible because the MSIV has closed and because the 2869A is ensured closed in EOP-1.0 attachment 3. The candidate may assume that this valve requires manual isolation.

   Incorrect because 2869A closes due to a Main Steam Isolation Signal.

B. Plausible because RB pressure is less than the HI-2 actuation for Main Steam Line Isolation.

   Incorrect because Main Steam Line Isolation occurred due to low SG pressure and the valve should be closed.

C. CORRECT. PVM-2869A should be closed due to a Main Steam Line Isolation Signal. PVM-2801A closed properly and PVT-2879A does not receive an auto isolation.

D. Plausible because PVM-2801A and PVM-2869A should have closed so the candidate may assume that a line drain of the main steam line should also close. This valve has an AUTO position and an auto closure due to high level in the moisture collector.

   Incorrect because this valve 2879A does not close due to a main steam line isolation signal.

K/A: 039 A3.02 Main and Reheat Steam - Ability to monitor automatic operation of the system, including: Isolation of the MRSS

K/A Match: The KA is matched because the candidate must determine if main steam line isolations occurred as required after an auto main steam line isolation.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 3.1 SRO 3.5
Technical Reference: **DRAWING 41-011-008**
DRAWING 208-067

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-9 030. DESCRIBE the actions that occur on a steamline isolation signal, specify equipment affected and its status.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

44. Initial conditions:

- A reactor and plant startup are in progress.
- The unit is at BOL.
- 10^{-3}% power and stable.
- The STM DUMP MODE SEL switch is in STM PRESS mode.

Current conditions:

- Steam Header pressure transmitter PT-464 has failed high.

Which ONE of the choices below completes the following statements?

As a result of the failure, RCS T_{AVG} will __(1)__ and that trend will continue until __(2)__.

ASSUME NO OPERATOR ACTIONS

A. 1) increase
2) steam generator PORVs actuate in relief mode.

B. 1) increase
2) steam generator pressure reaches the lowest safety setpoint.

C. 1) decrease
2) reactor thermal power is greater than the rate of secondary heat removal.

D.✓ 1) decrease
2) condenser steam dumps automatically close.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine the effect of a main steam header pressure transmitter failure on condenser steam dumps and RCS temperature.

A. The first part is plausible. If PT-464 fails were to fail low, the Steam Dump controller in steam pressure mode will close in error to bring pressure back to setpoint. TAVG would increase and steam generator PORVs would eventually open.

   Incorrect because TAVG does not decrease for a high failure of PT-464.

B. The first part is plausible. If PT-464 fails were to fail low, the Steam Dump controller in steam pressure mode will close in error to bring pressure back to setpoint. TAVG would increase and if the PORV were to fail to open, a safety would open. Since SG PORVs have function associated with the Main Steam Controller, the candidate may assume that these valves will not open.

   Incorrect because TAVG does not decrease for a high failure of PT-464.

C. The first part is correct. The second part is plausible because if steam dumps were to open and remain open, reactor power would increase to the power range and nuclear power would match the secondary heat removal. Since the unit is at BOL, reactivity addition due to temperature decrease would be small.

   Incorrect because the steam dumps will close due to P-12 before power reaches the power range.

D. CORRECT. If PT-464 fails high, the Steam Dump controller in steam pressure mode will open in error to bring pressure back to setpoint. This will continue until P-12 is met at 552°F which closes the steam dumps.

New for 2018 NRC - 5/15/17

K/A: 039 K3.05 Main and Reheat Steam - Knowledge of the effect that a loss or malfunction of the system will have on the following: RCS

K/A Match: The KA is matched because the candidate must determine the effect of a main steam header pressure transmitter failure on condenser steam dumps and RCS temperature.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 3.6 SRO 3.7
Technical Reference: **41-011-0010**

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-1 018. DESCRIBE the effects of various failures of the Steam Dump Control System during normal power operations, including the following: 1. Steam header pressure (PT-464) failure

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(5) and (7)
)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

45. Which ONE of the following will result in the automatic trip of **all** Feedwater Pumps?

A. The reactor trips and T_{AVG} lowers to 562°F and stabilizes.

B. The Main turbine bearing oil pressure transmitters fail to 3 psig.

C.✓ "B" Steam Generator reaches 85% with the other two generators at 60%.

D. Pressurizer pressure reaches 1860 psig and then increases to 2235 psig.

Revised without significant modification for 2018 NRC - rj

QUESTION USAGE:
Unavailable

REVISION HISTORY:
Rev. 1 submitted by RJ - adjusted distractors
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the applicant must recall conditions for feedwater isolation due to high SG levels.

A. Plausible because a Feedwater isolation signal that closes Feedwater Regulating Valves occurs when the reactor is tripped at a TAVG of 564°F.

   Incorrect because the Feedwater Pumps do not trip for this isolation signal.

B. Plausible because a trip of Feedwater pumps trips the Main Turbine. Since lube oil pressure is less than the Main Turbine trip setpoint of 6 psig, the candidate may recall this function in reverse and think that the Main Turbine will trip the Feedwater pumps.

   Incorrect because a Main Turbine lube oil pressure of 6 psig will not trip the Feedwater pumps.

C. CORRECT. "B" Steam Generator level greater than the setpoint of 79% will cause a trip of the Main Turbine and Feedwater pump.

D. Plausible because if Pressurizer pressure were to decrease to less than 1850, SI actuation would occur which trips Feedwater pumps.

   Incorrect because Pressurizer pressure lowering to 1860 does not trip Feedwater pumps.

K/A: 059 K1 .03 Main Feedwater - Knowledge of the physical connections and/or cause-effect relationships between the system and the following: S/GS

K/A Match: The KA is matched because the candidate must recall conditions for feedwater isolation due to high SG levels.

Selection criteria; **REVISED BANK**

Tier: 2 **Group:** 1
Importance Rating: RO 3.1 SRO 3.3
Technical Reference: **ARP-XCP-626, 1-4**
ARP-XCP-636, 1-1
SOP-112

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-9 034. STATE the conditions that actuate a feedwater isolation, include setpoint , coincidence, and a discussion of P-14.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

******************ORIGINAL FOR Q45************
Mods and revs

FEEDWATER SYS 018

Which ONE of the following conditions will trip all three (3) MFPs?

A. Reactor trip with T_{avg} at 558°F.

B.✓ S/G level 85% in 'B' S/G.

C. Pressurizer pressure 1860 psig.

D. Main turbine trip at 100% power.

46. Which ONE of the choices below answers both of the following question?

1) What is an activity that is performed during AB Lower Tech Spec Rounds because of the potential for check valve leakage from the steam generators back to the EFW system?

2) What is the concern if this leakage is occurring?

A. 1) EFW Pump discharge line temperatures are monitored.
2) EFW pump seal damage.

B.✔ 1) EFW Pump discharge line temperatures are monitored.
2) Steam binding of the pumps.

C. 1) EFW Pump vent valves are opened to check for a steam discharge.
2) EFW pump seal damage.

D. 1) EFW Pump vent valves are opened to check for a steam discharge.
2) Steam binding of the pumps.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the consequence of EFW check valve backleakage on EFW pumps and a check that is performed to detect occurrence.

A. The first part is correct. The second part is plausible because pump seal damage on components such as RHR pumps are dependent on the fluid in the pump.

   Incorrect because seal damage is not the concern.

B. CORRECT. The EFW Pump discharge line temperatures are monitored to detect back leakage through EFW check valves and the consequence of back leakage is steam binding of the EFW pumps.

C. Plausible because SOP-211 has actions to vent pumps if elevated line temperatures are detected. Pump seal damage on components such as RHR pumps are dependent on the fluid in the pump.

   Incorrect because vent valves are not opened during rounds and the concern is not pump seals.

D. The first part is plausible because steam binding of the pumps would occur if the pump casing contained steam and because SOP-211 has actions to vent pumps if elevated line temperatures are detected. The second part is correct.

   Incorrect because vent valves are not opened.

K/A: 061 K5.05 Auxiliary/Emergency Feedwater - Knowledge of the operational implications of the following concepts as they apply to the system: Feed line voiding and water hammer

K/A Match: The KA is matched because the candidate must recall the consequence of EFW check valve backleakage on EFW pumps.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 2.7 SRO 3.2
Technical Reference: **OAP-106.1**
SOP-211

Proposed references to be provided to candidates during examination: None

Learning Objective: IB-3 020. EX PLAIN the effects of various failures of the Emergency Feedwater System upon an auto-start signal, including the following : 4. EFW Discharge Check Valves Fail To Seat

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

47. Given the following plant conditions:

Time 0200:

- Breaker BUS 1EA FEED BKR is **closed**.
  - The breaker is **inoperable** and will not open remotely.
  - Operators have been dispatched to locally open and rack down the breaker.
- The following are indicated at the control switch for BUS 1EA FEED BKR.
  - The **red** light is LIT.
  - The **red** flag is present.

Time 0230:

- Breaker BUS 1EA FEED BKR is opened locally.

Time 0240:

- Both the Closing and Tripping DC control breakers inside of the BUS 1EA FEED BKR cubicle are opened locally.

Which of the choices below answers both of the following questions?

At time **0230**, the AMBER light at the BUS 1EA FEED BKR control switch __(1)__ illuminate.

At time **0240**, an indication that the local operator opened the control breakers was that __(2)__ .

A. 1) did
2) XCP-636, 5-1, 7KV ESF CHAN A LOSS OF DC came into alarm.

B✓ 1) did
2) indicating lights at the BUS 1EA FEED BKR switch extinguished.

C. 1) did **not**
2) XCP-636, 5-1, 7KV ESF CHAN A LOSS OF DC came into alarm.

D. 1) did **not**
2) indicating lights at the BUS 1EA FEED BKR switch extinguished.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine the proper indications in the control room while control power breakers are operated locally.

A. The first part is correct. The second part is plausible because the XCP-636, 5-1 alarms on a loss of control power for the associated switch gear. The indicating lights would extinguish in this case. The alarm could also be interpreted as an indication of power loss on a smaller portion of the switchgear such as a Train A ESF feeder to 1EA.

   Incorrect because XCP-636, 1-5 did not come into alarm.

B. CORRECT. At time 0230 the amber light illuminated at the BUS 1EA FEED BKR control switch due to the breaker being in disagreement control switch in the control room. At time 0240, the control power breakers removed power necessary for the indicating lights at the BUS 1EA FEED BKR switch which then extinguished.

C. The first part is plausible because for 7.2 KV pump breakers, the amber light indicates an overload condition. The second part is plausible because the XCP-636, 5-1 alarms on a loss of control power for the associated switch gear. The indicating lights would extinguish in this case. The alarm could also be interpreted as an indication of power loss on a smaller portion of the switchgear such as a Train A ESF feeder to 1EA.

   Incorrect because the lights illuminated at 0230 and because XCP-636, 1-5 did not come into alarm.

D. The first part is plausible because for 7.2 KV pump breakers, the amber light indicates an overload condition. The second part is correct.

   Incorrect because the lights illuminated at 0230.

K/A: 062 A4.04 AC Electrical Distribution - Ability to manually operate and/or monitor in the control room: Local operation of breakers

K/A Match: The KA is matched because the candidate must determine the proper indications in the control room while control power breakers are operated locally.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 2.6 SRO 2.7
Technical Reference: **DRAWING 208-037**
SOP-636, 5-1

Proposed references to be provided to candidates during examination: None

Learning Objective: ES-3-06 Explain how the indicator lights in each type of control circuit are controlled.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

48. Initial conditions:

- 100% power.
- DPN-1HB1 ED, Breaker 23, has tripped OPEN.
- BKR BUS 1DB NORM FEED has tripped OPEN due to a mechanical failure.

Which ONE of the choices below completes the following statements?

"B" EDG is __(1)__ .

A breaker fed from __(2)__ must be closed to restore power to 1DB.

REFERENCE PROVIDED

A. 1) running unloaded.
2) XTF-31, EMERGENCY AUXILIARY TRANSFORMER

B.✓ 1) running unloaded.
2) Bus 1DX

C. 1) **not** running.
2) XTF-31, EMERGENCY AUXILIARY TRANSFORMER

D. 1) **not** running.
2) Bus 1DX

Persons completing checklist (print)	Initials	EMERGENCY DIESEL GENERATOR B ELECTRICAL LINEUP
______________________	____________	
______________________	____________	
______________________	____________	
______________________	____________	

Reviewed by SS/CRS	Date/Time	
		Date/Time started___________/_________
________________________	________/________	Date/Time completed_________/_________

Electrical Lineup Initial Conditions

Positioning the following components to the REQUIRED POSITION partially prepares Diesel Generator B for startup with power available to essential components. Ensure completion of SOP-307, DG FUEL OIL SYSTEM TRAIN B ELECTRICAL LINEUP attachment to power all essential components.

COMPONENT	DESCRIPTION	REQUIRED POSITION	ACTUAL POSITION	INITIALS	VERIFIERS INITIALS
DPN-1HB1 ED (412 INTERMEDIATE BUILDING)					
06	B DIESEL GEN. OIL PUMP XES0007	ON			
13	1B DG CONT PNL XCX5202-DG CUBICLE 2	ON			
14	XPN5504-DG "B" RELAY & TERMINAL PANEL	ON			
15	XEX 4202-DG EXITER REG CUB DIESEL GEN B	ON			
23	XEX 4202 DSL GEN B "XEG 0001B" GEN FIELD FLASHING	ON			
MCC-XMC1DB2Z (436 DIESEL GENERATOR BUILDING)					
02FG	AIR COMP D STG AIR DG B XPT0007 XAC0008D	ON			
02HM	NO. 1B DIESEL POWER PNL XPN0048-DG	ON			
03IJ	AIR COMP C STG AIR DG B XPT0007 XAC0008C	ON			

PROVIDE SOP-306, ATTACHMENT IIB PAGE 1 of 2

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS
In order to answer this question, the candidate must determine the effect of the opening of a DC breaker and determine required subsequent breaker operations.

A. The first part is correct. The second part is plausible because the alternate supply to bus 1DA is from XTF-31.

   Incorrect because the alternate feeder to bus 1DB is not from XTF-31.

B. CORRECT. With DPN-1HB1 ED, Breaker 23 tripped open, field flash did not occur and the generator did not produce voltage. With the "B" diesel unloaded and the normal feed mechanically failed, the alternate feeder fed from bus 1DX must be closed to restore power.

C. The first part is plausible because if breaker 13 on DPN-1HB1 were opened the diesel would not start. The second part is plausible because the alternate supply to bus 1DA is from XTF-31.

   Incorrect because the diesel is running and the alternate feeder to bus 1DB is not from XTF-31.

D. The first part is plausible because if breaker 13 on DPN-1HB1 were opened the diesel would not start. The second part is correct.

   Incorrect because the diesel is running.

K/A: 063 A4.01 DC Electrical Distribution - Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses

K/A Match: The KA is matched because the candidate must determine the effect of the opening of a DC breaker and determine required subsequent breaker operations.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 2.8 SRO 3.1
Technical Reference: **SOP-306, Attachment IIB**
DRAWING-206-005

Proposed references to be provided to candidates during examination:

SOP-306, Attachment IIB Page 1 of 2

Learning Objective: IB-5 4. DESCRIBE the Emergency Diesel Generator System interfaces with the following systems or subsystems: 11. 125 VDC

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(8)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

49. Given the following plant conditions:

- 100% power
- Battery XBA1X BATTERY 1X has lost its connections to all downstream DC distribution panels.

Which ONE of the following describes a load that has lost a source of electrical power as a result of this event?

ASSUME NO OPERATOR ACTIONS

A✓ Emergency Seal Oil Pump.

B. "B" EDG control power.

C. Turning Gear Oil Pump.

D. Inverter XIT-5904.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the supply to a major DC load.

A. CORRECT. XBA1X supplies the Emergency Seal Oil pump.

B. Plausible because XBA1B supplies "B" EDG control power.

   Incorrect XBA1X does not supply "B" EDG Control Power.

C. Plausible because emergency bearing oil pumps for the main turbine and feedwater pumps are powered from this supply

   Incorrect XBA1X does not supply the turning gear oil pump.

D. Plausible because XIT5905 receives power from XBA1X.

   Incorrect XBA1X does not supply XIT-5904.

K/A: 063 K2.01 DC Electrical Distribution - Knowledge of the electrical power supplies to the following: Major DC loads

K/A Match: The KA is matched because the candidate must recall the supply to a major DC load.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 2.9 SRO 3.1
Technical Reference: **EOP-6.0, ATT 2**
DRAWING E-206-005

Proposed references to be provided to candidates during examination: None

Learning Objective: TB-4 013. IDENTIFY power supplies for the following Turbine and Generator Auxiliaries System components: 2. Emergency Seal Oil Pump

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(4)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

50. Initial conditions:

- 100% power.
- DPN-1HA, 125 VDC Main Distribution Panel, is deenergized

Current condition:

- A safety injection has occurred.

Which ONE of the choices below completes the following statement?

"A" EDG ____.

A. did start and is running as designed.

B. did start automatically but the generator field is deenergized.

C. did **not** start automatically but can be started with controls at the local control panel.

D✓ did **not** start automatically but can be started by manually actuating air start valves.

Modified dc26 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must identify a loss of control power.

A. Plausible because loss of other supplies such as 1HB and 1HX would not prevent "A" EDG from starting.

   Incorrect because the 1HB supplies the control power for "B" diesel which, if lost, would prevent the diesel from starting.

B. Plausible because the 1HA supplies field flashing and, if only that power is lost, the diesel would start without generating voltage.

   Incorrect because the diesel will not start if DPN-1HA is deenergized.

C. Plausible because without control power the diesel would not start automatically. Procedures such as EOP-6.0, ECA-0.0 have attachments to direct a local start.

   Incorrect because without control power the diesel would not start from the local control panel.

D. CORRECT. Without control power from the 1DPN-HA source, the diesel would not start automatically but would start by manually actuating air valves.

K/A: 064 K2.03 Emergency Diesel Generator - Knowledge of the electrical power supplies to the following: Control power

K/A Match: The KA is matched because the candidate must identify a loss of control power.

Selection criteria; **MODIFIED FROM DC26**

Tier: 2 **Group:** 1
Importance Rating: RO 3.2 SRO 3.6
Technical Reference: **SOP-306**
EOP-6.0

Proposed references to be provided to candidates during examination: None

Learning Objective: IB-5 4. DESCRIBE the Emergency Diesel Generator System interfaces with the following systems or subsystems: 1. Air Start Subsystem

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(8)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

DC ELECT DIST SYSTEM 026

What will be the response of the 'A' diesel generator if safety injection occurs while DPN-1HA, 125 VDC Main Distribution Panel, is deenergized?

A.✓ Will not start.

B. Will start and run unloaded.

C. Will start, but the exciter field will not flash.

D. Will start and tie on to bus 1DA.

51. Initial conditions:

- 100% power.
- RM-A1, CONTROL ROOM SUPPLY AIR ATMOSHPERIC MONITOR is in alarm.

Current conditions:

- A large break LOCA has occurred.
- EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION.

Which ONE of the choices below completes the following statements?

As a result of the RM-A1 alarm, Control Room Ventilation Train A Filter Bypass damper XDP-22A-AH will automatically __(1)__.

The Reference Page of EOP-1.0, will require operators to __(2)__ within 30 minutes.

A.✓ 1) go fully closed.
2) stop **either** XFN-30A, EMERG FLTR FAN A or XFN-30B, EMERG FLTR FAN B.

B. 1) go fully closed.
2) ensure XFN-30A, EMERG FLTR FAN A **and** XFN-30B, EMERG FLTR FAN B are running.

C. 1) close to a position that limits flow to 1000 cfm.
2) stop **either** XFN-30A, EMERG FLTR FAN A or XFN-30B, EMERG FLTR FAN B.

D. 1) close to a position that limits flow to 1000 cfm.
2) ensure XFN-30A, EMERG FLTR FAN A **and** XFN-30B, EMERG FLTR FAN B are running.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the operation of dampers after an ESF actuation caused by a process monitor and recall procedural requirements for operation of the associated system.

A. CORRECT. As a result of the RM-A1 alarm, Control Room Ventilation Train A Filter Bypass damper XDP-22A-AH will automatically close. The Reference Page of EOP-1.0, will require operators to reduce control room ventilation to one train which will require stopping either XFN-30A, EMERG FLTR FAN A or XFN-30B, EMERG FLTR FAN B.

B. The first part is correct. The second part is plausible because the main body of EOP-1.0 and Attachment 3 ensure operation of two trains of running equipment so the candidate may think that the reference to control room ventilation is to ensure both trains are running.

   Incorrect because the reference page does not provide guidance to keep both ventilation fans running.

C. The first part is plausible a 1000 cfm mechanical stop is contained in the emergency alignment. Under an emergency condition, return air is mixed with a 1000 cfm of make-up air which is ensured by mechanical stop. The candidate could assume that this mixture occurs by ensuring that the bypass damper closes to a 1000 cfm position. The second part is correct.

   Incorrect because the XDP-22A-AH will fully close.

D. The first part is plausible a 1000 cfm mechanical stop is contained in the emergency alignment. Under an emergency condition, return air is mixed with a 1000 cfm of make-up air which is ensured by mechanical stop. The candidate could assume that this mixture occurs by ensuring that the bypass damper closes to a 1000 cfm position. The second part is plausible because the main body of EOP-1.0 and Attachment 3 ensure operation of two trains of running equipment so the candidate may think that the reference to control room ventilation is to ensure both trains are running.

   Incorrect because the XDP-22A-AH will fully close and because the reference page does not provide guidance to keep both ventilation fans running.

K/A: 073 G2.2.44 Process Radiation Monitoring - Ability to interpret control room indications to verity the status and operation of a system, and understand how operator actions and directives affect plant and system conditions

K/A Match: The KA is matched because the candidate must recall the operation of dampers after an ESF actuation caused by a process monitor and recall procedural requirements for operation of the associated system.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 4.2 SRO 4.4
Technical Reference: **EOP-1.0**
STP-124.001

Proposed references to be provided to candidates during examination: None

Learning Objective: GS-8 07. DESCRIBE the normal and emergency operation of the following Control Building Ventilation System major components. Include component types and applicable setpoints: 1. Control Room Normal Fans 2. Control Room Emergency Filter Fans 3. Control Room Emergency Filters 4. Control Room Dampers

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(8)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

52. Given the following plant conditions:

- 100% power.
- "B" CCW loop is the active loop.
- A malfunction of Industrial Cooling water has occurred.
- The following RBCU fans are running:

  XFN 0064A-AH, 1A NORM.
  XFN 0065A-AH, 2A NORM.
  XFN 0064B-AH, 1B NORM.
- RB temperature is 115°F rising.
- The following valves are closed:

  MVG-3111A, RBCU 64A/65A TO IND CLG.
  MVG-3112A, RBCU 64A/65ATO IND CLG.
  MVB-3110A, IND CLG TO RBCU 64A/65A.

Which ONE of the choices below answers both of the following questions?

1) Is the current RB temperature in excess of the limit stated in T.S.3.6.1.5 CONTAINMENT SYSTEMS - AIR TEMPERATURE?

2) Will starting "A" Service Water Booster Pump (SWBP) decrease RB temperature without **any further operator action**?

A. 1) Yes
   2) No.

B. 1) Yes
   2) Yes.

C. 1) No.
   2) No.

D.✓ 1) No.
   2) Yes.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved: Robert Shane

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must predict the change in reactor building temperature after a change in a service water component.

A. The first part is plausible because if temperature were higher than 120°F this would be correct. The second part is plausible because the SW Booster pump is normally off with the discharge valve closed. The candidate may believe that manual action to open the discharge valve is required.

   Incorrect because 115°F is not greater than the T.S. 3.6.1.5 limit and starting "A" SW Booster pump will decrease temperature.

B. The first part is plausible because if temperature were higher than 120°F this would be correct. The second part is correct.

   Incorrect because 115°F is not greater than the T.S. 3.6.1.5 limit.

C. The first part is correct. The second part is plausible because the SW Booster pump is normally off with the discharge valve closed. The candidate may believe that manual action to open the discharge valve is required.

   Incorrect because starting "A" SW Booster pump will decrease temperature.

D. CORRECT. An RB temperature of 115°F is less than the LCO required limit if T.S. 3.6.1.5 of 120°F. Starting the "A" SW Booster Pump will cause its discharge valve 3106A to open automatically. As long as 3108A and B and 3109A and B are open, this will provide cooling flow to the running RBCUs 64A and 65A and decrease temperature.

K/A: 076 A1 .02 Service Water - Ability to predict and/or monitor changes in parameters associated with operating the system controls including: Reactor and turbine building closed cooling water temperatures.

K/A Match: the KA is matched because the candidate must predict the change in reactor building temperature after a change in a service water component.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 2.6 SRO 2.6
Technical Reference: **T.S. 3.6.1.5**
DRAWING 302-222-00

Proposed references to be provided to candidates during examination: None

Learning Objective: SB-4 015 Given a limiting condition for operation and a mode, DEFINE the requirements to satisfy the LCO, the actions if required within one hour or less, and describe the bases for the LCO.
AB-17 04. DESCRIBE the Reactor Building Ventilation System interfaces with the following systems and/or subsystems: 1. Service Water

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(10)

SSRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

53. Given the following plant conditions:

- 100% Power
- A rupture occurs in a Station Instrument Air system supply header.
- PI-8342, INST AIR HDR PRESS is at 65 psig and decreasing.

Which ONE of the following sets of positions for IPV-2659, INST AIR TO RB AIR SERV, and IPV-8324,STATION AIR SUPPLY HDR PRESS CONT VALVE are expected for the conditions above?

	IPV-2659	IPV-8324
A.✓	Closed.	Partially open.
B.	Closed.	Fully closed.
C.	Open.	Partially open.
D.	Open.	Fully closed.

2018 (1601) NRC test
Revised without significant modification for 2018 NRC - rj

QUESTION USAGE:
RO-11-01-Exam 7 (I&C)
RO-10-01-SYSTEMS Week 4.

REVISION HISTORY:
Rev 1. submitted by RJ - adjusted wording and gave pressure
Reviewed by: Danny Rhymer
Approved:

Rev 0. Submitted by Matthew R. Bender as a modified question of Instrument Air Sys18.
Rev. 1 Submitted by T Taffar 12/4/12
Changed stem to read "A rupture occurs in a Station Instrument Air system supply header" instead of "A large air leak occurs in the Station Instrument Air system"

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must recall the operation of a automatic valve based on instrument air header pressure.

A. CORRECT. With the leak in the Station Air header IPV-2659 will remain closed. IPV-8324 begins to modulate closed below 100 psig on the Instrument Air Header and fully closes at 60 psig. At 65 psig 8324 will be partially open..

B. The first part is correct. The second part is plausible because if Instrument Air Header pressure were less than 60 psig, IPV-8324 would be fully closed.

   Incorrect because IPV-8324 is partially opened.

C. The first part is plausible because 2659 would open if the leak was on the RB IA header. The second part is correct.

   Incorrect because IPV-2659 will not open for the given conditions.

D. The first part is plausible because 2659 would open if the leak was on the RB IA header. The second part is plausible because if Instrument Air Header pressure were less than 60 psig, IPV-8324 would be fully closed.

   Incorrect because IPV-2659 will not open for the given conditions and because IPV-8324 is partially opened.

K/A: 078 A3.01 Instrument Air - Ability to monitor automatic operation of the system, including: Air pressure

K/A Match: The KA is matched because the candidate must recall the operation of a automatic valve based on instrument air header pressure.

Selection criteria; **REVISED BANK**

Tier: 2 **Group:** 1
Importance Rating: RO 3.1 SRO 3.2
Technical Reference: **SOP-220**
DBD - IA INSTRUMENT AIR AND SERVICE AIR SYSTEM

Proposed references to be provided to candidates during examination: None

Learning Objective: TB-12 010. For the RB Instrument Air Backup Supply Isolation Valve XVA02659-IA: 1. STATE its purpose 2. DESCRIBE its automatic operation 3. DESCRIBE how it is locally opened.

011. For the Service Air Back Pressure Regulator IPV-8324, STATE its purpose and DESCRIBE its normal operation.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(4)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

INSTRUMENT AIR SYS 045

Given the following plant conditions:

- 100% Power
- A rupture occurs in a Station Instrument Air system supply header

Which ONE of the following combinations of IPV-2659, INST AIR TO RB AIR SERV, and IPV-8324,STATION AIR SUPPLY HDR PRESS CONT VALVE actions will result?

	IPV-2659	**IPV-8324**
A.	Remains Closed	Remains Open
B.✓	Remains Closed	Closes
C.	Opens	Remains Open
D.	Opens	Closes

54. Given the following plant conditions:

- Mode 6.
- Reactor cavity is at the normal level for refueling.
- Fuel Transfer Tube is **open**.
- The equipment hatch is **open**.
- Reactor Building Purge is in progress.
  - XFN-11A, SPLY FAN A is running.
  - XFN-11B, SPLY FAN B is OFF.
  - XFN-13A, EXH FAN A is running.
  - XFN-13B, EXH FAN B is running.

Which ONE of the following identifies an action that would cause Reactor Cavity level to **decrease**?

A. Closing the equipment hatch.

B. Stopping XFN-11A.

C✓ Stopping XFN-13A.

D. Starting XFN-20, FUEL BLDG SPLY FAN.

Proposed for use on 2018 NRC exam - rj

QUESTION USAGE:
Question 60 on **2016 NRC exam**.

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: SB
Approved: B. Moore 5/18/16

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must identify the effect of changing ventilation during a refueling condition with the fuel transfer tube open.

A. Plausible because closing the equipment hatch will change RB pressure due to closing a path for outside air to the RB at negative pressure which would affect refueling cavity level with the fuel transfer tube open.

   Incorrect because since there are more purge exhaust fans running than supply fans, RB pressure is initially negative. Closing a path for incoming air will cause RB pressure to trend more negative which would cause refueling cavity level to increase as water is transferred from the spent fuel pool.

B. Plausible because stopping the running purge supply fan will change RB pressure causing it to trend more negative. This would affect refueling cavity level with the fuel transfer tube open.

   Incorrect because RB pressure trends more negative which would cause refueling cavity level to increase as water is transferred from the spent fuel pool.

C. CORRECT. Stopping a running purge exhaust fan will cause RB pressure to increase which will cause water to be transferred from the reactor cavity to the spent fuel pool. This will cause reactor cavity level to decrease.

D. Plausible because starting XFN-20, FUEL BLDG SPLY FAN. will change FHB pressure and cause water to be transferred through the fuel transfer tube.

   Incorrect because starting XFN-20, FUEL BLDG SPLY FAN will raise FHB pressure and cause water to be transferred to the RB. This will cause refueling cavity level to increase.

K/A: 103 A1.01 Containment - Ability to predict and/or monitor changes in parameters associated with operating the system controls including: Containment pressure, temperature and humidity

K/A Match: The KA is matched because the candidate must determine the effect of stopping a ventilation fan on containment pressure.

Selection criteria; **BANK**

Tier: 2 **Group:** 1
Importance Rating: RO 3.7 SRO 4.1
Technical Reference: **SOP-114 REACTOR BUILDING VENTILATION SYSTEM**

Proposed references to be provided to candidates during examination: None

Learning Objective: AB017 - 016. DESCRIBE the normal operation of the Reactor Building Ventilation System including: 1. Purge and Supply

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(4)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

55. Given the following plant conditions:

Time 1600:

- Mode 6.
- The core has been reloaded.
- The Reactor Cavity is at the normal level for refueling.
- Mechanical maintenance reports that **neither** the inner or outer personnel hatch doors will close.

Time 1605:

- RHR has been lost.

Which ONE of the following identifies the **minimum** actions(s), if any, that must be done to meet the applicable requirement for containment penetrations?

A. **No** actions are required; Containment closure is **not** required for the above conditions.

B.✓ A **minimum** of one personnel hatch door must be restored to operation and closed to satisfy the requirements of OAP-108.4, OPERATIONS OUTAGE CONTROL OF CONTAINMENT PENETRATIONS.

C. **Both** personnel hatch doors must be restored to operation and closed to satisfy the requirements of OAP-108.4, OPERATIONS OUTAGE CONTROL OF CONTAINMENT PENETRATIONS.

D. A **minimum** of one personnel hatch door must be restored to operation and closed to satisfy the requirements of T.S. 3.6.1.1 CONTAINMENT INTEGRITY.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 Submitted by SAR
Ops Review:
Approved:

DISTRACTOR ANALYSIS:

A. Plausible because the unit is in mode 6 and below Technical Specification requirements for Containment Integrity.

Incorrect because OAP-108.4 requires closure of one penetration during a loss of RHR.

B. CORRECT. At least one penetration must be closed when containment closure is set.

C. Plausible because Technical Specification requirements of T.S. 3.6.1.1 typically requires two isolations methods operable in each penetration.

Incorrect because OAP-108.4 requires closure of only one penetration for the given conditions.

D. Plausible because Technical Specification 3.6.1.1 is applicable in Mode 4.

Incorrect because Technical Specification 3.6.1.1 is not in effect in Mode 6.

K/A: 103 K3.03 Containment - Knowledge of the effect that a loss or malfunction of the system will have on the following: Loss of containment integrity under refueling operations.

K/A Match: The KA is matched because the candidate must assess the effects of a loss of containment integrity and determine the minimum action required for restoration.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 3.7 SRO 4.1
Technical Reference: **OAP-108.4**
T.S. DEFINITIONS
T.S. 3.6.1.1

Proposed references to be provided to candidates during examination: None

Learning Objective: OAP-108.4 -04 DESCRIBE the penetration Status requirements for the following conditions: • Modes 5, 6 & defueled • During Core Alterations • When containment closure is set

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

56. Given the following plant conditions:

- 100% power
- One Control Bank D rod has just dropped.
- The following Power Range NI indications exist:

  PR N-41 - 96.6%
  PR N-42 - 101.8%
  PR N-43 - 102.1%
  PR N-44 - 103.1%

Which ONE of the choice below completes the following statements?

Rod withdrawal __(1)__ under the current conditions.

The rod block is caused by the power levels measured by __(2)__.

A. 1) can be done manually, but automatic rod withdrawal is blocked,
   2) N-44 **only**.

B. 1) can be done manually, but automatic rod withdrawal is blocked,
   2) N-43 **and** N-44.

C.✓ 1) is **not** possible manually or automatically
   2) N-44 **only**.

D. 1) is **not** possible manually or automatically
   2) N-43 **and** N-44.

Revised without significant modification for 2018 NRC - rj

QUESTION USAGE
RO-15-01-EXAM 8 (I&C 2)
RO-14-01-EXAM 8 (I&C 2)
RO-13-01-EXAM 8 (I&C 2)
RO-11-01-IPO-2 Exam
RO-10-01-IPO-2 Exam

REVISION HISTORY
Rev 2 submitted by RJ - restructured question
OPS Review: Danny Rhymer
Approval:

Rev. 1 Taffar 2/7/2013 Removed "A single dropped control rod will normally trip the reactor on high negative flux rate, but there are a few control rods that will NOT cause the negative flux rate trip." from answer C. They is no longer a negative rate trip. Removed "the reactor does NOT trip and " from the stem.

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate identify a rod block in effect due to an RPS channel actuation.

A. Plausible because control interlocks C-5 and C-11 block only automatic rod motion. The second part is correct.

   Incorrect because manual rod motion is blocked.

B. Plausible because control interlocks C-5 and C-11 block only automatic rod motion. The second part is plausible because N-43 and N-44 are the two highest and because rod blocks due to OT and OPDT require two channels.

   Incorrect because manual rod motion is blocked and because the block only is met by N-44.

C. CORRECT. At 103% power on any power range channel, a rod stop is initiated by C-2. Both manual and automatic rod withdrawal are blocked. Only PR-N44 is greater than the setpoint.

D. The first part is correct. The second part is plausible because N-43 and N-44 are the two highest and because rod blocks due to OT and OPDT require two channels.

   Incorrect because the block only is met by N-44.

K/A: 001 K1.05 Control Rod Drive - Knowledge of the physical connections and/or cause-effect relationships between the system and the following: NIS and RPS

K/A Match: The KA is matched because the candidate must identify a rod block in effect due to an RPS channel actuation.

Selection criteria; **REVISED BANK**

Tier: 2 **Group:** 2
Importance Rating: RO 4.5 SRO 4.4
Technical Reference: **1MS-41-011-0009**
SOP-401

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-9 03. DESCRIBE the Rod Control System interfaces with the following systems and/or subsystems: 3. Nuclear Instrumentation System

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

ROD CONTROL 132

Given the following plant conditions:

- The plant was at 100% power
- One Control Bank D rod is dropped.

The following NI indications exist:

- PR N-41 - 96.6%
- PR N-42 - 101.8%
- PR N-43 - 102.1%
- PR N-44 - 103.1%

Which ONE of the following describes the status of the Rod Control System (assuming no operator actions are taken)?

A. Manual rod withdrawal is available; automatic rod withdrawal is blocked due to the NIS POWER RANGE OVERPOWER ROD STOP caused by N-44 ONLY.

B. Manual rod withdrawal is available; automatic rod withdrawal is blocked due to the NIS POWER RANGE OVERPOWER ROD STOP caused by N-43 AND N-44.

C. ✓ Manual and automatic rod withdrawal are blocked due to the NIS POWER RANGE OVERPOWER ROD STOP caused by N-44 ONLY.

D. Manual and automatic rod withdrawal are blocked due to the NIS POWER RANGE OVERPOWER ROD STOP caused by N-43 AND N-44.

57. Given the following plant conditions:

- 100% power initially.
- A small break LOCA occurred.
- RCS Pressure stabilized at 1500 psig.
- EOP-2.1, ES-1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION is in progress.
- RCPs "A" and "C" were stopped due to high vibration.
- "B" RCP is running.
- Letdown is isolated.
- An RCS depressurization is required to refill the pressurizer.

Which ONE of the following describes the **first** method that will be directed for use to manually reduce Reactor Coolant System pressure in EOP-2.1?

A. Open PCV-444D **and** PCV-444C, PZR SPRAY.

B. Open **only** PCV-444D, PZR SPRAY.

C. Use Auxiliary Spray.

D.✓ Open a Pressurizer PORV.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must identify the pressure control that must be used after pressurizer spray is lost due to a loss of reactor coolant pumps.

A. Plausible because normally both Pressurizer Spray valves are in service and the master controller can be used to control both spray valves.

   Incorrect because pressurizer spray valves cannot be used with only "B" RCP running.

B. Plausible because only one Pressurizer spray valve is used in EOP-2.1.

   Incorrect because pressurizer spray valves cannot be used with only "B" RCP running.

C. Plausible because auxiliary spray is used after letdown is restored

   Incorrect because aux spray cannot be used at the step to refill the pressurizer.

D. CORRECT. Normal Spray is not available when only "B" RCP is running. A Pressurizer PORV is used for the depressurization to refill the pressurizer.

K/A: 002 K5.14 Reactor Coolant - Knowledge of the operational implications of the following concepts as they apply to the system: Consequences of forced circulation loss

K/A Match: The KA is matched because the candidate must identify the pressure control that must be used after pressurizer spray is lost due to a loss of reactor coolant pumps.

Selection criteria; NEW

Tier: 2 **Group:** 2
Importance Rating: RO 3.8 SRO 4.2
Technical Reference: **EOP-2.1**

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.1 07. RELATE any systems'/components' operation, indication, or malfunction to its effect on EOP-2.1

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

58. Initial conditions:

- Reactor startup is in progress.
- Power on Source Range NI channels is $5x10^2$ cps and increasing.
- Power on Intermediate Range NI channels is $3x10^{-7}$ % power and increasing.
- The NROATC placed **only** the SR TRAIN **A** Switch in BLOCK.

Current conditions:

- Time is **0404**.
- Power has been increasing steadily.
- Source Range Startup Rate is positive 0.5 DPM with **no** rod movement.
- Power on Source Range NI channels s is $4x10^4$ cps and increasing.
- Power on Intermediate Range NI channels is $8x10^{-5}$ % power and increasing.
- The NROATC placed **only** the SR TRAIN **B** Switch in BLOCK.

Which ONE of the following identifies the reading that is **closest** to the Source Range Startup Rate that will be present **two minutes** after the current conditions (time 0406)?

A. 0 DPM.

B.✓ 0.3 DPM negative.

C. 0.5 DPM negative.

D. 0.5 DPM positive.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall

A. Plausible because the normal post shutdown SUR is 0 after power has stabilized in the source range.

Incorrect because power will not have stabilized at 0406.

B. CORRECT. "A" train was not blocked satisfactorily since P-6 had not yet been met in the initial conditions. At time 0404, power was approximately 1/2 decade from the trip setpoint of 10^5 cps with a SUR of 0.5 dpm positive. The reactor trip occurred at approximately 0405. At 0406, or one minute after the trip, power would be lowering at 0.3 dpm negative.

C. Plausible because the SUR is negative and the bottom scale of the meter is 0.5 dpm. The candidate may believe that the reading is at the bottom peg due to the reactivity introduced by the trip.

Incorrect because a stable 0.5 dpm negative SUR is not possible after a trip.

D. Plausible because the candidate may believe that both trains of Source Range trip were successfully blocked and that a trip did not occur.

Incorrect because a trip occurred and SUR is not positive.

K/A: 015 A1 .02 Nuclear Instrumentation - Ability to predict and/or monitor changes in parameters associated with operating the system controls including: SUR

K/A Match: The KA is matched because the candidate must predict a SUR after a trip occurs from nuclear instrumentation signals.

Selection criteria; NEW

Tier: 2 **Group:** 2
Importance Rating: RO 3.5 SRO 3.6
Technical Reference: **GOP-3**
MAIN CONTROL BOARD LAYOUT
GFES - REACTOR OPERATION PHYSICS

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-8 015. DESCRIBE the component operation associated with each position for the following switches of the Intermediate Range Instrumentation System: 1. Intermediate range block switches

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(5)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

59. The following plant conditions exist:

- 100% power initially.
- A large rapid load reduction has occurred.
- Pressure is 2280 and increasing rapidly.

Given the conditions above, which of the choices below answers both of the following questions?

1) Which Pressurizer PORV is expected to open **first** to reduce pressure during a transient such as the one above?

2) What describes the expected control board meter reading for the associated pressure instrument when the first PORV lifts?

Assume an instrument error of 0 psig for the associated meter reading.

A. 1) PCV-444B.
   2) At the setpoint of 2335 psig.

B. 1) PCV-445A.
   2) At the setpoint of 2335 psig.

C.✓ 1) PCV-444B.
   2) Less than 2335 psig.

D. 1) PCV-445A.
   2) Less than 2335 psig.

Modified pzrpresscntlsys5 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine a PORV actuation and recall the anticipatory characteristic of the associated controller.

A. The first part is correct. The candidate may assume that the first PORV opens at the fixed setpoint of 2235 psig and that the others are at a higher setpoint such as the case with steam generator safeties.

   Incorrect because PCV-444B will open below the setpoint of 2235 psig.

B. The first part is plausible because one PORV will open first. The candidate may assume that the first PORV opens at the fixed setpoint of 2235 psig and that the others are at a higher setpoint such as the case with steam generator safeties.

   Incorrect because PCV-445A will not open first.

C. CORRECT. PCV-444B is controlled by the master pressure controller. As Pressurizer pressure increases, the integration of error will be added to the proportional control which will cause the PCV-444B to open in advance of the proportional setpoint of 2235 psig. The other two PORVs open at a fixed setpoint of 2235 psig.

D. The first part is plausible because PCV-445A will open at a high pressure and the candidate may assume that this PORV is controlled by the master pressure controller.

   Incorrect because PCV-445A will not open first.

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

K/A Match: The KA is matched because the candidate must determine a PORV actuation and recall the anticipatory characteristic of the associated controller.

Selection criteria; **MODIFIED FROM PZRPRESSCNTRLSYS5**

Tier: 2 **Group:** 2
Importance Rating: RO 2.9 SRO 2.9
Technical Reference: **DBD - REACTOR COOLING SYSTEM**
DRAWING 1MS-41-011-0016

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-3 07. DESCRIBE the normal operation of the following pressurizer pressure and level control system components. Include component types and applicable setpoints: 2. Pressurizer Power Operated Relief Valves

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

PZR PRESS CNTRL SYS 005

The following plant conditions exist:

- The plant was operating at 100% power.
- A large rapid load reduction has just occurred, causing pressurizer pressure to approach the PORV setpoints.
- NO additional operator action is taken.

Which of the following states which PORV, PCV-444B or PCV-445B, is expected to open first in response to the pressure increase and why?

A.✓ PCV-444B, because the Master Pressure Controller has a rate compensation circuit.

B. PCV-445B, because its bistable control is quicker-acting than the Master Pressure Controller.

C. PCV-444B, because its bistable control is quicker-acting than the Master Pressure Controller.

D. PCV-445B, because the Master Pressure Controller has a rate compensation circuit.

60. Given the following plant conditions:

- Mode 6.
- The Audio Count Rate circuit is in operation in accordance with SOP-404 EXCORE NUCLEAR INSTRUMENTATION SYSTEM.
- The CRS has directed the BOP to reduce the frequency of the audible beeping of the Audio Count Rate.
- Removal of the first fuel assembly from the core will be done on the next shift.

In order to change the time period between the audible beeps the operators will adjust the __(1)__ in accordance with SOP-404.

In accordance with T.S. 3.9.2, REFUELING OPERATIONS - INSTRUMENTATION, the Audio Countrate function __(2)__ required to be OPERABLE for the conditions above.

A. 1) SR COUNTER/SCALER thumbwheel
   2) is **not**

B. 1) SR COUNTER/SCALER thumbwheel
   2) is

C. 1) AUDIO MULTIPLIER switch
   2) is **not**

D✓ 1) AUDIO MULTIPLIER switch
   2) is

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the applicant must recall the operation of neutron monitoring instrumentation required in refueling mode.

A. The first part is plausible it is a control that is used while setting up the time interval for sampling in accordance with SOP-404. The second part is plausible because fuel is not yet being moved so the candidate and assume that the requirement only exists during core alterations such as the case for T.S. 3.9.5 for communications.

   Incorrect because thumbwheel is not used to adjust the frequency of beeps and the audio countrate circuit is required in Mode 6 at all times.

B. The first part is plausible it is a control that is used while setting up the time interval for sampling in accordance with SOP-404.

   Incorrect because thumbwheel is not used to adjust the frequency of beeps

C. The first part is correct. The second part is plausible because fuel is not yet being moved so the candidate and assume that the requirement only exists during core alterations such as the case for T.S. 3.9.5 for communications.

   Incorrect because the audio countrate circuit is required in Mode 6 at all times.

D. CORRECT. The time period between the audible beeps is adjusted by operators using the AUDIO MULTIPLIER switch. In accordance with T.S. 3.9.2, REFUELING OPERATIONS - INSTRUMENTATION, the Audio Countrate function is required to be OPERABLE in Mode 6.

K/A: 034 A4.02 Fuel Handling Equipment — - Ability to manually operate and/or monitor in the control room: Neutron levels

K/A Match: The KA is matched because the candidate must recall the operation of neutron monitoring instrumentation required in refueling mode.

Selection criteria; NEW

Tier: 2 **Group:** 2
Importance Rating: RO 3.5 SRO 3.9
Technical Reference: **T.S. 3.9.2**
SOP-404

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-8 042. DESCRIBE the operation of the following components of the Nuclear Instrumentation System: 3. Audio count rate circuit

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

61. Given the following plant conditions:

- 25% power.
- The "A" MSIV inadvertently closes.
- All other systems are normal.

Which ONE of the choices below answers **both** of the following questions?

1) Which SG(s) would show an **initial increase** in level following this event (within the first 15 seconds), and
2) How would "A" steam generator pressure relate to "B" and C" steam generators after conditions stabilize.

ASSUME NO OPERATOR ACTIONS

A. 1) SG "A" level would initially increase.
   2) SG "A" pressure will be higher than the pressure in the "B" and "C" SGs.

B. 1) SG "A" level would initially increase.
   2) SG "A" pressure will be the same as that in the "B" and "C" SGs.

C. 1) SG "B" and "C" levels would initially increase.
   2) SG "A" pressure will be the same as that in the "B" and "C" SGs.

D.✓ 1) SG "B" and "C" levels would initially increase.
   2) SG "A" pressure will be higher than the pressure in the "B" and "C" SGs.

Proposed for use on 2018 NRC - rj

QUESTION USAGE:
RO-14-01 Exam 2 (Secondary)

REVISION HISTORY:
Rev. 0 Submitted by MRB 2/13/13 FOR 2013 AUDIT.
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall

A. The first part is plausible because if feedwater remains constant, lower steam load would cause wide range level, and eventually narrow range, to increase. The second part is correct.

   Incorrect because "A" Steam Generator level will decrease due to shrink.

B. The first part is plausible because if feedwater remains constant, lower steam load would cause wide range level, and eventually narrow range, to increase. The second part is plausible because steam generator pressure is governed in part by RCS loop Tcolds. The candidate may assume that the generator pressures are governed by T_{HOTs} and they are the same going to the three Steam Generators.

   Incorrect because "A" Steam Generator level will decrease due to shrink and "A" Steam Generator pressure will increase above the other two steam generators.

C. The first part is correct. The second part is plausible because steam generator pressure is governed in part by RCS loop Tcolds. The candidate may assume that the generator pressures are governed by T_{HOTs} and they are the same going to the three Steam Generators.

   Incorrect because "A" Steam Generator pressure will increase above the other two steam generators.

D. CORRECT. With "A" MSIV closed, "B" and "C" steam generators will assume more steam load and decrease in pressure which will cause the NR levels to increase due to swell. Since "A" Steam generator is no longer steaming and is isolated, that generator pressure will increase.

K/A: 035 K6.01 Steam Generator - Knowledge of the of the effect of a loss or malfunction on the following will have on the system: MSIVs

K/A Match: The KA is matched because the candidate must determine the effect of a inadvertent closure on steam generator levels and pressures.

Selection criteria; **BANK**

Tier: 2 **Group:** 2
Importance Rating: RO 3.2 SRO 3.6
Technical Reference: **WESTINGHOUSE TECHNOLOGY SYSTEM MANUAL 11.1 (ML1123A293)**
E-3 BASIS

Proposed references to be provided to candidates during examination: None

Learning Objective: TS-6 316 EXPLAIN the phenomenon of swell citing its cause and relative duration.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(5)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

62. Given the following plant conditions:

Time 0945:

- 13% power.
- GOP-4B POWER OPERATION (MODE 1 - DESCENDING) in progress.
- The STM DUMP MODE SEL switch is in TAVG mode.
- Power Range channel N41 is inoperable and has been taken out of service.
  - N41 Control and Instrument power fuses have been removed.

Time 1000:

- All power output from XIT-5902 has been lost.

Which ONE of the choices below completes the following statement?

At 1000, Condenser steam dumps __(1)__ armed and they are currently __(2)__.

A. 1) are **not**
2) shut

B. 1) are **not**
2) open at a position based on the pot setting on STM DUMP CNTRL.

C.✓ 1) are
2) open at a position based on difference between T_{AVG} and 557°F.

D. 1) are
2) open at a position based on difference between T_{AVG} and 561°F.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine the effect of a loss of an inverter output on the reactor plant and the operation of steam dumps.

A. The first part is plausible because the candidate may not realize that the reactor has tripped due to not analyzing the effect of the loss of XIT-5902 on PR N-42. The second part is plausible because the steam dumps are normally shut after the Main Generator is placed in service.

   Incorrect because the reactor has tripped due which has armed the steam dumps and the steam dumps are open.

B. The first part is plausible because the candidate may not realize that the reactor has tripped due to not analyzing the effect of the loss of XIT-5902 on PR N-42. The second part is plausible because the pot setting determine the opening of steam dumps in steam pressure mode.

   Incorrect because the reactor has tripped due which has armed the steam dumps and the steam dumps opening is not controlled by STM DUMP CONTROL.

C. CORRECT. N-42 high flux bistables are tripped because the control power fuses have been removed. XIT5902 has deenergized which has removed control power from N-44 high flux bistables. The reactor has tripped on a 2/4 coincidence of the high flux trip function. The steam dumps have been armed by the operation of the P-4 permissive after the reactor tripped and are controlling the post trip mode which is based on a fixed TAVG setpoint of 557°F.

D. The first part is correct. The steam dumps have been armed by the operation of the P-4 permissive after the reactor tripped. The second part is plausible the candidate may believe that steam dumps are being controlled by the difference between TAVG and TREF. TREF for 13% -> 557°F +(0.13 x 30°F) = 561°F.

   Incorrect because steam dumps are controlling to a fixed setpoint of 557°F.

K/A: 041 K2.02 Steam Dump/Turbine Bypass Control - Knowledge of the electrical power supplies to the following: ICS inverter breakers

K/A Match: The KA is matched because the candidate must determine the effect of a loss of an inverter output on the reactor plant and the operation of steam dumps.

Selection criteria; NEW

Tier: 2 **Group:** 2
Importance Rating: RO 2.8 SRO 2.8
Technical Reference: **SOP-401**
DRAWING 1MA-41-011-0010

Proposed references to be provided to candidates during examination: None

Learning Objective: IC-1 012. DESCRIBE the operation of the following control systems associated with the Steam Dump Control System, including function, instrumentation and setpoints: 2. Tavg - plant trip control mode

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(7)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

63. Given the following plant conditions:

Time 0900:

- 35% power.
- Main Turbine control valve positions are as follows:

  CV1 - 12% and stable.
  CV2 - 12% and stable.
  CV3 - 12% and stable.
  CV4 - 0%

Time 0901:

- All alarms on FIRST OUT XCP-626 are **clear**.
- XCP-614, 5-3, LO EH FLUID PRESS TURB STP VLV CLOSD is in alarm.
- XCP-631, 1-1, EHC TURB TRP is in alarm.
- XCP-631, 1-2, EHC FLUID PRESS LO is in alarm.
- Main Turbine control valve positions are as follows:

  CV1 - 0%.
  CV2 - 0%.
  CV3 - 0%.
  CV4 - 0%.
- The crew has entered AOP-214.1 TURBINE TRIP.

Which ONE of the following is an **immediate action step** that will be directed by AOP-214.1 for the current condition?

A.✓ Trip the reactor.

B. Ensure XCB-8902 is open.

C. Establish Emergency Feedwater to all SGs.

D. Depress both MASTER TRIP/EMERGENCY TRIP pushbuttons.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine that a reactor trip is required due to a malfunction of EHC control.

A. CORRECT. The turbine has tripped due to a loss of EHC system pressure. An immediate action step is power was initially greater than 30% is to trip the reactor.

B. Plausible because an immediate action is to ensure that Main Generator output breaker are open and 8902 is downstream of that breaker.

   Incorrect because 8902 is not verified open.

C. Plausible because checking Main Feedwater is an immediate action step.

   Incorrect because checking operation of EFW is not an immediate action step.

D. Plausible because the MASTER TRIP/EMERGENCY TRIP pushbuttons are depressed if the stop valves are open.

   Incorrect because the MASTER TRIP/EMERGENCY TRIP pushbuttons will not be depressed for with the stop valves closed as given in the conditions.

K/A: 045 A2.17 Main Turbine Generator - Ability to (a) predict the impacts of the following on the system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunction of electrohydraulic control

K/A Match: The KA is matched because the candidate must determine that a reactor trip is required due to a malfunction of EHC control.

Selection criteria; NEW

Tier: 2 **Group:** 2
Importance Rating: RO 2.7 SRO 2.9
Technical Reference: **AOP-214.1**

Proposed references to be provided to candidates during examination: None

Learning Objective: AOP-214.1 02. STATE/IDENTIFY the following for AOP-214.1:
a. Required Immediate Actions

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

64. Given the following plant conditions:

- 100% power.
- Condenser Vacuum Pumps "A" and "C" are in service.

Which ONE of the following will cause a Main Turbine trip due to a loss of Main Condenser vacuum?

A. Loss of output from XTF0032, EMERG AUX XFMR.

B. ✓ Closure of MVB-102B, VAC PP C TO CNDSR B.

C. Trip of XPP-0042A, CO PUMP A.

D. Opening of MOV-1-5A TURB DRN VLV.

Proposed for use on 2018 NRC - rj

QUESTION USAGE:
2017 NRC

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

A. Plausible because XTF0032 is the alternate supply for busses 7.2 KV busses 1A and 1B which carry circulating water pumps. A trip of both of those pumps would cause a loss of vacuum.

   Incorrect because XTF0032 does not carry buses 1A and 1B in a normal lineup.

B. Plausible because the loss of a 7.2 KV circulating water pump could cause a loss of vacuum and XPP-0042A is a 7.2 KV condensate pump. Additionally, SOP-208 cites an air leakage concern then the condenser is under vacuum.

   Incorrect because a trip of that pump would not cause a loss of vacuum.

C. CORRECT. Both cross-over valves must be opened for the use of CVP “C” and closing MVB-102B would cause a loss of vacuum.

D. Plausible because SOP-210, FEEDWATER contains a caution that if only two circulating water pumps are operating that vacuum could be lost if MOV-1-5A is open.

   Incorrect because all three circulating water pumps are operating.

K/A: 055 K3.01 CondenserAirRemoval - Knowledge of the effect that a loss or malfunction of the system will have on the following: Main condenser

K/A Match: The KA is matched because the candidate must determine the effect that a loss of a condenser vacuum pump will have on condenser pressure.

Selection criteria; **BANK**

Tier: 2 **Group:** 2
Importance Rating: RO 2.5 SRO 2.7
Technical Reference: **SOP-206 MAIN AND AUXILIARY CONDENSER AIR REMOVAL SYSTEM**

Proposed references to be provided to candidates during examination: None

Learning Objective: TB-6 012 EXPLAIN the effects of various failures of the Condensate System during normal power operations, including the following: 1. Loss of a main condenser vacuum pump

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(4)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

65. Which ONE of the following conditions will close RCV00018-WL, LIQUID RADIOACTIVE WASTE CONTROL VALVE, during the release of a Waste Monitor Tank?

NOTE: RM-L5, as referenced below, is the LIQUID WASTE EFFLUENT LIQUID RADIATION MONITOR.

A. A low sample flow through RM-L5.

B. A low dilution flowrate from the Fairfield Penstocks.

C.✓ A complete loss of power to RM-L5.

D. A low level in the Waste Monitor Tank.

Modified lwr124 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall

A. Plausible because a low flow condition will activate the FAIL light for RM-L5

   Incorrect because a low flow condition will not close RCV-018.

B. Plausible because a low dilution flow rate will close PVD-6910.

   Incorrect because a low dilution flow rate does not close RCV-018.

C. CORRECT. A loss of power to RM-L5 will close RCV-018.

D. Plausible because a low level of 10% in the WMT will trip off the transfer pump and stop the release.

   Incorrect because this does not close RCV-018.

K/A: 068 K4.01 Liquid Radwaste - Knowledge of system design feature(s) and or interlock(s) which provide for the following: Safety and environmental precautions for handling hot, acidic and radioactive liquids

K/A Match: The KA is matched because the candidate must recall that a liquid release valve will close on interlock after a loss of power to an associated radiation monitor.

Selection criteria; **MODIFIED FROM LWR124**

Tier: 2 **Group:** 2
Importance Rating: RO 3.4 SRO 4.1
Technical Reference: **ARP-XCP-646**
DRAWING 208-059

Proposed references to be provided to candidates during examination: None

Learning Objective: GS-9 018. DESCRIBE the trips and automatic actions associated with the following radiation monitors. Include the purpose of each trip: 5. RM-L5, liquid waste effluent

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(11)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

LIQUID RAD WASTE 124

Which ONE of the following conditions will close PVD-6910, LIQUID EFFLUENTS TO FAIRFIELD PENSTOCKS during the release of a Waste Monitor Tank?

NOTE: RM-L5, as referenced below, is the LIQUID WASTE EFFLUENT LIQUID RADIATION MONITOR.

A. A low sample flow through RM-L5.

B✓ A low dilution flowrate from the Fairfield Penstocks.

C. A complete loss of power to RM-L5.

D. A low level in the Waste Monitor Tank.

66. Which ONE of the following identifies a position that may enter the Green Carpet Area in the Control Room without BOP or NROATC approval in accordance with SAP-200, CONDUCT OF OPERATIONS?

A✓ Shift Engineer.

B. Management Duty Supervisor.

C. Control Building Watch.

D. Work Control Center SRO.

Revised without significant modification for 2018 NRC - rj

QUESTION USAGE:
RO-15-01 ADMIN EXAM
RO-13-01 ADMIN EXAM
RO-11-01-NRC (**2013-RO NRC**)

REVISION HISTORY:
Rev 1 submitted by rj - replaced one distractor
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must recall the position that is allowed to enter the area adjacent to the Main Control Boards without obtaining permission from the control room watch standers.

A. CORRECT, The Shift Engineer may enter the Green Carpeted Area without RO or BOP approval.

B. Plausible because the Shift Manager is allowed to enter the Green Carpeted Area without RO or BOP approval.

   Incorrect because the Management Duty Supervisor is not allowed to enter without approval.

C. Plausible because the Control Building Watch desk is stationed in the control building envelope at a desk adjacent to the control area and is often tasked with clearing tags at the control boards.

   Incorrect because the Control Building Watch is not allowed to enter without approval.

D. Plausible because the Work Control Center has unrestricted access to the Area of Continuous Attention and the access to the Control Room is controlled by the Work Control Center.

   Incorrect because the Work Control Center SRO is not allowed to enter without approval.

K/A: G2.1.1 Conduct of operations - Knowledge of conduct of operations requirements.

K/A Match: the KA is matched because it requires the candidate to identify a position that is authorized to enter the Green Carpeted Area of the Control Room without prior authorization.

Selection criteria; **REVISED BANK**

Tier: 3 **Group:**
Importance Rating: RO 3.8 SRO 4.2
Technical Reference: **SAP-200, CONDUCT OF OPERATIONS**

Proposed references to be provided to candidates during examination: None

Learning Objective: SAP-200-03 DESCRIBE the requirements of SAP-200 for the following:• Control Room Access

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

Which ONE of the following identifies a position that may enter the Green Carpet Area in the Control Room without BOP or NROATC approval in accordance with SAP-200, CONDUCT OF OPERATIONS.

A.✓ Shift Engineer.

B. Management Duty Supervisor.

C. NRC Resident Inspector.

D. Work Control Center SRO.

67. Given the following conditions:

Time 0715:

- 100% power.
- The day shift turnover meeting started.

Time 0725:

- The meeting ended and the day shift assumed the watch.

Given the conditions above, which ONE of the choices completes the following statement?

In accordance with OAP-106.1, OPERATING ROUNDS, the **latest** time by which Tech Spec Rounds should be taken is by ____.

A. 0925.

B.✓ 0930.

C. 1315.

D. 1330.

NOTE TO EXAMINER:
THIS QUESTION WAS SUBMITTED AS PART OF SAMPLE AND HAS BEEN REVISED TO ADDRESS COMMENTS

Modified admin122 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall operator responsibilities for log taking during normal operation.

A. Plausible because the Tech Spec rounds are required to be completed in the first two hours of the shift and the day shift assumed the watch at 0725.

   Incorrect because the shift starts at 0730 for the purpose of log taking.

B. CORRECT. The shift starts at 0730 for the purpose of log taking and Tech Spec logs are required to be completed in the first two hours of the shift.

C. Plausible because the supervisory review is required to be completed in the first half of the shift and the shift meeting began at 0715. 0715 + 6 hours is 1315.

   Incorrect because Tech Spec rounds must be completed by 0730.

D. Plausible because the supervisory review is required to be completed in the first half of the shift and the shift begins at 0730 for the purpose of log taking. 0730 + 6 hours is 1330.

   Incorrect because Tech Spec rounds must be completed by 0730.

K/A: G2.1.2 Conduct of operations - Knowledge of operator responsibilities during all modes of plant operation.

K/A Match: The KA is matched because it requires the candidate to recall reactor operator responsibilities for log taking.

Selection criteria; **MODIFIED FROM ADMIN122**

Tier: 3 **Group:**
Importance Rating: RO 4.1 SRO 4.4
Technical Reference: **OAP-106.1, OPERATING ROUNDS**

Proposed references to be provided to candidates during examination:

Learning Objective: **OAP-100.6** 010 **STATE** the logkeeping expectations as described in OAP-100.6.

Question Cognitive Level: **Memory or Fundamental Knowledge** ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

ADMIN PROCEDURE 122

Due to operations in progress, shift turnover for the 1930-0730 shift has taken until 2015 to complete.

Which ONE of the following identifies the latest time that the Technical Specifications Logs must be completed by the on-coming shift in accordance with OAP-106.1, Operating Rounds?

A. 2100

B. 2115

C✓. 2130

D. 2145

68. Which ONE of the choices below completes the following statements in accordance with OAP-100.6 CONTROL ROOM CONDUCT AND CONTROL OF SHIFT ACTIVITIES?

OAP 100.6, Attachment IA – REACTIVITY CONTROL PARAMETERS should be maintained by __(1)__.

In accordance with OAP-100.6, the __(2)__ is an item that should be read at the Shift Turnover Meeting.

A. 1) Reactor Engineering.
   2) number of rod steps to change RCS T_{AVG} one degree

B. 1) Reactor Engineering.
   2) expected boric acid total gallons on a normal auto makeup

C. 1) the Reactor Operator.
   2) number of rod steps to change RCS T_{AVG} one degree

D.✓ 1) the Reactor Operator.
   2) expected boric acid total gallons on a normal auto makeup

NOTE TO EXAMINER:
THIS QUESTION WAS SUBMITTED AS PART OF SAMPLE AND HAS BEEN REVISED TO ADDRESS COMMENTS

Modified admin 426 for 2018 NRC- rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the position that maintains the attachment used in the control room for reactivity control parameters and the information that is available for immediate use on that attachment.

A. The first part is plausible because Reactor Engineering provides a review of the attachment on Monday mornings. The second part is plausible because the number of rod steps to change TAVG is contained on Attachment 1A and is discussed by the shift at the control boards.

   Incorrect because Reactor Engineering does not calculate values contained in Attachment IA and rods steps for temperature change is not discussed at the shift meeting.

B. The first part is plausible because Reactor Engineering provides a review of the attachment on Monday mornings. The second part is correct.

   Incorrect Reactor Engineering does not calculate values contained in Attachment IA.

C. The first part is correct. The second part is plausible because the number of rod steps to change TAVG is contained on Attachment 1A and is discussed by the shift at the control boards.

   Incorrect because rods steps for temperature change is not discussed at the shift meeting.

D. CORRECT. Attachment IA is maintained by the reactor operator and expected boric acid total gallons on a normal auto makeup is discussed as part of Attachment 1B part 1.

K/A: G2.1 .43 Conduct of operations - Ability to use procedures to determine the effects on reactivity of plant changes

K/A Match: The KA is matched because it requires the candidate to recall procedure use to determine reactivity additions during normal plant operations.

Selection criteria; **MODIFIED FROM ADMIN426**

Tier: 3 **Group:**
Importance Rating: RO 4.1 SRO 4.3
Technical Reference: **OAP-100.6**
SAP-205

Proposed references to be provided to candidates during examination:

Learning Objective: OAP-100.6 07 DISCUSS Reactivity Management expectations as described in OAP-100.6.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

ADMIN PROCEDURE 426

Given the following plant conditions:

- 50% power

Which ONE of the following identifies the plant position that maintains OAP-100.6, CONTROL ROOM CONDUCT AND CONTROL OF SHIFT ACTIVITIES, ATTACHMENT IA – REACTIVITY CONTROL PARAMETERS,
AND describes what will happen to the value of Moderator Temperature Coefficient/Differential Boron Worth (MTC/DBW = PPM/°F) as burnup increases at the End of Life?

A. Shift Engineer; lowers

B. Shift Engineer; rises

C. Reactor Operator; lowers

D.✓ Reactor Operator; rises

69. Given the following plant conditions:

- 100% power.
- The 7.2 KV breaker 1C ALT FEED was replaced and racked up.
- A test closure was attempted and the breaker failed to close.

Which ONE of the following identifies what will be performed by Operations for the **current condition** in accordance with OAP-100.5, GUIDELINES FOR CONFIGURATION CONTROL AND OPERATION OF PLANT EQUIPMENT?

A. Generate an R&R.

B.✔ Sequester the breaker.

C. Generate a troubleshooting plan.

D. Attempt a **maximum** of 1 reclosure.

Modified from admin389 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall that a breaker is sequestered after a closure failure to allow for troubleshooting.

A. Plausible because an R&R would be generated for a Technical Specification required component.

   Incorrect because an R&R is not required for the alt feed breaker to bus 1C.

B. CORRECT. For any 7.2 KV breaker malfunction, the breaker will be sequestered if possible and there is no given condition indicating that this not possible.

B. Plausible because a trouble shooting plan will be generated by Electrical Maintenance.

   Incorrect because operations will not generate the troubleshooting plan.

D. Plausible because some technical specification surveillances allow a repeat test.

   Incorrect because a reclosure attempt is not allowed for the given conditions.

K/A: G2.2.20 Equipment Control - Knowledge of the process for managing troubleshooting activities.

K/A Match: The KA is matched because it requires the candidate to recall that a breaker is sequestered after a closure failure to allow for troubleshooting.

Selection criteria; **MODIFIED FROM ADMIN389**

Tier: 3 **Group:**
Importance Rating: RO 2.6 SRO 3.8
Technical Reference: **OAP-100.5**

Proposed references to be provided to candidates during examination: None

Learning Objective: OAP-100.5 IDENTIFY the general guidelines operating 7.2 KV breakers and 480 Volt breakers, according ot OAP-100.5

Question Cognitive Level: **Memory or Fundamental Knowledge** __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

ADMIN PROCEDURE 389

If the breaker fails during a test start of a 480 Volt component operators must _______ in accordance with OAP-100.5, Guidelines for Configuration Control and Operation of Plant Equipment.

A. attempt 1 restart ONLY

B. replace the breaker and attempt a 1 restart ONLY

C.✓ sequester the breaker and initiate corrective actions

D. re-rack the break up or in and attempt 1 restart ONLY

70. Given the following plant conditions:

- The AFD Monitor Alarm identified in surveillance requirements under T.S. 3.2.1, AXIAL FLUX DIFFERENCE has been removed from service because it has been alarming with no actual adverse condition present.
- An R&R in accordance with SAP-0205 STATUS CONTROL AND REMOVAL AND RESTORATION has been written.
- A work order has been written for the repair.

Which ONE of the choices below completes both of the following statements?

After the work order is written, the R&R __(1)__ be closed out to the work order.

Operators __(2)__ be required to periodically perform an attachment in GTP-702 SURVEILLANCE ACTIVITY TRACKING AND TRIGGERING while the alarm is out of service.

A. 1) can
2) will

B. 1) can
2) will **not**

C.✓ 1) cannot
2) will

D. 1) cannot
2) will **not**

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate

A. The first part is plausible because if a technical specification does not apply, then the R&R can be closed to the work order. The second part is correct.

   Incorrect because the alarm in the given condition is a technical specification alarm and the R&R cannot be closed to the work order.

B. The first part is plausible because if a technical specification does not apply, then the R&R can be closed to the work order. The second part is plausible because most annunciators do not have a GTP-702 requirement.

   Incorrect because the alarm in the given condition is a technical specification alarm and the R&R cannot be closed to the work order and the XCP-620, 2-4 alarm does have a GTP-702 requirement.

C. CORRECT. XCP-620, 2-4 is a technical specification alarm and the R&R cannot be closed to the work order. The condition will require performance of a GTP-702 requirement.

D. The first part is correct. The second part is plausible because most annunciators do not have a GTP-702 requirement.

   Incorrect because the XCP-620, 2-4 alarm does have a GTP-702 requirement.

K/A: G2.2.43 Equipment Control - Knowledge of the process used to track inoperable alarms

K/A Match: The KA is matched because the candidate must recall administrative requirements for tracking an inoperable alarm.

Selection criteria; NEW

Tier: 3 **Group:**
Importance Rating: RO 3.0 SRO 3.3
Technical Reference: **GTP-702**
SAP-205

Proposed references to be provided to candidates during examination: None

Learning Objective: SAP-205 04. DESCRIBE the process for removal and restoration of systems or components.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

Given the following plant conditions:

- Unit 1 is in a refueling outage.
- Non-licensed operators report that they are unable to complete a Lockout-Tagout of a condensate isolation valve in the gland sealing system because the valve is leaking by.
- In accordance with OAP-100.5, GUIDELINES FOR CONFIGURATION CONTROL AND OPERATION OF PLANT EQUIPMENT, permission is obtained from the Shift Supervisor (SS) to use the appropriate sized valve wrench, the leakage is stopped, and the tagout is completed.

Based on the given conditions, operators are required to enter the affected valve in the ___(1)___ Log.

In addition to entering the affected valve in the above log, operators ___(2)___ also **required** to generate **both** a Work Order and a Condition Report describing the details surrounding the need to use a valve wrench on the affected valve.

Which ONE of the following choices completes the above statements, in accordance with OAP-100.5?

A. 1) R&R
   2) are

B. 1) R&R
   2) are **not**

C.✓ 1) Equipment Misalignment Status
   2) are

D. 1) Equipment Misalignment Status
   2) are **not**

71. Given the following plant conditions:

- Unit 1 is in an outage.
- The Personnel Hatch is closed.
- Reactor Building Purge is in service.
- RCS coolant is leaking to the Auxiliary Building.

Out of the Radiation Monitors listed below, which ONE will provide the first indication that airborne radiation in the Auxiliary Building is increasing due to the leak?

A. RM-A2, Reactor Building Sample Line Monitor.

B.✓ RM-A3, Main Plant Vent Exhaust Air Monitor.

C. RM-A6, Fuel Handling Building Exhaust Air Monitor.

D. RM-A10, Waste Gas Discharge Air Monitor.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall

A. Plausible because with Reactor Building Purge in service and the personnel hatch open, auxiliary building atmosphere could be drawn into the RB.

   Incorrect because RM-A2 will not sample auxiliary building atmosphere with the personnel hatch closed.

B. CORRECT. Auxiliary building ventilation discharges to the plant vent which is monitored by RM-A3.

C. Plausible because Fuel Handling building pressure is negative relative to the auxiliary building.

   Incorrect because without an opening between the two buildings given in the stem, RM-A6 would not be the first monitor.

D. Plausible because RM-A10 samples the waste gas release path prior to the plant vent. The candidate may believe that this monitor samples the plant vent upstream of the RM-A3 sample point.

   Incorrect because RM-A10 does not sample the plant vent.

K/A: G2.3.5 Radiation Control — - Ability to use radiation monitoring systems

K/A Match: The KA is matched because the candidate must identify a radiation monitor that can be used to detect a leak for the given conditions.

Selection criteria; NEW

Tier: 3 **Group:**
Importance Rating: RO 2.9 SRO 2.9
Technical Reference: **DBD - AH**
HANDOUT AB-18 HVAC BOARD SLIDE

Proposed references to be provided to candidates during examination: None

Learning Objective: GS-9 04. DESCRIBE the radiation monitoring system interfaces with the following systems and/or components: 11. Main plant vent exhaust

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 41(b)(11)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

72. Which ONE of the choices below answers the following questions regarding V.C. Summer Radiation Work Permits?

1) Can an entry into an area posted as a **Locked High Radiation Area** be entered under a General Radiation Work Permit in accordance with VCS-HPP-0151, USE OF THE RADIATION WORK PERMIT?

2) What information item can be found on a General Radiation Work Permit?

A. 1) Yes.
   2) The location of hot particles and contaminated areas in the rooms to be entered.

B. 1) Yes.
   2) The setpoint for personnel dosimetry dose and doserate alarms.

C. 1) No.
   2) The location of hot particles and contaminated areas in the rooms to be entered.

D.✓ 1) No.
   2) The setpoint for personnel dosimetry dose and doserate alarms.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must identify a restriction for the use of general radiation work permits and also identify information these permits contain that is used in the RCA.

A. The first part is plausible because a high radiation area can be entered under a general radiation work permit. The second part is plausible because this information is available on survey maps and the radiation work permit contains information regarding hot particles and contaminated areas.

   Incorrect because a locked high radiation area cannot be entered under a general radiation work permit and because the radiation work permit does not contain information regarding the specific location of hot particles or contaminated areas.

B. The first part is plausible because a high radiation area can be entered under a general radiation work permit. The second part is correct.

   Incorrect because a locked high radiation area cannot be entered under a general radiation work permit.

C. The first part is correct. The second part is plausible because this information is available on survey maps and the radiation work permit contains information regarding hot particles and contaminated areas.

   Incorrect because the radiation work permit does not contain information regarding the specific location of hot particles or contaminated areas.

D. CORRECT. The General Radiation Work Permit cannot be used to enter a locked high radiation area. The setpoint for personnel dosimetry dose and doserate alarms is indicated on the General Radiation Work Permit.

K/A: G2.3.7 Radiation Control - Ability to comply with radiation work permit requirements during normal or abnormal conditions

K/A Match: The KA is matched because the candidate must identify a restriction for the use of general radiation work permits and also identify information they contain that is used in the RCA.

Selection criteria; NEW

Tier: 3 **Group:**
Importance Rating: RO 3.5 SRO 3.6
Technical Reference: **VCS-HPP-151**

Proposed references to be provided to candidates during examination: None

Learning Objective: O-ILO-RP_RP Review for ILO: Given current conditions, discuss refresher training with ILO class on Helath Physics (HP) and Radiological Protection (RP) procedures, setpoints, and calculations

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(12)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

73. Given the following plant conditions:

- A loss of offsite power (115 KV and 230 KV) has occurred.
- The reactor has tripped.
- "A" and "B" EDGs have failed to start.

NOTE THE FOLLOWING PROCEDURE TITLES:

EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION
EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER

Which ONE of the choices below completes the following statements OAP-103.4, EOP/FSP/AOP USER'S GUIDE?

EOP-6.0 can be entered __(1)__.

Immediate actions in EOP-6.0 are identified by __(2)__ the step number.

A✓ 1) directly after the trip or by transitioning from an EOP-1.0 immediate action step.
2) a circle around

B. 1) directly after the trip or by transitioning from an EOP-1.0 immediate action step.
2) an asterisk next to

C. 1) **only** by transitioning from an EOP-1.0 immediate action step.
2) a circle around

D. 1) **only** by transitioning from an EOP-1.0 immediate action step.
2) an asterisk next to

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall that EOP-6.0 can be entered directly after a loss of all AC event and the method used to denote immediate action steps in that procedure

A. CORRECT. EOP-6.0 can be entered directly after the trip or by transitioning from an EOP-1.0 immediate action step. The immediate action steps will be indicated with a circle around the step number.

B. The first part is correct. The second part is plausible because continuous action steps are designated in the EOPs by use of an asterisk.

   Incorrect because immediate actions steps are not indicated using an asterisk.

C. The first part is plausible because the EOPs are normally entered from EOP-1.0 and that procedure provides a transfer to EOP-6.0 in the immediate action steps. The second part is correct.

   Incorrect because EOP-6.0 can be directly entered.

D. The first part is plausible because the EOPs are normally entered from EOP-1.0 and that procedure provides a transfer to EOP-6.0 in the immediate action steps. The second part is plausible because continuous action steps are designated in the EOPs by use of an asterisk.

   Incorrect because EOP-6.0 can be directly entered and because immediate actions steps are not indicated using an asterisk.

K/A: G2.4.1 Emergency Procedures/Plans - Knowledge of EOP entry conditions and immediate action steps.

K/A Match: The KA is matched because the candidate must recall that EOP-6.0 can be entered directly after a loss of all AC event and the method used to denote immediate action steps in that procedure.

Selection criteria; NEW

Tier: 3 **Group:**
Importance Rating: RO 4.6 SRO 4.8
Technical Reference: **OAP-103.4**

Proposed references to be provided to candidates during examination: None

Learning Objective: OAP-103.4 03. KNOW the format rules for EOP's and AOP's.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

74. Given the following plant conditions:

Time 0030:

- A electrical power transient with equipment failures occurred

Time 0040:

- An UNUSUAL EVENT was declared.

Time 0050:

- An ALERT was declared.

Time 0100:

- A SITE AREA EMERGENCY was declared.
- The Emergency Response Data System (ERDS) is **not** activated.

Which ONE of the choices below completes the following statements in accordance with VCS-EPP-002, COMMUNICATION AND NOTIFICATION?

The **first** notification of state and local authorities was required within a __(1)__ minute period after declaration of the UNUSUAL EVENT.

The Emergency Response Data System (ERDS) was **first** required to be activated due to declaration of the __(2)__.

A✓ 1) 15.
2) ALERT.

B. 1) 15.
2) SITE AREA EMERGENCY.

C. 1) 30.
2) ALERT.

D. 1) 30.
2) SITE AREA EMERGENCY.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the requirement for notifying state and local authorities and the emergency action level at which ERDS is activated.

A. CORRECT. The first notification of state and local authorities was required within a 15 minute period after declaration of the UNUSUAL EVENT. The Emergency Response Data System (ERDS) is required to be activated due to declaration of the Alert

B. The first part is correct. The second part is plausible because the Site Area Emergency Level is when accountability and Site Evacuation are required.

Incorrect because ERDS is activated at the Alert level.

C. The first part is plausible because the maximum time from initiation of the event to notification of state and local authorities is 30 minutes. The second part is correct.

Incorrect because the maximum time for notification of state and local authorities is not 30 minutes after declaration.

D. The first part is plausible because the maximum time from initiation of the event to notification of state and local authorities is 30 minutes. The second part is plausible because the Site Area Emergency Level is when accountability and Site Evacuation are required.

Incorrect because the maximum time is not 30 minutes and ERDS is activated at the Alert level.

K/A: G2.4.43 Emergency Procedures/Plans - Knowledge of emergency communications systems and techniques.

K/A Match: The KA is matched because the candidate must recall the requirement for notifying state and local authorities and the emergency action level at which ERDS is activated.

Selection criteria; NEW

Tier: 3 **Group:**
Importance Rating: RO 3.2 SRO 3.8
Technical Reference: **VCS-EPP-0001**
VCS-EPP-0001.1
VCS-EPP-0001.2
VCS-EPP-0001.3

Proposed references to be provided to candidates during examination: None

Learning Objective: EPP OVERVIEW – ILO 03 LIST the time requirements for initial and follow-up notifications. 04 STATE the level of classification that the Emergency Response Organization Notification System (ERONS) activated.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

75. Initial conditions:

- 100% power.
- A Steam Generator Tube Rupture began.
- The following radiation alarms sounded:

  RM-G19C, STMLN HI RNG GAMMA.
  RM-A9, CNDSR EXHAUST GAS ATMOS MONITOR.

- The CRS, while monitoring IPCS, announced that it appears that "C" SG is ruptured.
- The reactor automatically tripped.

Current conditions:

- The CRS is reading **step 6** of EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION.
- Containment conditions are normal.
- SG Narrow Range levels are as follows:

  "A" 37% and increasing.
  "B" 37% and increasing.
  "C" 45% and increasing.

Which ONE of the choices below completes the following statement?

Two items that the crew is allowed to do in the current conditions is to __(1)__ and __(2)__ in accordance with OAP-103.4, EOP/FSP/AOP USER'S GUIDE.

A. 1) close "C" MSIV
   2) throttle EFW to "C" SG to a minimum of 0 gpm.

B. 1) close "C" MSIV
   2) transfer to EOP-4.0, E-3 STEAM GENERATOR TUBE RUPTURE.

C.✓ 1) secure the TDEFP with an operator at the pump
   2) throttle EFW to "C" SG to a minimum of 0 gpm.

D. 1) secure the TDEFP with an operator at the pump
   2) transfer to EOP-4.0, E-3 STEAM GENERATOR TUBE RUPTURE.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall isolation steps during a steam generator tube rupture event and determine if they can be performed for the given conditions.

A. The first part is plausible because closing MSIVs for personnel safety concerns is allowed prior to written procedural guidance. The second part is correct.

   Incorrect because the ruptured steam generator MSIV will not be closed for the given conditions.

B. The first part is plausible because closing MSIVs for personnel safety concerns is allowed prior to written procedural guidance. The second part is plausible because EOP-1.0 immediate action steps are complete and that procedure has a step to transfer to EOP-4.0.

   Incorrect because the ruptured steam generator MSIV will not be closed for the given conditions and a transfer to EOP-4.0 does not occur in the next step after completion of the immediate action steps.

C. CORRECT. The TD EFW pump can be secured at any time if not needed and with an operator available at the pump to reset the overspeed trip mechanism. Both MD EFW pumps are running in the given conditions and generator levels above minimum required levels so the TD pump can be secured. The CRS previously identified the ruptured generator and RM-G19C was in alarm so EFW flow "C" SG can be throttled to 0 gpm.

D. The first part is correct. The second part is plausible because EOP-1.0 immediate action steps are complete and that procedure has a step to transfer to EOP-4.0.

   Incorrect because a transfer to EOP-4.0 does not occur in the next step after completion of the immediate action steps.

K/A: G2.4.6 Emergency Procedures/Plans - Knowledge symptom based EOP mitigation strategies.

K/A Match: The KA is matched because the candidate must recall isolation steps during a steam generator tube rupture event and determine if they can be performed for the given conditions.

Selection criteria; NEW

Tier: 3 **Group:**
Importance Rating: RO 3.7 SRO 4.7
Technical Reference: **OAP-103.4**

Proposed references to be provided to candidates during examination: None

Learning Objective: OAP-103.4 09. KNOW what "Deviations from Procedure" are allowed by OAP-103.4.

Question Cognitive Level: Memory or Fundamental Knowledge **__X__**
Comprehension or Analysis **_____**

10 CFR Part 55 Content: 41(b)(10)

SRO Justification: N/A

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

76. Initial conditions:

- 100% power initially.
- A Steam Generator Tube Rupture occurred on "A" Steam Generator.
- A Reactor trip and Safety Injection occurred.
- All offsite power (115 KV and 230 KV) was lost after the trip.
- The System Controller expects power to be returned in 8 hours.

Initial conditions:

- The CRS is reading the last step of EOP-4.0, E-3 STEAM GENERATOR TUBE RUPTURE.

Which ONE of the following identifies the **preferred** procedure transition, given the conditions above, and the reason for that preference in accordance with the bases for EOP-4.0?

A.✓ EOP-4.1A ES-3.1 POST-SGTR COOLDOWN BY BACKFILLING THE REACTOR COOLANT SYSTEM,
because the radioactive release will be minimized.

B. EOP-4.1A ES-3.1 POST-SGTR COOLDOWN BY BACKFILLING THE REACTOR COOLANT SYSTEM,
because this is the most rapid method to achieve COLD SHUTDOWN conditions.

C. EOP-4.1B ES-3.2 POST-SGTR COOLDOWN USING BLOWDOWN,
because the radioactive release will be minimized.

D. EOP-4.1C ES-3.3 POST-SGTR COOLDOWN USING STEAM DUMP,
because this is the most rapid method to achieve COLD SHUTDOWN conditions.

Revised version of eops693 without significant modification for 2018 - rj

QUESTION USAGE: (As eops693)
RO-10-01-Classroom Comp Exam

REVISION HISTORY
Rev 1 submitted by RJ - gave loss of offsite power in stem
Ops Review: Danny Rhymer
Approved:

Rev. 0 unknown

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must determine that the procedure that uses backfill will produce the least amount of release given that condenser steam dumps are not available due to a loss of offsite power.

A. CORRECT. The preferred procedure selection at the end of EOP-4.0 is EOP-4.1A because radioactive releases are minimized.

B. Plausible because EOP-4.1A is the preferred procedure.

   Incorrect because EOP-4.1A is not the most rapid method of achieving cold shutdown.

C. Plausible because EOP-4.1B does minimize radioactive releases.

   Incorrect because EOP-4.1B is not the preferred procedure.

D. Plausible because EOP4.1C does provide the fastest method of achieving cold shutdown.

   Incorrect because EOP-4.1C is not the preferred procedure.

K/A: 038 EA2.08 Steam Gen. Tube Rupture / 3 Ability to determine and interpret the following as the apply to the event: Viable alternatives for placing plant in safe condition when condenser is not available

K/A Match: The KA is matched because the candidate must determine that the procedure that uses backfill will produce the least amount of release given that condenser steam dumps are not available due to a loss of offsite power.

Selection criteria; **REVISED BANK**

Tier: 1 **Group:** 1
Importance Rating: RO 3.8 SRO 4.4
Technical Reference: **EOP-4.0**
ERG BASES E-3

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-4.0 05. STATE the bases or reasons for each action contained in EOP-4.0. This should include, but not be limited to, the following o. Advantages and disadvantages of cooldown methods

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

EOPS 693

Plant conditions:

- A Steam Generator Tube Rupture (SGTR) occurred on Steam Generator B.
- Off-site power is available.
- The crew has entered EOP-4.0, Steam Generator Tube Rupture.
- The crew is at Step 47, GO TO the appropriate Post-SGTR cooldown method.

Which ONE of the following identifies the preferred cooldown method and the reason for the preference?

A.✓ EOP-4.1A, Post-SGTR Cooldown by Backfilling the Reactor Coolant System, because the total radioactive release is minimized.

B. EOP-4.1A, Post-SGTR Cooldown by Backfilling the Reactor Coolant System, because only Safety Related components and controls are used in this procedure.

C. EOP-4.1C, Post-SGTR Cooldown Using Steam Dump, because there is no impact on RCS chemistry.

D. EOP-4.1C, Post-SGTR Cooldown Using Steam Dump, because CST inventory concerns are minimized.

77. Given the following plant conditions:

- 100% power initially.
- **"A"** Steam Generator faulted inside the RB.
- All EFW pumps failed to start manually or automatically.
- Upon transitioning out of EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION, the STA advised the CRS of a RED path on HEAT SINK.
- The crew entered EOP-15.0, FR-H.1 RESPONSE TO A LOSS OF SECONDARY HEAT SINK.
- Actions have been taken under steps to restore feed using Condensate flow.
- Operators are depressurizing **"B"** Steam Generator.
- The CRS is reading step 16, **"Check SG levels:"**.
- All other Critical Safety Functions are in a GREEN or YELLOW status.
- RB pressure is 8 psig and decreasing.

Which ONE of the choices below completes the following statement?

At the current steam generator levels, the CRS __(1)__ direct a transition out of EOP-15.0.

If the current equipment alignments are maintained, the next procedure transition will be to _____.

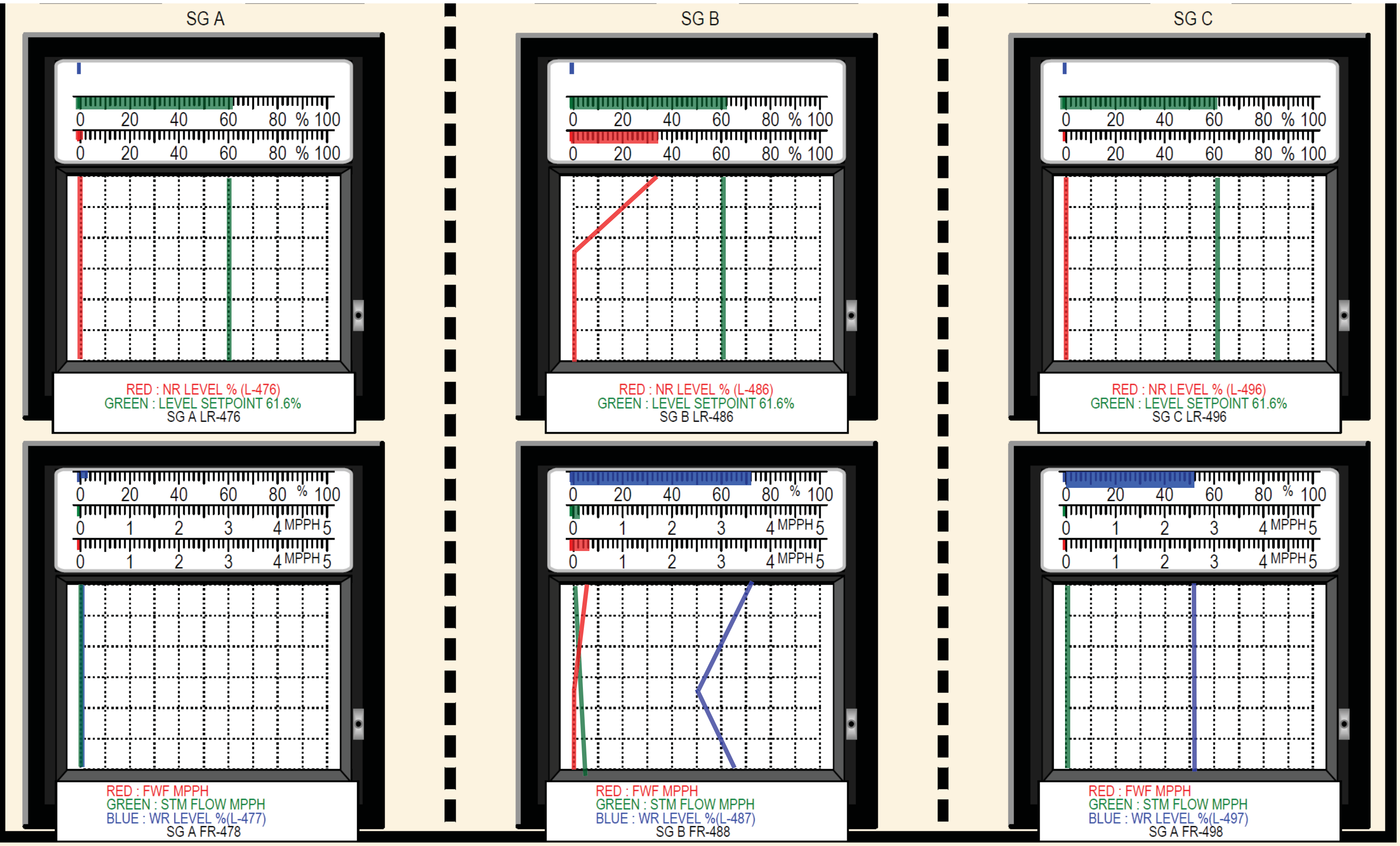
REFERENCE PROVIDED

A. 1) can
2) EOP-2.0.

B. 1) cannot
2) EOP-2.0.

C. 1) can
2) EOP-3.0.

D✓ 1) cannot
2) EOP-3.0.


SG A
SG B
SG C
RED : NR LEVEL % (L-476)
GREEN : LEVEL SETPOINT 61.6%
SG A LR-476
RED : NR LEVEL % (L-486)
GREEN : LEVEL SETPOINT 61.6%
SG B LR-486
RED : NR LEVEL % (L-496)
GREEN : LEVEL SETPOINT 61.6%
SG C LR-496
RED : FWF MPPH
GREEN : STM FLOW MPPH
BLUE : WR LEVEL %(L-477)
SG A FR-478
RED : FWF MPPH
GREEN : STM FLOW MPPH
BLUE : WR LEVEL %(L-487)
SG B FR-488
RED : FWF MPPH
GREEN : STM FLOW MPPH
BLUE : WR LEVEL %(L-497)
SG A FR-498

PROVIDE GRAPHIC OF MAIN CONTROL BOARD PANEL XCP6111.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 Submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must evaluated steam flow and feed flow recorder trends during a loss of feedwater to determine whether a procedure transition is allowed and predict the procedure transition.

A. The first part is plausible 0.4E6 lbm/hr is indicated on the feedwater line to "B" SG. This is approximately 800 gpm which is a greater value than the minimum heat sink requirement of 450 gpm. The second part is plausible because there is a transfer out of EOP-15.0 to EOP-2.0 later in the procedure and it is also a common transfer out of other procedures. EOP-14.0/EOP-14.1/EOP-2.5 are examples.

   Incorrect because transfer out of EOP-15.0 is not allowed for the given conditions and because the next transition will not be to EOP-2.0.

B. The first part is correct. The second part is plausible because there is a transfer out of EOP-15.0 to EOP-2.0 later in the procedure and it is also a common transfer out of other procedures. EOP-14.0/EOP-14.1/EOP-2.5 are examples.

   Incorrect because the next transition will not be to EOP-2.0.

C. The first part is plausible 0.4E6 lbm/hr is indicated on the feedwater line to "B" SG. This is approximately 800 gpm which is a greater value than the minimum heat sink requirement of 450 gpm. The second part is correct.

   Incorrect because transfer out of EOP-15.0 is not allowed for the given conditions/

D. CORRECT. EOP-15.0 will not allow a transfer out after restoration of feed using condensate until the minimum heat sink level is reached in the steam generator that is being fed. In the current alignments that level will be achieved and the transition will be back to procedure and step in effect which is step 1 of EOP-3.0 due to the faulted steam generator.

K/A: 054 AA2.08 Loss of Main Feedwater/ 4 Ability to determine and interpret the following as the apply to the event: Steam flow-feed trend recorder

K/A Match: The KA is matched because the candidate must evaluated steam flow and feed flow recorder trends during a loss of feedwater to determine whether a procedure transition is allowed.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 2.9 SRO 3.3
Technical Reference: **EOP-15.0**
EOP-1.0
EOP-14.0

Proposed references to be provided to candidates during examination:

GRAPHIC OF MAIN CONTROL BOARD PANEL XCP6111.

Learning Objective: EOP-15.0 08. SELECT an appropriate transition out of EOP-15.0 given a set of plant conditions.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

78. Given the following plant conditions:

- 100% power initially.
- Alternate Seal Injection was **inoperable** and removed from service.
- The plant has experienced a total loss of AC power.
- EOP-6.0 ECA-0.0 LOSS OF ALL ESF AC POWER is in progress.
- The 'A' D/G has been started and bus XSW-1DA is restored.
- The CRS is reading EOP-6.0, step 44, "**Verify the following to select the appropriate recovery procedure**"
- Plant conditions are as follows:
  - RCS subcooling is 45°F.
  - Pressurizer level is 18%.
  - There are **no** SI first out alarms lit on the first out annunciator panel.
  - There is **no** SI flow indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

Which ONE of the choices below completes the following statements?

The crew will transition to __(1)__ .

That procedure will direct operators to __(2)__.

A. 1) EOP-6.1, ECA-0.1 LOSS OF ALL ESF AC POWER RECOVERY WITHOUT SI REQUIRED.
2) isolate RCP seal injection before a Charging Pump is started.

B. 1) EOP-6.1, ECA-0.1 LOSS OF ALL ESF AC POWER RECOVERY WITHOUT SI REQUIRED.
2) start a Charging pump and then restore seal injection to RCPs.

C.✓ 1) EOP-6.2, ECA-0.2 LOSS OF ALL ESF AC POWER RECOVERY WITH SI REQUIRED.
2) isolate RCP seal injection before a Charging Pump is started.

D. 1) EOP-6.2, ECA-0.2 LOSS OF ALL ESF AC POWER RECOVERY WITH SI REQUIRED.
2) start a Charging pump and then restore seal injection to RCPs.

NOTE TO EXAMINER:
THIS QUESTION WAS SUBMITTED AS PART OF SAMPLE AND HAS BEEN REVISED TO ADDRESS COMMENTS

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must assess control room indications to select the procedure that will be used to restore plant equipment and recall an operator action that is performed in that procedure.

A. The first part is plausible because pressurizer level is on scale with no SI first outs or SI flow indicated and EOP-6.1 would be used if subcooling were higher. The second part is correct for the given conditions. Seal injection will be isolated before a charging pump is started.

   Incorrect because EOP-6.1 is not the correct procedure for the given conditions.

B. The first part is plausible because pressurizer level is on scale with no SI first outs or SI flow indicated and EOP-6.1 would be used if subcooling were higher. The second part is plausible because in SOP-102, seal injection is normally established after a charging pump has been started.

   Incorrect because EOP-6.1 is not the correct procedure for the given conditions and seal injection will not be restored after the charging pump is started.

C. CORRECT. With Subcooling at 50°F, EOP-6.2, will be selected. Seal injection will be isolated in that procedure before a charging pump is started.

D. The first part is correct. With Subcooling at 50°F, EOP-6.2, will be selected. The second part is plausible because in SOP-102, seal injection is normally established after a charging pump has been started.

   Incorrect because seal injection will not be restored under the given conditions.

K/A: 055 EG2.2.44 Station Blackout/6 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.

K/A Match: The KA is matched because it requires the candidate to assess plant conditions during a station blackout and determine the procedure that operators will use to restore plant equipment after power recovery.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 4.2 SRO 4.4
Technical Reference: **EOP-6.0**
EOP-6.2
SOP-101

Proposed references to be provided to candidates during examination:

Learning Objective: EOP-6.0 07 SELECT an appropriate transition out of Emergency Operating Procedure EOP-6.0 given a set of plant conditions.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

79.

Given the following plant conditions:

(Note the following equipment nomenclature)

XBC1A-1B, DC DISTRI BUS 1A-1B BACKUP BATTERY CHRG
XBA1A, DC DISTRIBUTION BUS 1A BATTERY
XBA1B, DC DISTRIBUTION BUS 1B BATTERY

6/10/17, Time 0645:

- Mode 3 with a heatup in progress.
- XBA**1B**, has been declared **inoperable**.
- Battery Charger XBC1A-1B was declared **inoperable**.

6/10/17, Time 0700:

- STP-501.002, QUARTERLY BATTERY SURVEILLANCE TEST has just been completed.
  - Two connected cells on XBA**1A** read 2.0 VDC.

6/10/17, Time 0745:

- The following annunciators are received on XCP-636:
  - 4-4, TRAIN A BATT CHGR TRBL XBC 1A/1A-1B.
  - 4-6, DC SYS OVRVOLT/UNDRVOLT.
- The BOP reports **1A** DC VOLTS indicates 125 VDC.

Which ONE of the choices below completes the following statements?

If the problems above **remain uncorrected**, the **earliest** time at which the plant must be placed in COLD SHUTDOWN in accordance with Technical Specifications is ______ on 6/11.

REFERENCE PROVIDED

A. 1300

B.✓ 1400

C. 1445

D. 2000

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery bank No. 1A and its associated full capacity charger.

b. 125-volt Battery bank No. 1B and its associated full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of 10 of the connected cells is $\geq 60°F$.

c. At least once per 18 months by verifying that:

1. The cells; cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

4. The battery charger will supply at least 300 amperes at 132 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1.d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A[1]	CATEGORY B[2]	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	≥ 1.200[b]	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[b]

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than (2) amps when on charge.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of 10 of the connected cells is $\geq$ 60°F.

c. At least once per 18 months by verifying that:

1. The cells; cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

4. The battery charger will supply at least 300 amperes at 132 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1.d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

NOTE TO EXAMINER:
THIS QUESTION WAS SUBMITTED AS PART OF SAMPLE AND HAS BEEN REVISED TO ADDRESS COMMENTS

PROVIDE TECHNICAL SPECIFICATION 3.8.2, PAGES 3/4 8-9 THROUGH 8-11.

Modified techspec470 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate assess 125V dc bus voltage, and alarms and determine operability and appropriate technical specification actions.

A. Plausible. 1B DC train was initially inoperable. At time 0700, two connected cells read less than the T.S. Table 4.8-2 Category B value of 2.07 volts. Note 3 states that this makes the battery inoperable. With two batteries inoperable, 3.0.3 is in effect. This time reflects an error that omits the 1 hour allowed by TS 3.0.3 to prepare to lower modes.

   Incorrect because CSD is not required to be reached until 1400.

B. CORRECT. 1B DC train was initially inoperable. At time 0700, two connected cells read less than the T.S. Table 4.8-2 Category B value of 2.07 volts. Note 3 states that this makes the battery inoperable. With two batteries inoperable, 3.0.3 is in effect. A TS 3.0.3 entry is required so 6/10, 0700 plus 1 hour for preparation plus 6 hours to HSD plus 24 hours to CSD is 6/11, 1400.

C. Alarms and battery voltage indications at terminal voltage rather than float voltage at 0745 indicate that the charger has been lost making 1A train inoperable. 1445 is the erroneous discovery time of 0745 plus the correct 31 hour reduction schedule.

   Incorrect because 1445 is after the earliest required time to reach CSD.

D. Plausible. 1B DC train was initially inoperable. At time 0700, two connected cells read less than the T.S. Table 4.8-2 Category B value of 2.07 volts. Note 3 states that this makes the battery inoperable. With two batteries inoperable, 3.0.3 is in effect. 6/10, 0700 plus an erroneous allowance of 6 hours to HSB plus 6 hours to HSD plus 24 hours to CSD is 6/11, 2000.

   Incorrect because 2000 is after the earliest required time to reach CSD.

K/A: 058 AA2.02 Loss of DC Power /6 Ability to determine and interpret the following as the apply to the event: 125V dc bus voltage, low/critical low, alarm

K/A Match: The KA is matched because it requires the candidate to assess DC systems power conditions, including voltage, to determine technical specification actions.

Selection criteria; **MODIFIED FROM TS470**

Tier: 1 **Group:** 1
Importance Rating: RO 3.3 SRO 3.6
Technical Reference: **T.S 3.0.3**
T.S 3.8.2 - DC SOURCES
ARP XCP-636, 4-4

Proposed references to be provided to candidates during examination:

TECHNICAL SPECIFICATION 3.8.2, PAGES 3/4 8-9 THROUGH 8-11.

Learning Objective: **SB04** 017. DETERMINE actions required if an LCO is not met including applicability of Specifications 3.0.3, 3.0.4, and 4.0.4.

SB04 08. STATE the conditions that would constitute noncompliance with the Technical Specifications.

Question Cognitive Level: **Memory or Fundamental Knowledge** ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(2)

SRO Justification: SRO question because it requires knowledge of facility operating limitations in the TS and their bases involving one or more of the following for TS, TRM or ODCM: Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1).

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

TECH SPECS 470

Given the following plant conditions:

- The Unit is operating in Mode 1 in a normal electrical lineup:
- The following annunciators are received:
  - DC SYS OVRVOLT/UNDRVOLT
  - TRAIN A BATT CHGR TRBL XBC 1A/1A-1B
  - DC Bus 1HA voltage indicates 125 volts and decreasing slowly

Which ONE of the following describes the status of the DC Bus and the action that will be required?

A.✓ Battery Charger XBC-1A has failed. Battery XBA-1A is supplying DC Bus 1HA. Battery Charger 1A-1B must be placed in service IAW SOP-311, *125 VDC System*, and operability of the DC Bus will be restored when voltage is > Technical Specification requirements.

B. Battery Charger XBC-1A has failed. Battery XBA-1A is supplying DC Bus 1HA. Battery Charger 1A-1B must be placed in service IAW SOP-311, *125 VDC System*. The DC Bus remains operable as long as it remains connected to the battery.

C. Battery XBA-1A output breaker has tripped. Determine corrective actions for Battery failure in accordance with the applicable ARPs. The bus remains operable as long as it is connected to the Battery Charger and voltage is maintained > Technical Specification requirements.

D. Battery XBA-1A output breaker has tripped. Determine corrective actions for Battery failure in accordance with the applicable ARPs. The bus is returned to operable status only when the Battery Charger, Battery, and DC Bus voltage is restored to > Technical Specification requirements.

80. Initial conditions:

- 100% power.
- XTF0004, UNIT 1 ENGINEERED SAFEGUARD TRANSFORMER is inoperable and tagged out of service.
- A **lockout** has occurred on XTF-5, UNIT 2 ENGINEERED SAFEGUARD TRANSFORMER.

Current conditions:

- All 230 KV power has been lost.
- The reactor has tripped.

Which ONE of the choices below answers both of the following questions regarding the XTF5052, ALTERNATE AC SOURCE TRANSFORMER?

1) Can XTF5052 be connected to either 7.2 KV buses 1DA or 1DB in the conditions above?

2) Is XTF5052 source credited in the safety analysis in accordance with the bases for T.S.3.8.1, A.C. SOURCES?

A. 1) Yes.
 2) Yes.

B.✓ 1) Yes.
 2) No.

C. 1) No.
 2) Yes.

D. 1) No.
 2) No.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must evaluate a path for electrical power restoration and recall whether a transformer that is a source for power restoration is credited in the Safety Analysis for AC power sources as discussed in the bases.

A. The first part is correct. The second part is plausible because XTF5052 is used in action statements and discussed in the bases for T.S. 3.8.1 as a power source for risk reduction during EDG maintenance.

   Incorrect because XTF5052 is not credited in the Safety Analysis as a safety grade power source.

B. CORRECT.

C. The first part is plausible because a lockout of a 115 KV transformer has occurred that feeds bus 1DX. In order to restore power using XTF5052 bus 1DX must be used in the power path. The second part is plausible because XTF5052 is used in action statements and discussed in the bases for T.S. 3.8.1 as a power source for risk reduction during EDG maintenance.

   Incorrect because XTF5052 can be used to restore power for the given conditions and because it is not credited in the Safety Analysis as a safety grade power source.

D. The first part is plausible because a lockout of a 115 KV transformer has occurred that feeds bus 1DX. In order to restore power using XTF5052 bus 1DX must be used in the power path. The second part is correct.

   Incorrect because XTF5052 can be used to restore power for the given conditions.

K/A: 077 AG2.2.25 Generator Voltage and Electric Grid Disturbances / 6 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A Match: The KA is matched because the candidate must recall the bases in technical specification regarding non safety-grade sources of power during a grid disturbance event.

Selection criteria; NEW

Tier: 1 **Group:** 1
Importance Rating: RO 3.2 SRO 4.2
Technical Reference: **ARP-XCP-639-1-3**
ARP-XCP-639-2-3
T.S. 3.8.1 BASES

Proposed references to be provided to candidates during examination: None

Learning Objective: SB-4 015 Given a limiting condition for operation and a mode, DEFINE the requirements to satisfy the LCO, the actions if required within one hour or less, and describe the bases for the LCO.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(2)

SRO Justification: SRO question because it requires knowledge of facility operating limitations in the TS and their bases involving one or more of the following for TS, TRM or ODCM: Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1).

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

AOPS 628

Given the following plant conditions:

- 100% power.
- XTF-4 and XTF-6 are out of service for maintenance.
- XTF-5 is in service.

A grid disturbance occurs causing the following line voltages:

- 230 KV source is at 236.1 KV.
- 115 KV source is at 115.4 KV.
- The normal feeder for bus 1C opens spuriously.
- VARs are at 335 MVARs lagging at the request of the System Controller.

Which ONE of the following identifies the 7.2 KV ESF busses that have **inoperable** offsite sources in accordance with AOP-301.1, RESPONSE TO ELECTRICAL GRID ISSUES?

REFERENCE PROVIDED

A✓ 1DA **only.**

B. 1DB **only.**

C. Both 1DA and 1DB.

D. Neither bus.

81. Which ONE of the following choices completes the following statement:

Technical Specification bases for 3.7.1.2, EMERGENCY FEEDWATER SYSTEM, states that each Motor Driven Emergency Feed Water pump is capable of delivering a total feedwater flow of __________ gpm at a pressure of 1211 psig to the entrance of at least __________.

A. 450 gpm; ONE (1) Steam Generator

B. 450 gpm; TWO (2) Steam Generators

C. 380 gpm; ONE (1) Steam Generator

D.✓ 380 gpm; TWO (2) Steam Generators

Proposed for use on 2018 NRC - rj

QUESTION USAGE:

SRO-11-01-Classroom Comp MU Exam
SRO-10-01-SYSTEMS COMP MU

REVISION HISTORY:

Rev.1 (dow 5/28/05) Moved this bases question to the open reference Exam bank as SRO question because NUREG 1021 suggests asking SROs bases questions on annual Exams.
Rev. 2 (03/02/99) - Added reference to the Design Bases of the Technical Specifications in the stem of the question.
Rev. 3 wdb 2/17/10 Changed 425 gpm to 510, no reference for 425 gpm.
Rev. 4 ML 2/28/13 Changed to a 2 part question due to similarity to question EFW 58
Rev. 5, rj changed the order of distractors.

DISTRACTOR ANALYSIS:

In order to answer this question, the candidate must recall the minimum actual flow required by technical specification that satisfies the EOP heat sink critical safety function.

A. Plausible because 450 gpm to one generator is the minimum heat sink requirement contained in the EOPs for the Heat Sink CSF.

   Incorrect because this is not the value stated in the bases.

B. Plausible because 450 gpm to one generator is the minimum EFW flow for the heat sink requirement contained in the EOPs for the Heat Sink CSF and the basis for T.S. 3.7.1.2 states that a minimum capability is to two SGs..

   Incorrect because this is not the value stated in the bases.

C. Plausible because 380 gpm is the correct flow value as stated in the bases and because the minimum EOP heat sink is to one SG.

   Incorrect because flow must be applied to two SGs.

D. CORRECT. Technical Specification bases for 3.7.1.2, EMERGENCY FEEDWATER SYSTEM, states that each Motor Driven Emergency Feed Water pump is capable of delivering a total feedwater flow of 380 gpm at a pressure of 1211 psig to the entrance of at least two SGs.

K/A: WEO5 EG2.4.2 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4
Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

K/A Match: The KA is matched because the candidate must recall the minimum actual flow required by technical specification that satisfies the EOP heat sink critical safety function.

Selection criteria; **BANK**

Tier: 1 **Group:** 1
Importance Rating: RO 4.5 SRO 4.6
Technical Reference: **T.S 3.7.1.2 BASES**
EOP-12.0

Proposed references to be provided to candidates during examination: None

Learning Objective: SB-4 015 Given a limiting condition for operation and a mode, DEFINE the requirements to satisfy the LCO, the actions if required within one hour or less, and describe the bases for the LCO.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 43(b)(2)

SRO Justification: SRO question because it requires knowledge of facility operating limitations in the TS and their bases involving one or more of the following for TS, TRM or ODCM: Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1).

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

82. Given the following plant conditions:

- 100% power.
- A fuel element failure has occurred.
- XCP-642, 1-5, RC LTDN HI RNG RM-L1 HI RAD is in alarm.
- The CRS is referring to T.S. 3.4.8, REACTOR COOLANT SYSTEM - SPECIFIC ACTIVITY.

Which ONE of the choices below answers both of the following questions?

1) What action will operators take as a result of the fuel failure in accordance with XCP-642, 1-5?

2) What is the consequence of high RCS activity if the T.S. 3.4.8 L.C.O limit is violated in accordance with the associated basis?

ASSUME NO OPERATOR ACTIONS

A.✓ 1) Normal Letdown flow will be maximized.
2) The assumed offsite dose during a steam generator tube rupture could be exceeded.

B. 1) Normal Letdown flow will be maximized.
2) The assumed offsite dose during a main steam line break could be exceeded.

C. 1) Normal Letdown will be isolated.
2) Access of vital areas in the Auxiliary Building could be prohibited.

D. 1) Normal Letdown will be isolated.
2) Limits on gaseous waste release via the plant vent could be exceeded.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the procedural action for maximizing letdown during high RCS activity and the technical specification basis for maintaining activity below the limit.

A. CORRECT. In accordance with XCP-642, 1-5, Normal Letdown will be maximized to cleanup RCS activity. A high RCS activity in excess of T.S. 3.4.1 limits could cause the assumed offsite dose during a steam generator tube rupture to be exceeded.

B. Plausible because normal Letdown flow will be maximized and the limits for secondary activity in accordance with the bases for T.S. 3.7.1.4, the assumed offsite dose during a main steam line break could be exceeded.

   Incorrect because a steam line break is incorrect.

C. Plausible because the EOPs caution against restoring letdown for the case of high RCS activity. The associate statement of limiting access of vital areas in the Auxiliary Building is plausible.

   Incorrect because letdown will be maximized.

D. Plausible because leakage of small amounts of RCS fluid to the Auxiliary building could cause conditions of the standing release limit to the plant vent to be exceeded.

   Incorrect because letdown will be maximized.

K/A: 076 AA2.05 High Reactor Coolant Activity / 9 Ability to determine and interpret the following as the apply to the event: CVCS letdown flow rate indication

K/A Match: The KA is matched because the candidate must recall the procedural action for maximizing letdown during high RCS activity and the technical specification basis for maintaining activity below the limit.

Selection criteria; NEW

Tier: 1 **Group:** 2
Importance Rating: RO 2.2 SRO 2.5
Technical Reference: **ARP-XCP-642, 1-5**
FSAR

Proposed references to be provided to candidates during examination: None

Learning Objective: SB-4 015 Given a limiting condition for operation and a mode, DEFINE the requirements to satisfy the LCO, the actions if required within one hour or less, and describe the bases for the LCO.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 43(b)(2)

SRO Justification: SRO question because it requires knowledge of facility operating limitations in the TS and their bases involving one or more of the following for TS, TRM or ODCM: Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1).

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

83. Initial conditions:

- LOCA occurred.
- All offsite power (115 KV and 230 KV) was lost.
- XCP-610, 1-1, RHR PP A TRIP is in alarm.
- RCS pressure 1100 psig, decreasing.

Current conditions:

- EOP-2.0, LOSS OF REACTOR OR SECONDARY COOLANT in progress.
  - Operators just checked for a potential transition to EOP-1.2, SAFETY INJECTION TERMINATION.
- RB pressure is 13 psig, decreasing.
- Pressurizer level is 0%.
- RWST level is 55%, decreasing.
- RHR Sump Level is 412 ft, increasing.
- RCS pressure 600 psig and stable.

Which ONE of the following describes the **next** procedure transition and an action that will be taken in that procedure based on the current conditions?

A. EOP-2.4 LOSS OF EMERGENCY COOLANT RECIRCULATION.
Align RB Spray for recirculation.

B. EOP-2.4 LOSS OF EMERGENCY COOLANT RECIRCULATION.
Stop RB Spray pumps and Charging Pumps.

C. EOP-2.1, POST- LOCA COOLDOWN AND DEPRESSURIZATION.
Commence an RCS cooldown using Condenser steam dumps.

D.✓ EOP-2.1, POST- LOCA COOLDOWN AND DEPRESSURIZATION.
Commence an RCS cooldown using Steam Generator PORVs.

Modified eops 2 963 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must select a procedure and recall steps for a post loca cooldown and depressurization.

A. Plausible because an RHR pump has been lost and, in addition, EOP-2.4 would be selected if the RWST was at the switchover point of 18% with the RHR sump level less than 414 feet. Aligning Spray for recirculation is an action contained in that procedure.

   Incorrect because conditions are not met for entry into EOP-2.4.

B. Plausible because an RHR pump has been lost and, in addition, EOP-2.4 would be selected if the RWST was at the switchover point of 18% with the RHR sump level less than 414 feet. Stopping spray and charging pumps could be performed in that procedure for a low RWST level.

   Incorrect because conditions are not met for entry into EOP-2.4.

C. Plausible because with RCS pressure stable at 600 psig a post loca cooldown and depressurization is required. That transition would occur from EOP-2.0. The cooldown in that procedure would preferentially be performed to the condenser if the condenser steam dumps are available.

   Incorrect because condenser steam dumps are not available because power to the circulating water pumps has been lost.

D. CORRECT. With RCS pressure stable at 600 psig a post loca cooldown and depressurization is required. That transition would occur from EOP-2.0. The cooldown in that procedure will be performed using SG PORVs because power is not available to the circulating water pumps to maintain the condenser steam dumps available.

K/A: WEO3 EG2.l.20 LOCA Cooldown - Depress. / 4 Ability to execute procedure steps.

K/A Match: The KA is matched because the candidate must recall steps from the procedure for post loca cooldown and depressurization.

Selection criteria; **MODIFIED FROM EOPS 2 963**

Tier: 1 **Group:** 2
Importance Rating: RO 4.6 SRO 4.6
Technical Reference: **EOP-2.0**
EOP-2.1
EOP-2.2
EOP-2.4

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.0 08. SELECT an appropriate transition out of EOP-2.0 given a set of plant conditions

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

EOPS 2 963

Given the following plant conditions:

Initial conditions:

- LOCA occurred.
- All offsite power was lost (115KV and 230KV).
- XCP-610, 1-1 and 2-1, RHR PP A(B) TRIP are in alarm.
- Both RHR pumps are tripped.
- RCS pressure 1100 psig, decreasing.

Current conditions:

- EOP-2.0, LOSS OF REACTOR OR SECONDARY COOLANT in progress.
  - Operators just checked for a potential transition to EOP-1.2, SAFETY INJECTION TERMINATION.
- RB pressure is 13 psig, decreasing.
- RWST level is 55%, decreasing.
- RHR Sump Level is 415 ft, increasing.
- RCS pressure 300 psig, lowering.

Which ONE of the following describes the **next** procedure transition and an action that will be taken in that procedure based on the current conditions?

A. EOP-2.1, POST- LOCA COOLDOWN AND DEPRESSURIZATION.
Commence a RCS cooldown using Condenser steam dumps.

B. EOP-2.1, POST- LOCA COOLDOWN AND DEPRESSURIZATION.
Commence a RCS cooldown using Steam Generator PORVs.

C.✓ EOP-2.4 LOSS OF EMERGENCY COOLANT RECIRCULATION.
Align RB Spray for recirculation.

D. EOP-2.4 LOSS OF EMERGENCY COOLANT RECIRCULATION.
Stop RB Spray pumps and Charging Pumps.

84. Initial conditions:

- 100% power.
- A loss of 230 KV occurred.
- The reactor tripped.
- Immediate actions were completed.
- Operator entered EOP-1.1, ES-0.1 REACTOR TRIP RESPONSE
- Operators then transitioned to EOP-1.3, NATURAL CIRCULATION COOLDOWN at the direction of plant Management.

Current conditions:

- EOP-1.3, NATURAL CIRCULATION COOLDOWN is in progress.
- A cooldown is in progress at 70°F/hour.
- Power has just been restored to bus 1B.
- Conditions are satisfied to start "B" RCP.
- Management has given direction to expedite entry into Mode 5.

Which ONE of the following describes the course of action that the CRS will direct?

A✓ Start "B" RCP and transfer to GOP-6, PLANT SHUTDOWN FROM HOT STANDBY TO COLD SHUTDOWN (MODE 3 TO MODE 5) and increase the cooldown rate.

B. Start "B" RCP and transfer back to EOP-1.1, REACTOR TRIP RESPONSE at the step in effect and complete the actions in that procedure.

C. Start a CRDM Shroud Exhaust Fan, remain in EOP-1.3, and increase the cooldown rate.

D. Start a CRDM Shroud Exhaust Fan, transfer to EOP-1.4, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL and increase the cooldown rate.

NOTE TO EXAMINER:
THIS QUESTION FOR THIS KA THAT WAS SUBMITTED AS PART OF SAMPLE AND HAS BEEN REPLACED PER DISCUSSION WITH CHIEF EXAMINER.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must The K/A is met because the candidate must know the actions that are performed in the natural circulation procedure and select the next procedure for use.

A. CORRECT. When conditions are met an RCP is started while using EOP-1.3, step 1 is repeated in accordance with the reference page. In that step, a transition is made to the appropriate GOP. For this case, with a cooldown in progress that would be GOP-6, PLANT SHUTDOWN FROM HOT STANDBY TO COLD SHUTDOWN (MODE 3 TO MODE 5) and in that procedure the cooldown rate can be increased to 100°F/ hr.

B. Plausible because when conditions are met an RCP is started while using EOP-1.3 and because several EOPs have a transition to procedure and step in effect.

   Incorrect because the transition out of EOP-1.3 after an RCP is started is not to procedure and step in effect.

C. Plausible because with power restored to bus 1B, a CRDM Shroud Exhaust fan can be started. With a fan started, the cooldown rate can be increased to 100°F/hr.

   Incorrect because an RCP will be started and operators will transition out of EOP-1.3 to GOPs.

D. Plausible because with power restored to bus 1B, a CRDM Shroud Exhaust fan can be started. In addition, there is note stating that transition to EOP-1.4 can be performed if a [high] rate of cooldown will cause a void in the head. The cooldown rate in that procedure is 100°F/hr.

   Incorrect because an RCP will be started and operators will transition out of EOP-1.3 to GOPs.

<u>K/A:</u> WEO9 EG2.I .30 Natural Circ. / 4 Ability to locate and operate components, including loc controls.

<u>K/A Match:</u> The K/A is met because the candidate must know the actions that are performed in the natural circulation procedure.

<u>Selection criteria;</u> NEW

Tier: 1 **Group** 2
Importance Rating: RO 4.4 SRO 4.0
Technical Reference: **EOP-1.3**

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-1.3 08. SELECT an appropriate transition out of EOP-1.3 given a set of plant conditions

Question Cognitive Level: **Memory or Fundamental Knowledge** __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 43(b)(5)

<u>SRO Justification:</u> SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

<u>NRC Form ES-401-9 Comments:</u>

Facility Response:

Comments;

85. Given the following plant conditions:

Time 1200:

- 100% power, stable.

Time 1205:

- "B" main steam line breaks inside of the Reactor Building.
- RB pressure is 13 psig and rising.
- Both RB Spray pumps fail to start automatically or manually.

Time 1305:

- The "B" Steam Generator PORV sticks open.

Time 1315:

- A rupture begins on "B" Steam Generator.
- Operators manually re-initiate SI.
- RM-G7/ RM-G18,CNTMT HI RNG Gamma are 2R/ hr and increasing.

Which ONE of the choices below completes the following statement?

The **highest** Emergency Action Level (EAL) classification that the Shift Manager was required to declare for this event was __(1)__ ;

Conditions for that declaration were **first met** at time __(2)__.

Do **not** consider Emergency Director Judgment as a basis for emergency classification.

REFERENCE PROVIDED

A. 1) an ALERT.
   2) 1205.

B. 1) an ALERT.
   2) 1305.

C. 1) a SITE AREA EMERGENCY.
   2) 1305.

D. 1) a SITE AREA EMERGENCY.
   2) 1315.

PROVIDE VCS-EPP-0001, ATTACHMENT I, EAL CLASSIFICATION MATRICES - UNIT 1

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recognize that a loss of the containment barrier exists and determine the appropriate event declaration in accordance with the given procedural reference.

SEE VCS-EPP-0001, Table F-1 for the discussion below.

A. The first part is plausible because at time 1205 a Safety Injection is in progress which the candidate could misapply F-1 item 1.A. .

   Incorrect because an Alert is not the highest declaration.

B. Plausible because at time 1305 a Safety Injection is in progress which the candidate could misapply F-1 item 1.A. In addition, the Containment barrier has been lost because an SG PORV is stuck open on a generator that is faulted to containment. RB pressure is also greater than 12 psig with no RB spray pumps running. This is an ORANGE path of Containment and is also contained as a consideration for F-1 item 4.C potential loss. The candidate could misapply containment barrier loss or potential loss as a basis for Alert.

   Incorrect because an Alert is not the highest declaration.

C. The first part is plausible because at time 1305 a Safety Injection is in progress which the candidate could misapply F-1 item 1.A. In addition, the Containment barrier has been lost because an SG PORV is stuck open on a generator that is faulted to containment. RB pressure is also greater than 12 psig with no RB spray pumps running. This is an ORANGE path of Containment and is also contained as a consideration for F-1 item 4.C potential loss.

   Incorrect because Conditions for a Site Area Emergency are not met at 1305.

D. CORRECT. At time 1315 a rupture begins on a generator that is faulted outside of containment. This is a Site Area Emergency under F-1 Containment 1.A.

K/A: WE14 EA2.2 Loss of CTMT Integrity / 5 Ability to determine and interpret the following as the apply to the event: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A Match: The KA is matched because the candidate must recognize that a loss of the containment barrier exists and determine the appropriate event declaration in accordance with the given procedural reference.

Selection criteria; NEW

Tier: 1 **Group:** 2
Importance Rating: RO 3.3 SRO 3.8
Technical Reference: **VCS-EPP-0001**

Proposed references to be provided to candidates during examination:

VCS-EPP-0001, ATTACHMENT I, EAL CLASSIFICATION MATRICES - UNIT 1

Learning Objective: EPP-001 01 For a given set of conditions, determine which Emergency Action Level applies.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires knowledge "unique to the SRO position" as documented within SAT process as ties the knowledge/ability to the licensee's SRO job position duties.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

86. Initial conditions:

- Mode 4.
- RCS T_{AVG} is stable.
- Pressurizer level is 8% and decreasing at 1%/ minute.
- The FCV-122, CHG FLOW is at 140 gpm and increasing.
- RB pressure is 1 psig and increasing slowly.
- RCS pressure is 380 psig and decreasing.
- RB sump levels are increasing.
- VCT level is 21% and decreasing.

Which ONE of the choices identifies the procedure that will be entered for the conditions given?.

1) Which procedure will be implemented to address the conditions above?
2) What is the appropriate alignment of the flowpath from CVCS to the RCS for the given conditions in that procedure?

A✓ 1) AOP-112.1, SHUTDOWN LOCA.
2) Cold Leg Injection.

B. 1) AOP-112.1, SHUTDOWN LOCA.
2) Normal Charging.

C. 1) AOP-101.1 , LOSS OF REACTOR COOLANT NOT REQUIRING SAFETY INJECTION.
2) Cold Leg Injection.

D. 1) AOP-101.1 , LOSS OF REACTOR COOLANT NOT REQUIRING SAFETY INJECTION.
2) Normal Charging.

NOTE TO EXAMINER:
THIS QUESTION WAS SUBMITTED AS PART OF SAMPLE AND HAS BEEN REVISED TO ADDRESS COMMENTS.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer question correctly, the candidate must determine that the correct procedure for the stated condition and recall the method for CVCS alignment while determining the leak location.

A. CORRECT. The correct procedure for RCS leakage in Mode 4 is AOP-112.1 and charging flow is at 140 gpm which is almost at the top of the scale. With pressurizer level decreasing at 1% minute, pressurizer level cannot be restored. The correct alignment for charging is cold leg injection in step 4 of AOP-112.1.

B. Plausible because the correct procedure for RCS leakage in Mode 4 is AOP-112.1 and because if pressurizer level could be maintained greater than 10%, normal charging would be maintained

   Incorrect because FCV-122 is almost full open and pressurizer level will not be restored.

C. Plausible because various entry conditions are met for entry into AOP101.1 and there is criteria in that procedure to initiate safety injection.

   Incorrect because the correct procedure is not AOP-101.1.

D. Plausible because various entry conditions are met for entry into AOP101.1 and because if that procedure uses normal charging.

   Incorrect because the correct procedure is not AOP-101.1.

K/A: 004 G2.4.47 Chemical and Volume Control Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

K/A Match: the K/A is met because the candidate must select and use a control room procedure to align CVCS components.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 4.2 SRO 4.2
Technical Reference: **AOP-112.1 SHUTDOWN LOCA**
AOP-101.1 LOSS OF REACTOR COOLANT NOT REQUIRING SI HANDOUT

Proposed references to be provided to candidates during examination: None

Learning Objective: AOP-112.1 03.Given a set of plant conditions, DETERMINE the appropriate plant response and operator actions in accordance with AOP-112.1.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

87. Given the following plant conditions:

- 100% power initially.
- A large break LOCA is in progress.
- A lockout occurs on XTF0004, UNIT 1 ENGINEERED SAFEGUARD TRANSFORMER has just occurred.
- The BOP reports that "A" EDG has stopped.
- EOP-2.2, ES-1.3 TRANSFER TO COLD LEG RECIRCULATION is in progress.
    - Operators are reading step 6, **"Verify at least one train of RHR has been aligned to the RHR Sumps**".
    - MVG-8811B RHR SUMP B TO RHR PP B will **not** open automatically or manually.
    - MVG-8812B, RHR SUMP B TO RHR PP B is open.

Which one of the following identifies the procedure action, **out of the ones listed below**, that will be implemented **next** to maintain core cooling?

A. Remain in EOP-2.2 and use AOP-304.1A LOSS OF BUS 1DA WITH THE DIESEL NOT AVAILABLE to reenergize bus 1DX to allow a restart of "A" RHR pump.

B. Remain in EOP-2.2 and use AOP-304.1A LOSS OF BUS 1DA WITH THE DIESEL NOT AVAILABLE to locally start "A" EDG to allow a restart of "A" RHR pump.

C. Transfer to EOP-2.4, LOSS OF EMERGENCY COOLANT RECIRCULATION and stop a Charging Pump to preserve the RWST.

D✓ Transfer to EOP-2.4, LOSS OF EMERGENCY COOLANT RECIRCULATION and add makeup to the RWST to maintain a water source for Charging.

QUESTION USAGE:
2016 NRC

REVISION HISTORY:
Rev. 1 submitted by RJ - Changed stem to include valve failures
Ops Review: Danny Rhymer
Approved:

Rev. 0 submitted by RJ

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine the correct procedural action after a loss of recirculation cause by failure of RHR suction valves.

A. Plausible because the procedures such as EOP-2.0 and EOP-6.0 direct the use of AOP-304 to restore offsite power to ESF buses. Restoration of 1DA would allow restart of "A" RHR pump.

   Incorrect because a transition to EOP-2.4, LOSS OF EMERGENCY COOLANT RECIRCULATION is required for the conditions given.

B. Plausible because loss of power AOP-304 directs operators to investigate the cause of the loss of power including malfunctions of the diesels. Restoring the A EDG to service would allow the restart to the "A" RHR pump.

   Incorrect because a transition to EOP-2.4, LOSS OF EMERGENCY COOLANT RECIRCULATION is required for the conditions given.

C. Plausible because a transfer is required to EOP-2.4 since both RHR pumps have been lost and if two charging pumps are running, one is stopped to preserve the RWST inventory.

   Incorrect because with only one Charging pump running, it will not be stopped.

D. CORRECT. In accordance with EOP-2.2, if at least one train of recirculation cannot be established and maintained, a transition to EOP-2.4 is required. EOP-2.4 will direct actions to makeup to the RWST which will allow the Charging pump to continue to inject.

K/A: 005 A2.04 Residual Heat Removal Ability to (a) predict the impacts of the following on the system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction

K/A Match: The KA is matched because the candidate must determine the correct procedural action after a loss of recirculation cause by failure of RHR suction valves.

Selection criteria; **REVISED BANK (FROM 2016 NRC)**

Tier: 2 **Group:** 1
Importance Rating: RO 2.9 SRO 2.9
Technical Reference: **EOP-2.2, ES-1.3 TRANSFER TO COLD LEG RECIRCULATION**
SDD EOP-2.2
EOP-2.4

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-2.2 07 RELATE any systems/components operation, indication, or malfunction to its effect on EOP-2.2

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

78. Given the following plant conditions:

Time 0100

- 100% power initially.
- A large break LOCA occurred.
- "B" RHR pump failed to start manually or automatically.

Time 0140:

- The crew entered EOP-2.2, ES-1.3 TRANSFER TO COLD LEG RECIRCULATION.
  - The CRS has just begun reading Step 1.
- A lockout occurs on XTF0004, UNIT 1 ENGINEERED SAFEGUARD TRANSFORMER.
- The BOP reports that "A" EDG cannot be started from the Main Control Board.

Which one of the following identifies the procedure action, out of the ones listed below, that will be implemented **next** to maintain core cooling?

A. Remain in EOP-2.2 and use AOP-304.1A LOSS OF BUS 1DA WITH THE DIESEL NOT AVAILABLE to reenergize bus 1DX to allow a restart of "A" RHR pump.

B. Remain in EOP-2.2 and use AOP-304.1A LOSS OF BUS 1DA WITH THE DIESEL NOT AVAILABLE to locally start "A" EDG to allow a restart of "A" RHR pump.

C. Transfer to EOP-2.4, LOSS OF EMERGENCY COOLANT RECIRCULATION and stop a Charging Pump to preserve the RWST.

D. Transfer to EOP-2.4, LOSS OF EMERGENCY COOLANT RECIRCULATION and add makeup to the RWST to maintain a water source for Charging.

88. Given the following plant conditions:

- Mode 5.
- "A" loop is supplying non-essential CCW loads.
- "A" CCW pump is running with the associated switch in NORMAL AFTER START.
- "B" CCW pump is OFF with the associated switch in NORMAL AFTER STOP.
- "C" CCW is OFF, aligned electrically to the "B" loop, and the supply breaker **cannot** be racked down.

Which ONE of the choices below answers both of the following questions?

1) Which of the CCW loops is OPERABLE?
2) Can the plant be moved to MODE 4 within the **next** 48 hours in the current condition in accordance with Technical Specifications?

REFERENCE PROVIDED

A. 1) "A" **only**.
   2) Yes.

B.✓ 1) "A" **only**.
   2) No.

C. 1) "B" **only**.
   2) Yes.

D. 1) "B" **only**.
   2) No.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

PROVIDE TECHNICAL SPECIFICATION 3.7.3, PAGE 3/4 7-11.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine technical specification actions required for inoperability of CCW pumps.

A. The first part is correct. The second part is plausible because action statement of T.S. 3.7.3 states that 72 hours is allowed for restoration and the candidate may not apply T.S. 3.0.4 and think that the plant can be escalated in mode.

   Incorrect because the plant cannot be escalated in mode.

B. CORRECT. Only the "A" loop is operable. With the "B" CCW pump switch in normal after stop and the "C" CCW pump racked up on that train, "B" loop is inoperable. A T.S. Action statement of T.S. 3.7.3 and T.S. 3.0.4 apply so that plant mode cannot be escalated.

C. The first part is plausible because C CCW pump is aligned to B loop and A loop is the active loop. The C CCW pump is normally aligned to the active loop in standby. The second part is plausible because action statement of T.S. 3.7.3 states that 72 hours is allowed for restoration and the candidate may not apply T.S. 3.0.4 and think that the plant can be escalated in mode.

   Incorrect because "B" is not operable because of switch and breaker positions for B and C CCW pumps and the plant cannot be escalated in mode.

D. The first part is plausible because C CCW pump is aligned to B loop and A loop is the active loop. The second part is correct when one either loop alone is inoperable.

   Incorrect because "B" is not operable because of switch and breaker positions for B and C CCW pumps.

K/A: 008 G2.I .23 Component Cooling Water Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A Match: The KA is matched because the candidate must determine technical specification actions required for inoperability of CCW pumps.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 4.3 SRO 4.4
Technical Reference: **T.S. 3.7.3**
SOP-118

Proposed references to be provided to candidates during examination:

TECHNICAL SPECIFICATION 3.7.3, PAGE 3/4 7-11.

Learning Objective: SB-4 017 DETERMINE actions required if an LCO is not met including applicability of Specifications 3.0.3, 3.0.4, and 4.0.4.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(2)

SRO Justification: SRO question because it requires knowledge of facility operating limitations in the TS and their bases involving one or more of the following for TS, TRM or ODCM: Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1).

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

89. Initial conditions:

- 100% power.
- XCP-6117, ESF XFMR FEED KV increased to 117 KV.
- A malfunction occurred on XTF0006, XTF0004 7.2KV VOLTAGE REGULATOR causing output voltage to lower excessively.
- "A" EDG started automatically.
- The 1A EDG output breaker has just **closed**.

Which ONE of the choices below completes the following statements?

The CRS will **first** implement __(1)__ as a result of this event.

Prior to adjusting bus 1DA voltage and frequency, the EMERG START OVRRIDE pushbutton __(2)__ be depressed.

A.✓ 1) SOP-306, EMERGENCY DIESEL GENERATOR
2) will

B. 1) SOP-306, EMERGENCY DIESEL GENERATOR
2) will **not**

C. 1) AOP-301.1, RESPONSE TO ELECTRICAL GRID ISSUES
2) will

D. 1) AOP-301.1, RESPONSE TO ELECTRICAL GRID ISSUES
2) will **not**

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the applicant must select the procedure to ensure proper equipment alignments and operations of the diesel within limits and recall guidance in that procedure for maintain bus frequency and voltage.

A. CORRECT. An automatic start of an EDG requires the use of SOP-306 to ensure alignments of equipment and to maintain bus voltage and frequency. Before voltage and frequency adjustment is made, the EMERG START OVERRIDE pushbutton will be depressed.

B. The first part is correct. The second part is plausible because the diesel has experienced an emergency start and the candidate may think that the adjustment must be attempted without depressing the pushbutton.

   Incorrect because the EMERG START OVERRIDE pushbutton will be depressed.

C. The first part is plausible because 115 KV voltage decreased has occurred which started in a chain of events that resulted in an automatic start of an EDG. The second part is correct.

   Incorrect because AOP-301.1 will not be entered because entry conditions are not met.

D. The first part is plausible because 115 KV voltage decreased has occurred which started in a chain of events that resulted in an automatic start of an EDG. The second part is plausible because no guidance is contained in that procedure to depress the EMERG START OVERRIDE.

   Incorrect because AOP-301.1 will not be entered because entry conditions are not met.

K/A: 064 A2.02 Emergency Diesel Generator Ability to (a) predict the impacts of the following on the system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Load, VARS, pressure on air compressor, speed droop, frequency, voltage, fuel oil level temperatures

K/A Match: the KA is matched because the candidate must select the procedure to ensure proper equipment alignments and operations of the diesel within limits and recall guidance in that procedure for maintain bus frequency and voltage.

Selection criteria; NEW

Tier: 2 **Group:** 1
Importance Rating: RO 2.7 SRO 2.9
Technical Reference: **SOP-306**
AOP-301.1

Proposed references to be provided to candidates during examination: None.

Learning Objective: IB-5 19. DESCRIBE the normal operation of the Diesel Generator System including: 5. Operation in Emergency Start Mode

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

90. Given the following plant conditions:

- Mode 4
- RCS temperature is 325°F, stable.
- RM-A2, REACTOR BUILDING SAMPLE LINE MONITOR has been declared inoperable due to a detector failure at **2345, 8/19**.
- Manual leakrate calculations are being done to satisfy Surveillance 4.4.6.2.1.d in accordance with the associated action statement in T.S.3.4.6.1, REACTOR COOLANT SYSTEM LEAKAGE, LEAKAGE DETECTION SYSTEMS.
- The last calculation was performed at **0215, 8/20**.

Which ONE of the following describes the **latest** time for completion of the next leakrate calculation that will comply with Technical Specifications?

REFERENCE PROVIDED

A. 2345, 8/20.

B.✓ 0215, 8/21.

C. 0545, 8/21.

D. 0815, 8/21.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. One reactor building sump level,

b. One reactor building atmosphere radioactivity monitor (gaseous or particulate), and

c. One reactor building cooling unit condensate flow rate monitor.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the reactor building sump level monitor inoperable, perform surveillance requirement 4.4.6.2.1.d (Reactor Coolant System water inventory balance) at least once per 24 hours[1] and restore the required reactor building sump level monitor to OPERABLE status within 30 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the required reactor building atmosphere radioactivity monitor inoperable, analyze grab samples of the containment atmosphere at least once per 24 hours or perform surveillance requirement 4.4.6.2.1.d (Reactor Coolant System water inventory balance) at least once per 24 hours[1] and restore the required reactor building atmosphere radioactivity monitor to OPERABLE status or verify the reactor building cooling unit condensate flow rate monitor is OPERABLE within 30 days; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the reactor building cooling unit condensate flow rate monitor inoperable, perform a CHANNEL CHECK of the required reactor building atmosphere radioactivity monitor at least once per 8 hours or perform surveillance requirement 4.4.6.2.1.d (Reactor Coolant System water inventory balance) at least once per 24 hours[1]; otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With the reactor building sump level monitor and the reactor building cooling unit condensate flow rate monitor inoperable and with the reactor building atmosphere gaseous radioactivity monitor being the only remaining OPERABLE leakage

---

(1) Not required to be performed/completed until 12 hours after establishment of steady state operation.

PROVIDE T.S. 3.4.6.1, PAGE 3/4 4-18.

Proposed for use on 2018 NRC - rj

QUESTION USAGE:
SRO-14-01-EXAM 7 (I&C 1)
SRO-11-01-NRC (**2013-SRO NRC**)

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the SRO candidate must determine that Surveillance requirement 4.4.6.2.1.d must be performed every 24 hours and that the 25% extension allowed by T.S. 4.0.2 does not apply because the surveillance is being performed to satisfy an action statement. The maximum time before the next performance must then be selected.

A. Plausible because 2345 is 24 hours after the declaration of INOPERABLITY.

Incorrect because the correct time is 24 hours after the last performance of the leakrate calculation.

B. CORRECT. The last performance of 4.4.6.2.1.d was at 0215, 8/20/13 and the 25% extension allowed by 4.0.2 does **not** apply. The surveillance must next be performed at 0215, 8/21/13. (0215, 8/20/13 + 24 hours).

C. Plausible because 0545 is 30 hours after the declaration of INOPERABLITY. This would represent the 24 hours after INOPERABLITY plus an additional 6 hours for a 25% extension.

Incorrect because the correct time is 24 hours after the last performance of leakrate calculation and the 25% extension is not allowed when the surveillance is being performed to satisfy an action statement.

D. Plausible because 0815 is 30 hours after the last performance of 4.4.6.2.1.d. This would represent the 24 hours after the last performance plus an additional 6 hours for a 25% extension.

Incorrect because the 25% extension is not allowed when the surveillance is being performed to satisfy an action statement.

K/A: 073A2.02 K/A: 073 Process Radiation Monitoring (PRM) System A2: Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: 2.02: Detector failure

K/A Match: the KA is matched because it requires the use of Technical Specifications to determine surveillance requirements for the case of a detector failure of Process monitoring channel RM-A2 Reactor Building Sample Line Monitor.

Selection criteria; **BANK**

Tier: 2 **Group:** 1
Importance Rating: RO 2.7 SRO 3.2
Technical Reference: **T.S. 3.4.6.1., Surveillance 4.4.6.2.1.d, and TSR 1007 for Section 4.0.2.**

Proposed references to be provided to candidates during examination:

T.S. 3.4.6.1, PAGE 3/4 4-18.

Learning Objective: SB-4 021 PLAN a course of action that will ensure compliance with the limiting conditions for operation and/or action statements.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(2)

SRO Justification: SRO question because it requires knowledge of facility operating limitations in the TS and their bases involving one or more of the following for TS, TRM or ODCM: Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1).

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

91. Given the following plant conditions:

Time 0400:

- 75% power.
- T_{AVG} is on program and matched with T_{REF}.
- Pressurizer level indications are as follows:

  LI-459A 51%, stable.
  LI-460 51%, stable.
  LI-461 45%, stable.

- FCV-122, CHG FLOW controller is in AUTO, output is 36%, stable.

Time 0500:

- Pressurizer level transmitter LT-459 fails low.

Current time is 1300:

- I&C reports that the problem(s) cannot be corrected for 24 hours.

Which ONE of the choices below completes the following statements?

At the current time of 1300, the **highest** power level that Technical Specifications will allow to be maintained is __(1)__.

The **earliest** time at which this condition was required to be established was __(2)__.

REFERENCE PROVIDED

A. 1) 5%.
   2) 1100.

B✓ 1) 5%.
   2) 1200.

C. 1) 0%.
   2) 1100.

D. 1) 0%.
   2) 1200.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1	6#
12. A. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6#
13. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	6#
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in each loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch in same loop or 2/loop-level and 1/loop-flow mismatch in same loop	1, 2	6#

PROVIDE T.S. TABLE 3.3.1, PAGE 3/4 3-3 AND OAP-106.1, ATT 1 PAGES 29, 30

Modified pzrlvlcntrlsys50 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must determine that a pressurizer level channel has drifted off of the temperature-dependent level program and is inoperable along with another channel. The candidate must then determine the time schedule for power reduction.

A. The first part is correct. The second part is plausible because at time 0400, channels 459A and 461 are both at 51% and the candidate may think that both these channel are inoperable at that time. If 3.0.3 in entered the candidate may compute a power reduction to mode 3 in 7 hours (1 hour to prepare and 6 to reduce power). This leads to an answer of 1100.

   Incorrect because the plant is not required to be at 5% at 1100.

B. CORRECT. Programmed level for 75% is 0.75 x 35% span + 25% = 51%. At 0400, channel 460 is out of tolerance (>4%) and fails a channel check. Then at 0500, channel 459 fails. 3.0.3 is entered at 0500 which results in a power reduction to the point at which the specification can be exited in 7 hours (1 hour to prepare and 6 to reduce power). The mode of applicability for the high pressurizer level trip function is Mode 1 only. This leads to an answer of 5% by 1200.

C. Plausible because at time 0400 channels 459A and 461 are both at 51% and the candidate may think that both these channel are inoperable at that time. If 3.0.3 in entered at that time the candidate may compute a power reduction to mode 3 in 7 hours (1 hour to prepare and 6 to reduce power). This leads to an answer of 1100.

   Incorrect two channels are not inoperable at 0400 and because the power must be reduced only to the level at which the specification no longer applies; in this case 5%.

D. Plausible because at 0400, channel 460 is out of tolerance and then at 0500, 459 fails. If 3.0.3 in entered at that time the candidate may compute reduction to mode 3 in 7 hours (1 hour to prepare, 6 to reduce power). This leads to an answer of 1100.

   Incorrect because the power must be reduced only to mode 2 or 5%.

KA: 011 G2.2.44 Pressurizer Level Control Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions

K/A Match: the KA is matched because the candidate must determine that a pressurizer level channel has drifted off of the temperature-dependent level program by and is inoperable. The candidate must then determine the time schedule for power reduction.

Selection criteria; **MODIFIED FROM PZRLVLCNTRLSYS50**

Tier: 2 **Group:** 2
Importance Rating: RO 4.2 SRO 4.4
Technical Reference: **T.S. 3.3.1**
T.S. DEFINITIONS
T.S. 3.0.3

Proposed references to be provided to candidates during examination:

PROVIDE TECHNICAL SPECIFICATION TABLE 3.3.1, PAGE 3/4 3-3 AND OAP-106.1, ATTACHMENT 1 PAGES 29 AND 30.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

Learning Objective: SB-4 021 PLAN a course of action that will ensure compliance with the limiting conditions for operation and/or action statements.

10 CFR Part 55 Content: 43(b)(2)

SRO Justification: SRO question because it requires knowledge of facility operating limitations in the TS and their bases involving one or more of the following for TS, TRM or ODCM: Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1).

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

PZR LEVEL CNTRL SYS 050

Given the following plant conditions:

- Mode 3.
- T_{AVG} 557°F, stable.
- Pressurizer level indications are as follows:

  LI-459A 25%, stable.

  LI-460 20%, stable.

  LI-461 25%, stable.
- FCV-122, CHG FLOW controller is in AUTO, output is 36%, stable.

Which ONE of the following describes the status of the High Pressurizer Level trip **Functional Unit** for the current Mode and the **highest** plant Mode that can be attained in accordance which Technical Specifications if all applicable action statements **are satisfied**?

REFERENCE PROVIDED

A.✓	OPERABLE;	Mode 1.
B.	OPERABLE;	Mode 3.
C.	INOPERABLE;	Mode 2.
D.	INOPERABLE;	Mode 3.

92. Given the following plant conditions:

Time 0300:

- 75% power and decreasing with a plant shutdown in progress.
- A primary leak to "B" steam generator of 30 gallons per day is present.
- RM-A9, CNDSR EXHAUST GAS ATMOS MONITOR High Rad alarm is in.

Time 0315:

- Pressurizer level and pressure began decreasing.
- The RM-G19B, STMLN HI RNG GAMMA red alarm light is lit.
- Operators manually tripped the reactor.

Time 0320:

- EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION was in progress.
- All steam generators began depressurizing uncontrollably.

Time 0325:

- EOP-3.1 ECA-2.1 UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS is in progress.
- The CRS is reading step 7, "**Check if Secondary radiation levels are normal**".
- The BOP reports that all radiation alarms are clear and show no elevated radiation levels.

Which ONE of the choices completes the following statements?

In accordance with the bases for EOP-3.1, the term "normal", as it applies to the RM-G19B radiation monitor readings checked in step 7, means the value of readings __(1)__.

The CRS __(2)__ direct a transfer to EOP-4.0, E-3 STEAM GENERATOR TUBE RUPTURE at step 7 **prior** to obtaining results of steam generator samples.

A✓ 1) during routine plant operations.
2) will

B. 1) during routine plant operations.
2) will **not**

C. 1) that are expected after a reactor trip.
2) will

D. 1) that are expected after a reactor trip.
2) will **not**

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the basis discussion for the use of radiation alarms that clear after the trip for the purpose of procedure selection.

A. CORRECT. The basis for the EOP step (E-3 ERG step 23) states that normal for the purpose of this step means the value of the process parameter experienced during routine operations The CRS can direct a transition based on the standing previously experienced RM-G19C alarm at 0315.

D. The first part is correct. The second part is plausible because the candidate may either think that a transition to EOP-4.0 is not allowed until sample results are obtained or that the step in EOP-3.1 does not direct transition to EOP-4.0.

   Incorrect because the basis for the EOP step states that normal for the purpose of this step means the value of the process parameter experienced during routine operations and the crew will transition to EOP-4.0.

C. The first part is plausible the reactor has tripped and most indications for EOP compliance are taken for the condition at which at the step is read. The second part is plausible because there was a primary to secondary leak at time 0300 with a power reduction due to AOP usage. The candidate may believe that this is sufficient justification for the transition.

   Incorrect because the basis for the EOP step states that normal for the purpose of this step means the value of the process parameter experienced during routine operations.

D. The first part is plausible the reactor has tripped and most indications for EOP compliance are taken for the condition at which at the step is read. The second part is plausible because the candidate may either think that a transition to EOP-4.0 is not justified until sample results are obtained or that the step in EOP-3.1 does not direct transition to EOP-4.0..

   Incorrect because the basis for the EOP step states that normal for the purpose of this step means the value of the process parameter experienced during routine operations and the crew will transition to EOP-4.0.

K/A: 072 G2.4.18 Area Radiation Monitoring Knowledge of the specific bases for EOPs.

K/A Match: The KA is matched because the candidate must recall the basis discussion for the use of radiation alarms that clear after the trip for the purpose of procedure selection.

Selection criteria; NEW

Tier: 2 **Group:** 2
Importance Rating: RO 3.3 SRO 4.0
Technical Reference: **EOP-1.0**
E-0 ERG BASES

Proposed references to be provided to candidates during examination: None

Learning Objective: EOP-1.0 05. STATE the bases or reasons for each action contained in EOP-1.0 This should include, but not be limited to, the following: 5. Definition of "normal"

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires the candidate to assess plant conditions and then select a procedures to mitigate, recover, or with which to proceed.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

93. Time 0630:

- A plant cooldown in progress.
- RCS T_{AVG} is 425°F and decreasing.
- The Diesel Driven Fire Service pump is **inoperable**.

Time 0700:

- A small fire with open flames on the IB-427' elevation is reported to the control room.

Time 0705:

- An AO reports that a Fire Brigade member is responding and should extinguish the fire within the next five minutes.

Time 0707:

- A loss of 230 KV power has occurred.
- The fire has reflashed and is now growing.
- An AO reports that Fuel oil lines from the Diesel Generator "A" Fuel Oil Storage tank to the "A" EDG Fuel Oil Day Tank have been damaged.

Which ONE of the choices below completes the following statements?

A start of the Alternate Fire Pumps __(1)__ required to maintain Fire Service header pressure.

The **highest** classification in accordance with VCS-EPP-0001, CLASSIFICATION OF EMERGENCIES was an __(1)__.

Do **not** consider Emergency Director Judgment as a basis for emergency classification.

REFERENCE PROVIDED

A. 1) is
2) Unusual Event.

B✓ 1) is
2) Alert.

C. 1) is **not**
2) Unusual Event.

D. 1) is **not**
2) Alert.

PROVIDE VCS-EPP-0001, ATTACHMENT I, EAL CLASSIFICATION MATRICES - UNIT

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine the correct EAL declaration for a fire event in which the fire header has depressurized.

Refer to VCS-EPP-0001, Attachment 1 for this discussion.

A. The first part is correct. The second part is plausible because the highest available declaration on row H4 on page 1 of 3 of attachment 1 is Unusual Event.

   Incorrect because the highest declaration is not UE.

B. CORRECT. With the loss of power for the electric-driven fire pump due to the loss of 230 KV, start of the alternate fire pumps is required to maintain fire service header pressure. With visible damage reported to the diesel fuel oil storage system due to the fire, conditions for an Alert on item SA9.1 on Attachment 1 page 2 of 3 are met.

C. The first part is plausible because there is an electric-driven fire pump that will start on low header pressure. The second part is plausible because the highest available declaration on row H4 on page 1 of 3 of attachment 1 is Unusual Event.

   Incorrect because the electric driven fire pump has lost power and the alternate fire pumps must be started and because the highest declaration is not UE.

D. The first part is plausible because there is an electric-driven fire pump that will start on low header pressure. The second part is correct.

   Incorrect because the electric driven fire pump has lost power and the alternate fire pumps must be started.

K/A: 086 A2.02 Fire Protection Ability to (a) predict the impacts of the following on the system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low FPS header pressure

K/A Match: The KA is matched because the candidate must determine the correct EAL declaration for a fire event in which the fire header has depressurized.

Selection criteria; NEW

Tier: 2 **Group:** 2
Importance Rating: RO 3.0 SRO 3.3
Technical Reference: **VCS-EPP-0001**
SOP-509

Proposed references to be provided to candidates during examination:

VCS-EPP-0001, ATTACHMENT I, EAL CLASSIFICATION MATRICES - UNIT 1

Learning Objective: EPP-001 01 For a given set of conditions, determine which Emergency Action Level applies.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires knowledge "unique to the SRO position" as documented within SAT process as ties the knowledge/ability to the licensee's SRO job position duties.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

94. Given the following condition:

- A new Special Order was implemented today.
- The following positions will be assumed by personnel who have the licenses specified:

(1) Shift Manager	SRO license
(2) CRS	SRO license
(3) NROATC	RO license
(4) BOP	RO license
(5) Shift Engineer	SRO license

Which of the on-coming watchstanders **must** initial the Special Order cover page to acknowledge receipt in accordance with OAP-100.4 COMMUNICATIONS?

A. (1) **only**.

B. (1) and (2) **only**.

C.✓ (1), (2) and (5) **only**.

D. (1), (2), (3), (4) and (5).

NOTE TO EXAMINER:
THE QUESTION THAT WAS SUBMITTED FOR THIS KA AS PART OF SAMPLE WAS RE-WRITTEN PER DISCUSSION WITH CHIEF EXAMINER.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall he requirements for signing special orders.

A. Plausible because review of special orders appears on the Shift Manager turnover checklist contained in OAP-100.5 and that position maintains overall responsibility for the conduct of the shift.

   Incorrect because the other personnel besides Shift Manager must sign when a new special order is issued.

B. Plausible because the review of special orders only appears on the Shift Manager and CRS turnover checklist contained in OAP-100.5.

   Incorrect because the other personnel besides the Shift Manager and CRS must sign when a new special order is issued.

C. CORRECT. Items (1), (2) and (5) identify licensed SRO positions. Licensed SROs must sign a signature block on the first watch stood after a new special order is issued.

D. Plausible because all positions are licensed and these personnel should be aware of the contents of special orders.

   Incorrect because only SRO licensed operators must sign new special orders.

K/A: G2.1.15 Conduct of operations Knowledge of administrative requirements for temporary management directives such as standing orders, night orders, Operations memos, etc

K/A Match: The K/A is met because the candidate must know the requirements for signing special orders.

Selection criteria; NEW

Tier: 3
Importance Rating: RO 2.7 SRO 3.4
Technical Reference: **OAP-100.4**
OAP-100.6

Proposed references to be provided to candidates during examination: None

Learning Objective: **OAP-100.6** 11 DISCUSS shift relief expectations as described in OAP-100.6.

Question Cognitive Level: **Memory or Fundamental Knowledge** __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:

SRO Justification: SRO because the question requires knowledge “unique to the SRO position” as documented within SAT process as ties the knowledge/ability to the licensee’s SRO job position duties.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

ADMIN PROCEDURE 595

Given the following plant conditions:

Unit is in Mode 1.

Which ONE of the following contains two items that are required to be reviewed by the **<u>Control Room Supervisor</u>** and initialed for on the Control Room Supervisor Relief checklist?

A. BISI and SPDS displays.
   FEP Manning Sheet.

B.✓ BISI and SPDS displays.
   ESF Systems Status.

C. Status of Security Keys.
   FEP Manning Sheet.

D. Status of Security Keys.
   ESF Systems Status.

95. Initial Conditions:

- MODE 6.
- The refueling team was raising a fuel assembly from the core.
  - The fuel assembly had **not** yet cleared the top of the core.
- A red OVERLOAD IL-4 light was received.
- All activities for fuel movement in the RB were stopped.
- The digital Load Readout indicated that the load cell has failed to a **higher** output.

Current Conditions:

- Technicians have determined that the load cell output did **not** trip the Master Overload interlock.
- The decision has been made to complete the fuel assembly transfer and then maintain the Manipulator Crane at the Transfer Canal to allow repairs.
- The Overload Bypass switch TS-1 was placed in the BYPASS position.
- Lifting of the fuel assembly has resumed.

Which ONE of the choices below completes the following statements?

In accordance with FHP-601, REFUELING ORGANIZATION, the use of TS-1 required __(1)__.

While TS-1 is in the BYPASS position, the Master Overload __(2)__ provide protection against an overload condition.

A✓ 1) authorization by the Refueling Operations Coordinator (ROC) **and** concurrence by the Core Loading Supervisor (SRO).

2) does

B. 1) authorization by the Refueling Operations Coordinator (ROC) **and** concurrence by the Core Loading Supervisor (SRO).

2) does **not**

C. 1) authorization by the Refueling Operations Coordinator (ROC); concurrence by the Core Loading Supervisor (SRO) is **not** required.

2) does

D. 1) authorization by the Refueling Operations Coordinator (ROC); concurrence by the Core Loading Supervisor (SRO) is **not** required.

2) does **not**

Proposed for use on 2018 NRC - rj

2015 NRC exam question
SRO Question 92 - NRC authored justification appears below

KA: 034K6.01: Fuel Handling Equipment System (FHES). Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System: Fuel handling equipment.
CFR: 41.7 / 45.7
IR RO/SRO: 2.1 / 3.0

K/A Match Analysis
This question matches the K/A statement by requiring the applicant to have knowledge of the impact of a malfunction involving an OVERLOAD condition with the manipulator crane.

Answer Choice Analysis
A. Incorrect. The Overload bypass switch will only bypass the Overload setpoint set with the LOAD SELECT switch; the Master Overload setpoint can't be bypassed. Plausible because an Overload bypass switch does exist and the applicant may not be aware of its limitation.
B. CORRECT. The Master Overload setpoint can't be bypassed and the emergency manual drive must be used.
C. Incorrect. The first part is correct; the Master Overload setpoint can't be bypassed and the emergency manual drive must be used. The second part is incorrect; plausible because the machine needs to be moved to the transfer canal location and the interlock involves the hoist; however as long as the hoist is not being moved, the refueling machine can be moved without any bypass; also this interlock can't be bypassed.
D. Incorrect. The first part is incorrect, see answer A for explanation. The second part is correct

Supporting References
1. FHP-611.09, Refueling Machine Operation, Rev 12
2. GS-04, Rev 15, Fuel Handling System Handout

References Provided to Applicant
None

Notes
New
Learning Objective: _
GS-04 Fuel Handling System Handout Obj. 010 (RO and SRO) and 012 (SRO only)

Question Cognitive Level:
Memory or Fundamental Knowledge: ____
Comprehension or Analysis: __X__

K/A: G2.1.40 Conduct of operations Knowledge of refueling administrative requirements

K/A Match: The K/A is met because the candidate must recall the interlock which provides overload protection during refueling machine operation and also know who will authorize the bypass of an refueling machine interlock.

Selection criteria; **BANK**

Tier: 3 **Group:**
Importance Rating: RO 2.8 SRO 3.9
Technical Reference: **FHP- 601, REFUELING ORGANIZATION**
FHP- 611.09, REFUELING MACHINE OPERATION

Proposed references to be provided to candidates during examination: None

Learning Objective: FHP-601 - 03 STATE personnel who authorize or concur with the bypass of an interlock on Refueling Equipment.

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 43(b)(7)

SRO Justification: SRO Only because the question tests knowledge of Fuel-Handling Facilities and Procedures [10 CFR 55.43(b)(7)] including:
refuel floor SRO responsibilities

NRC Form ES-401-9 Comments (2011 NRC Exam):

Facility Response:

comments;

96. Given the following plant conditions:

- Mode 3.
- Cooldown in progress for a refueling outage.
- The "A" RB Spray Pump is being returned to an OPERABLE status after corrective maintenance.

Which ONE of the choices below completes the following statements in accordance with SAP-205, STATUS CONTROL AND REMOVAL AND RESTORATION?

1) Which position in Operations will update BISI and EOOS to ensure that they reflect actual equipment status?

2) Which procedural tool will document the update of BISI and EOOS after the "A" RB Spray Pump is returned to service?

A. 1) Reactor Operator.
   2) SAP-205, Attachment 1, REMOVAL AND RESTORATION CHECKLIST.

B. 1) Reactor Operator.
   2) SAP-205, Attachment 6, OUTAGE REMOVAL AND RESTORATION CHECKLIST.

C.✓ 1) Control Room Supervisor.
   2) SAP-205, Attachment 1, REMOVAL AND RESTORATION CHECKLIST.

D. 1) Control Room Supervisor.
   2) SAP-205, Attachment 6, OUTAGE REMOVAL AND RESTORATION CHECKLIST.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must recall the administrative tools for tracking inoperability of a safety-related pump and the required update to BISI and EOOS.

A. The first part is plausible because an RO is a control room watchstander who acknowledges and announces BISI alarms.

   Incorrect because an the CRS updates BISI and EOOS.

B. The first part is plausible because an RO is a control room watchstander who acknowledges and announces BISI alarms. The second part is plausible because the stated conditions are that the plant is in an outage with a cooldown in progress. Additionally, if the component was not a Technical Specification required item for Mode 4 or above, the outage R&R would be appropriate.

   Incorrect because an the CRS updates BISI and EOOS and because an outage R&R is not used for an RB spray pump.

C. CORRECT. The CRS will update BISI and EOOS to reflect plant status and when a Technical Specification required item for Mode 4 or above is taken out of service, the SAP-205, Attachment 1 R&R is correct.

D. The first part is correct. The second part is plausible the stated conditions are that the plant is in an outage with a cooldown in progress. Additionally, if the component was not a Technical Specification required item for Mode 4 or above, the outage R&R would be appropriate.

   Incorrect because the outage R&R will not be used for an RB spray pump.

K/A: G2.2.23 Equipment Control Ability to track Technical Specification limiting conditions for operations.

K/A Match: The KA is matched because the candidate must recall the administrative tool for tracking inoperability of a safety-related pump and the required update to BISI and EOOS.

Selection criteria; NEW

Tier: 3 **Group:**
Importance Rating: RO 3.1 SRO 4.6
Technical Reference: **SAP-205**

Proposed references to be provided to candidates during examination: None

Learning Objective: SAP-205 03. DEFINE the following terms as used in SAP-205: • Action R&R • Outage R&R

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content:

SRO Justification: SRO because the question requires knowledge "unique to the SRO position" as documented within SAT process as ties the knowledge/ability to the licensee's SRO job position duties.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

97. Initial conditions:

- A charge of "A" Battery, "XBA1A," is in progress.
- XFN-38A **and** XFN-39A were found tripped.
- I&C has determined that a faulty control circuit input has caused the problem.

Current condition:

- An electrical lead was lifted to allow operation of XFN-38A/XFN-39A SPLY & EXH FANS using the provisions of SAP-148, TEMPORARY BYPASS, JUMPER, AND LIFTED LEAD CONTROL.
- A review and concurrence has been provided by Design Engineering.

Which ONE of the choices below completes the following statements?

The **longest** time that the lead can remain in that condition without requiring additional approvals is __(1)__;

The **highest** position in the Operations chain of command that must approve maintaining the lead lifted after that time is the __(2)__.

A. 1) 14 days.
   2) General Manager, Nuclear Plant Operations.

B. 1) 14 days.
   2) Manager, Operations.

C.✓ 1) 90 days.
   2) General Manager, Nuclear Plant Operations.

D. 1) 90 days.
   2) Manager, Operations.

Modified adm148 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 Submitted by RJ
Ops Rev:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must recall time limits and approvals for an inoperability resulting in a change to the facility.

A. Plausible because 14 days is the amount of time that Engineering has to review the lifted lead. The second part is correct.

   Incorrect because the lead may be left for up to 90 days.

B. Plausible because 14 days is the amount of time that Engineering has to review the lifted lead. The second part is plausible because Operations Manager provides the initial approval.

   Incorrect because the lead may be left for up to 90 days and because SAP-148 requires GMNPO approval to have a lead lifted for more than 90 days

C. CORRECT. The lead may be left for up to 90 days and SAP-148 requires GMNPO approval to have a lead lifted for more than 90 days

D. The first part is correct. The second part is plausible because Operations Manager provides the initial approval.

   Incorrect because SAP-148 requires GMNPO approval to have a lead lifted for more than 90 days

K/A: G2.2.5 Equipment Control Knowledge of the process for making design or operating changes to the facility

K/A Match: The K/A is met because the candidate must recall time limits and approvals for an inoperability resulting in a change to the facility.

Selection criteria; **MODIFIED FROM ADMIN148**

Tier: 3 **Group:**
Importance Rating: RO 2.2 SRO 3.2
Technical Reference: **SAP-148**

Proposed references to be provided to candidates during examination: None

Learning Objective: SAP-148 04. DESCRIBE the review process of a Bypass Authorization Request. 05.DESCRIBE the Implementation Process of SAP-148

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content: 43(b)(3)

SRO Justification: SRO Only because the question tests knowledge of Facility Licensee Procedures Required To Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)] including administrative processes for temporary modifications.

NRC Form ES-401-9 Comments:

Facility Response:

comments;

Mods and revs

ADMIN PROCEDURE 148

Given the following plant conditions:

Initial Conditions:

- A charge of "A" Battery, "XBA1A," is in progress.
- XFN-38A **and** XFN-39A SPLY & EXH FANS are in service.

Final Conditions:

- XFN-38A **and** XFN-39A were found tripped.
- I&C has determined that a faulty control circuit input has caused the problem.
- Technicians advise that lifting an electrical lead will allow operation of XFN-38A/XFN-39A SPLY & EXH FANS.
- The lead will be lifted using the provisions of SAP-148, TEMPORARY BYPASS, JUMPER, AND LIFTED LEAD CONTROL.

Which ONE of the choices below answers both of the following:

1) What is the ventilation alignment that will allow the charge of "XBA1A" to continue?

2) If the lead is lifted on the XFN-38A/XFN-39A control circuit, what is the longest that it can remain in that condition without GMNPO approval?

A. The charge may continue by placing XFN-38B/XFN-39B, SPLY & EXH FANS in service; The lead must be restored to normal within 14 days.

B. The charge may continue **only** if the control circuit lead is lifted and XFN-38A/ XFN-39A are restarted; The lead must be restored to normal within 14 days.

C.✓ The charge may continue by placing XFN-38B/XFN-39B, SPLY & EXH FANS in service; The lead must be restored to normal within 90 days.

D. The charge may continue **only** if the control circuit lead is lifted and XFN-38A/ XFN-39A are restarted; The lead must be restored to normal within 90 days.

98. Given the following plant conditions:

Time 0400:

- A LOCA with a leak outside of the RB occurred.

Time 0430

- The OSC, TSC and EOF are fully manned and activated.
- The Shift Manager has been relieved of duties as the Interim Emergency Director.
- An action by an AO is required to operate plant equipment.
  - It is **not** required to save a valuable equipment in the Auxiliary Building.
  - The action is **not** vital for the protection of the public.
- The AO volunteer has received 500 mrem of exposure in the current year.
- Emergency exposure limits in accordance with EPP-020, EMERGENCY PERSONNEL EXPOSURE CONTROL will be used for this exposure.

Which ONE of the choices below answers both of the following questions in accordance with EPP-020?

1) What is the **highest** TEDE dose the AO can receive for the exposure described above?
2) Can the Shift Manager approve the use of the planned emergency exposure limits at time **0430**?

A. 1) 4.5 REM.
   2) No.

B. 1) 4.5 REM.
   2) Yes.

C.✓ 1) 5 REM.
   2) No.

D. 1) 5 REM.
   2) Yes.

Modified epp284 for 2018 NRC - rj

QUESTION USAGE:

ILO 13-01 Audit (2015)
Modified from epps27 for 2015 Audit. RJ 4/26/14

REVISION HISTORY:

Rev. 0 Submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:

A. Plausible. 4.5 REM is the 5 REM limit to operate plant equipment in an emergency minus the 0.5 rem of current exposure (current exposure is not considered for emergency dose limits). The second half is correct.

Incorrect because the limit is not 4,5 REM.

B. Plausible. 4.5 REM is the 5 REM limit to operate plant equipment in an emergency minus the 0.5 rem of current exposure (current exposure is not considered for emergency dose limits). If the Shift Supervisor were acting as the IED, he could authorize the exposure but in this case has been relieved by the ED.

Incorrect because the limit is not 4,5 REM and the SM cannot approve planned exposure limits after being relieved of IED duties.

C. CORRECT. 5 REM limit is the correct limit in accordance with EPP-020. If the Shift Supervisor were acting as the IED, he could authorize the exposure but in this case has been relieved by the ED and cannot give the authorization.

D. The first part is correct. The second part is plausible because if the Shift Supervisor were acting as the IED, he could authorize the exposure.

Incorrect because the SM cannot approve planned exposure limits after being relieved of IED duties.

K/A: G2.3.4 Radiation Control Knowledge of radiation exposure limits under normal and emergency conditions

K/A Match: The KA is met because the candidate must determine the exposure limit for a worker during an emergency event.

Selection criteria; **MODIFIED FROM EPPS284**

Tier: 3 **Group:**
Importance Rating: RO 3.2 SRO 3.7
Technical Reference: **VCS-EPP-20**

Proposed references to be provided to candidates during examination: None

Learning Objective: EPP-20, 02.STATE the maximum exposure limits for company personnel

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis _____

10 CFR Part 55 Content:

SRO Justification: SRO because the question requires knowledge “unique to the SRO position” as documented within SAT process as ties the knowledge/ability to the licensee’s SRO job position duties.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

EPPS/FEPS 284

Given the following plant conditions:

Time 0400:

- A LOCA with a leak outside of the RB occurred.

Time 0600

- The OSC, TSC and EOF are fully manned and activated.
- An action by an AO can be used to save a valuable piece of equipment in the Auxiliary Building.
- The action is **not** vital for the protection of the public.
- The AO volunteer has received 1000 mrem of exposure in the current year.
- Emergency exposure limits in accordance with EPP-020, EMERGENCY PERSONNEL EXPOSURE CONTROL will be used for this exposure.

Which ONE of the choices below answers both of the following questions?

What is the **highest** TEDE dose the AO can receive for the exposure described above?

What position provides the **minimum** level of approval for this use of emergency dose limits, in accordance with EPP-020?

A. 9 REM; Shift Supervisor.

B. 9 REM; Emergency Director.

C. 10 REM; Shift Supervisor.

D.✓ 10 REM; Emergency Director.

99. Given the following plant conditions:

- 100% power
- Various personnel have called in sick due to a flu epidemic.
- The on-coming Shift Manager is reviewing the EP/FEP Manning Sheet and the personnel available to assume the shift.
- The on-coming shift CRS, NROATC and BOP are performing shift relief with the off-going operators.
- There are four AOs that have reported for the on-coming shift.
  - **All** four are fully qualified AOs.
  - **One** is a qualified Fire Brigade Operations Advisor.
  - **One** is a qualified Fire Brigade member.
  - The **other two** AOs are qualified for their watchstations **only**.
- There are no additional personnel qualified Fire Brigade member or Fire Brigade Operations Advisor in the on-coming shift.

Which ONE of the following describes what additional action must be taken, if any, to ensure shift manning is met in accordance with OAP-100.6, CONTROL ROOM CONDUCT AND CONTROL OF SHIFT ACTIVITIES?

Assume that there are will be no impact due to Fatigue rules.

A. **No** additional actions are required.

B. The shift manager must fill a vacant AO watch position within a **maximum** of two hours.

C. The shift manager must fill a vacant Fire Brigade position within a **maximum** of two hours.

D✓ The shift manager must hold a Fire Brigade member qualified AO over from the previous shift.

New for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review: Danny Rhymer
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question correctly, the candidate must determine personnel requirements for the fire brigade given a loss of shift personnel.

A. Plausible because Technical Specification manning requires 2 AO and that are 2 fully qualified AOs and 2 watchstation qualified AOs available.

   Incorrect because OAP-100.6 requires 5 AOs to fully man the shift.

B. Plausible because an additional AO is required for full manning and because Technical Specifications allow 2 hours for unexpected absences. Additionally, the candidate may believe that the Fire Brigade Operations Advisor can also fill a Fire Brigade member position.

   Incorrect because the 2 hour allowance does not allow the position to go unfilled during shift change.

C. Plausible because there are two more AOs than technical specifications require, because an additional Fire Brigade Member is required for full manning, and because Technical Specifications allow 2 hours for unexpected absences.

   Incorrect because a fire brigade member qualified AO is still needed and because the 2 hour allowance does not allow the position to go unfilled during shift change.

D. CORRECT. An Additional AO is required to complete the 5 required and additional Fire Brigade member is also required. The position cannot go unfilled because of shift change.

K/A: G2.4.26 Emergency Procedures/Plans Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage.

K/A Match: The KA is matched because the candidate must determine personnel requirements for the fire brigade given a loss of shift personnel.

Selection criteria; NEW

Tier: 3 **Group:**
Importance Rating: RO 3.1 SRO 3.6
Technical Reference: **OAP-100.6**

Proposed references to be provided to candidates during examination: None

Learning Objective: OAP-100.6 011. DISCUSS shift relief expectations as described in OAP-100.6.

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis __X__

10 CFR Part 55 Content:

SRO Justification: SRO because the question requires knowledge "unique to the SRO position" as documented within SAT process as ties the knowledge/ability to the licensee's SRO job position duties.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

100. Given the following plant conditions:

Time 1200:

- 100% power.
- Anti-nuclear protesters were gathered at the first guard post off of route 213.
- Two of the protesters ran past the guard post.

Time 1230.

- The Security Team Leader reported the following:
  - One of the protesters was intercepted just prior to reaching the rock barriers.
  - This person threatened the apprehending officer with a knife.
  - The apprehending officer immediately disarmed and detained this person.

Time 1400.

- The Security Team Leader reported the following:
  - An intruder was intercepted at the base of an electrical tower in the field between the rock barrier and the switchyard fence.
    - This person was attempting to place a bomb and a detonator at the tower base and attempted to choke the security guard before he was subdued.
  - There are **no** additional security challenges.

Which ONE of the choices completed the following statements in accordance with EPP-001, ACTIVATION AND IMPLEMENTATION OF THE EMERGENCY PLAN?

The **highest** Emergency Action Level declaration was an __(1)__ and conditions were **first** met for the highest declaration at __(2)__.

Do **not** consider Emergency Director Judgment as a basis for emergency classification

REFERENCE PROVIDED

A. 1) ALERT
   2) 1230.

B✓ 1) ALERT
   2) 1400.

C. 1) SITE AREA EMERGENCY
   2) 1230.

D. 1) SITE AREA EMERGENCY
   2) 1400.

PROVIDE VCS-EPP-0001, ATTACHMENT I, EAL CLASSIFICATION MATRICES - UNIT

Modified epps87 for 2018 NRC - rj

QUESTION USAGE:

REVISION HISTORY:
Rev. 0 submitted by RJ
Ops Review:
Approved:

DISTRACTOR ANALYSIS:
In order to answer this question, the candidate must determine an EAL declaration during a attempted sabotage hostile action.

A. The first part is plausible when considered with the second part. A knife-brandishing assailant intruded past the guarded vehicular checkpoint and committed a hostile action (see B below). If the candidate does not know what constitutes the protected area this is plausible.

   Incorrect because an Alert declaration is not required at 1230.

B. CORRECT. At 1400 the assailant intruded past the rock barrier which is the border for the Owner Controlled area at 1400. At that time and met and attempted to plant explosives. The definition is "An act toward a nuclear power plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidates the licensee to achieve an end."

C. Plausible because an SAE would be warranted if the assailant were to enter the protected area and the assailant intruded past a guarded vehicular checkpoint at 1230 and met the procedural definition of HOSTILE ACTION. If the candidate does not know what constitutes the protected area this is plausible.

   Incorrect because an SAE is not warranted because the assailant did not enter the protected area.

D. Plausible because an SAE would be warranted if the assailant were to enter the protected area and the assailant intruded into the Owner Controlled area at 1400 and met the procedural definition of HOSTILE ACTION. If the candidate does not know what constitutes the protected area this is plausible.

   Incorrect because an SAE is not warranted because the assailant did not enter the protected area.

K/A: G2.4.28 Emergency Procedures/Plans Knowledge of procedures relating to emergency response to sabotage.

K/A Match: the KA is matched because it requires the candidate to determine an EAL declaration during a attempted sabotage hostile action.

Selection criteria; **MODIFIED FROM EPPS87**

Tier: 3 **Group:**
Importance Rating: RO 3.2 SRO 4.1
Technical Reference: **VCS-EPP-0001**

Proposed references to be provided to candidates during examination:

VCS-EPP-0001, ATTACHMENT I, EAL CLASSIFICATION MATRICES - UNIT 1

Learning Objective: EPP-001 01 For a given set of conditions, determine which Emergency Action Level applies.

Question Cognitive Level: Memory or Fundamental Knowledge _____
Comprehension or Analysis __X__

10 CFR Part 55 Content: 43(b)(5)

SRO Justification: SRO because the question requires knowledge “unique to the SRO position” as documented within SAT process as ties the knowledge/ability to the licensee’s SRO job position duties.

NRC Form ES-401-9 Comments:

Facility Response:

Comments;

EPPS/FEPS 087

Given the following plant conditions:

Time 1203:

- 100% power.
- The plant has been in Mode 1 for the last seven months.
- Anti-nuclear protesters were gathered at the guard post.
- One person was seen running past the gate carrying a pistol and backpack and climbing over the rock barrier.

Time 1205.

- Security observed the same person jump over the fence and run past Warehouse C.

Time 1216.

- A loud explosion was heard in the Control Room.

Time 1228.

- Spent Fuel Pool level LI-7431/ 7433 are at bottom of the scale at 447 feet.

Time 1235.

- RM-A13, MAIN PLANT VENT EXHAUST reads 15 mr/hr, rising.

Which ONE of the choices contains the times at which conditions were **first** met for declaration of Site Area Emergency and General Emergency?

Do **not** consider Emergency Director Judgment as a basis for emergency classification

REFERENCE PROVIDED

	SAE	GE
A.	1203	1228
B.	1203	1235
C.	1205	1228
D.✓	1205	1235

Facility: VC SUMMER	Scenario No: 1		Op Test No: NRC-ILO-15-01

Examiners: ____________ ____________ ____________

Operators: CRS: ____________ RO: ____________ BOP: ____________

Initial Conditions:

- The Reactor power is 75%.
- "B" train work week.
- "B" MDEFW pump is OOS.
- XFN-0065B RBCU is OOS.

Turnover:

- Lower power to 65% to take "A" Main Feed Pump out of service. The pump will be evaluated by Engineering.

Critical Tasks:

- Swap controlling steam flow channels without a reactor trip on SG level.
- Swap controlling Pressurizer Level channels without a reactor trip on Pressurizer Level.
- Insert control rods or emergency borate prior to the reactor being locally tripped.

Event No.	Malf No.	Event Type*	Event Description
1	N/A	N-BOP, CRS R-RO	Lower power to 65% IAW GOP-4B, Power Operation (Mode 1 - Descending).
2#	MAL-PRS002A	I-RO, CRS TS-CRS	LT-459 fails low, causing Letdown to isolate.
3	MAL-CRF004F10	C-RO, CRS TS-CRS	Dropped Rod, F-10.
4	XMT-MS003O	I-BOP, CRS TS-CRS	FT-484 (STM FLOW) fails LOW causing "B" SG level to lower.

5	XMT-FW017O	I-BOP, CRS	PT-508 (FW PP DISCH HDR PRESS) fails LOW causing Main Feed Pumps to speed up.
6	MAL-PCS009AB MAL-PCS009BB	M-ALL	ATWS
7	MAL-RCS006A	M-ALL	SBLOCA
	PMP-CC001T PMP-CC003F		"A" CCW pump trips. "C" Fails to AUTO start.
	PMP-CC002F		"B" CCW pump fails to start in AUTO.
	VLV-SP006F		MVG-3003B, Spray header isolation valve, fails to open
	PMP-CS006S		"B" Charging Pump sheared shaft.
TERMINATE: The scenario may be terminated once the crew has initiated a cooldown in EOP-2.1, ES-1.2 Post LOCA Cooldown and Depressurization or at the Examiners discretion.			
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

Used on previous two NRC Exams. Event 2 was used on the 2017 NRC Exam.

The following notation is used in the ES-D-2 form "Time" column:

IOA designates Immediate Operator Action steps.

***** designates Continuous Action steps.

<u>TURNOVER:</u>

The crew will assume the watch having been pre-briefed on the Initial Conditions, the plan for this shift and any related operating procedures. The "B" Motor Driven EFW pump will be inoperable for scheduled preventive maintenance. Tech Spec 3.7.1.2, Emergency Feedwater System action a (restore "B" pump within 72 hours) has been in effect for 6 hours with pump return to service is expected 6 hours from now. Train "B" RBCU, XFN-0065B is tagged out for breaker maintenance and is to be returned to service in 10 hours. The Crew will be instructed to lower power to 65% power to secure the "C" Main Feedwater pump.

- **PRE-LOAD**

  - OVR-AH022A
    CS-AH280 RBCU FAN 65B FAST SPEED GREEN L
    FINAL = OFF

  - OVR-AH023A
    CS-AH279 RBCU FAN 65B SLOW SPEED GREEN L
    FINAL = OFF

  - OVR-EF010A
    CS-EF02 MOTOR DRIVEN EMERG FW PP B(XPP-
    FINAL = OFF

EVENT 1: Lower Reactor Power.

The crew will be prepared to commence the power reduction following a panel walk down and short briefing on the power reduction. The turnover stated that the “A” Feedwater pump is to be removed from service to allow an inspection that was not performed at the last shutdown. There is no concern that the pump is in imminent danger of failure. The CRS will direct the power reduction using GOP-4B, POWER OPERATION (MODE 1-DESCENDING). The RO will borate in accordance with SOP-106 and monitor Control Rod operation. The BOP will decrease turbine load at the rate directed by the CRS (1/2% per minute).

EVENT 2: LT-459 fails low, Letdown isolates.

- **TRIGGER 2**

    - MAL-PRS002A
      PRESSURIZER LEVEL CHANNEL 459 FAILURE
      FINAL = 0

On cue from the Examiner, LT-459 will fail low. The RO will swap controlling channels to the two operable channels, LT-460 and LT-461. The CRS will enter AOP-401.6, Pressurizer Level Control and Protection Channel Failure. The RO will restore Letdown using Attachment 4 of AOP-401.6. The CRS will refer to Tech Specs 3.4.3.1, Reactor Trip System Instrumentation, and enter action statement 6 and must place the inoperable channel in a tripped condition within 72 hours.

It took 13 minutes with no operator action to reach the high Pressurizer Level reactor trip of 92%.

EVENT 3: Dropped Rod, F-10.

- **TRIGGER 3**

    - MAL-CRF004F10
      DROPPED ROD F10
      FINAL VALUE = STATIONARY
      DELETE IN = 25 sec

On cue from the Examiner Control Bank D rod F-10 will fully insert into the core. The CRS will implement AOP-403.6, DROPPED CONTROL ROD. When contacted I&C will inform the CRS that the rod was dropped due to a faulty fuse. The CRS determines that TS 3.1.3.1.d is the only action statement that will be entered. TS 3.1.3.1: All full length (shutdown and control) rods which are inserted in the core shall be OPERABLE and positioned within 12 steps (indicated position) of their group step-counter demand position. d. With one full length rod inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either: 1. The rod is restored to OPERABLE status within the above alignment requirements. Following replacement of the faulty fuse by I&C, the rod is recovered to its original position.

EVENT 4: FT-484 Fails Low, causing "B" SG level to lower.

- **TRIGGER 4**

    - XMT-MS003O
      SG B STEAM FLOW FAIL TO POSN
      Ramp = 30 sec
      Final Value = 0

On cue from the Examiner, Steam Flow transmitter FT-484 will fail low. "B" SG level will start to lower. The BOP will select the operable steam flow and feed flow channels, FT-485 and FT-486 in accordance with AOP-401.3. The operator may take manual control of the feed regulating valves to control SG level if necessary. Once SG level has stabilized, the CRS will refer to Tech Spec 3.4.3.1 Item 14, action 6 and Tech Spec 3.4.3.2, Item 4d, action 24. Both of these action statements require the inoperable channel to be placed in the tripped condition within 72 hours.

It took two minutes and six seconds to reach 40% SG level. At this point, the crew would be directed to trip the reactor based on Reference page criteria in AOP-401.3 and XCP-624, 3-6, SG LVL HI/LO LIMIT.

EVENT 5: PT-508 fails low, causing Main Feedwater Pumps to speed up.

- **TRIGGER 5**

  - XMT-FW017O
    IPT00508 FW PP DSCHG HDR PRESS PI-508 FAIL TO POSN
    FINAL VALUE = 200 psig
    RAMP = 30 sec

On cue from the Examiner, PT-508 will fail low, causing main feedwater pumps to speed up. The BOP will take manual control of the Main Feedwater Pump Master Controller to manual and lower main feedwater pump speed. The crew will enter AOP-210.3, Feedwater Pump Malfunction. They will restore main feedwater pump D/P to the proper program band in manual.

EVENT 6/7: ATWS followed by a Small Break LOCA including failures after the break.

- **PRE-LOAD,**

  - VLV-SP006F (MVG-3003B fails to open)
    XVG0300B-SP SPR HDRS ISO CIRCUIT B FAIL AS IS

  - PMP-CC003F
    XPP0001CAL CCW PMP C TRAIN A FAIL TO START

  - PMP-CC002F
    CCW PMP B FAIL TO START.

  - PMP-CS006S
    XPP0043B CHRG/SI PMP B SHEARED SHAFT

  - MAL-PCS009AB
    REACTOR TRIP BREAKER A FAILURE (FAIL TO OPEN)
    FAIL TO: BOTH

  - MAL-PCS009BB
    REACTOR TRIP BREAKER B FAILURE (FAIL TO OPEN)
    FAIL TO: BOTH

- **TRIGGER 6**

  - MAL-TUR001
    Inadvertent turbine trip
    DELAY = 5 sec (Delay is to ensure that BST-RC039 goes in before the turbine trips.)

  - BST-RC039
    ISB00408C1 AUTO RODS IN
    FINAL = FAIL AS IS

- **TRIGGER 7**
  - MAL-RCS006A
    REACTOR COOLANT SYSTEM LEAK COLD LEG (LOOP 1)
    FINAL VALUE = 2500 gpm
- **TRIGGER 12**
  - MAL-PCS009AA
    REACTOR TRIP BREAKER A FAILURE (INADVERTENT OPEN)
- **TRIGGER 13**
  - MAL-PCS009BA
    REACTOR TRIP BREAKER B FAILURE (INADVERTENT OPEN)
- **TRIGGER 14,** X02I102O==1 (Deletes MVG-3003B failure when valve is opened)
  - VLV-SP006F (NEW)
    DELETE IN = 1 sec
- **TRIGGER 15,** X06O013A==1 (“A” CCW Pump trips when “A” RX Trip Breaker opens)
  - PMP-CC001T
    XPP0001AL CCW PMP A TRIP ON COMMAND

On cue from the Examiner, a turbine trip will occur leading to an ATWS. The crew will enter into EOP-13.0. The crew will have to insert control rods or emergency borate prior to the reactor being locally tripped. We will trip the reactor three minutes after being sent to locally trip the reactor. They will then exit out of EOP-13.0. At this point we will insert a Small Break LOCA. They will now enter EOP-1.0 (E-0), Reactor Trip/Safety Injection Actuation. Several failures will occur on the trip. “A” CCW pump will trip, “B” CCW pump will not automatically start on the sequencer. MVG-3003B, “B” Train spray header isolation valve will not open automatically when containment pressure reaches 12 psig. The “B” Charging pump will experience a sheared shaft and “C” Charging pump breaker cannot be racked up. The crew will transition from EOP-1.0 to EOP-2.0, Loss of Reactor or Secondary Coolant to EOP-2.1, ES-1.2 Post LOCA Cooldown and Depressurization.

The Critical Task of inserting control rods or emergency borating will need to be performed within three minutes of the crew telling an operator to locally trip the reactor. Once the AO is informed to trip the reactor locally, we will trip the reactor at three minutes.

CRITICAL TASKS:

- Swap controlling steam flow channels without a reactor trip on SG level.
- Swap controlling Pressurizer Level channels without a reactor trip on Pressurizer Level.
- Insert control rods or emergency borate prior to the reactor being locally tripped.

TERMINATION:

The scenario may be terminated once the crew has transitioned from EOP-2.0, E-1 Loss of Reactor or Secondary Coolant, to EOP-2.1, ES-1.2 Post LOCA Cooldown and Depressurization, or at the Examiners discretion.

Scenario Attributes		Events
Total Malfunctions (5-8)	10	· LT-459 fails LOW. · Dropped Rod, F-10. · FT-484 (STM FLOW) fails LOW. · PT-508 (FW PP DISCH HDR PRESS) fails LOW. · Small Break LOCA. · “A” CCW pump trips, “C” CCW pump fails to AUTO start. · “B” CCW pump fails to start in AUTO. · MVG-3003B fails to open. · “B” Charging pump sheared shaft. · ATWS
Malfunctions after EOP entry (1-2)	4	· “A” CCW pump trips, “C” CCW pump fails to AUTO start. · “B” CCW pump fails to start in AUTO. · MVG-3003B fails to open. · “B” Charging pump sheared shaft.
Abnormal Events (2-4)	4	· LT-459 fails low. · Dropped Rod, F-10. · FT-484 (STM FLOW) fails LOW. · PT-508 (FW PP DISCH HDR PRESS) fails LOW.
Major Transients (1-2)	2	· Small Break LOCA. · ATWS
EOPs Entered (1-2)	2	· EOP-2.0, E-1 Loss of Reactor or Secondary Coolant. · EOP-13.0, FR-S.1 Response to Abnormal Nuclear Power Generation.
EOP Contingencies (0-2)	1	· EOP-13.0, FR-S.1 Response to Abnormal Nuclear Power Generation.
Critical Tasks (≥ 2)	3	· Swap controlling steam flow channels without a reactor trip on SG level. · Swap controlling Pressurizer Level channels without a reactor trip on Pressurizer Level. · Insert control rods or emergency borate prior to the reactor being locally tripped.

SIMULATOR SCENARIO SETUP

INITIAL CONDITIONS:

- IC Set 300
- 75% Power, MOL
- Burnup = 10,033 MWD/MTU
- RCS Boron Concentration = 1076 ppm
- FCV-113 Pot Setting = 4.53
- Rod Position: Group D = 187
- Tavg = 579.8˚F
- Xe = -0.0 pcm
- Prior to the scenario, the crew should pre-brief conditions and their expectations for the shift.

PRE-EXERCISE:

- Ensure simulator has been checked for hardware problems (DORT, burnt out light bulbs, switch malfunctions, chart recorders, etc.).
- Complete VCS-TQP-0807 Attachment I-A, Unit 1 Booth Instructor Checklist.
- Verify plant aligned for "B1" work week IAW PTP-101.004, Safety Related Train Swap Checklist.
- Verify red hold tag and R&R tag on "B" MDEFW Pump **AND** XFN-0065B RBCU and ensure they are in P-T-L. XFN-65B can’t be taken to P-T-L.
- Verify red Placard on “A” CCW Pump and “B” Charging Pump.
- Verify the Hard Card for Turbine operation is in its proper storage location and cleaned.
- Verify the Hard Card for borating via MVT-8104 is in its proper storage location and cleaned.
- Update EOOS for “B” MDEFW Pump being out of service.
- Verify Rod Bank Update set correctly: 187 steps on Control Bank D and 228 steps on all other Banks.
- Ensure you have the following pre-marked up procedures:
    - GOP-4B, Power Operation (Mode 1 - Descending)
- Ensure you have a turnover sheet for each position.
- Conduct two-minute drill.

Op Test No:	NRC-ILO-16-01	Scenario #	1	Event #	1	Page:	10	of	56

Event Description: Lower Reactor Power

Time	Position	Applicant's Actions or Behavior
BOOTH OPERATOR:		No TRIGGER for this event.
Indications Available: N/A		
EVALUATOR NOTE: The crew will have briefed a power reduction to 65% at 1/2% per minute prior to assuming the watch. Procedure guidance for borating as found in SOP-106, Reactor Makeup Water System is included in this scenario guide beginning at page 53 of 55.		
	CRS	Enters GOP-4B, POWER OPERATION (MODE 1 - DESCENDING)

GOP-4B

GOP- 4B REFERENCE PAGE

GENERAL NOTES

A. Procedure steps should normally be performed in sequence. However, it is acceptable to perform steps in advance after thorough evaluation of plant conditions and impact by the Shift Supervisor or Control Room Supervisor.

B. Axial Flux Difference, ΔI, should be maintained within limits per V.C. Summer Curve Book, Figure I-4.1 during Reactor Power Operation above 50% per Tech Spec 3.2.1.

C. After any Thermal Power change of greater than 15% within any one hour, Attachment III.H. of GTP-702 must be completed.

D. If time allows, all load changes should be discussed with the Load Dispatcher prior to commencing the load change.

E. If Reactor Power is stabilized during this procedure for the purpose of raising power per GOP-4A, a Power Range Heat Balance shall be performed.

REACTOR CONTROL

A. During operation with a positive Moderator Temperature Coefficient:

1) Power and temperature changes should be slow and will require constant operator attention.

2) All power and load changes should be performed in small increments.

B. Rod Control should be maintained in Automatic if any Pressurizer PORV is isolated.

C02→ C. If at any time, power decreases unexpectedly below 0.1% on any Power Range NI (computer indication available) OR below 1.0% on any Power Range NI control board indication (computer not available): CHG B

1) No positive reactivity will be added by rods or dilution.

2) A complete reactor shutdown shall be performed per GOP-5.

3) A controlled reactor startup may be commenced per GOP-3 once the event has been reviewed by Reactor Engineering.

TURBINE CONTROL

A. If during power descension plant stabilization is required, HOLD should be selected on the EHC HMI: Control/Load screen. CHG E

B. To resume power descension select the recommended Load Ramp Rate

C. Turbine Load values are approximate and provided as initial starting points for load changes. When desired Reactor or Turbine parameters are achieved stabilize (if necessary) and proceed as directed. CHG A

D. The load limit "ramp rate" buttons only affect how fast the Load Limit Ref. moves to the new Load Limit Setpoint. Load reductions made using the limiter will always occur at 30% per minute. CHG B

E. The load limiter will reduce turbine load if it is set more than 2% below the current Load Reference value. Load will only be shed until the Load Reference value is once again within 2% of Load Limit Ref.

MSR CONTROL

A. Do not exceed 50°F ΔT between the inlets to the Low Pressure Turbine.

B. When in Manual, do not exceed 25°F per half-hour temperature change rate for the tube side of the Moisture Separator/Reheater.

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 1	Page: 11 of 56
Event Description: Lower Reactor Power			

Time	Position	Applicant's Actions or Behavior	
		NOTE 3.2 a. Step 3.2 lowers Reactor Power from 90% to 48%. b. While the plant is being maneuvered, total condensate flow through the Blowdown Heat Exchangers must be maintained greater than 450 gpm, which should maintain condensate outlet temperature at least 30°F below the DA temperature.	GOP-4B
	BOP	3.2. Reduce Reactor Power to 48% as follows: a. Using the EHC HMI, Control/Load screen, reduce load per SOP-214, Main Turbine And Controls, Section III.D, Turbine Load Reduction/Shutdown, at a rate of 1% per minute or less.	GOP-4B
		3.2. NOTE 3.2.b The System Controller should be notified prior to manually changing MVARs by more than 50 MVARs in a five minute period, unless the change is needed to prevent equipment damage.	GOP-4B
	BOP	3.2. Reduce Reactor Power to 48% as follows: b. As load decreases, adjust Megavars using GEN FIELD VOLT ADJ as requested by the System Controller and within the Estimated Generator Capability curve (Enclosure A).	GOP-4B
		NOTE 3.2.c a. When securing Main Feedwater Pumps, it may be desired to perform PTP-125.020, Main Feedwater Pump Timed Trip Test. b. Due to the physical location of the start-up drain, securing Feedwater Booster Pump D last will ensure better cooling flow for the Deaerator.	GOP-4B
	BOP	3.2. Reduce Reactor Power to 48% as follows: c. When Reactor Power is between 60% and 80%, reduce to the following pumps in service: 1. Two Main Feedwater Pumps per SOP-210, Feedwater System, Section III.H, Feedwater Pump Shutdown. 2. Three Feedwater Booster Pumps per SOP-210, Feedwater System, Section III.I, Feedwater Booster Pump Shutdown.	GOP-4B

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 1	Page: 12 of 56
Event Description: Lower Reactor Power			

Time	Position	Applicant's Actions or Behavior	
	BOP	3.2. Reduce Reactor Power to 48% as follows: d. When Reactor Power is between 60% and 75% perform PTP-102.001, Main Turbine Tests (Power Operated Extraction System Check Valve portion only).	GOP-4B
NOTE 2.2 The turbine will come off the limiter and turbine load will lower once Load Set Reference is less than Load Limit Reference. Acknowledging dialog boxes is "skill of the Craft".			SOP-214
	BOP	2.2 To lower Turbine Load using Load Set, perform the following: a. If directed by Operations Management, disable the Turbine Vibration Trips per Section III. b. Select (or enter) the desired Rate %/min on Load Set. c. Select Load on Load Set (a dialog box will open). d. Enter the desired load and confirm. e. Verify proper system response. f. If during a load reduction, it is desired to stop the load reduction, perform the following: 1) Select Hold on Load Set. 2) Select the desired Rate %/min to resume load reduction. 3) If desired, place LOAD LIMIT in service per Section III.	SOP-214

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 1	Page: 13 of 56

Event Description: Lower Reactor Power

Time	Position	Applicant's Actions or Behavior
BOOTH OPERATOR:		If called to adjust Blowdown Cooler flow use the following remotes: • LOA-CND044, COND TO S/G BD TC-3062A AUTO-MANUAL MODE SELECTOR - position to MANUAL • LOA-CND045, COND TO S/G BD TC-3062B AUTO-MANUAL MODE SELECTOR - position to MANUAL • LOA-CND046, COND TO S/G BD TC-3062C AUTO-MANUAL MODE SELECTOR - position to MANUAL • LOA-CND-047, COND TO S/G BD TV-3062A MANUAL POSITION – adjust final value to obtain flow as directed. • LOA-CND-048, COND TO S/G BD TV-3062B MANUAL POSITION – adjust final value to obtain flow as directed. • LOA-CND-049, COND TO S/G BD TV-3062C MANUAL POSITION – adjust final value to obtain flow as directed.

EVALUATOR NOTE: The next event may be inserted following completion of the power reduction, or at any time per the discretion of the Lead Examiner.

Op Test No: NRC-ILO-16-01 Scenario # 1 Event # 2 Page: 14 of 56			
Event Description: LT-459 fails LOW			
Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: LT-459 will fail LOW. Charging flow will increase and actual Pressurizer Level will rise.			
BOOTH OPERATOR:		When directed - Initiate Event 2 (TRIGGER 2).	
Indication Available: XCP-616 1-3, BLCK HTRS ISOL LTDN PZR LCS LO XCP-616 1-5, PZR LCS DEV HI/LO XCP-616 3-1, PZR HTR CNTRL OR BU GRP 1/2 TRIP XCP-616 4-6, SCR OUTPT LOSS XCP-642 4-4, RC LTDN LO RNG RM-L1 TRBL			
	CRS	Enters AOP-401.6, Pressurizer Level Control and Protection Channel Failure.	
IOA **Critical Task**	RO	1. Place PZR LEVEL CNTRL Switch to the position with two operable channels.	AOP-401.6
	RO	2. Select an operable channel on PZR LEVEL RCDR.	AOP-401.6
	RO	3. Control the PZR Heaters as necessary to maintain PZR pressure: • CNTRL GRP Heaters. • BU GRP 1 Heaters • BU GRP 2 Heaters.	AOP-401.6
EVALUATOR NOTE: Energizing Pressurizer Heaters is done in accordance with SOP-101, Reactor Coolant System. This can be seen on page 55 of 55.			
	RO	4. Verify Letdown is in service. **(NO)**	AOP-401.6
	RO	**Alternative Action Step:** 4. Re-establish Normal Letdown using ATTACHMENT 4, ESTABLISHING NORMAL LETDOWN.	AOP-401.6
EVALUATOR NOTE: Attachment 4 can be seen on page 15 of 55.			
	RO	5. Check if FCV-122, CHG FLOW is in AUTO. **(NO)**	AOP-401.6
	RO	**Alternative Action Step:** 5. Place FCV-122, CHG FLOW in AUTO using ATTACHMENT 5, RESTORING AUTOMATIC CHARGING FLOW CONTROL.	AOP-401.6
EVALUATOR NOTE: Attachment 5 can be seen on page 16 of 55.			

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 2	Page: 15 of 56
Event Description: LT-459 fails LOW			

Time	Position	Applicant's Actions or Behavior	
	RO	6. Check if the PZR LVL MASTER CONTROLLER is responding appropriately: • Verify Charging flow is normal and responding to PZR level error. • Verify PZR level is stable at or trending to program level.	AOP-401.6
NOTE – Step 7			
Compliance with T.S. requires tripping failed instrument channel bistables within 72 hours of the channel failure. Time should be allowed for troubleshooting of the failed channel prior to tripping the bistables.			AOP-401.6
	CRS	7. Place the failed channel protection bistables in a tripped condition within 72 hours of the channel failure: a. Write an R&R for the failed channel. b. Select the attachment for the failed channel from the back of this procedure. c. Record the R&R number on the attachment. d. Determine the cause of the channel failure. e. Notify the I&C Department to place the identified bistables in trip using the attachment.	AOP-401.6

AOP-401.6
ATTACHMENT 1
PAGE 1 of 1
REVISION 4

LEVEL TRANSMITTER LT-459

TRIPPED BISTABLE STATUS

INSTRUMENT	ASSOCIATED BISTABLE	BISTABLE LOCATION	TRIP STATUS LIGHT
LT-459	LB-459A	C1-442-BS-1	CHAN I PZR LVL HI

R&R#____________

TRIP

APPLICABLE STPS	STP # USED TO TRIP BISTABLE	TRIP STATUS LIGHT ON	TRIPPED BY	VERIFIED BY
302.007				
345.018				
OATC LOG				

AOP-401.6

RESTORATION

APPLICABLE STP(S) COMPLETED	STP # USED TO RESTORE BISTABLE	TRIP STATUS LIGHT OFF	RESTORED BY	VERIFIED BY

REVIEWED BY:____________ / ____________
SHIFT MANAGER DATE

TECH SPECS
TABLE 3.3-1 ITEM 11

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 2	Page: 16 of 56	
Event Description: LT-459 fails LOW				

Time	Position	Applicant's Actions or Behavior	
	CRS	Enters T.S. 3.4.3.1, REACTOR TRIP SYSTEM INSTRUMENTATION. Function 11; Pressurizer Water Level—High, Action 6; With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 72 hours; and b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1.	Tech Specs
EVALUATOR NOTE: The next event may be inserted following the CRS assessment of Tech Specs, or at any time per the discretion of the Lead Examiner.			
	RO	1. Establish Normal Letdown: a. Adjust FCV-122, CHG FLOW, to obtain 70 gpm Charging flow. b. Set PCV-145, LO PRESS LTDN, to 70%. c. Open TCV-144, CC TO LTDN HX. d. Open PVT-8152, LTDN LINE ISOL. e. Place TCV-143, LTDN TO VCT OR DEMIN, in VCT position. f. Open both LCV-459 and LCV-460, LTDN LINE ISOL. g. Open desired Orifice Isolation Valve(s) to obtain 60 gpm to 120 gpm: • PVT-8149A, LTDN ORIFICE A ISOL (45 gpm). • PVT-8149B, LTDN ORIFICE B ISOL (60 gpm). • PVT-8149C, LTDN ORIFICE C ISOL (60 gpm). h. Adjust FCV-122, CHG FLOW, to maintain TI-140, REGEN HX OUT TEMP °F, between 250°F and 350°F while maintaining PZR level. i. Adjust PCV-145, LO PRESS LTDN, to maintain PI-145, LO PRESS LTDN PRESS PSIG, between 300 psig and 400 psig. j. Place PCV-145, LO PRESS LTDN, in AUTO. k. Place TCV-144, CC TO LTDN HX, in AUTO. l. WHEN Letdown temperatures are stable, place TCV-143, LTDN TO VCT OR DEMIN, in DEMIN/AUTO.	Attachment 4

Op Test No: NRC-ILO-16-01 Scenario # 1 Event # 2 Page: 17 of 56		
Event Description: LT-459 fails LOW		
Time	Position	Applicant's Actions or Behavior
		1. Place FCV-122, CHG FLOW in AUTO as follows: a. Place the following in MAN: 1. PZR LEVEL MASTER CONTROL 2. FCV-122, CHG FLOW b. Adjust FCV, CHG FLOW, in MAN to establish Pressurizer level at or near programmed level. c. Establish automatic FCV-122, CHG FLOW, control as follows: 1. Adjust FCV-122, CHG FLOW, to establish 75 gpm flow as indicated on FI-122A, CHG FLOW GPM. 2. Manually adjust the PZR LEVEL MASTER CONTROL to 50% demand. 3. Place FCV-122, CHG FLOW, in AUTO. d. Adjust PZR LEVEL MASTER CONTROL in MAN, as necessary, to maintain Pressurizer level at or near programmed level. e. When Pressurizer level is within 1% and trending toward programmed level, place PZR LEVEL MASTER CONTROL in AUTO.

Attachment 5

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 3	Page: 18 of 56	
Event Description: Dropped Rod, F-10				
Time	Position	Applicant's Actions or Behavior		
EVALUATOR NOTE: Control Bank D rod F-10 will fully insert into the core. The CRS will implement AOP-403.6, DROPPED CONTROL ROD.				
BOOTH OPERATOR:		When directed - Initiate Event 3 (TRIGGER 3).		
Indication Available: XCP-620 2-3, CMPTR NIS PR TILTS XCP-620 2-5, CMPTR ROD DEV XCP-620 1-4, PR CHAN DEV XCP-621 3-1, ONE ROD ON BOTTOM Rod Bottom light for Bank D rod F-10 is lit.				
EVALUATOR NOTE: IF at any time RCS pressure lowers to less than 2206 PSIG in Mode 1, then T.S. 3.2.5, DNB PARAMETERS must be entered. T.S. 3.2.5, DNB PARAMETERS: With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.				
	CRS	Enters AOP-403.6, Dropped Control Rod		
IOA	RO	1. Verify only one Control Rod has dropped.		AOP-403.6
IOA	RO	2. Place ROD CNTRL BANK SEL Switch in MAN.		AOP-403.6
	CREW	3. Stabilize the plant: a. Decrease Main Turbine load to maintain Tavg within 5°F of Tref. b. Verify PZR pressure is stable at OR trending to 2230 psig (2220 psig to 2250 psig). c. Verify PZR level is stable at OR trending to program level.		AOP-403.6
	RO	4. Check if Reactor power is LESS THAN 75%.		AOP-403.6
	RO	5. Initiate GTP-702, Attachments IV.A, IV.B, and IV.C. • ATTACHMENT IV.A - INOPERABLE CONTROL ROD. • ATTACHMENT IV.B - INOPERABLE ROD POSITION DEVIATION MONITOR. • ATTACHMENT IV.C - INOPERABLE ROD INSERTION LIMIT MONITOR.		AOP-403.6
	CRS	6. Notify the following plant personnel prior to moving Control Rods: • Management Duty Supervisor. • Rod Control System Engineer. • Reactor Engineering		AOP-403.6

Op Test No: NRC-ILO-16-01 Scenario # 1 Event # 3 Page: 19 of 56		
Event Description: Dropped Rod, F-10		
Time	**Position**	**Applicant's Actions or Behavior**
	CRS	7. Provide Reactor Engineering with the following information: • Time rod dropped: ________. • Dropped rod location: ________. • Initial Reactor power level: ________. • Current Reactor power level: ________. • Current QPTR: ________.
	CRS	8. Determine and correct the cause of the failure.
NOTE - Step 9 This Step must be completed before continuing with Step 10.		
BOOTH OPERATOR:		NOTE: No action is necessary to reset the dropped rod. When contacted as I&C: - Acknowledge request for support. - WAIT 5 minutes and report that a Stationary Gripper fuse is blown. Request permission to replace the blown fuse. - WAIT 1 minute and notify the CRS that the Stationary Gripper Fuse has been replaced. - If called to get permission from the SM, report back “replace the blown fuse”.

AOP-403.6 (Step 7)
AOP-403.6 (Step 8)
AOP-403.6 (Note - Step 9)
AOP-403.6 (Booth Operator)

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 3	Page: 20 of 56
Event Description: Dropped Rod, F-10			

Time	Position	Applicant's Actions or Behavior	
	CRS	Enters Tech Spec 3.4.1.3, Moveable Control Assemblies, Action d. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either: 1. The rod is restored to OPERABLE status within the above alignment requirements, or 2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT (COLR); the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation. or 3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that: a. A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions. b. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once Per 12 hours. c. A core power distribution measurement is obtained and Fq(z) and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and d. The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following ·4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. Enters Tech Spec 3.4.2.4, Quadrant Power Tilt Ratio, Action a. a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than *or* equal to 1.09: 1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either: a) The QUADRANT POWER TILT RATIO is reduced to within its limit. or b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER	Tech Specs

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 3	Page: 21 of 56
Event Description: Dropped Rod, F-10			

Time	Position	Applicant's Actions or Behavior	
		Tech Spec 3.4.2.4, Quadrant Power Tilt Ratio, Action a. Continued 2. Within 2 hours either: a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours. 3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours. 4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.	
	CRS	9. Obtain the following information from Reactor Engineering: • Power level at which recovery is to be performed: ________. • Rate of Control Rod movement during recovery: ________.	AOP-403.6
BOOTH OPERATOR:		When contacted as Rx Engineering for this information: - WAIT 2 minutes and notify the CRS "Maintain current power while the rod is being withdrawn. There is no speed limitation on the rate of rod withdrawal during the recovery,"	
	CRS	10. If necessary, reduce Reactor Power to the power level determined in Step 9. REFER TO GOP-4B, POWER OPERATION (MODE 1 - DESCENDING) OR GOP-4C, RAPID POWER REDUCTION.	AOP-403.6

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 3	Page: 22 of 56
Event Description: Dropped Rod, F-10			

Time	Position	Applicant's Actions or Behavior	
		NOTE - Steps 11 through 15 Throughout the following steps, "AFFECTED" refers to any Control Rod Bank which contains a dropped Control Rod.	AOP-403.6
	CRS	11. Record the AFFECTED Bank readings: a. Group Step Counter demands: • AFFECTED Bank: __________. • Group 1 reading: __________. • Group 2 reading: __________. b. Dispatch an operator with Key #91, Rod Control Cabinets, to the Rod Control Cabinet room (IB-463). c. Locally at XCA4-CR, P/A CONVERTER CABINET (IB-463), record the P/A CONVERTER reading for the AFFECTED Bank:	AOP-403.6
BOOTH OPERATOR:		When contacted as AO: - Notify the Control Room that you have Key #91 - Wait 2 minutes and report rod position indicates (the value noted on the MCB by the RO) steps.	AOP-403.6
	RO	12. Rotate ROD CNTRL BANK SEL Switch clockwise to the AFFECTED Bank position.	AOP-403.6

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 3	Page: 23 of 56

Event Description: Dropped Rod, F-10

Time	Position	Applicant's Actions or Behavior	
	RO	13. Withdraw the dropped Control Rod: a. Reset the Step Counter for the AFFECTED Group to zero. b. At the CONTROL ROD DISCONNECT SWITCH BOX inside the MCB, place all Lift Coil Disconnect Switches for the AFFECTED Bank, except the switch for the dropped Control Rod, to the ROD DISCONNECTED position. (KEY #10) **NOTE - Step 13.c** **ROD CNTRL SYS FAIL URGENT (XCP‑620 5‑1), annunciator will alarm when the dropped rod is moved in this step.** c. Move the dropped Control Rod at least six steps out. d. Verify dropped rod movement on the associated Digital Rod Position Indicator. e. Verify ONE ROD ON BOTTOM (XCP‑621 3‑1), annunciator clears. f. Adjust Main Turbine load to maintain Tavg within 5°F of Tref. g. Using the rate of Control Rod movement determined in Step 9, continue withdrawal of the dropped rod until the demand position recorded in Step 11.a is reached. h. Verify DRPI indicates the dropped rod at the same position as the other Control Rods within the bank.	AOP-403.6
	CRS	14. Locally at XCA4‑CR, P/A CONVERTER CABINET (IB‑463), reset the P/A CONVERTER as follows: a. Ensure the Bank Position Display Switch is in the AFFECTED Bank position. b. Place MANUAL/AUTOMATIC Switch in MANUAL. c. Depress the DOWN Pushbutton to reset the P/A CONVERTER to the reading recorded in Step 11.c. d. Place MANUAL/AUTOMATIC Switch in AUTOMATIC. e. Place the Bank Position Display Switch in DISPLAY OFF.	AOP-403.6

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 3	Page: 24 of 56

Event Description: Dropped Rod, F-10

Time	Position	Applicant's Actions or Behavior	
BOOTH OPERATOR:		When Contacted as field operator to reset the P/A Converter acknowledge request and perform the following: Insert: LOA-CRF005 P/A MAN HEIGHT VALUE **(USE BEFORE SETTING LOA CRF1) -** insert value previously recorded. Insert: LOA-CRF001 P/A MAN BANK SELECT **(USE AFTER SETTING LOA CRF5)** CB "D" - select Control Bank "D". Notify the Control Room that the P/A Converter is reset.	
	RO	15. Restore the Rod Control System to normal alignment: a. Place all Lift Coil Disconnect Switches for the AFFECTED Bank to the ROD CONNECTED position. b. Rotate ROD CNTRL BANK SEL Switch counter −clockwise to MAN. c. Depress the ROD CNTRL ALARM RESET Pushbutton. d. Verify the ROD CNTRL SYS FAIL URGENT (XCP-620 5-1), annunciator clears. e. Update the control rod bank positions per OAP-107.1, CONTROL OF IPCS FUNCTIONS. f. Notify the I&C Department to perform ICP-500.023, ROD CONTROL TROUBLESHOOTING AND REPAIR, to verify proper Master Cycler setup prior to moving rods. g. COMPLETE STP-106.001, MOVEABLE ROD INSERTION TEST.	AOP-403.6
BOOTH OPERATOR:		When contacted as I&C to perform ICP-500.023, ROD CONTROL TROUBLSHOOTING AND REPAIR report that the Cycler Setup has been completed satisfactorily.	

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 3 Page: 25 of 56
Event Description: Dropped Rod, F-10		

Time	Position	Applicant's Actions or Behavior
EVALUATOR NOTE: If necessary, the crew will use the procedure steps below to update the control rod bank position on IPCS.		
NOTE 6.2.b Running the Rod Bank Update function will correct the computer and reinstate the normal Rod Deviation Alarm functions.		
	CRS	b. Update the control rod bank positions as follows: 1. Activate RBU. 2. Obtain the correct Group 1 step counter positions from the RO. 3. Verify control rod step counts are correct, and select F3 (bottom left of the display page) to save the data. a. IF no changes were made to the step counts, proceed to 6.2.b 6). 4. The following Main Control Board annunciators should clear after about one minute: a. CMPTR ROD DEV (XCP-620 2-5). b. CMPTR ROD SEQ (XCP-620 2-6). 5. If only one annunciator clears, select F3 again. 6. When both annunciators are clear, press ESC.
EVALUATOR NOTE: The next event may be inserted after recovery of the dropped rod is complete or at any time per the discretion of the Lead Examiner.		

OAP-107.1

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 4	Page: 26 of 56
Event Description: FT-484 fails LOW.			

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: FT-484 will fail LOW. This will cause "B" SG level to lower.			
BOOTH OPERATOR:		When directed - Initiate Event 4 (TRIGGER 4).	
Indication Available: XCP-624 2-5, SG B LVL DEV. XCP-624 5-4, SG B FWF>STF MISMATCH.			
	CRS	Enters AOP-401.3, Steam Flow-Feedwater Flow Protection Channel Failure.	AOP-401.3
REFERENCE PAGE FOR AOP-401.3 1 LOSS OF MAIN FEEDWATER FLOW IF Feedwater flow is lost while Reactor Power is GREATER THAN 10%, and cannot be quickly restored from the MCB, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1. 2 STEAM GENERATOR LEVEL CONTROL a. IF Narrow Range Steam Generator Level decreases to LESS THAN 40% in any SG, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1. b. IF Reactor Power is GREATER THAN 15% and Narrow Range Steam Generator Level exceeds 75% in any SG, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.			AOP-401.3
NOTE Throughout this procedure, "AFFECTED" refers to any SG experiencing level control problems.			AOP-401.3
IOA	BOP	1. Verify the failed channel is the controlling channel.	AOP-401.3
NOTE - Step 2 FW and STEAM CONTROL CHANNEL SEL Switches for a SG should be selected to the same direction (both to the left or both to the right).			AOP-401.3
IOA **Critical Task**	BOP	2. Select the operable flow channel: • Place FW CONTROL CHANNEL SEL Switch to the operable channel. • Place STEAM CONTROL CHANNEL SEL Switch to the operable channel.	AOP-401.3

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 4	Page: 27 of 56
Event Description: FT-484 fails LOW.			

Time	Position	Applicant's Actions or Behavior	
		NOTE - Step 3 CTRL+ALT+S on either EHC HMI is equivalent to 50 MWe, and is the preferred method to accomplish a rapid load reduction.	AOP-401.3
IOA	BOP	3. Verify Turbine Load is LESS THAN 950 MWe.	AOP-401.3
IOA	BOP	4. Verify only one SG is AFFECTED.	AOP-401.3
IOA	BOP	5. Adjust the Feedwater Flow Control Valve as necessary to restore feed flow to the AFFECTED SG.	AOP-401.3
IOA	BOP	6. Check if Feedwater Pump speed control is operating properly: • Feedwater Header pressure is GREATER THAN Main Steam Header pressure. • Feed flow is normal for steam flow and power level. • All operating Feedwater Pump speeds and flows are balanced.	AOP-401.3
	BOP	7. Verify Narrow Range levels in all SGs are between 60% and 65%.	AOP-401.3
	BOP	8. Restore the AFFECTED SG control systems to normal: • Place the Feedwater Flow Control Valve in AUTO. • Place the Feedwater Pump Speed Control System in AUTO. REFER TO SOP-210, FEEDWATER SYSTEM.	AOP-401.3
		NOTE - Step 9 Steam flow transmitters FT-474, FT-484, FT-494, FT-475, FT-485, and FT-495 are density compensated by steam pressure transmitters PT-475, PT-485, PT-495, PT-476, PT-486, and PT-496.	AOP-401.3
	CRS	9. Perform the following: a. Determine the failed instrument channel. ____________________ b. Record the time of the channel failure. ____________________	AOP-401.3

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 4	Page: 28 of 56
Event Description: FT-484 fails LOW.			

Time	Position	Applicant's Actions or Behavior	
NOTE - Step 10 Compliance with T.S. requires tripping failed instrument channel bistables within 72 hours of the channel failure. Time should be allowed for troubleshooting of the failed channel prior to tripping the bistables.			AOP-401.3
*	CRS	10. Place the failed channel protection bistables in a tripped condition within 72 hours of the channel failure: a. Write an R&R for the failed channel. b. Select the attachment for the failed channel from the back of this procedure. c. Record the R&R number on the attachment. d. Determine the cause of the channel failure. e. Notify the I&C Department to place the identified bistables in trip using the attachment.	AOP-401.3

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 4	Page: 29 of 56
Event Description: FT-484 fails LOW.			

Time	Position	Applicant's Actions or Behavior	
	CRS	Enters T.S. 3.4.3.1, Reactor Trip System Instrumentation. Function 14; Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level, Action 6; With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 72 hours; and b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1. Enters T.S. 3.4.3.2, Engineered Safety Feature Actuation System Instrumentation Function 4d. STEAM LINE ISOLATION: Steam Flow in Two Steam Lines—High Coincident with Tavg—Low—Low, Action 24; With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: a. The inoperable channel is placed in the tripped condition within 72 hours. b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.2.1.	Tech Spec

EVALUATOR NOTE: The next event may be inserted following the CRS assessment of Tech Specs, or at any time per the discretion of the Lead Examiner.

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 5	Page: 30 of 56
Event Description: PT-508 fails LOW.			

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: PT-508 will fail LOW. This will cause the Main Feedwater Pumps to speed up.			
BOOTH OPERATOR:		When directed - Initiate Event 3 (TRIGGER 3).	
Indication Available: XCP-624 1-5, SG A LVL DEV XCP-624 2-5, SG B LVL DEV XCP-624 3-5, SG C LVL DEV PT-508 failing Low.			
	CRS	Enters AOP-210.3, Feedwater Pump Malfunction.	AOP-210.3

REFERENCE PAGE FOR AOP-210.3

1 MAIN FEEDWATER REGULATING VALVE MANUAL CONTROL

Manual Control of Main Feedwater Regulating Valves is permissible at any time as deemed necessary during the performance of this procedure. If a Main Feedwater Regulating Valve has been placed in Manual it should be returned to Automatic as soon as possible.

2 LOSS OF MAIN FEEDWATER FLOW

IF Feedwater flow is lost while Reactor Power is GREATER THAN 10%, and cannot be quickly restored from the MCB, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.

AOP-210.3

3 STEAM GENERATOR LEVEL CONTROL

a. IF Narrow Range Steam Generator Level decreases to LESS THAN 40% in any SG, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.

b. IF Reactor Power is GREATER THAN 15% and Narrow Range Steam Generator Level exceeds 75% in any SG, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.

4 MAIN FEED PUMP SPEED CONTROL

IF IPT00464 has failed with the Steam Dumps in Steam Pressure Mode, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.

IOA	BOP	1. Verify at least one Feedwater Pump is running.	AOP-210.3

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 5	Page: 31 of 56	
Event Description: PT-508 fails LOW.				

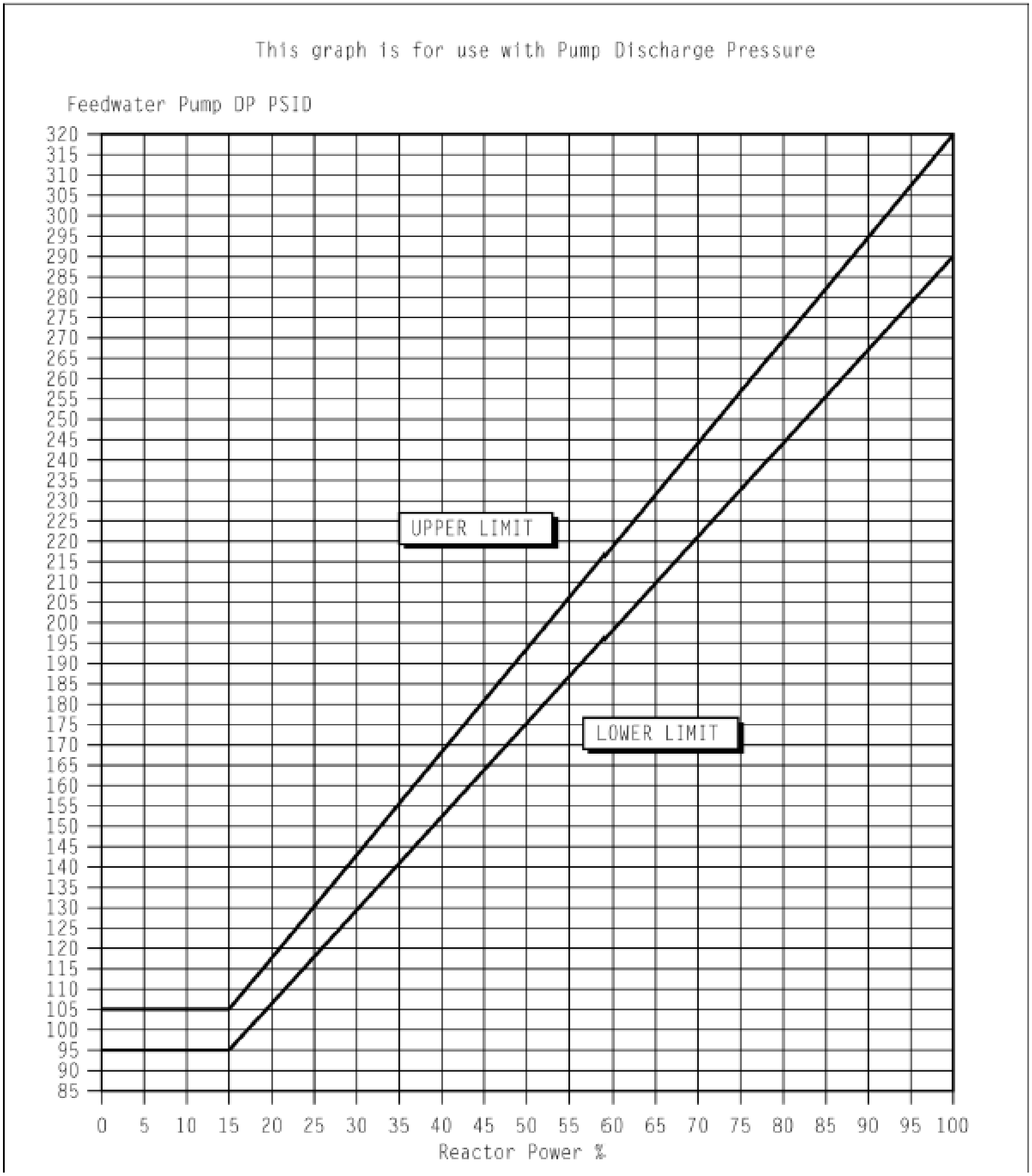
Time	Position	Applicant's Actions or Behavior	
IOA	BOP	2. Check if a Feedwater Pump trip has occurred. **(NO)**	AOP-210.3
IOA	BOP	**Alternative Action Step:** 2. GO TO Step 4.	AOP-210.3
IOA	BOP	4. Check Main Feedwater Pump operation. a. Verify all Main Feedwater Pumps are affected. b. Check if Reactor Power is GREATER THAN 50%. c. Place the MCB MASTER SPEED CNTRL in MAN and adjust to between 50% and 60% demand OR as needed to control Feedwater Flow. d. Verify all Main Feedwater Pump speeds are stable.	AOP-210.3
NOTE - Step 5 Due to the slow operation of the Main Feedwater Pump Recirculation Valves, a constant Main Feedwater Pump speed should be maintained until the recirculation valves have become relatively stable while adjusting Feedwater Flow.			AOP-210.3
	BOP	5. Check if Main Feedwater Flow matches Main Steam Flow for each Steam Generator.	AOP-210.3
*	BOP	6. Maintain Narrow Range Steam Generator Level between 60% and 65%.	AOP-210.3
	BOP	7. If necessary, place Main Feed Regulating valves in AUTO.	AOP-210.3
NOTE - Step 8 Main Feedwater Program ΔP should be established using the following as available: • PI-508, FW PP DICH HDR PRESS PSIG. • Any operating Main Feedwater Pump Discharge Pressure. • PI-464C, MS HDR PRESS PSIG. • Any available MCB Main Steam Header Pressure. • IPCS (ZZMENU S/G SU Trend or FW Start)			AOP-210.3
	BOP	8. Restore Feedwater Pump D/P to program. a. Using the Feedwater Pump Speed Control method established in Step 4, slowly adjust Feedwater Pump discharge header pressure to within the limits of ATTACHMENT 1, FEEDWATER PUMP D/P LIMITS. b. Adjust PUMP A(B)(C) SPEED CNTRL (MCB M/A Stations) as necessary to balance all operating Feedwater Pumps speed to within 120 rpm of each other.	AOP-210.3

Op Test No:	NRC-ILO-16-01	Scenario #	1	Event #	5	Page:	32	of	56

Event Description: PT-508 fails LOW.

Time	Position	Applicant's Actions or Behavior

FEEDWATER PUMP D/P LIMITS



AOP-210.3

Op Test No: NRC-ILO-16-01 Scenario # 1 Event # 5 Page: 33 of 56		
Event Description: PT-508 fails LOW.		
Time	Position	Applicant's Actions or Behavior
	BOP	9. Determine and correct the cause of the Feedwater Pump speed control malfunction. AOP-210.3
EVALUATOR NOTE: At this point the crew has done everything they can in this procedure since they will not get PT-508 back. The next event may be inserted at any time per the discretion of the Lead Examiner.		

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 6	Page: 34 of 56

Event Description: ATWS

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: A turbine trip will occur. The reactor will not trip and the crew will insert negative reactivity with rods and emergency boration. The reactor will be locally tripped. Once the reactor is tripped and the crew has exited EOP-13.0, a small break LOCA will be put in. The "A" CCW pump will trip, "B" and "C" CCW pumps will not start automatically. MVG-3003B, Spray Header Isolation valve will fail to open automatically on the Phase "A" isolation and the "B" Charging pump will experience a sheared shaft.			
BOOTH OPERATOR:		When directed - Initiate Event 6 (TRIGGER 6).	
Indication Available: Pressurizer Level lowering RCS Pressure lowering Charging Flow increasing			
	CRS	Enters EOP-1.0, E-0 Reactor Trip or Safety Injection.	EOP-1.0
IOA	RO	1. Verify Reactor Trip: • Trip the Reactor using either Reactor Trip Switch. • Verify all Reactor Trip and Bypass Breakers are open. **(NO)** • Verify all Rod Bottom Lights are lit. • Verify Reactor Power level is decreasing.	EOP-1.0
IOA	RO	**Alternative Action Step:** 1. Trip the Reactor using both Reactor Trip Switches. If the Reactor will NOT trip OR is NOT subcritical, THEN GO TO EOP-13.0, FR-S.1, RESPONSE TO ABNORMAL NUCLEAR POWER GENERATION, Step 1.	EOP-1.0
	CRS	Enters EOP-13.0, FR-S.1 Response to Abnormal Nuclear Power Generation.	
CAUTION RCPs should NOT be tripped with Reactor power GREATER THAN 5%, to prevent core damage due to low flow.			EOP-13.0
NOTE Steps 1 and 2 are Immediate Operator Actions.			EOP-13.0
NOTE - Step 1 Manual or Automatic Rod Control may be used to perform Alternative Action Step 1, whichever provides the fastest Control Rod insertion rate.			EOP-13.0

Op Test No: NRC-ILO-16-01 Scenario # 1 Event # 6 Page: 35 of 56			
Event Description: ATWS			
Time	**Position**	**Applicant's Actions or Behavior**	
IOA	RO	1. Verify Reactor Trip: • Trip the Reactor using either Reactor Trip Switch. • Verify all Reactor Trip and Bypass Breakers are open. **(NO)** • Verify all Rod Bottom Lights are lit. • Verify Reactor Power level is decreasing.	EOP-13.0
IOA **Critical Task**	CRS/RO	**Alternative Action Step:** 1. IF the Reactor will NOT trip OR is NOT subcritical, THEN insert Control Rods. Trip the Reactor per ATTACHMENT 1, TRIPPING THE REACTOR LOCALLY.	EOP-13.0
EVALUATOR NOTE: The critical task is to insert negative reactivity using rods or emergency boration.			
BOOTH OPERATOR:		When contacted as local operator to manually trip the reactor, acknowledge the request, wait 3 minutes, THEN insert the following: **TRIGGER 12** - Open Reactor Trip Breaker "A" **TRIGGER 13** - Open Reactor Trip Breaker "B" After inserting Trigger 12 and 13 contact the Control Room and report: "I have completed EOP-13.0 Attachment 1, Tripping the Reactor Locally".	EOP-13.0
IOA	BOP	2. Verify Turbine/Generator Trip: a. Verify all Turbine STM STOP VLVs are closed. b. Ensure Generator Trip (after 30 second delay): 1. Ensure the GEN BKR is open. 2. Ensure the GEN FIELD BKR is open. 3. Ensure the EXC FIELD CNTRL is tripped.	EOP-13.0
	BOP	3. Ensure EFW Pumps are running: a. Ensure both MD EFW Pumps are running. b. Verify the TD EFW Pump is running if necessary to maintain SG levels.	EOP-13.0

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 6	Page: 36 of 56
Event Description: ATWS			

Time	Position	Applicant's Actions or Behavior	
CRITICAL TASK	RO	4. Initiate emergency boration of the RCS: a. Ensure at least one Charging Pump is running. b. Verify PZR pressure is LESS THAN 2330 psig. c. Verify SI ACT status light is NOT lit. d. Open MVT-8104, EMERG BORATE. e. Verify XPP-13B, BA XFER PP B, is running. f. Verify GREATER THAN 30 gpm flow on FI-110, EMERG BORATE FLOW GPM.	EOP-13.0
	BOP	5. Verify Containment Ventilation Isolation Valves closed by verifying the following SAFETY INJECTION monitor lights are dim: • XCP-6103 3-4 (POST ACCID HR EXH 6057 & 6067). • XCP-6103 2-1 (POST ACCID HR EXH 6056/6066).	EOP-13.0
CAUTION - Step 6 If an automatic SI exists or occurs, Steps 1 through 8 of EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, should be performed to verify proper SI actuation, while continuing with this procedure.			EOP-13.0
	BOP	6. Check if all Turbine STM STOP VLVs are closed.	EOP-13.0
*	RO	7. Verify the Reactor is subcritical: a. Power Range channels indicate LESS THAN 5%. b. Intermediate Range channels indicate a negative startup rate. c. GO TO Step 16. Observe the CAUTION prior to Step 16.	EOP-13.0
*	BOP	8. Check SG levels: a. Verify Narrow Range level is GREATER THAN 26% [41%] in at least one SG. b. Control EFW flow to maintain Narrow Range SG levels between 26% [41%] and 60%.	EOP-13.0
	CRS	Recognizes reactor trip breakers are now open and goes to Step 16.	
CAUTION - Step 16 Boration should be continued to obtain adequate shutdown margin during subsequent actions.			EOP-13.0
	CRS	16. RETURN TO the Procedure and Step in effect.	EOP-13.0
EVALUATOR NOTE: The crew will transition back to EOP-1.0 now. The Small Break LOCA should be inserted at this point.			

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 37 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
BOOTH OPERATOR:		When directed - Initiate Event 7 (TRIGGER 7).	
IOA	RO	1. Verify Reactor Trip: • Trip the Reactor using either Reactor Trip Switch. • Verify all Reactor Trip and Bypass Breakers are open. • Verify all Rod Bottom Lights are lit. • Verify Reactor Power level is decreasing.	EOP-1.0
IOA	BOP	2. Verify Turbine/Generator Trip: a. Verify all Turbine STM STOP VLVs are closed. b. Ensure Generator Trip (after 30 second delay): 1. Ensure the GEN BKR is open. 2. Ensure the GEN FIELD BKR is open. 3. Ensure the EXC FIELD CNTRL is tripped.	EOP-1.0
IOA	BOP	3. Verify both ESF buses are energized.	EOP-1.0
IOA	RO	4. Check if SI is actuated: a. Check if either: • SI ACT status light is bright on XCP-6107 1-1. OR • Any red first-out SI annunciator is lit onXCP-626 top row. b. Actuate SI using either SI ACTUATION Switch. c. GO TO Step 6.	EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 38 of 56

Event Description: Small Break LOCA with failures. (Major)

Time	Position	Applicant's Actions or Behavior

REFERENCE PAGE FOR EOP-1.0

1 RCP TRIP CRITERIA

a. IF Phase B Containment Isolation has actuated (XCP-612 4-2), THEN trip all RCPs.

b. IF both of the following conditions occur, THEN trip all RCPs:

- SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

AND

- RCS Wide Range pressure is LESS THAN 1418 psig.

2 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.

3 MONITOR SPENT FUEL COOLING

Periodically check status of Spent Fuel Cooling by monitoring the following throughout event recovery:

- Spent Fuel Pool level.
- Spent Fuel Pool temperature.

4 RUPTURED STEAM GENERATOR

IF a RUPTURED Steam Generator has been positively identified, THEN throttle EFW to the RUPTURED Steam Generator WHEN its Narrow Range Level is GREATER THAN 26%[41%].

5 FAULTED STEAM GENERATOR

- IF a FAULTED Steam Generator has been positively identified, THEN isolate EFW to the faulted Steam Generator as soon as possible UNLESS all three Steam Generators are FAULTED.
- IF all three Steam Generators are FAULTED, THEN throttle EFW flow to all three Steam Generators to 50 gpm.

EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 39 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
	BOP	6. Initiate ATTACHMENT 3, SI EQUIPMENT VERIFICATION.	EOP-1.0
EVALUATOR NOTE: Attachment 3 can be found on page 50 of 55.			
	CRS	7. Announce plant conditions over the page system.	EOP-1.0
*	RO	8. Verify RB pressure has remained LESS THAN 12 psig on PR-951, RB PSIG (P-951), red pen.	EOP-1.0
BOOTH OPERATOR:		When called to rack up the breaker for the "C" Charging pump, Wait 2 minutes and tell them you are standing by to rack up the breaker for "C" Charging pump. When told to rack up the breaker for "C" Charging Pump, Wait 1 minute, and inform them that the breaker for "C" is mechanically bound and cannot be racked up.	
BOOTH OPERATOR:		If called to look at "B" Charging pump, wait 2 minutes and report back that it has a sheared shaft.	
EVALUATOR NOTE: RB Pressure will reach 12 psig, however, it will be later in the scenario in which case the crew will come back to this step.			

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 40 of 56

Event Description: Small Break LOCA with failures. (Major)

Time	Position	Applicant's Actions or Behavior	
*	RO	**Alternative Action Step:** 8. Perform the following: a. Verify both the following annunciators are lit: • XCP-612 3-2 (RB SPR ACT). • XCP-612 4-2 (PHASE B ISOL). IF either annunciator is NOT lit, THEN actuate RB Spray by placing the following switches to ACTUATE: • Both CS-SGA1 and CS-SGA2. OR • Both CS-SGB1 and CS-SGB2. b. For any valve status light on XCP-6105 that is NOT bright, ensure its associated valve is closed. c. Ensure the following are open: • MVG-3001A(B), RWST TO SPRAY PUMP A(B) SUCT. • MVG-3002A(B),NAOH TO SPRAY PUMP A(B) SUCT. • MVG-3003A(B), SPRAY HDR ISOL LOOP A(B). d. Ensure both RB Spray Pumps are running. IF any RB Spray Pump will not start OR trips, THEN close MVG-3003A(B), SPRAY HDR ISOL LOOP A(B) for the AFFECTED RB Spray Pump. e. Verify RB Spray flow is GREATER THAN 2500 gpm for each operating train on: • FI-7368, SPR PP A DISCH FLOW GPM. • FI-7378, SPR PP B DISCH FLOW GPM. f. Stop all RCPs.	EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 41 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
	RO	9. Check RCS temperature: • With any RCP running, RCS Tavg is stable at OR trending to 557°F. OR • With no RCP running, RCS Tcold is stable at OR trending to 557°F.	EOP-1.0
*	RO	**Alternative Action Step:** 9. IF RCS temperature is LESS THAN 557°F AND decreasing, THEN stabilize temperature by performing the following as required: a. Close IPV-2231, MS/PEGGING STM TO DEAERATOR. b. Perform one of the following: • IF Narrow Range SG level is LESS THAN 26% [41%] in all SGs, THEN reduce EFW flow as necessary to stop cooldown, while maintaining total EFW flow GREATER THAN 450 gpm. OR • WHEN Narrow Range SG level is GREATER THAN 26% [41%] in at least one SG, THEN control EFW flow as necessary to stabilize RCS temperature at 557°F. c. Initiate ATTACHMENT 6, STEAM VALVE ISOLATION, while continuing with this procedure. d. IF RCS cooldown continues, THEN close: • MS Isolation Valves, PVM-2801A(B)(C). • MS Isolation Bypass Valves, PVM-2869A(B)(C).	EOP-1.0
	RO	10. Check PZR PORVs and Spray Valves: a. PZR PORVs are closed. b. PZR Spray Valves are closed. c. Verify power is available to at least one PZR PORV Block Valve: • MVG-8000A, RELIEF 445 A ISOL. • MVG-8000B, RELIEF 444 B ISOL. • MVG-8000C, RELIEF 445 B ISOL. d. Ensure one of the following Block Valves is open unless it was closed to isolate an open PZR PORV: • MVG-8000A, RELIEF 445 A ISOL. • MVG-8000B, RELIEF 444 B ISOL.	EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 42 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
NOTE - Step 11 Seal Injection flow should be maintained to all RCPs.			EOP-1.0
	RO	11. Check if RCPs should be stopped: a. Check if either of the following criteria is met: • Annunciator XCP-612 4-2 is lit (PHASE B ISOL). OR • RCS pressure is LESS THAN 1418 psig AND SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. b. Stop all RCPs.	EOP-1.0
	RO	12. Verify no SG is FAULTED: • No SG pressure is decreasing in an uncontrolled manner. • No SG is completely depressurized.	EOP-1.0
	RO	13. Verify Secondary radiation levels indicate SG tubes are NOT RUPTURED: • RM-G19A(B)(C), STMLN HI RNG GAMMA. • RM-A9, CNDSR EXHAUST GAS ATMOS MONITOR. • RM-L3, STEAM GENERATOR BLOWDOWN LIQUID MONITOR. • RM-L10, SG BLOWDOWN CW DISCHARGE LIQUID MONITOR.	EOP-1.0
	RO	14. Check if the RCS is INTACT: a. RB radiation levels are normal on: • RM-G7, CNTMT HI RNG GAMMA. • RM-G18, CNTMT HI RNG GAMMA. b. RB Sump levels are normal. **(NO)** c. RB pressure is LESS THAN 1.5 psig. **(NO)** d. The following annunciators are NOT lit: • XCP-606 2-2 (RBCU 1A/2A DRN FLO HI). • XCP-607 2-2 (RBCU 1B/2B DRN FLO HI).	EOP-1.0
	RO	**Alternative Action Step:** 14. GO TO EOP-2.0, E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.	EOP-1.0
EVALUATOR NOTE: Crew will transition to EOP-2.0, E-1 Loss of Reactor or Secondary Coolant based on Sump levels or RB Pressure.			

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 43 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
	CRS	Transition to EOP-2.0, LOSS OF REACTOR OR SECONDARY COOLANT	EOP-2.0

REFERENCE PAGE FOR EOP-2.0

1 SI REINITIATION CRITERIA

IF either of the following conditions occurs, THEN start Charging Pumps and operate valves as necessary:

- RCS subcooling on TI-499A(B), A(B) TEMP °F, is LESS THAN 52.5°F [67.5°F].
- PZR level can NOT be maintained GREATER THAN 10% [28%].

2 RCP TRIP CRITERIA

IF either of the following criteria is met, THEN trip all RCPs:

- Annunciator XCP-612 4-2 is lit (PHASE B ISOL).
- RCS pressure is LESS THAN 1418 psig AND SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

3 SECONDARY INTEGRITY TRANSITION CRITERIA

IF any unisolated SG pressure is decreasing in an uncontrolled manner OR is completely depressurized, THEN **GO TO EOP-3.0, FAULTED STEAM GENERATOR ISOLATION, Step 1.** EOP-2.0

4 TUBE RUPTURE TRANSITION CRITERIA

IF any SG level increases in an uncontrolled manner OR if any SG has abnormal radiation, THEN start Charging Pumps and operate valves as necessary, and **GO TO EOP-4.0, STEAM GENERATOR TUBE RUPTURE, Step 1.**

5 COLD LEG RECIRCULATION TRANSITION CRITERION

IF RWST level decreases to LESS THAN 18%, THEN **GO TO EOP-2.2, TRANSFER TO COLD LEG RECIRCULATION, Step 1.**

6 LOSS OF EMERGENCY COOLANT RECIRCULATION TRANSITION CRITERION

IF Emergency Coolant Recirculation is established and subsequently lost, THEN **GO TO EOP-2.4, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.**

7 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. **REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.**

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 44 of 56

Event Description: Small Break LOCA with failures. (Major)

Time	Position	Applicant's Actions or Behavior	
		NOTE Seal Injection flow should be maintained to all RCPs.	
	RO	1. Check if RCPs should be stopped: a. Check if either of the following criteria is met: • Annunciator XCP-612 4-2 is lit (PHASE B ISOL). OR • RCS pressure is LESS THAN 1418 psig AND SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. b. Stop all RCPs	EOP-2.0
	BOP	2. Verify no SG is FAULTED: • No SG pressure is decreasing in an uncontrolled manner. • No SG is completely depressurized.	EOP-2.0
*	BOP	3. Check INTACT SG levels: a. Verify Narrow Range level in INTACT SGs is GREATER THAN 26% [41%]. b. Control EFW flow to maintain Narrow Range level in each INTACT SG between 26% [41%] and 60%.	EOP-2.0
	RO	4. Reset both SI RESET TRAIN A(B) Switches.	EOP-2.0
	RO	5. Reset Containment Isolation: • RESET PHASE A - TRAIN A(B) CNTMT ISOL. • RESET PHASE B - TRAIN A(B) CNTMT ISOL.	EOP-2.0
	BOP	6. Check if Secondary radiation levels are normal: a. Check radiation levels normal on: • RM-G19A(B)(C), STMLN HI RNG GAMMA. • RM-A9, CNDSR EXHAUST GAS ATMOS MONITOR. • RM-L3, STEAM GENERATOR BLOWDOWN LIQUID MONITOR. • RM-L10, SG BLOWDOWN CW DISCHARGE LIQUID MONITOR. b. Place SVX-9398A(B)(C), SG A(B)(C) SMPL ISOL, in AUTO. c. Notify Chemistry to sample all SG secondary sides, and screen samples for abnormal activity using a frisker.	EOP-2.0

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 45 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
*	RO	7. Check PZR PORVs and Block Valves: a. Verify power is available to the PZR PORV Block Valves: 1. MVG-8000A, RELIEF 445 A ISOL. 2. MVG-8000B, RELIEF 444 B ISOL. 3. MVG-8000C, RELIEF 445 B ISOL. **CAUTION - Step 7.b** **If any PZR PORV opens because of high PZR pressure, Step 7.b should be repeated after pressure decreases to LESS THAN 2330 psig, to ensure the PORV recloses.** b. Verify all PZR PORVs are closed. c. Verify at least one PZR PORV Block Valve is open.	EOP-2.0
	BOP	8. Place both ESF LOADING SEQ A(B) RESETS to: a. NON-ESF LCKOUTS. b. AUTO-START BLOCKS.	EOP-2.0
	RO	9. Establish Instrument Air to the RB: a. Start one Instrument Air Compressor and place the other in Standby. b. Verify PI-8342, INSTR AIR HDR PRESS PSIG, indicates GREATER THAN 60 psig. c. Open PVA-2659, INST AIR TO RB AIR SERV. d. Open PVT-2660, AIR SPLY TO RB.	EOP-2.0

Op Test No: NRC-ILO-16-01 Scenario # 1 Event # 7 Page: 46 of 56			
Event Description: Small Break LOCA with failures. (Major)			
Time	**Position**	**Applicant's Actions or Behavior**	
*	RO	10. Check if SI flow should be reduced: a. RCS subcooling on TI-499A(B), A(B) TEMP °F, is GREATER THAN 52.5°F [67.5°F]. b. Secondary Heat Sink is adequate: • Total EFW flow to INTACT SGs is GREATER THAN 450 gpm. OR • Narrow Range level is GREATER THAN 26% [41%] in at least one INTACT SG. c. RCS pressure is stable OR increasing. **NOTE - Step 10.d** **If PZR level is LESS THAN 10% [28%], the PZR should refill from SI flow after pressure is stabilized.** d. PZR level is GREATER THAN 10% [28%]. e. GO TO EOP-1.2, ES-1.1, SAFETY INJECTION TERMINATION, Step 1.	EOP-2.0
*	RO	11. Check if RB Spray should be stopped: a. Check if any RB Spray Pumps are running. b. Verify RB pressure is LESS THAN 11 psig. c. Depress both RESET TRAIN A(B) RB SPRAY. **NOTE - Step 11.d** • **RB Spray must run for a minimum of four hours.** • **Anytime RB Spray Pumps are stopped, MVG-3003A(B), SPRAY HDR ISOL LOOP A(B), should be closed for containment isolation.** d. Consult with TSC personnel concerning RB Spray System operation.	EOP-2.0

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 47 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
CAUTION - Step 12 RCS pressure should be monitored. If RCS pressure decreases in an uncontrolled manner to LESS THAN 325 psig, the RHR Pumps must be manually restarted to supply water to the RCS.			EOP-2.0
*	RO	12. Check if RHR Pumps should be stopped: a. Check RCS pressure: 1. RCS pressure is GREATER THAN 325 psig. 2. RCS pressure is stable OR increasing. b. Check if any RHR Pump is running with suction aligned to the RWST. c. Stop any RHR Pump which is running with suction aligned to the RWST and place in Standby.	EOP-2.0
	RO	13. Check if RCS pressure is stable OR decreasing.	EOP-2.0
	RO	14. Check if pressure in all SGs is stable OR increasing.	EOP-2.0
CAUTION - Step 15 The DGs should NOT be run at a minimum load or unloaded for extended periods of time, to prevent carbon fouling.			EOP-2.0
	BOP	15. Check if DGs should be stopped: a. Verify both ESF buses are energized by offsite power. b. Stop any unloaded DG. REFER TO SOP-306, EMERGENCY DIESEL GENERATOR.	EOP-2.0
EVALUATOR NOTE: We expect to get to 12 psig in containment (Spray actuation) around steps 10-15 of EOP-2.0. Once we reach 12 psig in containment, the crew will use EOP-1.0, step 8 to verify spray has actuated properly.			

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 48 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
	RO	16. Verify equipment is available for Cold Leg Recirculation: a. Verify A Train equipment available: 1. XPP0031A-RH, RESIDUAL HEAT REMOVAL PUMP A 2. XVG08809A-SI, REFUEL WTR STG TK RH PUMP A SUCT VLV 3. XVG08811A-SI, CNTMT SUMP RH PUMP A SUCT ISOL VALVE. 4. XVG08812A-SI, RH PUMP A SUCTION HEADER VALVE. 5. One Train A Charging Pump: • XPP0043A CHARGING/SI PUMP A. OR • XPP0043C CHARGING/SI PUMP C (Train A). 6. LCV00115B-CS, CHG PUMP A SUCTION HDR RWST HDR ISOL VALVE. 7. XVG08706A-RH, CHG/SI PUMP SUCT HDR RH HDR A(B) INLET VLV. b. Verify B Train equipment available: 1. XPP0031B-RH, RESIDUAL HEAT REMOVAL PUMP B. 2. XVG08809B-SI, REFUEL WTR STG TK RH PUMP B SUCT VLV. 3. XVG08811B-SI, CNTMT SUMP RH PUMP B SUCT ISOL VALVE. 4. XVG08812B-SI, RH PUMP B SUCTION HEADER VALVE. 5. One Train B Charging Pump: • XPP0043B CHARGING/SI PUMP OR • XPP0043C CHARGING/SI PUMP C (Train B). 6. LCV00115D-CS, CHG PUMP B SUCTION HDR RWST HDR ISOL VALVE. 7. XVG08706B-RH, CHG/SI PUMP SUCT HDR RH HDR B INLET VLV. c. Open both MVB-9503A(B), CC TO RHR HX A(B).	EOP-2.0

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # 7	Page: 49 of 56
Event Description: Small Break LOCA with failures. (Major)			

Time	Position	Applicant's Actions or Behavior	
	RO	Step 16 continued **CAUTION - Step 16.d** • **If the swing CCW Pump is NOT available, the running pump should NOT be secured to shift it to fast speed, to prevent damage to the Charging Pump on that train.** • **If CCW can NOT be shifted to fast speed, this procedure should be continued. CCW alignment will be addressed in EOP-2.2, ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.** d. Shift the CCW Train to fast speed in the Active Loop. REFER TO SOP-118, COMPONENT COOLING WATER. e. Consult with TSC personnel to determine if equipment required for Cold Leg Recirculation is available.	EOP-2.0
NOTE - Step 17 Presence of abnormally high levels of radioactivity in the AB indicates that a Containment breach may be in progress. Conditions for upgrading the Emergency status should be evaluated using EPP-001, ACTIVATION AND IMPLEMENTATION OF EMERGENCY PLAN.			EOP-2.0
	BOP	17. Check the AB for evidence of ECCS leakage: a. Verify AB radiation levels are normal on: • RM-A3, MAIN PLANT VENT EXH ATMOS MONITOR: PARTICULATE, IODINE, GAS. • RM-A13, PLANT VENT HI RANGE. • RM-A11, AB VENT GAS ATMOS MONITOR. • Local area monitors. b. Verify annunciator XCP-631 6-1 is NOT lit (AB SMP LVL HI). c. Verify annunciators XCP-606 3-4 and XCP-607 3-4 are NOT lit (LD TRBL AB SMP/FLDRN LVL HI). d. Verify annunciators XCP-606 3-1 and XCP-607 3-1 are NOT lit (LTDN LD TEMP HI).	EOP-2.0
	RO	18. Obtain necessary Chemistry samples: a. Ensure all RCS sample valves are in AUTO: • SVX-9364B and SVX-9365B, RCS LP B SMPL ISOL. • SVX-9364C and SVX-9365C, RCS LP C SMPL ISOL. b. Notify Chemistry to sample the following: • RCS. • All SGs for isotopic activity.	EOP-2.0

Op Test No: NRC-ILO-16-01 Scenario # 1 Event # 7 Page: 50 of 56		
Event Description: Small Break LOCA with failures. (Major)		
Time	Position	Applicant's Actions or Behavior
	RO	19. Shut down and stabilize the Secondary Plant. REFER TO AOP-214.1, TURBINE TRIP. EOP-2.0
	RO	20. Check if RCS cooldown and depressurization is required: a. RCS pressure is GREATER THAN 325 psig. b. GO TO EOP-2.1, ES-1.2, POST-LOCA COOLDOWN AND DEPRESSURIZATION, Step 1. EOP-2.0
	CRS	Transitions to EOP-2.1, ES-1.2 Post-LOCA Cooldown and Depressurization.
EVALUATOR NOTE: The scenario may be terminated when the crew transitions from EOP-2.0 to EOP-2.1.		

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # N/A	Page: 51 of 56
Event Description: EOP-1.0, Attachment 3			

Time	Position	Applicant's Actions or Behavior	
	BOP	1. Ensure EFW Pumps are running: a. Ensure both MD EFW Pumps are running. b. Verify the TD EFW Pump is running if necessary to maintain SG levels.	EOP-1.0 Attachment 3
	BOP	2. Ensure the following EFW valves are open: • FCV-3531(3541)(3551), MD EFP TO SG A(B)(C). • FCV-3536(3546)(3556), TD EFP TO SG A(B)(C). • MVG-2802A(B), MS LOOP B(C) TO TD EFP.	Attachment 3
	BOP	3. Verify total EFW flow is GREATER THAN 450 gpm.	Attachment 3
	BOP	4. Ensure FW Isolation: a. Ensure the following are closed: • FW Flow Control, FCV-478(488)(498). • FW Isolation, PVG-1611A(B)(C). • FW Flow Control Bypass, FCV-3321(3331)(3341). • SG Blowdown, PVG-503A(B)(C). • SG Sample, SVX-9398A(B)(C). b. Ensure all Main FW Pumps are tripped.	Attachment 3
	BOP	5. Ensure SI Pumps are running: • Two Charging Pumps are running. • Both RHR Pumps are running.	Attachment 3
	BOP	6. Ensure two RBCU Fans are running in slow speed (one per train).	Attachment 3

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # N/A	Page: 52 of 56
Event Description: EOP-1.0, Attachment 3			

Time	Position	Applicant's Actions or Behavior	
	BOP	7. Verify Service Water to the RBCUs: a. Ensure two Service Water Pumps are running. b. Verify Service Water Booster Pump A is stopped. **(NO)** **Alternative Action Step:** b. GO TO Step 7.e. 7e. Verify that Service Water Booster Pump B is stopped. **(NO)** **Alternative Action Step:** e. GO TO Step 7.h. 7h. Verify GREATER THAN 2000 gpm flow for each train on: • FI-4466, SWBP A DISCH FLOW GPM. • FI-4496,SWBP B DISCH FLOW GPM.	Attachment 3
	BOP	8. Verify two CCW Pumps are running.	Attachment 3
	BOP	9. Ensure two Chilled Water Pumps and Chillers are running.	Attachment 3
	BOP	10. Verify both trains of Control Room Ventilation are running in Emergency Mode.	Attachment 3
	BOP	11. Check if Main Steamlines should be isolated: a. Check if any of the following conditions are met: • RB pressure GREATER THAN 6.35 psig. OR • Steamline pressure LESS THAN 675 psig. OR • Steamline flow GREATER THAN 1.6 MPPH AND Tavg LESS THAN 552°F. b. Ensure all the following are closed: • MS Isolation Valves, PVM-2801A(B)(C). • MS Isolation Bypass Valves, PVM-2869A(B)(C).	Attachment 3

Op Test No: NRC-ILO-16-01 Scenario # 1 Event # N/A Page: 53 of 56		
Event Description: EOP-1.0, Attachment 3		
Time	**Position**	**Applicant's Actions or Behavior**
	BOP	12. Ensure Excess Letdown Isolation Valves are closed: • PVT-8153, XS LTDN ISOL. • PVT-8154, XS LTDN ISOL.
	BOP	13. Verify ESF monitor lights indicate Phase A AND Containment Ventilation Isolation on XCP-6103, 6104, and 6106. REFER TO ATTACHMENT 4, CONTAINMENT ISOLATION VALVE MCB STATUS LIGHT LOCATIONS, as needed.
	BOP	14. Verify proper SI alignment: a. Verify SI valve alignment by verifying SAFETY INJECTION/PHASE A ISOL monitor lights are bright on XCP-6104. b. Verify all SAFETY INJECTION monitor lights are dim on XCP-6106. c. Verify SI flow on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. d. Check if RCS pressure is LESS THAN 325 psig.
	BOP	Report completion of Attachment 3.
EVALUATOR NOTE: ATTACHMENT 3 is complete.		

Attachment 3
Attachment 3
Attachment 3
Attachment 3

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # N/A	Page: 54 of 56
Event Description: SOP-106, Borate Operations			

Time	Position	Applicant's Actions or Behavior	
		NOTE 2.0 1. Energizing additional Pressurizer Heaters will enhance mixing. 2. LCV-115A, LTDN DIVERT TO HU-TK, will begin to modulate to the HU-TK position at 70% level on LI-115, VCT LEVEL %.	SOP-106
	RO	2.1. Ensure at least one Reactor Coolant Pump is running.	SOP-106
	RO	2.2. Place RX COOL SYS MU switch to STOP.	SOP-106
	RO	2.3. Place RX COOL SYS MU MODE SELECT switch to BOR. **(Peer ✓)**	SOP-106
	RO	2.4. Adjust FCV-113 A&B, BA FLOW SET PT, for desired flow rate. **(Peer ✓)**	SOP-106
	RO	2.5. Set FIS-113, BA TO BLNDR FLOW, batch integrator to the desired volume. **(Peer ✓)**	SOP-106
	RO	2.6. Place RX COOL SYS MU switch to START.	SOP-106
		NOTE 2.7 Step 2.7 may be omitted when borating less than 10 gallons.	SOP-106
	RO	2.7. Place FCV-113 A&B, BA FLOW, controller in AUTO.	SOP-106
		NOTE 2.8 The AUTO setpoint dial for FCV-113A&B, BA FLOW, controller may be adjusted slowly to obtain the desired flow rate.	SOP-106
	RO	2.8. Verify the desired Boric Acid flow rate on FR-113, BA TO BLNDR GPM (F-113).	SOP-106
	RO	2.9. When the preset volume of boric acid has been reached, perform the following: a. Place FCV-113A&B, BA flow controller in MAN. b. Verify boration stops.	SOP-106
	RO	2.10. Place RX COOL SYS MU switch to STOP.	SOP-106

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # N/A	Page: 55 of 56
Event Description: SOP-106, Borate Operations			

Time	Position	Applicant's Actions or Behavior	
		NOTE 2.11 a. If plant conditions require repeated borations, Step 2.11 may be omitted. b. The volume in the piping between the blender and the VCT outlet is approximately 3.8 gallons.	SOP-106
		2.11. Alternate Dilute 4 to 6 gallons of Reactor Makeup Water to flush the line downstream of the blender by performing the following: a. Place RX COOL SYS MU MODE SELECT switch to ALT DIL.**(Peer ✓)** b. Adjust FCV-168, TOTAL MU FLOW SET PT, to desired flow rate. **(Peer ✓)** c. Set FIS-168, TOTAL MU FLOW, batch integrator to desired volume. **(Peer ✓)** d. Place RX COOL SYS MU switch to START. e. Verify desired flow rate on FR-113, TOTAL MU GPM (F-168). f. Verify alternate dilution stops when preset volume is reached on FIS-168, TOTAL MU FLOW, batch integrator. g. Place RX COOL SYS MU switch to STOP.	SOP-106
		2.12. Place RX COOL SYS MU MODE SELECT switch to AUTO. **(Peer ✓)**	SOP-106
		2.13. Adjust FCV-168, TOTAL MU FLOW SET PT, to 7.5 (120 gpm). **(Peer ✓)**	SOP-106
		2.14. In MAN, adjust FCV-113 A&B, BA FLOW OUTPUT, to the required position which will ensure proper Boric Acid addition for subsequent Automatic Makeup operations.	SOP-106
		2.15. Adjust FCV-113 A & B, BA FLOW, SET PT per one of the following: a. OAP-100.6, Attachment IA, Reactivity Control Parameters. b. Desired position to ensure proper boric acid addition based on current RCS conditions.	SOP-106
		2.16. Place RX COOL SYS MU switch to START. **(Peer ✓)**	SOP-106
		2.17. Perform the following: a. Start XPP-13A(B), BA XFER PP A(B), for the in-service Boric Acid Tank. b. If necessary, start XPP-13A(B), BA XFER PP A(B), for the Boric Acid Tank on recirculation.	SOP-106

Op Test No: NRC-ILO-16-01	Scenario # 1	Event # N/A	Page: 56 of 56
Event Description: SOP-101, Reactor Coolant System			

Time	Position	Applicant's Actions or Behavior
		NOTE 2.2 Operation of Pressurizer Back Up Heaters for a long period of time may result in a large integral signal built into the demand of the Pressurizer Master Pressure Controller while in automatic. In order to clear this signal, the Pressurizer Master Pressure Controller should be place in MAN and then back to AUTO.
	RO	2.2. Energize a set of Pressurizer Back Up Heaters. a. Place the BU GRP to be energized in the After Close position. b. Monitor RCS Pressure for proper Pressurizer Master Pressure Controller response. c. When the Pressurizer Back Up Heater Group is no longer needed, Place the BU GRP in the After Trip position.

SOP-101

SOP-101

Facility: VC SUMMER U1 Scenario No: 2 Op Test No: NRC-ILO-16-01

Examiners: ____________________ Operators: CRS: ____________________

____________________ RO: ____________________

____________________ BOP: ____________________

Initial Conditions:
- The Reactor is at 100% power
- “B” train work week.
- “B” MDEFW pump is OOS.
- XFN-0065B RBCU is OOS.

Turnover:
- Perform stroke test of PZR Block valves.

Critical Tasks:
- Establish EFW flow before a transition to EOP-15.0, FR-H.1 Response to loss of Secondary Heat Sink is required.
- Take manual control of Pressurizer Heaters and Spray before lifting a Pressurizer PORV.

Event	Malf No.	Event Type*	Event Description
1	N/A	N-BOP, CRS	STP-127.001, PZR PORV Block valve stroke test.
2	MAL-PRS001A	I-RO, CRS TS-CRS	PT-444 Fails low, heaters turn on, RCS pressure rises.
3#	OVR-CW020B MAL-TUR002A MAL-TUR002B MAL-TUR002C MAL-TUR002D MAL-TUR002E	I-BOP CRS	TLO TCV-4211 Fails Open in AUTO with increased turbine vibration.
4#	MAL- FWM001B	C-BOP, CRS R-RO	Main Feedwater Pump “B” Trip
5#	MAL- RCS002A	C-RO, CRS TS-CRS	SG “A” Tube Leak

6	MAL- RCS002A	M-ALL	SG “A” tube leak becomes a Tube Rupture.
	MAL-EPS013 OVR-EG020B OVR-EG021B		Main Generator and Voltage Regulator Breakers fail to trip.
	PMP-EF001F		“A” Motor Driven EFW pump fails to auto start, can be started manually
	MAL-FWM003C		Turbine Driven EFW pump trips.
	PMP-AH022F PMP-AH025F PMP-AH023F		Control Room emergency Ventilation fails to start.
7	MAL- MSS004A	M-ALL	SG “A” becomes faulted
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

Used on previous two NRC Exams. Event 3 used on the 2016 exam, scenario 2 and event 4 was used on the 2017 exam, scenario 2. Event 5 was used on the 2017 exam, scenario 1.

The following notation is used in the ES-D-2 form “Time” column:

IOA designates Immediate Operator Action steps.

* designates Continuous Action steps.

TURNOVER:

The crew will assume the watch having been pre-briefed on the Initial Conditions, the plan for this shift and any related operating procedures. The “B” Motor Driven EFW pump will be inoperable for scheduled preventive maintenance. Tech Spec 3.7.1.2, Emergency Feedwater System action a (restore “B” pump within 72 hours) has been in effect for 6 hours with pump return to service is expected 6 hours from now. Train “B” RBCU, XFN-0065B is tagged out for breaker maintenance and is to be returned to service next shift. The crew will be instructed to perform STP-127.001, Pressurizer Block Valve Operability Test as soon as they take the shift.

- **PRE-LOAD**

  - OVR-AH022A
    CS-AH280 RBCU FAN 65B FAST SPEED GREEN L
    FINAL = OFF

  - OVR-AH023A
    CS-AH279 RBCU FAN 65B SLOW SPEED GREEN L
    FINAL = OFF

  - OVR-EF010A
    CS-EF02 MOTOR DRIVEN EMERG FW PP B(XPP-
    FINAL = OFF

EVENT 1: Perform Stroke test of Pressurizer Block Valves.

The crew will be prepared to take the shift and commence STP-127.001, Pressurizer Block Valve Operability Test. They will perform the test and record the appropriate data for all three Pressurizer Block valves. Once this is complete, the next event may be inserted.

EVENT 2: PT-444 Fails low, heaters turn on, RCS pressure rises.

- **TRIGGER 2**

  - MAL-PRS001A
    PRESSURIZER PRESSURE CHANNEL 444 FAILURE
    FINAL = 1700

On cue from the Examiner, PI-444 will fail low. This will cause all Pressurizer heaters to turn on and increase pressure. Pressure will continue to increase until the PORVs open or the operator takes manual control of Pressurizer heaters and spray. The CRS will enter AOP-401.5, Pressurizer Pressure Control Channel Failure. The CRS will apply TS 3.4.4.4, Relief Valves, Action a, for PCV-444B being inoperable. They will need to close the associated block valve and maintain power to the block valve within 1 hour.

It took six minutes and 28 seconds until PCV-445A and PCV-445B, Pressurizer PORVs, lifted.

EVENT 3: Failure of automatic Turbine Lube Oil Temperature Control.

- **TRIGGER 3**

  - OVR-CW020B
    TC-4211 TURBINE ROOM CLOSE CYCLE HX POT SIGNAL
    FINAL = 0

  - MAL-TUR002A
    TURBINE VIBRATION (BEARING 1)
    SEVERITY = 3.1
    RAMP = 120 sec

  - MAL-TUR002B
    TURBINE VIBRATION (BEARING 3)
    SEVERITY = 5.9
    RAMP = 120 sec

  - MAL-TUR002C
    TURBINE VIBRATION (BEARING 5)
    SEVERITY = 6.1
    RAMP = 120 sec

  - MAL-TUR002D
    TURBINE VIBRATION (BEARING 7)
    SEVERITY = 4.9
    RAMP = 120 sec

  - MAL-TUR002E
    TURBINE VIBRATION (BEARING 9)
    SEVERITY = 5.2
    RAMP = 120 sec

NOTE: Trigger 13: Causes turbine bearing vibrations to lower once temperature controller is in manual and oil temperature has been raised to greater than or equal to 100°F.

- **TRIGGER 13** x11o064m==1 & x11d016m>=100

  - MAL-TUR002A
    TURBINE VIBRATION (BEARING 1)
    SEVERITY = 1.32
    RAMP = 180

  - MAL-TUR002B
    TURBINE VIBRATION (BEARING 3)
    SEVERITY = 4
    RAMP = 180

  - MAL-TUR002C
    TURBINE VIBRATION (BEARING 5)
    SEVERITY = 3.2
    RAMP = 180

  - MAL-TUR002D
    TURBINE VIBRATION (BEARING 7)
    SEVERITY = 2.9
    RAMP = 180

  - MAL-TUR002E
    TURBINE VIBRATION (BEARING 9)
    SEVERITY = 2.9
    RAMP = 180

- **TRIGGER 19** (Resets the Generator Aux Panel Alarm)

  - LOA-TUR005
    GEN AUX PNL XPN-7201 ALARM RESET

On cue from the Examiner the Main Turbine Lube Oil control valve that automatically controls lube oil temperature will fully open. The Main Turbine lube oil temperature will lower as a result of the open control valve and Main Turbine vibrations will rise. The BOP will take manual control of the Turbine Lube Oil controller and will lower cooling flow which will raise oil temp and cause vibrations to lower.

EVENT 4: Feedwater Pump "B" trips, power reduction.

- **TRIGGER 4**

  - MAL- FWM001B
    MAIN FEEDWATER PUMP B TRIP

On cue from the Examiner, the "B" Main Feedwater pump will trip. This failure will result in 2 Main Feedwater Pumps and 4 Feedwater Booster pumps in service at 100% power. The crew will perform immediate actions of AOP-210.3, Feedwater Pump Malfunction. The procedures will require a reduction of Reactor Power to less than 91%. The CRS will implement GOP-4C, Rapid Power Reduction to direct actions needed to accomplish the power reduction. The RO and BOP will perform actions necessary to reduce power from 100% to 90%.

EVENT 5: "A" Steam Generator Tube Leak.

- **TRIGGER 5**

  - MAL- RCS002A
    STEAM GENERATOR A TUBE LEAK
    FINAL VALUE = 120 gpm
    RAMP = 30 sec

- **TRIGGER 20**

  - LOA-CND145
    MN&AUX COND VAC PP CHAR EXH DISCH VALVE – 110
    FINAL = 1

  - LOA-CND144
    MN&AUX COND VAC PUMP ATMOS DISCH VALVE – 109
    FINAL = 0
    Delay = 30 sec

On cue from the Examiner, "A" Steam Generator will develop a tube leak. The crew will enter into AOP-112.2, Steam Generator Tube Leak Not Requiring SI. The crew will isolate Letdown and stabilize Pressurizer level. The CRS will enter Tech Spec 3.4.6.2, Reactor Coolant System Operational Leakage, Action a. Action "a" says With any PRESSURE BOUNDARY LEAKAGE or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. Once the CRS has addressed Tech Specs a Tube Rupture will go in and they will then meet conditions to trip the reactor and actuate Safety Injection.

EVENT 6: "A" Steam Generator Tube Rupture.

- **PRE-LOAD**

  - MAL-EPS013
    GENERATOR BREAKER FAILS TO TRIP

  - OVR-EG020B
    CS-XE01 MAIN FIELD BREAKER CLOSE
    FINAL = TRUE

  - PMP-EF001F
    XPP0021A MOTOR DRIVEN EFW PMP A FAIL TO START

  - MAL-FWM003C
    EMERGENCY FEEDWATER PUMP C TRIP

  - PMP-AH022F
    XFN0030A CNTRL ROOM EMERG FAN A FAIL TO START

  - PMP-AH025F
    XFN0032B CNTRL ROOM VENT FAN B FAIL TO START

  - PMP-AH023F
    XFN0030B CNTRL ROOM EMERG FAN B FAIL TO START

- **TRIGGER 6**

  - MAL- RCS002A
    STEAM GENERATOR A TUBE LEAK
    FINAL VALUE = 600 gpm
    RAMP = 120 sec

  - MAL- RCS002A (new)
    STEAM GENERATOR A TUBE LEAK
    FINAL VALUE = 120 gpm
    DELETE = 1 sec

- **TRIGGER 14 (Allows CR EMERG Fan "A" to start)**

  - PMP-AH022F
    XFN0030A CNTRL ROOM EMERG FAN A FAIL TO START
    DELETE = 1 sec
    X16I036T==1

- **TRIGGER 15 (Allows XFN-32B CR EMERG Fan "B" to start)**

  - PMP-AH025F
    XFN0032B CNTRL ROOM VENT FAN B FAIL TO START
    DELETE = 1 sec
    X16I037T==1

- **TRIGGER 16 (Allows XFN-30B CR EMERG Fan "B" to start)**

  - PMP-AH023F
    XFN0030B CNTRL ROOM EMERG FAN B FAIL TO START
    DELETE = 1 sec
    X16I038T==1

- **TRIGGER 17 (Allows the Generator Breaker to be opened)**

  - MAL-EPS013(NEW)
    GENERATOR BREAKER FAILS TO TRIP
    DELETE = 1 sec
    X12I072T == 1

- **TRIGGER 18 (Starts the Diesel Air Compressor.)**

  - LOA-AUX130
    DIESEL AIR COMPRESSOR LOCAL CONTROL
    FINAL = START

On cue from the Examiner, "A" Steam Generator will develop a tube leak. The crew will enter into AOP-112.2, Steam Generator Tube Leak Not Requiring SI. The crew will attempt to control Pressurizer level and then meet conditions to trip the reactor and actuate Safety Injection. The crew will make their way through EOP-1.0, E-0 Reactor Trip or Safety Injection. The Main Generator and Voltage Regulator breakers will not trip and they must manually trip them. The "A" Motor Driven Emergency Feedwater Pump will fail to auto start and will need to be manually started. The Turbine Driven Emergency Feedwater Pump will trip. The crew will then transition to EOP-4.0, E-3 Steam Generator Tube Rupture.

It takes approximately three minutes to get a red path condition on Heat Sink once the Reactor Trip.

EVENT 7: Steam Generator “A” becomes faulted outside containment.

- **TRIGGER 7** X10I019R == 1(Automatically puts in the SG “A” Fault when they cooldown)

  - MAL- MSS004A
    STEAM GENERATOR A Fault(outside containment)
    FINAL VALUE = 70,000 lbm/hr

  Once the crew starts their cooldown in EOP-4.0, “A” SG will automatically become faulted. After the cooldown, the crew will analyze the faulted Steam Generators pressure and see it is within 250 psig of the intact Steam Generators. The crew will transition to EOP-4.2, ECA-3.1 SGTR With Loss of Reactor Coolant Subcooled Recovery Desired at this point.

CRITICAL TASKS:

- Establish EFW flow before a transition to EOP-15.0, FR-H.1 Response to loss of Secondary Heat Sink is required.
- Take manual control of Pressurizer Heaters and Spray before lifting a Pressurizer PORV.

TERMINATION:

The scenario may be terminated once the crew has depressurized and refilled the Pressurizer in EOP-4.2, ECA-3.1 SGTR With Loss of Reactor Coolant Subcooled Recovery Desired.

Scenario Attributes		Events
Total Malfunctions (5-8)	10	· PT-444 Fails low. · TLO TCV-4211 Fails Open in AUTO · “B” Main Feedwater Pump trip. · SG “A” develops a tube leak. · SG “A” develops a tube rupture. · Main Generator and Voltage Regulator Breakers fail to trip. · Motor Driven EFW Pump “A” shaft shear. · Turbine Driven EFW pump fails to auto start. · SG “A” becomes faulted. · Control Room Emergency Ventilation doesn't start automatically.
Malfunctions after EOP entry (1-2)	4	· Main Generator and Voltage Regulator Breakers fail to trip. · “A” Motor Driven EFW pump develops a sheared shaft. · Turbine Driven EFW pump fails to auto start. · Control Room Emergency Ventilation doesn't start automatically.
Abnormal Events (2-4)	4	· PT-444 Fails low. · TLO TCV-4211 Fails Open in AUTO · “B” Main Feedwater Pump trip. · SG “A” tube leak.
Major Transients (1-2)	2	· SG “A” tube leak that becomes ruptured. · SG “A” becomes faulted.
EOPs Entered (1-2)	2	· EOP-4.0, E-3 Steam Generator Tube Rupture. · EOP-4.2, ECA-3.1 SGTR With Loss of Reactor Coolant Subcooled Recovery Desired.
EOP Contingencies (0-2)	1	· EOP-4.2, ECA-3.1 SGTR With Loss of Reactor Coolant Subcooled Recovery Desired.
Critical Tasks (2-3)	2	· Establish EFW flow before a transition to EOP-15.0, FR-H.1 Response to loss of Secondary Heat Sink is required. · Take manual control of Pressurizer Heaters and Spray before lifting a Pressurizer PORV.

SIMULATOR SCENARIO SETUP

INITIAL CONDITIONS:

- IC Set 301
- 100% Power, MOL
- Burnup = 10,030 MWD/MTU
- RCS Boron Concentration = 993 ppm
- FCV-113 Pot Setting = 4.28
- Rod Position: Group D = 228
- Tavg = 587.4˚F
- Xe = -2731.6 pcm
- Prior to the scenario, the crew should pre-brief conditions and their expectations for the shift.

PRE-EXERCISE:

- Ensure simulator has been checked for hardware problems (DORT, burnt out light bulbs, switch malfunctions, chart recorders, etc.).
- Complete VCS-TQP-0807 Attachment I-A, Unit 1 Booth Instructor Checklist.
- Verify plant aligned for "B1" work week IAW PTP-101.004, Safety Related Train Swap Checklist.
- Verify red Placard on “A” CCW Pump and “B” Charging Pump.
- Verify red hold tag and R&R tag on "B" MDEFW Pump **AND** XFN-0065B RBCU and ensure they are in P-T-L. XFN-65B Fast speed can’t be taken to P-T-L.
- Verify the Hard Card for Turbine operation is in its proper storage location and cleaned.
- Verify the Hard Card for borating via MVT-8104 is in its proper storage location and cleaned.
- Verify Rod Bank Update set correctly: 228 steps on all Banks.
- Update EOOS for “B” MDEFW Pump being out of service.
- Ensure you have the following procedure:
    - STP-127.001, PRESSURIZER BLOCK VALVE OPERABILITY TEST
- Ensure you have a turnover sheet for each position.
- Conduct two-minute drill.

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 1	Page: 12 of 52

Event Description: Perform STP-127.001, Pressurizer Block Valve Operability Test.

Time	Position	Applicant's Actions or Behavior
EVALUATOR NOTE: The crew will have briefed the stroke test of the Pressurizer Block Valves prior to assuming the watch.		
BOOTH OPERATOR:		No TRIGGERS for this event.
Indications Available: N/A		
NOTE 6.3, 6.4, 6.5 To prevent preconditioning, the Block Valves open and closed exercises may be performed out of sequence.		
	BOP	6.3. Stroke test XVG08000A-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, as follows: a. Verify PCV-445A, PWR RELIEF, is closed. **(PEER ✓)** b. Close MVG-8000A, RELIEF 445A ISOL, and measure the stroke time to the nearest 0.1 second, from switch actuation until the closed light is lit and the open light goes out. *c. Record the stroke time for XVG08000A-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, on Attachment I. d. Open MVG-8000A, RELIEF 445A ISOL, and measure the stroke time to the nearest 0.1 second, from switch actuation until the open light is lit and the closed light goes out. *e. Record the stroke time for XVG08000A-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, on Attachment I.
	BOP	6.4. Stroke test XVG08000B-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, as follows: a. Verify PCV-444B, PWR RELIEF, is closed. **(PEER ✓)** b. Close MVG-8000B, RELIEF 444B ISOL, and measure the stroke time to the nearest 0.1 second, from switch actuation until the closed light is lit and the open light goes out. *c. Record the stroke time for XVG08000B-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, on Attachment I. d. Open MVG-8000B, RELIEF 444B ISOL, and measure the stroke time to the nearest 0.1 second, from switch actuation until the open light is lit and the closed light goes out. *e. Record the stroke time for XVG08000B-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, on Attachment I.

STP-127.001

STP-127.001

STP-127.001

Op Test No: NRC-ILO-16-01 Scenario # 2 Event # 1 Page: 13 of 52			
Event Description: Perform STP-127.001, Pressurizer Block Valve Operability Test.			
Time	Position	Applicant's Actions or Behavior	
	BOP	6.5. Stroke test XVG08000C-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, as follows: a. Verify PCV-445B, PWR RELIEF, is closed. **(PEER ✓)** b. Close MVG-8000C, RELIEF 445B ISOL, and measure the stroke time to the nearest 0.1 second, from switch actuation until the closed light is lit and the open light goes out. *c. Record the stroke time for XVG08000C-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, on Attachment I. d. Open MVG-8000C, RELIEF 445B ISOL, and measure the stroke time to the nearest 0.1 second, from switch actuation until the open light is lit and the closed light goes out. *e. Record the stroke time for XVG08000C-RC, PRZ PWR OPER RELIEF INLET ISOL VLV, on Attachment I.	STP-127.001
	BOP	*6.6. Perform a REQUIRED OPERABLE POSITION equipment lineup per Attachment II.	STP-127.001
	RO	*6.7. Perform a REQUIRED OPERABLE POSITION equipment lineup independent verification per Attachment II.	STP-127.001
EVALUATOR NOTE: The next event may be inserted following completion of the PORV Block Valve test, or at any time per the discretion of the Lead Examiner.			

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 2	Page: 14 of 52
Event Description: PT-444 fails LOW.			

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: PT-444 will fail LOW causing Pressurizer Heaters to turn on and RCS pressure to rise.			
BOOTH OPERATOR:		When directed - Initiate Event 2 (TRIGGER 2).	
Indications Available: XCP-616, 3-6, PZR PCS LO BU HTRS ON			
	CRS	Enters AOP-401.5, Pressurizer Pressure Control Channel Failure.	AOP-401.5
NOTE Throughout this procedure, "AFFECTED" refers to any PZR PORV that is controlled by the failed instrument.			AOP-401.5
IOA	RO	1. Verify the PZR PORVs are closed: • PCV-445A, PWR RELIEF. • PCV-445B, PWR RELIEF. • **PCV-444B, PWR RELIEF. (AFFECTED PORV)**	AOP-401.5
IOA	RO	2. Check if PI-444, CNTRL CHAN PRESS PSIG, indication is NORMAL. **(NO)**	AOP-401.5
IOA **CRITICAL TASK**	RO	**Alternative Action Step:** 2. IF PT-444 is failed, THEN perform the following: a. Place both PZR Spray Valves in MAN and closed: • PCV-444C, PZR SPRAY. • PCV-444D, PZR SPRAY. b. Place the PZR PRESS MASTER CONTROL in MAN. c. Operate the PZR Heaters and Spray Valves in manual to control RCS pressure between 2220 psig and 2250 psig.	AOP-401.5
EVALUATOR NOTE: Crews may control spray valves in manual independently or may place them in Auto and control them using the Master Pressure Controller in Manual.			
	RO	3. Ensure ROD CNTRL BANK SEL Switch is in AUTO.	AOP-401.5
	RO	4. Place the switch for the AFFECTED PZR PORV(s) in CLOSE: • PCV-445A, PWR RELIEF. • PCV-445B, PWR RELIEF. • **PCV-444B, PWR RELIEF. (AFFECTED PORV)**	AOP-401.5

Op Test No: NRC-ILO-16-01 Scenario # 2 Event # 2 Page: 15 of 52

Event Description: PT-444 fails LOW.

Time	Position	Applicant's Actions or Behavior	
		NOTE - Step 5 Operations management decision is to conservatively declare the AFFECTED PORV(s) inoperable.	AOP-401.5
	RO	5. Within one hour of the instrument failure, close the affected PORV Block Valve: • MVG-8000A, RELIEF 445 A ISOL. • MVG-8000C, RELIEF 445 B ISOL. • **MVG-8000B, RELIEF 444 B ISOL. (AFFECTED PORV BLOCK Valve)**	AOP-401.5
	RO	6. If plant conditions allow, place the PZR Spray Valves in AUTO: • PCV-444C, PZR SPRAY. • PCV-444D, PZR SPRAY.	AOP-401.5
*	RO	7. Maintain RCS pressure between 2220 psig and 2250 psig.	AOP-401.5
	CRS	8. Notify the I&C Department to determine and correct the cause of the channel failure.	AOP-401.5
	CRS	9. WHEN the failed channel has been returned to service, THEN continue with this procedure.	AOP-401.5
	CRS	Enters T.S. 3.4.4.4, Relief Valves, Action a: With one or more PORV(s) inoperable and capable of being manually cycled, within 1 hour: 1. restore the PORV(s) to OPERABLE status or 2. close the associated block valve(s) and maintain power to the block valve; Otherwise, be in at least HOT STANDBY Within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.	Tech Specs
		EVALUATOR NOTE: The next event may be inserted following the CRS assessment of Tech Specs, or at any time per the discretion of the Lead Examiner.	

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 3	Page: 16 of 52

Event Description: Failure of automatic Turbine Lube Oil Temperature Control.

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: The automatic temperature controller for the Main Turbine lube oil control valve will fail causing the flow control valve to fully open. The Main Turbine lube oil temperature will lower as a result of the open control valve and Main Turbine vibrations will rise. The BOP will take manual control of the Turbine Lube Oil controller and will lower cooling flow which will raise oil temp and cause vibrations to lower. Once the event is put in, it takes approximately 2 minutes until you get the first alarm.			
BOOTH OPERATOR:		When directed - Initiate Event 3 (TRIGGER 3).	
Available Indications: XCP-632, 1-4, TURB SUPERVISORY INSTR. XCP-632, 4-2, MN TURB VIB HI. XCP-632, 4-2, MN TURB VIB HI			
	BOP	Respond to ANNUNCIATOR XCP-632, 4-2, MN TURB VIB HI.	
	BOP	1. Evacuate all unnecessary personnel from the Turbine Building.	XCP-632, 4-2
	BOP	2. Monitor Main Turbine vibration levels: a. At the HMI: Select Monitor and as applicable: 1. LP Hoods TEMP 2. Lube – Hyd Oil. 3. Vibration. 4. Prox. b. IPCS, type in TURBRG.	XCP-632, 4-2
	BOP	(see table below) 3. If any of the above vibration trip conditions are exceeded, perform the following: a. Trip the Main Turbine. b. Implement AOP-214.1 while monitoring for indications of imminent Turbine damage per Step 4.	XCP-632, 4-2

SPEED	TRIP IMMEDIATELY IF JOURNAL 1-8 VIBRATION EXCEEDS	TRIP IMMEDIATELY IF JOURNAL 9-10 VIBRATION EXCEEDS
LESS THAN 800 RPM	8 MILS	8 MILS
800-1800 RPM	11 MILS	11 MILS

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 3	Page: 17 of 52

Event Description: Failure of automatic Turbine Lube Oil Temperature Control.

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: The highest bearing vibration is Bearing 5 at 6.1 mils per the scenario design. The lowest vibration level that calls for turbine trip is 9 mils on bearing 9 or 10.			
	BOP	5. If a Turbine trip is **NOT** required perform the following: a. Monitor TI-4211, TURB OIL CLR TEMP °F, to determine if Turbine oil temperature is in the normal operating range between 100°F and 120°F. b. If required, change Turbine load per SOP-214 to reduce vibration levels. c. Dispatch an operator to verify oil flow to Turbine bearings. d. Monitor IPCS Group Display (TSI). e. On the Turbine HMI, select Monitor/LP Hoods and monitor Exhaust Hood A/B Temperature. f. Verify proper MSR operation per SOP-204.	XCP-632, 4-2
	BOP	Supplemental Actions: 2. Refer to SOP-215 for abnormal oil temperature control.	XCP-632, 4-2
BOOTH OPERATOR:		• Call up the TURBRG screen in SIPCS and be prepared to report back Bearing Oil temperatures as displayed when called. • When contacted to check Main Turbine oil cooler conditions wait 2 minutes and report as an AO; "All bearing oil temperatures are reading (as displayed) °F"	XCP-632, 4-2
BOOTH OPERATOR:		If called to reset the GEN AUX PNL alarm, insert TRIGGER 19 and report back that the GEN AUX PNL alarm has been reset.	

Op Test No: NRC-ILO-16-01	Scenario # 2 Event # 3	Page: 18 of 52
Event Description: Failure of automatic Turbine Lube Oil Temperature Control.		

Time	Position	Applicant's Actions or Behavior
	BOP	2.2. If TI-4211, TURB OIL CLR TEMP °F, reads less than 100°F, perform the following: a. Verify temperature on the following (TB-436): 1. ITI15633, MN TB LUBE OIL CLR LO INLET TEMP IND. 2. ITI15634, MN TB LUBE OIL CLR LO OUT TEMP IND. 3. ITI04197, MAIN TURB OIL CLR A TC OUTLET TEMP IND. 4. ITI04207, MAIN TURB OIL CLR B TC OUTLET TEMP IND. b. Verify the position of ITV04211-TC, TURBINE OIL CLR TC OUTLET TEMP CONT VLV, and perform the following as necessary (TB-412): 1. If the valve is not fully closed, take manual control of the valve and restore temperature to normal.
BOOTH OPERATOR:		When contacted about ITV-4211 position, wait 2 minutes and report as an AO: "Valve appears to be fully open with no obvious problem with the operator".
EVALUATOR NOTE: After the BOP takes manual control of the TLO Temp Controller and restores TLO temp (control in manual with Lube Oil temp > 100°F) high turbine vibrations will ramp back to normal. The next Event may be inserted after Turbine Lube Oil Temperature control has been re-established or at any time per the discretion of the Lead Examiner.		

SOP-215

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 4	Page: 19 of 52

Event Description: Main Feedwater Pump "B" Trip, down power to 90%.

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: The "B" Main Feedwater pump will trip. This failure will result in 2 Main Feedwater Pumps and 4 Feedwater Booster pumps in service at 100% power. The crew will refer to XCP-625 2-1, which will require a reduction of Reactor Power to less than 91%.			
BOOTH OPERATOR:		When directed - Initiate Event 2 (TRIGGER 2).	
Indications Available: XCP-625 2-1, FWP A/B/C TRIP			
	CRS	Enters ARP-001- XCP-625 2-1.	XCP-625 2-1
	CRS	CORRECTIVE ACTIONS: 1. Reduce Reactor power per GOP-4C, Rapid Power Reduction, below the following limits: a. With 2 Feedwater Pumps in operation, 91% Reactor power. 2. Go to AOP-210.3, Feedwater Pump Malfunction.	XCP-625 2-1

REFERENCE PAGE FOR AOP-210.3

1 MAIN FEEDWATER REGULATING VALVE MANUAL CONTROL

Manual Control of Main Feedwater Regulating Valves is permissible at any time as deemed necessary during the performance of this procedure. If a Main Feedwater Regulating Valve has been placed in Manual it should be returned to Automatic as soon as possible.

2 LOSS OF MAIN FEEDWATER FLOW

IF Feedwater flow is lost while Reactor Power is GREATER THAN 10%, and cannot be quickly restored from the MCB, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.

3 STEAM GENERATOR LEVEL CONTROL

a. IF Narrow Range Steam Generator Level decreases to LESS THAN 40% in any SG, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.

b. IF Reactor Power is GREATER THAN 15% and Narrow Range Steam Generator Level exceeds 75% in any SG, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.

4 MAIN FEED PUMP SPEED CONTROL

IF IPT00464 has failed with the Steam Dumps in Steam Pressure Mode, THEN Trip the reactor and GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1.

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 4	Page: 20 of 52

Event Description: Main Feedwater Pump "B" Trip, down power to 90%.

Time	Position	Applicant's Actions or Behavior	
IOA	BOP	1. Verify at least one Feedwater Pump is running.	AOP-210.3
IOA	BOP	2. Check if a Feedwater Pump trip has occurred.	AOP-210.3
IOA	BOP	3. GO TO Step 14.	AOP-210.3
BOOTH OPERATOR:		When called to check out "B" Main Feedwater Pump, wait 2 minutes and report back "The pump is tripped with no visible problems".	
*	BOP	14. Verify Narrow Range Steam Generator Level in all Steam Generators is GREATER THAN 40%.	AOP-210.3
	BOP	15. Verify two Feedwater Pumps are running.	AOP-210.3
	CRS	16. Reduce Reactor Power to LESS THAN 91% Reactor Power at a maximum rate of 3%/minute. REFER TO GOP-4C, RAPID POWER REDUCTION.	AOP-210.3
*	BOP	17. Verify Narrow Range Steam Generator Levels are stable at or trending to 60%.	AOP-210.3
	BOP	18. Verify the high pressure and low pressure stop valves close on the AFFECTED Feedwater Pump (GRAPHIC 310 SCREEN).	AOP-210.3
	BOP	19. Verify proper operation of the AFFECTED Feedwater Pump Turning Gear: a. Check if the Feedwater Pump Turbine is stopped. b. Ensure the Feedwater Pump Turning Gear is engaged and running.	AOP-210.3
	BOP	20. Place Main Feed Regulating valves in AUTO.	AOP-210.3
	BOP	21. Restore Main Feedwater System to pre-event conditions. a. Determine and correct the cause of the Feedwater Pump trip. b. Start the AFFECTED Feedwater Pump.	AOP-210.3
	CRS	Enters GOP-4C, RAPID POWER REDUCTION.	

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 4	Page: 21 of 52

Event Description: Main Feedwater Pump "B" Trip, down power to 90%.

Time	Position	Applicant's Actions or Behavior	

GOP-4C REFERENCE PAGE

GENERAL NOTES

A. Procedure steps should normally be performed in sequence. However, it is acceptable to perform steps in advance after thorough evaluation of plant conditions and impact by the Shift Supervisor or Control Room Supervisor.

B. After any Thermal Power change of greater than 15% within any one hour, Attachment III.H. of GTP-702 must be completed.

C. If Reactor Power is stabilized during this procedure for the purpose of raising power per GOP-4A, a Power Range Heat Balance shall be performed.

D. Once a Rapid Power Reduction has begun, every effort should be made to prevent the Turbine from reaching "AT SET LOAD" unless it is desired to stabilize the plant.

REACTOR CONTROL

A. During operation with a positive Moderator Temperature Coefficient, power and temperature changes will require constant operator attention.

B. Rod Control should be maintained in Automatic if any Pressurizer PORV is isolated.

C02→ C. If at any time, power decreases unexpectedly below 0.1% on any Power Range NI (computer indication available) OR below 1.0% on any Power Range NI control board indication (computer not available):

1) No positive reactivity will be added by rods or dilution.

2) A complete reactor shutdown shall be performed per GOP-5.

3) A controlled reactor startup may be commenced per GOP-3 once the event has been reviewed by Reactor Engineering.

REACTOR TRIP CRITERIA DURING RAPID LOAD REDUCTION

A. If any of the following conditions occur, trip the Reactor and implement EOP-1.0:

1) RCS T_{avg} is less than 551°F for greater than 15 minutes.

2) T_{avg}/T_{ref} mismatch exceeds 10°F.

3) Pressurizer pressure approaches 1870 psig.

4) Power reduction at 5% per minute is not sufficient to mitigate the event.

GOP-4C

NOTE 3.0

If time allows, load reductions should be discussed with the Load Dispatcher.

GOP-4C

CAUTION 3.1 through 3.12

a. Thermal Power changes of greater than 15% in any one-hour period requires completion of GTP-702 Attachment III.H.

b. VCS PID Report, POWER CHANGE SEARCH, should be periodically performed to ensure a thermal power change of greater than 15% in any one-hour period is detected.

GOP-4C

Time	Position	Applicant's Actions or Behavior	
	RO	3.1. Commence rapid Plant Shutdown as follows: a. Energize all Pressurizer Heaters.	GOP-4C

EVALUATOR NOTE: The boration volume required will be in accordance with Reactivity Plans provided at turnover. The crew may use MVT-8104 and Emergency Borate. If the crew uses SOP-106 to borate, procedure guidance for borating is found in SOP-106, Reactor Makeup Water System is included in this scenario guide beginning at **page 50 of 52**.

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 4	Page: 22 of 52
Event Description: Main Feedwater Pump "B" Trip, down power to 90%.			

Time	Position	Applicant's Actions or Behavior	
		NOTE 3.1.b Setting FCV-113A&B, BA FLOW SET PT to 8.3 will yield 33 gpm Boration flow rate.	GOP-4C
	RO	3.1. Commence rapid Plant Shutdown as follows: b. Maintain the following with rod motion or boron concentration changes: 1. Tavg within 10°F and trending to Tref. 2. ΔI within limits. 3. Control Rods above the rod insertion limit.	GOP-4C
	BOP	3.1. Commence rapid Plant Shutdown as follows: c. Using the Turbine HMI, Control/Load screen, reduce to the desired load, as low as 5% (50 MWe), as follows: 1. Under Rate %/min, select desired ramp rate up to 5% per minute. 2. Select Load (a dialog box opens). 3. Enter desired load. 4. Select OK. 5. Confirm setpoint. 6. Select OK. 7. Verify proper plant response.	GOP-4C
	BOP	3.2. Continue Rapid Plant Shutdown as follows: a. Adjust Megavars using GEN FIELD VOLT ADJ to maintain less than 300 MVARS.	GOP-4C

Op Test No: NRC-ILO-16-01 Scenario # 2 Event # 4 Page: 23 of 52			
Event Description: Main Feedwater Pump "B" Trip, down power to 90%.			
Time	**Position**	**Applicant's Actions or Behavior**	
BOOTH OPERATOR:		If called to adjust Blowdown Cooler flow use the following remotes: • LOA-CND044, COND TO S/G BD TC-3062A AUTO-MANUAL MODE SELECTOR - position to MANUAL • LOA-CND045, COND TO S/G BD TC-3062B AUTO-MANUAL MODE SELECTOR - position to MANUAL • LOA-CND046, COND TO S/G BD TC-3062C AUTO-MANUAL MODE SELECTOR - position to MANUAL • LOA-CND-047, COND TO S/G BD TV-3062A MANUAL POSITION – adjust final value to obtain flow as directed. • LOA-CND-048, COND TO S/G BD TV-3062B MANUAL POSITION – adjust final value to obtain flow as directed. • LOA-CND-049, COND TO S/G BD TV-3062C MANUAL POSITION – adjust final value to obtain flow as directed.	GOP-4C
	BOP	3.2. Continue Rapid Plant Shutdown as follows: b. When Reactor Power is less than 90% reduce to two operating Feedwater Pumps by placing the desired FWP A(B)(C) TRIP/RESET switch in TRIP (MCB).	GOP-4C
NOTE 3.2.c It is preferred to maintain D FWBP in operator for better Start-Up drain flow.			GOP-4C
	BOP	3.2. Continue Rapid Plant Shutdown as follows: c. When only two Main Feedwater Pumps are operating reduce to three Feedwater Booster Pumps.	GOP-4C
	BOP	3.2. Continue Rapid Plant Shutdown as follows: d. Perform one of the following: 1. Adjust ITV-3062A(B)(C), BD COOLER A(B)(C) CDSTE OUT TEMP to raise total Blowdown heat exchanger flow to between 1000 gpm and 1100 gpm (AB-436'). OR 2. Secure Steam Generator Blowdown per SOP-212.	GOP-4C
	BOP	3.2. Continue Rapid Plant Shutdown as follows: e. Adjust FC-3136, FLOW TO DEAERATOR setpoint, to maintain DA level between 2.5 and 5.0 feet narrow range level (raising potentiometer set point raises DA level).	GOP-4C
EVALUATOR NOTE: The next event may be inserted following completion of the power reduction, or at any time per the discretion of the Lead Examiner.			

Op Test No:	NRC-ILO-16-01	Scenario #	2	Event #	5	Page:	24	of	52

Event Description: Tube Leak on SG “A”

Time	Position	Applicant’s Actions or Behavior	
EVALUATOR NOTE: “A” SG will develop a tube leak.			
BOOTH OPERATOR:		When directed - Initiate Event 5 (TRIGGER 5).	
Available Indications: XCP-646 2-1, MN STM LINE RM-G19 HI RAD XCP-646 3-1, MN STM LINE RM-G19A TRBL			
	CRS	Enters AOP-112.2, Steam Generator Tube Leak Not Requiring SI.	
NOTE • Conditions for implementing Emergency Plan Procedures should be evaluated using EPP-001, ACTIVATION AND IMPLEMENTATION OF EMERGENCY PLAN. • Due to N-16 gamma radiation effects, RM-G19A(B)(C), STMLN HI RNG GAMMA, will display elevated readings and should not be used for classification of EAL while the Reactor is critical.			AOP-112.2
*	RO	1. Check if PZR level can be maintained: a. Fully open FCV-122, CHG FLOW. b. Verify PZR level is STABLE OR INCREASING. **(NO)** c. Control FCV-122, CHG FLOW, as necessary to maintain PZR level.	AOP-112.2
*	RO	**Alternative Action Step:** 1. Check if PZR level can be maintained: b. Perform the following: 1. Isolate Letdown as follows: a. Close PVT-8149A(B)(C), LTDN ORIFICE A(B)(C) ISOL. 2. IF PZR level continues to decrease, THEN perform the following: a. Start a second CCW pump. b. Start a second Charging Pump.	AOP-112.2
BOOTH OPERATOR:		If called to align Condenser Exhaust Gas to Aux Building Charcoal Exhaust, Wait 2 minutes, insert TRIGGER 20 and report back that “Condenser Exhaust Gas is aligned to Aux Building Charcoal Exhaust”.	

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 5	Page: 25 of 52
Event Description: Tube Leak on SG "A"			

Time	Position	Applicant's Actions or Behavior	
*	RO	2. Check if SI is required: a. Check if any of the following criteria are met: • PZR level is decreasing with Charging maximized and Letdown isolated. OR • PZR level is approaching 8%. OR • PZR pressure is approaching 1870 psig.	AOP-112.2
	RO	3. Verify VCT level is being maintained between 20% and 40%.	AOP-112.2
	CRS	4. IF Steam Generator primary to secondary tube leakage has not been determined, THEN perform the following: a. Estimate the RCS leak rate. Refer to IPCS CHGNET. b. Calculate the RCS leak rate. REFER TO STP-114.002, OPERATIONAL LEAK TEST. c. Comply with the applicable Tech Spec 3.4.6.2 action statement.	AOP-112.2
	CRS	Enters Tech Spec 3.4.6.2. RCS Operational Leakage, Action a: With any PRESSURE BOUNDARY LEAKAGE or with primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.	Tech Specs
	CRS	5. Notify Chemistry to sample all SG secondary sides for activity.	AOP-112.2
	BOP	6. Check if RM-A9 is available.	AOP-112.2
	CRS	7. GO TO Step 12.	AOP-112.2
	CRS	12. Check if Steam Generator Tube Leakage is GREATER THAN OR EQUAL TO 75 gpd (0.05 gpm) using one of the following: • UR1019, S/G LEAKAGE FROM RMA9 (in gpd). OR • RM-A9, using the RM-A9 Total Count Rate vs. Calculated Primary to Secondary Leakrate graph.	AOP-112.2
*		13. Evaluate plant shutdown requirements per ATTACHMENT 3, PLANT SHUTDOWN REQUIREMENTS BASED ON RMA-9 EVALUATION.	AOP-112.2

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 5	Page: 26 of 52
Event Description: Tube Leak on SG "A"			

Time	Position	Applicant's Actions or Behavior	

PLANT SHUTDOWN REQUIREMENTS BASED ON RMA-9 EVALUATION

IPCS METHOD (Preferred)

Steam Generator primary to secondary tube leakage rate, and rate of increase, is represented by the following IPCS Computer points:

- UR1019, S/G LEAKAGE FROM RMA9 (in GPD)
- UR1019-R, S/G LEAKAGE FROM RMA9-RATE (in GPD/HR)

1. Obtain the Steam Generator Tube Leakage rate of change from the IPCS.
2. Determine the shutdown requirements using Steam Generator Tube Leakage rate of change to the flow chart and table on page 2.

RM-A9 METER (IPCS not available)

1. Obtain the RM-A9 Total Count Rate vs. Calculated Primary to Secondary Leakrate graph from the RAD MON SETPOINTS AND VALVE LOCATIONS book in the Control Room.
2. Record initial RM-A9 reading: ________________ CPM.
3. Using the RM-A9 Total Count Rate vs. Calculated Primary to Secondary Leakrate graph, convert the RM-A9 initial reading to the initial Leakrate: ________________ GPD. AOP-112.2, Attachment 3
4. Wait 15 minutes.
5. Record 15 minute RM-A9 reading: ________________ CPM.
6. Using the RM-A9 Total Count Rate vs. Calculated Primary to Secondary Leakrate graph, convert the RM-A9 15 minute reading to the 15 minute Leakrate: ________________ GPD.
7. Determine the Steam Generator Tube Leakage rate of change in GPD/HR:

   a. Calculate the rate of Steam Generator Tube Leakage change in a 15 minute period:

      ________________ - ________________ = ________ GPD/15 min.
      (15 minute Leakrate - initial Leakrate) = GPD/15 min.

   b. Multiply the GPD/15 min. leakrate change by 4:

      ________________ * 4 = ________ GPD/HR.
      GPD/15 min. * 4 = GPD/HR

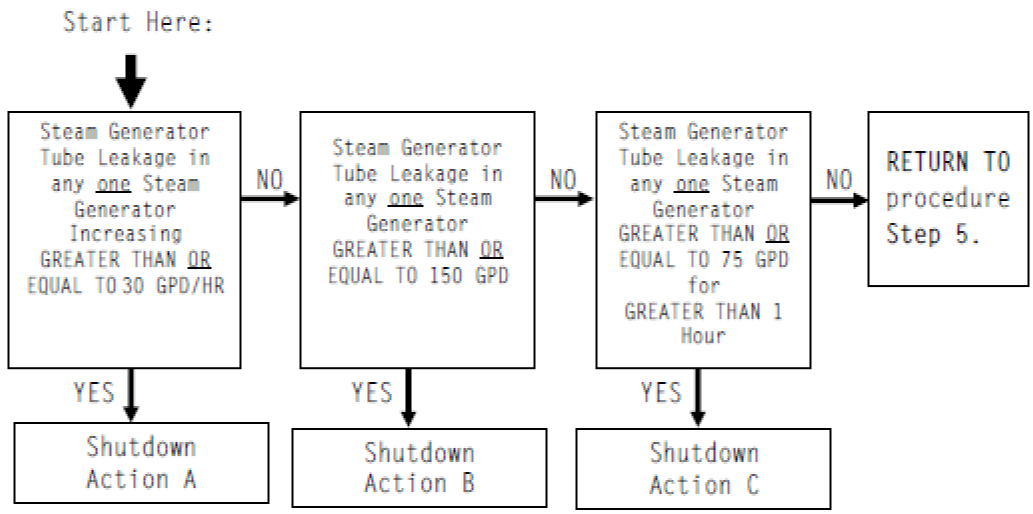
8. Determine the shutdown requirements using Steam Generator Tube Leakage rate of change to the flow chart and table on page 2.

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 5	Page: 27 of 52
Event Description: Tube Leak on SG "A"			

Time	Position	Applicant's Actions or Behavior

PLANT SHUTDOWN REQUIREMENTS BASED ON RMA-9 EVALUATION

NOTE: If unable to determine leakage from individual steam generators, the total leakage is assumed to be coming from one steam generator.



AOP-112.2, Attachment 3

Shutdown Action	Requirement:
A	Be ≤ 50% power within <u>one</u> hour at 1% or 3% per minute and be in Mode 3 within the next <u>two</u> hours. (Total of <u>three</u> hours)
B	Be in Mode 3 within <u>six</u> hours at 1/2% or 1% per minute.
C	Be in Mode 3 within 23 hours at 1/2% or 1% per minute. (Total of 24 hours after exceeding 75 GPD)

EVALUATOR NOTE: The next event may be initiated once the CRS has addressed Tech Specs or at any time per the discretion of the Lead Evaluator.

Op Test No: NRC-ILO-16-01 Scenario # 2 Event # 6 Page: 28 of 52

Event Description: SG Tube Rupture on “A” SG that becomes Faulted during the cooldown.

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: “A” SG will develop a tube rupture.			
BOOTH OPERATOR:		When directed - Initiate Event 6 (TRIGGER 6).	
Available Indications: Pressurizer Level/Pressure lowering. “A” SG level rising uncontrollably.			
*	CRS/RO	Applies continuous action step 1 of AOP-112.2: 1. Check if PZR level can be maintained: a. Fully open FCV-122, CHG FLOW. b. Verify PZR level is STABLE OR INCREASING. **(NO)**	AOP-112.2
*	RO	**Alternative Action step:** b. Perform the following: 1. Isolate Letdown as follows: a. Close PVT-8149A(B)(C), LTDN ORIFICE A(B)(C) ISOL. 2. IF PZR level continues to decrease, THEN perform the following: a. Start a second CCW pump. b. Start a second Charging Pump.	AOP-112.2
*	CRS	Applies continuous action step 2 of AOP-112.2: 2. Check if SI is required: a. Check if any of the following criteria are met: • PZR level is decreasing with Charging maximized and Letdown isolated. OR • PZR level is approaching 8%. OR • PZR pressure is approaching 1870 psig. b. Perform the following: 1. Trip the Reactor. 2. GO TO EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION. WHEN EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION Immediate Actions are complete, THEN actuate SI.	AOP-112.2
	CRS	Enters EOP-1.0, E-0 Reactor Trip or Safety Injection	EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 29 of 52

Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.

Time	Position	Applicant's Actions or Behavior	
		NOTE Steps 1 through 5 are Immediate Operator Actions.	EOP-1.0
IOA	RO	1. Verify Reactor Trip: • Trip the Reactor using either Reactor Trip Switch. • Verify all Reactor Trip and Bypass Breakers are open. • Verify all Rod Bottom Lights are lit. • Verify Reactor Power level is decreasing.	EOP-1.0
IOA	BOP	2. Verify Turbine/Generator Trip: a. Verify all Turbine STM STOP VLVs are closed. b. Ensure Generator Trip (after 30 second delay): 1. Ensure the GEN BKR is open. 2. Ensure the GEN FIELD BKR is open. 3. Ensure the EXC FIELD CNTRL is tripped.	EOP-1.0
IOA	BOP	3. Verify both ESF buses are energized.	EOP-1.0
IOA	RO	4. Check if SI is actuated: a. Check if either: • SI ACT status light is bright on XCP-6107 1-1. OR • Any red first-out SI annunciator is lit on XCP-626 top row. b. Actuate SI using either SI ACTUATION Switch. c. GO TO Step 6.	EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 30 of 52

Event Description: SG Tube Rupture on “A” SG that becomes Faulted during the cooldown.

Time	Position	Applicant’s Actions or Behavior

REFERENCE PAGE FOR EOP-1.0

1 RCP TRIP CRITERIA

a. IF Phase B Containment Isolation has actuated (XCP-612 4-2), THEN trip all RCPs.

b. IF both of the following conditions occur, THEN trip all RCPs:

- SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

AND

- RCS Wide Range pressure is LESS THAN 1418 psig.

2 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.

EOP-1.0

3 MONITOR SPENT FUEL COOLING

Periodically check status of Spent Fuel Cooling by monitoring the following throughout event recovery:

- Spent Fuel Pool level.
- Spent Fuel Pool temperature.

4 RUPTURED STEAM GENERATOR

IF a RUPTURED Steam Generator has been positively identified, THEN throttle EFW to the RUPTURED Steam Generator WHEN its Narrow Range Level is GREATER THAN 26%[41%].

5 FAULTED STEAM GENERATOR

- IF a FAULTED Steam Generator has been positively identified, THEN isolate EFW to the faulted Steam Generator as soon as possible UNLESS all three Steam Generators are FAULTED.
- IF all three Steam Generators are FAULTED, THEN throttle EFW flow to all three Steam Generators to 50 gpm.

Op Test No: NRC-ILO-16-01 Scenario # 2 Event # 6 Page: 31 of 52

Event Description: SG Tube Rupture on “A” SG that becomes Faulted during the cooldown.

Time	Position	Applicant’s Actions or Behavior	
	BOP	6. Initiate ATTACHMENT 3, SI EQUIPMENT VERIFICATION.	EOP-1.0
EVALUATOR NOTE: EOP-1.0, Attachment 3 can be found on **page 48 of 52.**			
	CRS	7. Announce plant conditions over the page system.	EOP-1.0
*	RO	8. Verify RB pressure has remained LESS THAN 12 psig on PR-951, RB PSIG (P-951), red pen.	EOP-1.0
	RO	9. Check RCS temperature: • With any RCP running, RCS Tavg is stable at OR trending to 557°F. **(NO)** OR • With no RCP running, RCS Tcold is stable at OR trending to 557°F.	EOP-1.0
*	RO	**Alternative Action Step:** 9. IF RCS temperature is LESS THAN 557°F AND decreasing, THEN stabilize temperature by performing the following as required: a. Close IPV-2231, MS/PEGGING STM TO DEAERATOR. b. Perform one of the following: • IF Narrow Range SG level is LESS THAN 26% [41%] in all SGs, THEN reduce EFW flow as necessary to stop cooldown, while maintaining total EFW flow GREATER THAN 450 gpm. OR • WHEN Narrow Range SG level is GREATER THAN 26% [41%] in at least one SG, THEN control EFW flow as necessary to stabilize RCS temperature at 557°F. c. Initiate ATTACHMENT 6, STEAM VALVE ISOLATION, while continuing with this procedure. d. IF RCS cooldown continues, THEN close: • MS Isolation Valves, PVM-2801A(B)(C). • MS Isolation Bypass Valves, PVM-2869A(B)(C).	EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 32 of 52
Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.			

Time	Position	Applicant's Actions or Behavior	
	RO	10. Check PZR PORVs and Spray Valves: a. PZR PORVs are closed. b. PZR Spray Valves are closed. c. Verify power is available to at least one PZR PORV Block Valve: • MVG-8000A, RELIEF 445 A ISOL. • MVG-8000B, RELIEF 444 B ISOL. • MVG-8000C, RELIEF 445 B ISOL. d. Ensure one of the following Block Valves is open unless it was closed to isolate an open PZR PORV: • MVG-8000A, RELIEF 445 A ISOL. • MVG-8000B, RELIEF 444 B ISOL.	EOP-1.0
NOTE - Step 11 Seal Injection flow should be maintained to all RCPs.			EOP-1.0
	RO	11. Check if RCPs should be stopped: a. Check if either of the following criteria is met: • Annunciator XCP-612 4-2 is lit (PHASE B ISOL). OR • RCS pressure is LESS THAN 1418 psig AND SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. b. Stop all RCPs.	EOP-1.0
	RO	12. Verify no SG is FAULTED: • No SG pressure is decreasing in an uncontrolled manner. • No SG is completely depressurized.	EOP-1.0

Op Test No:	NRC-ILO-16-01	Scenario #	2	Event #	6	Page:	33	of 52

Event Description: SG Tube Rupture on “A” SG that becomes Faulted during the cooldown.

Time	Position	Applicant’s Actions or Behavior	
	RO	13. Verify Secondary radiation levels indicate SG tubes are NOT RUPTURED: • RM-G19A(B)(C), STMLN HI RNG GAMMA. **(NO)** • RM-A9, CNDSR EXHAUST GAS ATMOS MONITOR. • RM-L3, STEAM GENERATOR BLOWDOWN LIQUID MONITOR. • RM-L10, SG BLOWDOWN CW DISCHARGE LIQUID MONITOR.	EOP-1.0
	CRS	**Alternative Action Step:** 13. GO TO EOP-4.0, E-3 STEAM GENERATOR TUBE RUPTURE, Step 1.	EOP-1.0
	CRS	Enters EOP-4.0, E-3 STEAM GENERATOR TUBE RUPTURE.	

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 34 of 52
Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.			

Time	Position	Applicant's Actions or Behavior

REFERENCE PAGE FOR EOP-4.0

1 SI REINITIATION CRITERIA

IF either of the following conditions occurs, THEN start Charging Pumps and operate valves as necessary:

- RCS subcooling on TI-499A(B), A(B) TEMP °F, can NOT be maintained GREATER THAN 52.5°F [67.5°F].

OR

- PZR level can NOT be maintained GREATER THAN 10% [28%].

IF SI Reinitiation occurs after procedure Step 28, THEN GO TO EOP-4.2, ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT: SUBCOOLED RECOVERY DESIRED, Step 1.

2 SECONDARY INTEGRITY TRANSITION CRITERIA

IF any unisolated SG pressure is decreasing in an uncontrolled manner OR is completely depressurized, THEN GO TO EOP-3.0, E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1, unless it is needed for RCS cooldown. EOP-4.0

3 COLD LEG RECIRCULATION TRANSITION CRITERION

IF RWST level decreases to LESS THAN 18%, THEN GO TO EOP-2.2, ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1.

4 MULTIPLE TUBE RUPTURE CRITERIA

IF any INTACT SG level increases in an uncontrolled manner OR any INTACT SG has abnormal radiation, THEN stop any cooldown or depressurization in progress and RETURN TO EOP-4.0, E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.

5 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 35 of 52

Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.

Time	Position	Applicant's Actions or Behavior	
		NOTE Seal Injection flow should be maintained to all RCPs.	EOP-4.0
	RO	1. Check if RCPs should be stopped: a. Check if either of the following criteria is met: • Annunciator XCP-612 4-2 is lit (PHASE B ISOL). OR • RCS pressure is LESS THAN 1418 psig AND SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. **(NO)**	EOP-4.0
	CRS	**Alternative Action Step:** a. GO TO Step 2. Observe the NOTE prior to Step 2.	EOP-4.0
		NOTE - Step 2 Cycling of multiple PZR PORVs should be minimized to conserve operating air.	EOP-4.0
	RO	2. Perform the following: a. Check if both of the following are available: • PCV-445A, PWR RELIEF. • PCV-444B, PWR RELIEF. b. Close one of the following: • PCV-445A, PWR RELIEF. • PCV-444B, PWR RELIEF.	EOP-4.0

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 36 of 52
Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.			

Time	Position	Applicant's Actions or Behavior	
		CAUTION - Step 3 Radiation levels may have increased in steamlines. Proper radiological precautions must be taken when obtaining samples to minimize personnel exposure.	EOP-4.0
	BOP	3. Identify the RUPTURED SG(s): • Narrow Range level in any SG increasing in an uncontrolled manner. OR • High Radiation on any of the following: a. RM-G19A(B)(C), STMLN HI RNG GAMMA. b. Local hand held radiation monitor readings taken by Health Physics on the blowdown lines at the following penetrations: • XRP0326, SG A Blowdown Line (AB-412 West Pen). • XRP0224, SG B Blowdown Line (IB-412 East Pen). • XRP0219, SG C Blowdown Line(IB-412 East Pen). OR • As determined by Chemistry sample analysis for abnormal activity using a frisker.	EOP-4.0
	BOP	4. For each RUPTURED SG, initiate the appropriate isolation attachment: • ATTACHMENT 1A, ISOLATION OF ASTEAM GENERATOR	EOP-4.0
	CRS	5. Locally, start XAC0014, Diesel Driven Air Compressor. REFER TO SOP-220, STATION AND BACKUP INSTRUMENT AIR SYSTEMS (YD-436).	EOP-4.0
BOOTH OPERATOR:		When requested to locally start the Diesel Driven Air Compressor, Wait 2 minutes, Insert **TRIGGER 18** and report back that the Diesel Driven Air Compressor is running.	
		CAUTION - Step 6 If the TD EFW Pump is the only available source of feed flow, the steam supply to the TD EFW Pump must be maintained from at least one SG, to maintain a secondary heat sink.	EOP-4.0
		NOTE - Step 6 If the TD EFW Pump is tripped, it should be reset as time permits.	EOP-4.0

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 37 of 52

Event Description: SG Tube Rupture on “A” SG that becomes Faulted during the cooldown.

Time	Position	Applicant’s Actions or Behavior	
	CRS	6. IF SG B OR SG C is RUPTURED, THEN perform the following: a. IF at least one MD EFW Pump is running, THEN isolate the TD EFW Pump by placing PVG-2030, STM SPLY TO TD EFP TRN A(B), to CLOSE. b. Notify local Operators to perform Alternative Action Step 6, while continuing with this procedure.	EOP-4.0
CAUTION - Step 7 If any RUPTURED SG is FAULTED, feed flow to that SG should remain isolated during subsequent recovery actions unless needed for RCS cooldown, to prevent excessive cooldown due to the FAULT.			EOP-4.0
	BOP	7. Check level in each RUPTURED SG: a. Verify Narrow Range level in each RUPTURED SG is GREATER THAN 26%[41%]. b. Stop EFW flow to each RUPTURED SG: 1. Close FCV-3531(3541)(3551), MD EFP TO SG A(B)(C). 2. Close FCV-3536(3546)(3556), TD EFP TO SG A(B)(C). 3. Maintain Narrow Range level in each RUPTURED SG GREATER THAN 26%[41%].	EOP-4.0
CAUTION - Step 8 The major flowpaths from each RUPTURED SG (MSIV, TD EFW Pump, and PORV) must be isolated before performing Step 8, to minimize radiological releases and ensure RCS subcooling is maintained.			EOP-4.0
	BOP	8. Verify each RUPTURED SG pressure is GREATER THAN 460 psig.	EOP-4.0
	CRS	9. Determine the required core exit TC temperature for RCS cooldown from the table below:	EOP-4.0
	RO	10. Check if any RCP is running.	EOP-4.0

LOWEST RUPTURED SG PRESS (PSIG)	CORE EXIT TC TEMP (°F)	CONTROLLER SETPOINT
1101-1200	494 [478]	4.9
1001-1100	482 [466]	4.4
901-1000	469 [453]	3.8
801-900	455 [439]	3.4
701-800	439 [423]	2.8
601-700	421 [405]	2.3
460-600	392 [376]	1.6

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 38 of 52

Event Description: SG Tube Rupture on “A” SG that becomes Faulted during the cooldown.

Time	Position	Applicant’s Actions or Behavior	
		NOTE - Step 11 Before the Low Steamline Pressure SI signal is blocked, Main Steam Isolation will occur if the Low Steam Pressure rate setpoint is exceeded.	EOP-4.0
	BOP	11. Dump steam from each INTACT SG: a. WHEN RCS Tavg is LESS THAN P-12 (552°F), THEN: • Place both STM DUMP INTERLOCK Switches to BYP INTLK. • Place both STMLN SI TRAIN A(B) Switches to BLOCK. b. Dump steam from each INTACT SG to the Condenser: 1. Verify PERMISV C-9 status light is bright on XCP-6114 1-3. 2. Perform the following: • Verify the MS Isolation Valves, PVM-2801A(B)(C), are open for the INTACT SGs. OR • IF the RUPTURED SG(s) MSIV is closed, THEN open MS Isolation Bypass Valves : a. Depress both MAIN STEAM ISOL VALVES RESET TRAIN A(B). b. Open MS Isolation Bypass Valves, PVM-2869A(B)(C), for only the INTACT SGs. 3. Place the STM DUMP CNTRL Controller in MAN and closed. 4. Place the STM DUMP MODESELECT Switch in STM PRESS. 5. Adjust the STM DUMP CNTRL Controller to fully open the Bank 1 Steam Dump Valves, (Approximately 14% Demand).	EOP-4.0
		NOTE - Step 12 Steps 14 through 20 should be performed as time permits, while the cooldown is in progress.	EOP-4.0
	RO	12. Verify core exit TC temperature is LESS THAN the value determined in Step 9. **(NO)**	EOP-4.0

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 39 of 52
Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.			

Time	Position	Applicant's Actions or Behavior	
	CRS	**Alternative Action Step:** 12. WHEN core exit TC temperature is LESS THAN the value determined in Step 9, THEN COMPLETE Step 13. Observe the NOTE prior to Step 13. CONTINUE WITH Step 14.	EOP-4.0
EVALUATOR NOTE: Once the cooldown has started, the failure for the faulted SG will automatically go in. This will allow them to transition out to EOP-4.2 when the ruptured SG pressure decreases to less than 250 psig above the intact SG(s) used for cooldown.			
	BOP	14. Check INTACT SG levels: a. Verify Narrow Range level in INTACT SGs is GREATER THAN 26% [41%]. b. Control EFW flow to maintain Narrow Range level in INTACT SGs between 40% and 60%.	EOP-4.0
	RO	15. Check PZR PORVs and Block Valves: a. Verify power is available to the PZR PORV Block Valves: 1. MVG-8000A, RELIEF 445 A ISOL. 2. MVG-8000B, RELIEF 444 B ISOL. 3. MVG-8000C, RELIEF 445 B ISOL. **CAUTION - Step 15.b** **If any PZR PORV opens because of high PZR pressure, Step 15.b should be repeated after pressure decreases to LESS THAN 2330 psig, to ensure the PORV recloses.** b. Verify all PZR PORVs are closed. c. Check if PCV-445A, PWR RELIEF switch is in close. d. Ensure MVG-8000A, RELIEF 445 A ISOL, is Open e. GO TO Step 16.	EOP-4.0
	RO	16. Reset both SI RESET TRAIN A(B) Switches.	EOP-4.0
	RO	17. Reset Containment Isolation: • RESET PHASE A - TRAIN A(B) CNTMT ISOL. • RESET PHASE B - TRAIN A(B) CNTMT ISOL.	EOP-4.0
	RO	18. Establish Instrument Air to the RB: a. Open PVA-2659, INST AIR TO RB AIR SERV. b. Open PVT-2660, AIR SPLY TO RB.	EOP-4.0

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 40 of 52

Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.

Time	Position	Applicant's Actions or Behavior	
		CAUTION - Step 19 RCS pressure should be monitored. If RCS pressure decreases in an uncontrolled manner to LESS THAN 325 psig, the RHR Pumps must be manually restarted to supply water to the RCS.	EOP-4.0
*	RO	19. Check if RHR Pumps should be stopped: a. Check if any RHR Pump is running with suction aligned to the RWST. b. Verify RCS pressure is GREATER THAN 325 psig. c. Stop any RHR Pump which is running with suction aligned to the RWST and place in Standby.	EOP-4.0
	BOP	20. Verify core exit TC temperature is LESS THAN the value determined in Step 9.	EOP-4.0
	BOP	21. Stop the RCS cooldown: a. Stop the RCS cooldown to the Condenser: 1. Adjust the STM DUMP CNTRL Controller to closed. 2. Adjust the setpoint to maintain core exit TC temperature LESS THAN the required temperature per Step 9. 3. Place the STM DUMP CNTRL Controller in AUTO. **NOTE - Step 21.b** **With no RCPs running, it may be necessary to manually open steam dumps or Steamline PORVs to maintain desired TC temperature.** b. Adjust controller setpoints as necessary to maintain core exit TC temperature LESS THAN the required temperature per Step 9.	EOP-4.0
	BOP	22. Verify each RUPTURED SG pressure is stable OR increasing. **(NO)**	EOP-4.0
	BOP	**Alternative Action Step:** 22. Monitor SG pressures. IF any RUPTURED SG pressure decreases to LESS THAN 250 psi above the INTACT SG(s) used for cooldown, THEN GO TO EOP-4.2, ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT: SUBCOOLED RECOVERY DESIRED, Step 1.	EOP-4.0
	CRS	Enters EOP-4.2, ECA-3.1 SGTR WITH LOSS OF REACTOR COOLANT: SUBCOOLED RECOVERY DESIRED.	EOP-4.2

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 41 of 52

Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.

Time	Position	Applicant's Actions or Behavior	
		REFERENCE PAGE FOR EOP-4.2 1 SI REINITIATION CRITERIA IF either of the following conditions occurs, THEN start Charging Pumps and operate valves as necessary: • RCS subcooling on TI-499A(B), A(B) TEMP °F, is LESS THAN 52.5°F [67.5°F]. OR • PZR level can NOT be maintained GREATER THAN 10% [28%]. 2 SECONDARY INTEGRITY TRANSITION CRITERIA IF any unisolated SG pressure is decreasing in an uncontrolled manner OR is completely depressurized, THEN GO TO EOP-3.0, E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1, unless it is needed for RCS cooldown. 3 COLD LEG RECIRCULATION TRANSITION CRITERION IF RWST level decreases to LESS THAN 18%, THEN GO TO EOP-2.2, ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1. 4 REDUCING CONTROL ROOM EMERGENCY VENTILATION Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.	EOP-4.2
	RO	1. Reset both SI RESET TRAIN A(B) Switches.	EOP-4.2
	RO	2. Reset Containment Isolation: • RESET PHASE A - TRAIN A(B) CNTMT ISOL. • RESET PHASE B - TRAIN A(B) CNTMT ISOL.	EOP-4.2
	BOP	3. Place both ESF LOADING SEQ A(B) RESETS to: a. NON-ESF LCKOUTS. b. AUTO-START BLOCKS.	EOP-4.2

Op Test No: NRC-ILO-16-01 Scenario # 2 Event # 6 Page: 42 of 52

Event Description: SG Tube Rupture on “A” SG that becomes Faulted during the cooldown.

Time	Position	Applicant’s Actions or Behavior	
	RO	4. Establish Instrument Air to the RB: a. Start one Instrument Air Compressor and place the other in Standby. b. Verify PI-8342, INSTR AIR HDR PRESS PSIG, indicates GREATER THAN 60 psig. c. Open PVA-2659, INST AIR TO RB AIR SERV. d. Open PVT-2660, AIR SPLY TO RB.	EOP-4.2
*	BOP	5. Verify all AC buses are energized by offsite power: • ESF AC buses. • BOP AC buses.	EOP-4.2
CAUTION - Step 6 PZR Heaters should NOT be energized until PZR water level is GREATER THAN the minimum level recommended by TSC personnel to ensure the heaters are covered.			EOP-4.2
	RO	6. Deenergize PZR Heaters: a. Place both BU GRP 1(2) Switches in PULL TO LK NON-A. b. Secure the CNTRL GRP Heaters. c. Consult TSC personnel for a minimum indicated PZR water level that will ensure heaters are covered.	EOP-4.2
*	RO	7. Check if RB Spray should be stopped: a. Check if any RB Spray Pumps are running. **(NO)**	EOP-4.2
	CRS	**Alternative Action Step:** 7.a. GO TO Step 8. Observe the CAUTION prior to Step 8.	
CAUTION - Step 8 If any RUPTURED SG is FAULTED, feed flow to that SG should remain isolated during subsequent recovery actions unless needed for RCS cooldown, to prevent excessive cooldown due to the FAULT.			EOP-4.2
*	BOP	8. Check level in each RUPTURED SG: a. Verify Narrow Range level in each RUPTURED SG is GREATER THAN 26% [41%]. b. Stop EFW flow to each RUPTURED SG: 1. Close FCV-3531(3541)(3551), MD EFP TO SG A(B)(C). 2. Close FCV-3536(3546)(3556), TD EFP TO SG A(B)(C). 3. Maintain Narrow Range level in each RUPTURED SG GREATER THAN 26% [41%].	EOP-4.2

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 43 of 52

Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.

Time	Position	Applicant's Actions or Behavior	
		CAUTION - Step 9 RCS pressure should be monitored. If RCS pressure decreases in an uncontrolled manner to LESS THAN 325 psig, the RHR Pumps must be manually restarted to supply water to the RCS.	EOP-4.2
*	RO	9. Check if RHR Pumps should be stopped: a. Check if any RHR Pump is running with suction aligned to the RWST. b. Check RCS pressure: • RCS pressure is GREATER THAN 325 psig. • RCS pressure is stable OR increasing. c. Stop any RHR Pump which is running with suction aligned to the RWST and place in Standby.	EOP-4.2
	CRS	10. Verify radiation levels are normal outside the RB: a. Check the Radiation Monitoring System. b. Notify Health Physics to survey for activity levels and for radioactive leakage.	EOP-4.2
	RO/CRS	11. Obtain necessary Chemistry samples: a. Ensure all the following sample valves are in AUTO: • SVX-9364B and SVX-9365B, RCS LP B SMPL ISOL. • SVX-9364C and SVX-9365C, RCS LP C SMPL ISOL. • SVX-9398A(B)(C), SG A(B)(C) SMPL ISOL. b. Notify Chemistry to sample the following: • RCS. • All SGs for isotopic activity.	EOP-4.2
	CRS	12. Consult with TSC personnel to determine what additional equipment will be required for cooldown.	EOP-4.2
	BOP	13. Verify no SG is FAULTED: • No SG pressure is decreasing in an uncontrolled manner. **(NO)** • No SG is completely depressurized.	EOP-4.2
	BOP	**Alternative Action Step:** Verify each FAULTED SG has been isolated unless needed for RCS cooldown. IF any FAULTED SG has NOT been isolated, THEN GO TO EOP-3.0, E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.	EOP-4.2

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 44 of 52

Event Description: SG Tube Rupture on “A” SG that becomes Faulted during the cooldown.

Time	Position	Applicant’s Actions or Behavior
EVALUATOR NOTE: Crew may transition to EOP-3.0, E-2 Faulted Steam Generator Isolation; however, it is not necessary since this steam generator has already been isolated. If the crew does transition, EOP-3.0 can be found on **page 51 of 52.**		
*	BOP	14. Check INTACT SG levels: a. Verify Narrow Range level in INTACT SGs is GREATER THAN 26% [41%]. b. Control EFW flow to maintain Narrow Range level in INTACT SGs between 40% [41%] and 60%.

EOP-4.2

Op Test No:	NRC-ILO-16-01	Scenario #	2	Event #	6	Page:	45	of	52

Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.

Time	Position	Applicant's Actions or Behavior

NOTE - Step 15

- Before the Low Steamline Pressure SI signal is blocked, Main Steam Isolation will occur if the Low Steam Pressure rate setpoint is exceeded. EOP-4.2
- Shutdown margin should be monitored during RCS cooldown.

Time	Position	Applicant's Actions or Behavior
	BOP	15. Initiate RCS cooldown to Cold Shutdown: a. Maintain the cooldown rate in the RCS Cold Legs LESS THAN 100°F/hr. b. Use the RHR System if it is in service. REFER TO SOP-115, RESIDUAL HEAT REMOVAL. c. WHEN RCS Tavg is LESS THAN P-12 (552°F), THEN: • Place both STM DUMP INTERLOCK Switches to BYP INTLK. • Place both STMLN SI TRAIN A(B) Switches to BLOCK. **NOTE - Step 15.d** **If no INTACT SG is available, TSC personnel should be consulted to determine a release rate prior to using a RUPTURED SG.** d. Dump steam from each INTACT SG to the Condenser: 1. Verify PERMISV C-9 status light is bright on XCP-6114 1-3. 2. Perform the following: • Verify the MS Isolation Valves, PVM-2801A(B)(C), are open for the INTACT SGs. OR • Open MS Isolation Bypass Valves: a. Depress both MAIN STEAM ISOL VALVES RESET TRAIN A(B). b. Open MS Isolation Bypass Valves, PVM-2869A(B)(C), for only the INTACT SGs. 3. Place the STM DUMP CNTRL Controller in MAN and closed. 4. Place the STM DUMP MODE SELECT Switch in STM PRESS. 5. Adjust the STM DUMP CNTRL Controller to establish the desired cooldown rate.

EOP-4.2

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # 6	Page: 46 of 52

Event Description: SG Tube Rupture on "A" SG that becomes Faulted during the cooldown.

Time	Position	Applicant's Actions or Behavior	
*	RO/BOP	16. Check if a subcooled recovery is appropriate: a. Verify RWST level is GREATER THAN 59%. b. Verify Narrow Range level in each RUPTURED SG is LESS THAN 90% [83%].	EOP-4.2
	RO	17. Verify RCS subcooling on TI-499A(B), A(B) TEMP °F, is GREATER THAN 52.5°F [67.5°F].	EOP-4.2
		18. Check if the SI System is in service: • Any Charging Pump is running with flow indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. OR • Any RHR pump running in the SI Mode with flow indicated on: • FI-605A, RHR DISCHARGE PUMP A FLOW GPM. • FI-605B, RHR DISCHARGE PUMP B FLOW GPM.	EOP-4.2
NOTE - Step 19 If no RCP is running, the Reactor Vessel Head Upper Plenum may void during depressurization resulting in a rapidly increasing PZR level.			EOP-4.2
		19. Depressurize the RCS to refill the PZR: a. Establish Normal PZR Spray: • Using RCP A: 1. Open PCV-444D, PZR SPRAY. 2. Close PCV-444C, PZR SPRAY. OR • Using RCPs B and C: 1. Open PCV-444C, PZR SPRAY. 2. Close PCV-444D, PZR SPRAY. b. Verify PZR level is GREATERTHAN 22% [39%]. c. Stop RCS depressurization.	EOP-4.2
EVALUATOR NOTE: The scenario may be terminated at any point after they have refilled the Pressurizer.			

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # N/A	Page: 47 of 52
Event Description: EOP 1.0, Attachment 3			

Time	Position	Applicant's Actions or Behavior	
CRITICAL TASK	BOP	1. Ensure EFW Pumps are running: a. Ensure both MD EFW Pumps are running. b. Verify the TD EFW Pump is running if necessary to maintain SG levels.	EOP-1.0 Attachment 3
	BOP	2. Ensure the following EFW valves are open: • FCV-3531(3541)(3551), MD EFP TO SG A(B)(C). • FCV-3536(3546)(3556), TD EFP TO SG A(B)(C). • MVG-2802A(B), MS LOOP B(C) TO TD EFP.	Attachment 3
	BOP	3. Verify total EFW flow is GREATER THAN 450 gpm.	Attachment 3
	BOP	4. Ensure FW Isolation: a. Ensure the following are closed: • FW Flow Control, FCV-478(488)(498). • FW Isolation, PVG-1611A(B)(C). • FW Flow Control Bypass, FCV-3321(3331)(3341). • SG Blowdown, PVG-503A(B)(C). • SG Sample, SVX-9398A(B)(C). b. Ensure all Main FW Pumps are tripped.	Attachment 3
	BOP	5. Ensure SI Pumps are running: • Two Charging Pumps are running. • Both RHR Pumps are running.	Attachment 3
	BOP	6. Ensure two RBCU Fans are running in slow speed (one per train).	Attachment 3
	BOP	7. Verify Service Water to the RBCUs: a. Ensure two Service Water Pumps are running. b. Verify Service Water Booster Pump A is stopped. **(NO)** **Alternative Action Step:** b. GO TO Step 7.e. 7e. Verify that Service Water Booster Pump B is stopped. **(NO)** **Alternative Action Step:** e. GO TO Step 7.h. 7h. Verify GREATER THAN 2000 gpm flow for each train on: • FI-4466, SWBP A DISCH FLOW GPM. • FI-4496,SWBP B DISCH FLOW GPM.	Attachment 3

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # N/A	Page: 48 of 52

Event Description: EOP 1.0, Attachment 3

Time	Position	Applicant's Actions or Behavior	
	BOP	8. Verify two CCW Pumps are running.	Attachment 3
	BOP	9. Ensure two Chilled Water Pumps and Chillers are running.	Attachment 3
	BOP	10. Verify both trains of Control Room Ventilation are running in Emergency Mode.	Attachment 3
	BOP	11. Check if Main Steamlines should be isolated: a. Check if any of the following conditions are met: • RB pressure GREATER THAN 6.35 psig. OR • Steamline pressure LESS THAN 675 psig. OR • Steamline flow GREATER THAN 1.6 MPPH AND Tavg LESS THAN 552°F. b. Ensure all the following are closed: • MS Isolation Valves, PVM-2801A(B)(C). • MS Isolation Bypass Valves, PVM-2869A(B)(C).	Attachment 3
	BOP	12. Ensure Excess Letdown Isolation Valves are closed: • PVT-8153, XS LTDN ISOL. • PVT-8154, XS LTDN ISOL.	Attachment 3
	BOP	13. Verify ESF monitor lights indicate Phase A AND Containment Ventilation Isolation on XCP-6103, 6104, and 6106. REFER TO ATTACHMENT 4, CONTAINMENT ISOLATION VALVE MCB STATUS LIGHT LOCATIONS, as needed.	Attachment 3
	BOP	14. Verify proper SI alignment: a. Verify SI valve alignment by verifying SAFETY INJECTION/PHASE A ISOL monitor lights are bright on XCP-6104. b. Verify all SAFETY INJECTION monitor lights are dim on XCP-6106. c. Verify SI flow on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. d. Check if RCS pressure is LESS THAN 325 psig.	Attachment 3
	BOP	Report completion of Attachment 3.	Attachment 3

EVALUATOR NOTE: ATTACHMENT 3 is complete.

Op Test No: NRC-ILO-16-01 Scenario # 2 Event # N/A Page: 49 of 52

Event Description: SOP-106, Reactor Makeup Water System.

Time	Position	Applicant's Actions or Behavior	
NOTE 2.0 1. Energizing additional Pressurizer Heaters will enhance mixing. 2. LCV-115A, LTDN DIVERT TO HU-TK, will begin to modulate to the HU-TK position at 70% level on LI-115, VCT LEVEL %.			SOP-106
	RO	2.1. Ensure at least one Reactor Coolant Pump is running.	SOP-106
	RO	2.2. Place RX COOL SYS MU switch to STOP.	SOP-106
	RO	2.3. Place RX COOL SYS MU MODE SELECT switch to BOR. **(Peer ✓)**	SOP-106
	RO	2.4. Adjust FCV-113 A&B, BA FLOW SET PT, for desired flow rate. **(Peer ✓)**	SOP-106
	RO	2.5. Set FIS-113, BA TO BLNDR FLOW, batch integrator to the desired volume. **(Peer ✓)**	SOP-106
	RO	2.6. Place RX COOL SYS MU switch to START.	SOP-106
NOTE 2.7 Step 2.7 may be omitted when borating less than 10 gallons.			SOP-106
	RO	2.7. Verify desired flow rate on FR-113, TOTAL MU GPM (F-168).	SOP-106
NOTE 2.8 The AUTO setpoint dial for FCV-113A&B, BA FLOW, controller may be adjusted slowly to obtain the desired flow rate.			
	RO	2.8. Verify the desired Boric Acid flow rate on FR-113, BA TO BLNDR GPM (F-113).	SOP-106
	RO	2.9. When the preset volume of boric acid has been reached, perform the following: a. Place FCV-113A&B, BA flow controller in MAN. b. Verify boration stops.	SOP-106
	RO	2.10. Place RX COOL SYS MU switch to STOP.	SOP-106
NOTE 2.11 a. If plant conditions require repeated borations, Step 2.11 may be omitted. b. The volume in the piping between the blender and the VCT outlet is approximately 3.8 gallons.			SOP-206

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # N/A	Page: 50 of 52

Event Description: SOP-106, Reactor Makeup Water System.

Time	Position	Applicant's Actions or Behavior	
	RO	2.11. Alternate Dilute 4 to 6 gallons of Reactor Makeup Water to flush the line downstream of the blender by performing the following: a. Place RX COOL SYS MU MODE SELECT switch to ALT DIL. **(Peer ✓)** b. Adjust FCV-168, TOTAL MU FLOW SET PT, to desired flow rate. **(Peer ✓)** c. Set FIS-168, TOTAL MU FLOW, batch integrator to desired volume. **(Peer ✓)** d. Place RX COOL SYS MU switch to START. e. Verify desired flow rate on FR-113, TOTAL MU GPM (F-168). f. Verify alternate dilution stops when preset volume is reached on FIS-168, TOTAL MU FLOW, batch integrator. g. Place RX COOL SYS MU switch to STOP.	SOP-106
	RO	2.12. Place RX COOL SYS MU MODE SELECT switch to AUTO. **(Peer ✓)**	SOP-106
	RO	2.13. Adjust FCV-168, TOTAL MU FLOW SET PT, to 7.5 (120 gpm). **(Peer ✓)**	SOP-106
	RO	2.14. In MAN, adjust FCV-113 A&B, BA FLOW OUTPUT, to the required position which will ensure proper Boric Acid addition for subsequent Automatic Makeup operations.	SOP-106
	RO	2.15. Adjust FCV-113 A & B, BA FLOW, SET PT per one of the following: a. OAP-100.6, Attachment IA, Reactivity Control Parameters. b. Desired position to ensure proper boric acid addition based on current RCS conditions.	SOP-106
	RO	2.16. Place RX COOL SYS MU switch to START. **(Peer ✓)**	SOP-106
	RO	2.17. Perform the following: a. Start XPP-13A(B), BA XFER PP A(B), for the in-service Boric Acid Tank. b. If necessary, start XPP-13A(B), BA XFER PP A(B), for the Boric Acid Tank on recirculation.	SOP-106

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # N/A	Page: 51 of 52
Event Description: EOP-3.0, E-2 Faulted Steam Generator Isolation			

Time	Position	Applicant's Actions or Behavior	
		CAUTION • At least one SG must be maintained available for RCS cooldown. • Any FAULTED SG or secondary break should remain isolated during subsequent recovery actions unless needed for RCS cooldown, to prevent reinitiating the break.	EOP-3.0
	BOP	1. Ensure all the following are closed: • MS Isolation Valves, PVM-2801A(B)(C). • MS Isolation Bypass Valves, PVM-2869A(B)(C).	EOP-3.0
	BOP	2. Check if any SG is NON-FAULTED: • Pressure in any SG is stable OR increasing. • Any SG is NOT completely depressurized.	EOP-3.0
	BOP	3. Identify any FAULTED SG(s): • Any SG pressure decreasing in an uncontrolled manner. OR • Any SG completely depressurized.	EOP-3.0
	BOP	4. Close the following for each FAULTED SG: • FW Flow Control, FCV-478(488)(498). • FW Isolation, PVG-1611A(B)(C). • SG Blowdown, PVG-503A(B)(C). • FW Flow Control Bypass, FCV-3321(3331)(3341).	EOP-3.0
	BOP	5. Complete the isolation of each FAULTED SG: a. Close SG Chemical Feed Isolation, MVK-1633A(B)(C). b. Close MS Drain Isolation, PVT-2843A(B)(C). c. Close MS Drain Isolation, PVT-2877A for SG A, PVT-2877B for SG C. d. Place the Steamline PWR RELIEF A(B)(C) SETPT Controller(s) in MAN and closed. e. Place the Steamline Power Relief A(B)(C) Mode Switch(s) in PWR RLF. f. Close FCV-3531(3541)(3551), MD EFP TO SG A(B)(C). g. Close FCV-3536(3546)(3556), TD EFP TO SG A(B)(C). h. Locally unlock and close XVGO1017A(B)(C)-EF, SG A(B)(C) MTR DR EF PUMP SUPPLY HEADER VALVE (IB-423). i. Locally unlock and close XVG01018A(B)(C)-EF. SG A(B)(C) TURB DR EF PUMP SUPPLY HDR VALVE (IB-423).	EOP-3.0

Op Test No: NRC-ILO-16-01	Scenario # 2	Event # N/A	Page: 52 of 52
Event Description: EOP-3.0, E-2 Faulted Steam Generator Isolation			

Time	Position	Applicant's Actions or Behavior	
		NOTE - Step 6 Any high radiation level received on a radiation monitor that was unisolated at event initiation may be considered a valid alarm.	EOP-3.0
	BOP	6. Check if Secondary radiation levels are normal: a. Check radiation levels normal on all unisolated radiation monitors: • RM-G19A(B)(C), STMLN HI RNG GAMMA. **(NO)** • RM-L3, STEAM GENERATOR BLOWDOWN LIQUID MONITOR. • RM-L10, SG BLOWDOWN CW DISCHARGE LIQUID MONITOR. • RM-A9, CNDSR EXHAUST GAS ATMOS MONITOR. b. Notify Chemistry to sample all SG secondary sides, and screen samples for abnormal activity using a frisker.	EOP-3.0
	CRS	**Alternative Action Step:** 6. GO TO EOP-4.0, E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.	EOP-3.0

Facility: VC SUMMER U1 Scenario No: 3 Op Test No: NRC-ILO-16-01

Examiners: ____________ Operators: CRS: ____________

____________ RO: ____________

____________ BOP: ____________

Initial Conditions:
- The Reactor power is 10^{-3}%.
- “B” train work week.
- XFN-0065B RBCU is OOS.

Turnover:
- Increase reactor power to between 1% and 3% in accordance with GOP-3, Reactor Startup From Hot Standby to Startup (MODE 3 to MODE 2).

Critical Tasks:
- Manually trip the reactor when it fails to AUTO trip with three faulted Steam Generators before SG WR <12%.
- Control the EFW flowrate (minimum of 50 gpm to each s/g < 26% [41%] level) and terminate SI, in order to minimize the RCS cooldown rate before a severe (orange-path) challenge develops to the integrity CSF.

Event	Malf No.	Event Type*	Event Description
1	N/A	N-BOP, CRS	Restore “B” MDEFW Pump to service.
2#	MAL-AUX014A PMP-IA002F	C-RO, CRS	Instrument Air Compressor “A” trips, “B” Fails to auto start.
3	N/A	R-RO, CRS	Raise Power to between 1% and 3%.
4	ANN-ES001	C-BOP, CRS TS-CRS	Elevated temperatures on XTF-31, transfer power to alternate source.
5	MAL-FWM012	I-BOP, CRS	Condensate Flow to the Deaerator goes high due to a failure of controller SC-3136 to control in auto.
6	MAL-CVC016A	I-RO, CRS	FCV-122 fails closed in AUTO, isolating charging flow.

7	ANN-FW018	TS-CRS	Feedwater Isolation Valve Accumulator low pressure alarm.
8	MAL-AUX009BA MAL-AUX009BB MAL-AUX009BC	M-ALL	Earthquake
	MAL-MSS004A MAL-MSS004B MAL-MSS004C		Three Faulted Steam Generators
	MAL-PCS005A MAL-PCS005B OVR-SI009(FALSE)		SI doesn't auto initiate, must be actuated from BOP side.
	MAL-PCS009AB MAL-PCS009BB		Reactor fails to automatically trip, must be tripped manually.
	VLV-CS051F VLV-CS042F		Phase "A" valves don't close, 8100 and 8112.
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

Used on previous two NRC Exams. Event 2 was used on the 2017 NRC Exam.

The following notation is used in the ES-D-2 form "Time" column:

IOA designates Immediate Operator Action steps.

* designates Continuous Action steps.

TURNOVER:

The crew will assume the watch having been pre-briefed on the Initial Conditions, the plan for this shift and any related operating procedures. Train "B" RBCU, XFN-0065B is tagged out for breaker maintenance and is to be returned to service next shift. The crew will take the shift and start the "B" MDEFW Pump.

EVENT 1: Start "B" MDEFW Pump from service.

The crew will be turned over to start "B" MDEFW pump in accordance with SOP-211, Emergency Feedwater System, section III.A, step 2.1. The BOP will start the pump and commence feeding the Steam Generators with both "A" and "B" Motor Driven Emergency Feedwater pumps.

EVENT 2: Instrument Air Compressor "A" trips, "B" Fails to auto start.

- **TRIGGER 2**
  - MAL-AUX014A
    INSTRUMENT AIR COMPRESSOR A TRIP
- **PRE-LOAD**
  - PMP-IA002F(NEW)
    XAC0003B INST AIR COMP B FAIL TO START
- **TRIGGER 11** X02I072S == 1 (Allows Instrument Air Compressor "B" to be started)
  - PMP-IA002F(NEW)
    XAC0003B INST AIR COMP B FAIL TO START
    DELETE = 1 sec

On cue from the Examiner, Instrument Air Compressor "A" will trip. XCP-606 2-1, INSTR AIR CMPR A TRBL, will alarm. Once Instrument Air Header pressure gets below 90 psig, the "B" Instrument Air Compressor will fail to auto start. The candidate will respond with the alarm response procedure and start the "B" Instrument Air Compressor.

EVENT 3: Raise Power to between 1% and 3%.

The crew will be prepared to commence a power ascension following a panel walk down and short briefing on the power ascension. The turnover stated that the crew will raise power to between 1% and 3% in accordance with GOP-3, Reactor Startup From Hot Standby to Startup (MODE 3 to MODE 2).

EVENT 4: Elevated temperatures on XTF-31, transfer power to alternate source.

- **TRIGGER 4**
  - ANN-ES001
    EMERG AUX XFMR XTF-31 TRBL
    FINAL = ON

- **TRIGGER 12** X13I071N == 1 (Clears the alarm 2 minutes after 1DB is put on its alt power source based on the bus 1DB XFER INIT switch going to N-E.)
  - ANN-ES001 (NEW)
    EMERG AUX XFMR XTF-31 TRBL
    DELETE = 120 sec

On cue from the Examiner, annunciator XCP 633, 1-4, EMERG AUX XFMR XTF-31 TRBL, will alarm. The crew will respond to this annunciator and send an AO out to investigate. The report from the field will be that several fans are not running and oil temperatures read 93°C and increasing. The crew will transfer loads from XTF-31 to 1DX in accordance with SOP-304, 115KV/7.2KV Operations. The CRS will refer to Tech Spec 3.4.8.1, AC Source and will apply Action "a" which is to Demonstrate the OPERABILITY of the remaining offsite AC. sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter, and If either EDG has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.3 separately for each such EDG within 24 hours unless the diesel is already operating, and Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

EVENT 5: Condensate flow to the Deaerator fails High.

- **TRIGGER 5**
  - MAL- FWM012
    CONDENSATE FLOW TO DEAERATOR FLOW CONTROLLER (SC-3136) FAILURE
    FINAL VALUE = 15%

On cue from the Examiner, the flow controller for Condensate to the Deaerator will fail high in automatic. As flow rises, level in the Deaerator will rise. The BOP will take action in accordance with OAP-100.5 guidance for equipment not responding properly in automatic, and the OP CRIT alarm which will indicate Deaerator level rising. The BOP will place the flow control in manual and lower condensate flow as necessary to maintain Deaerator level.

EVENT 6: FCV-122 fails closed in AUTO, isolating charging flow.

- **TRIGGER 6**

  - MAL-CVC016A
    CHARGING FLOW CONTROL VALVE FAILURE (AUTO ONLY)
    FINAL VALUE = 0

  On cue from the Examiner, FCV-122 will fail closed. XCP-614, 5-1, CHG LINE FLO HI/LO will alarm. The RO will take manual control of FCV-122 and maintain TI-140, REGEN HX OUT TEMP °F, between 250°F and 350°F while maintaining Pressurizer level.

EVENT 7: Feedwater Isolation Valve Accumulator low pressure alarm.

- **TRIGGER 7**

  - ANN-FW018
    FIV A/B/C ACCUM PRESS LO
    FINAL VALUE = ON

  On cue from the Examiner, XCP-625, 3-3, FIV A/B/C ACCUM PRESS LO, will alarm. The crew will send an AO out to look at pressure for the accumulators. The report from the field will be that accumulator pressure for PVG-1611A, A ISOL, is 0 psig. The CRS will refer to Tech Spec 3.7.1.6, Feedwater Isolation Valves, and apply the action for MODE 2 and 3, With one feedwater isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided: a. The isolation valve is maintained closed. b. The provisions of Specification 3.0.4. are not applicable. Otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

EVENT 8: Earthquake leading to all three Steam Generators being faulted and an ATWS.

- **PRE-LOADS:**

  - MAL-PCS009AB
    REACTOR TRIP BREAKER A FAILURE (FAIL TO OPEN)
    FINAL = AUTO

  - MAL-PCS009BB
    REACTOR TRIP BREAKER B FAILURE (FAIL TO OPEN)
    FINAL = AUTO

  - MAL-PCS005A
    SAFETY INJECTION FAILURE TRAIN A
    FINAL = TOTAL FAILURE

  - MAL-PCS005B
    SAFETY INJECTION FAILURE TRAIN B
    FINAL = TOTAL FAILURE

- VLV-CS042F
  XVT08100-CS RCP SEAL WTR ISO VLV FAIL AS IS

- VLV-CS051F
  XVT08112-CS RCP SEAL WTR ISO VLV FAIL AS IS

- **TRIGGER 8**

  - MAL-AUX009BA
    SEISMIC EVENT EARTHQUAKE FULL O.B.E.(NORTH/SOUTH HORIZONTAL)
    FINAL VALUE = 25.4

  - MAL-AUX009BB
    SEISMIC EVENT EARTHQUAKE FULL O.B.E.(UP/DOWN VERTICAL)
    FINAL VALUE = 25.4

  - MAL-AUX009BC
    SEISMIC EVENT EARTHQUAKE FULL O.B.E.(EAST/WEST HORIZONTAL)
    FINAL VALUE = 25.4

  - MAL-MSS004A
    STEAMLINE S/G A BREAK OUTSIDE CONTAINMENT
    RAMP = 15 sec
    FINAL = 5.15E5 lbm/hr

  - MAL-MSS004B
    STEAMLINE S/G B BREAK OUTSIDE CONTAINMENT
    RAMP = 15 sec
    FINAL = 5.15E5 lbm/hr

  - MAL-MSS004A
    STEAMLINE S/G A BREAK OUTSIDE CONTAINMENT
    RAMP = 15 sec
    FINAL = 5.15E5 lbm/hr

- **TRIGGER 14** X09I073A==1 (Allows SI to be manually actuated from the BOP side)

  - MAL-PCS005A (NEW)
    SAFETY INJECTION FAILURE TRAIN A
    FINAL = TOTAL FAILURE
    DELETE = 1 sec

  - MAL-PCS005B (NEW)
    SAFETY INJECTION FAILURE TRAIN B
    FINAL = TOTAL FAILURE
    DELETE = 1 sec

- **TRIGGER 15** X04I101C==1 (Allows 8112 to be manually closed)

  - VLV-CS051F (NEW)
    XVT08112-CS RCP SEAL WTR ISO VLV FAIL AS IS
    DELETE = 1 sec

  On cue from the Examiner, an earthquake will occur followed by all three steam generators being faulted. The reactor will not trip automatically. The crew should manually trip the reactor and enter EOP-1.0, E-0 Reactor Trip or Safety Injection. The crew will transition out to EOP-3.0, E-2 Faulted Steam Generator Isolation. They will then transition to EOP-3.1, ECA-2.1 Uncontrolled Depressurization of all Steam Generators.

  It took twenty two minutes to reach <12% Wide Range in all Steam Generators with the reactor failing to auto trip and the RO not inserting rods. During this time, power peaked at approximately 22%.

  It took forty six minutes to get to <250F in the Cold legs presenting an orange path on Integrity.

TERMINATION:

The scenario may be terminated once the crew has terminated Safety Injection in accordance with EOP-3.1, ECA-2.1 Uncontrolled Depressurization of all Steam Generators.

PRE-LOADS:

- OVR-AH023A
  CS-AH279 RBCU FAN 65B SLOW SPEED GREEN L
  FINAL = OFF

- OVR-AH022A
  CS-AH280 RBCU FAN 65B FAST SPEED GREEN L
  FINAL = OFF

Scenario Attributes		Events
Total Malfunctions (5-8)	10	• Instrument Air Compressor “A” trips, “B” Fails to auto start. • Elevated temperatures on XTF-31, transfer power to alternate source. • DA Level controller fails high. • FCV-122 fails closed in AUTO, isolating charging flow. • Feedwater Isolation Valve Accumulator low pressure alarm. • Earthquake. • Three Faulted Steam Generators. • SI doesn’t auto initiate, must be actuated from BOP side. • Reactor fails to automatically trip, can be manually tripped. • Containment Isolation Valves fail to close.
Malfunctions after EOP entry (1-2)	3	• SI doesn’t auto initiate, must be actuated from BOP side. • Reactor fails to automatically trip, can be manually tripped. • Containment Isolation Valves fail to close.
Abnormal Events (2-4)	5	• Instrument Air Compressor “A” trips, “B” Fails to auto start. • Elevated temperatures on XTF-31, transfer power to alternate source. • DA Level controller fails high. • FCV-122 fails closed in AUTO, isolating charging flow. • Feedwater Isolation Valve Accumulator low pressure alarm.
Major Transients (1-2)	1	• Three Faulted Steam Generators.
EOPs Entered (1-2)	1	• EOP-3.1, ECA-2.1 Uncontrolled Depressurization of all Steam Generators.
EOP Contingencies (0-2)	1	• EOP-3.1, ECA-2.1 Uncontrolled Depressurization of all Steam Generators.
Critical Tasks (2-3)	2	• Manually trip the reactor when it fails to AUTO trip with three faulted Steam Generators before SG WR <12%. • Control the EFW flow rate (minimum of 50 gpm to each S/G < 30% [50%] level) and terminate SI, in order to minimize the RCS cooldown rate before a severe (orange-path) challenge develops to the integrity CSF.

SIMULATOR SCENARIO SETUP

INITIAL CONDITIONS: (Example below)

- IC Set 302
- 10^{-3}% Power, MOL
- Burnup = 10,025 MWD/MTU
- RCS Boron Concentration = 1481 ppm
- FCV-113 Pot Setting = 6.35
- Rod Position: Group D = 100
- Tavg = 558.3˚F
- Xe = -0.0 pcm
- Prior to the scenario, the crew should pre-brief conditions and their expectations for the shift.

PRE-EXERCISE: (Example below)

- Ensure simulator has been checked for hardware problems (DORT, burnt out light bulbs, switch malfunctions, chart recorders, etc.).
- Complete VCS-TQP-0807 Attachment I-A, Unit 1 Booth Instructor Checklist.
- Verify plant aligned for "B1" work week IAW PTP-101.004, Safety Related Train Swap Checklist.
- Verify red hold tag and R&R tag on XFN-0065B RBCU and ensure they are in P-T-L. XFN-65B can't be taken to P-T-L.
- Verify red Placard on "A" CCW Pump and "B" Charging Pump.
- Verify the Hard Card for Turbine operation is in its proper storage location and cleaned.
- Verify the Hard Card for borating via MVT-8104 is in its proper storage location and cleaned.
- Set RO SIPCS station to ZZREAC.
- Verify Rod Bank Update set correctly: 100 steps on Control Bank D and 228 steps on all other Banks.
- Verify NR-45 is set to One Intermediate Range and One Source Range channel and is set to fast speed.
- Reset Digital Reactivity on SIPCS (disable calc, select N35 and N36, re-enable and calculate)
- Ensure no NI's are removed from service, on SIPCS type in "add/omit" to verify.
- Ensure 115kV & 230kV setpoints are set to appropriate values for the shutdown condition IAW SOP-304, Enclosure B.
- Ensure you have the following pre-marked up procedures:
    - o GOP-3, Reactor Startup From Hot Standby To Startup (Mode 3 To Mode 2)
    - o GOP-4A, Power Operation (Mode 1 - Ascending)
- Ensure you have a turnover sheet for each position.
- Conduct two-minute drill.

PRE-LOAD: (These are traditionally the pre-loads from the initial IC)

STANDARD SIMULATOR SETUP:

- PMP-LD003P, XPP0138 Leak Detection Sump Pmp Loss of Power
- VLV-FW028W, XVG01676-FW FW Hdr Recirc Isol Vlv Loss of Power
- VLV-FW029W, XVG01679-FW FW HTR Recirc Isol Vlv Loss of Power

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 1 Page: 11 of 35		
Event Description: Restore “B” MDEFW Pump to service.		
Time	Position	Applicant’s Actions or Behavior
EVALUATOR NOTE: The crew will be turned over to start “B” Motor Driven EFW pump in accordance with SOP-211, Emergency Feedwater System.		
BOOTH OPERATOR:		No TRIGGERS for this event.
Available Indications: N/A		
	CRS	2.1. Contact the Primary Chemist to place SS-SS24B, SG EMERG FW BYPASS SWITCH, to BYPASS at XPN0036 (CB-412). SOP-211
BOOTH OPERATOR:		When called to place SS-SS24B in BYPASS, wait 2 minutes and report back that SS-SS24B is in BYPASS. If called to look at the pump for start, wait 2 minutes and report back that the pump looks good for start.
	BOP	2.2. Momentarily place PVG-503ABC, A,B&C ISOL, to OPEN/BYPASS. SOP-211
	BOP	2.3. Momentarily place the following control switches, to OPEN/BYPASS: a. PVG-503A, A ISOL. b. PVG-503B, B ISOL. c. PVG-503C, C ISOL. SOP-211
	BOP	2.4. Hold the MD EFP RESET Switch in RESET for at least one second and then release. SOP-211
	BOP	2.5. To operate flow control valves using hand controllers, place the following switches, in MAN: a. FCV-3531, MD EFP TO SG A. b. FCV-3541, MD EFP TO SG B. c. FCV-3551, MD EFP TO SG C. SOP-211
	BOP	2.6. Close the following flow control valves using hand controllers: a. IFV-3531, MD EFP TO SG A. b. IFV-3541, MD EFP TO SG B. c. IFV-3551, MD EFP TO SG C. SOP-211
EVALUATOR NOTE: Step 2.6 was marked N/A prior to the start of this procedure since this step is not necessary to restore the “B” MDEFW pump.		
NOTE 2.7 and 2.8 XVM01072A(B)-EF, MTR DRIVEN EF PUMP A(B) RECIRC CV, maintains recirculation line flow between 110 gpm and 140 gpm when flow to the Steam Generators is isolated. If total pump flow is <110 gpm, a computer generated low EFW Pump flow alarm will occur. SOP-211		

Op Test No: NRC-ILO-16-01	Scenario # 3	Event # 1	Page: 12 of 35

Event Description: Restore “B” MDEFW Pump to service.

Time	Position	Applicant’s Actions or Behavior	
	BOP	2.8. Start Motor Driven Emergency Feedwater Pump B as follows: a. Place PUMP B switch, to START. **PEER** ✓ b. Verify starting current decays to less than 49 amps.	
NOTE 2.9 Enclosure B, Guidance To Prevent Cavitating Flow Vs. Steam Generator Pressure During Normal Operations, should be referenced when throttling flow.			SOP-211
	BOP	2.9. Adjust the following flow control valves to control Steam Generator levels: a. IFV-3531, MD EFP TO SG A. b. IFV-3541, MD EFP TO SG B. c. IFV-3551, MD EFP TO SG C.	SOP-211
NOTE 2.10 a. Steam Generator cavitating venturies should limit flow to each Steam Generator to ≤ 380 gpm at normal operating pressure. b. If the running EFW Pump total flow is greater than 190 gpm and recirculation line flow is >5 gpm, a computer generated alarm will occur indicating the failure of the recirculation valve to properly close.			SOP-211
	BOP	2.10. Monitor flow on the following indicators: a. FI-3561, TO SG A FLOW GPM. b. FI-3571, TO SG B FLOW GPM. c. FI-3581, TO SG C FLOW GPM.	SOP-211
EVALUATOR NOTE: The next event may be inserted once “B” MDEFW Pump is restored to service.			

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 2 Page: 13 of 35

Event Description: Instrument Air Compressor “A” trips, “B” Fails to auto start.

Time	Position	Applicant’s Actions or Behavior	
EVALUATOR NOTE: Instrument Air Compressor “A” trips, “B” Fails to auto start.			
BOOTH OPERATOR:		When directed - Initiate Event 2 (TRIGGER 2).	
Available Indications: XCP-606 2-1, INSTR AIR CMPR ATRBL			
	RO	Corrective Actions 1. If Instrument Air Compressor A trips, ensure the standby air compressor starts. 2. Dispatch an operator to Instrument Air Compressor A to determine the cause of the alarm.	XCP-606, 2-1
BOOTH OPERATOR:		If contacted as an operator to check the air compressors, wait 2 minutes and report "No obvious problem detected on "A" compressor" and if asked to check status on standby compressor report "The "B" compressor is properly aligned for auto start and is ready for start". If asked, the “B” Compressor looks good after start.	XCP-606, 2-1
EVALUATOR NOTE: The crew will continue on with the power escalation immediately following the completion of this event.			XCP-606, 2-1

Op Test No: NRC-ILO-16-01	Scenario # 3 Event # 3	Page: 14 of 35
Event Description: Raise power to between 1% - 3%.		

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: The crew will increase reactor power to between 1% and 3%.			
BOOTH OPERATOR:		No TRIGGERS for this event.	
Available Indications: Reactor Power			
CAUTION 3.14 While operating with a positive Moderator Temperature Coefficient: a. All reactivity additions should be slow and controlled. b. A stable Startup Rate of 0.3 decade per minute should not be exceeded. c. Rods should be moved in 1/2 step increments until the effect of rod motion has been evaluated.			GOP-3
NOTE 3.14 Ensure sufficient Emergency Feedwater Flow exists prior to raising power.			GOP-3
	RO	3.14. Increase Reactor Power to between 1% and 3%.	GOP-3
	RO	3.15. At the Point of Adding Heat, if NR-45, NIS RECORDER, had previously been selected to HI speed place the recorder in LO speed.	GOP-3
CAUTION 3.16 a. Adjustment of Tavg with the Rod Control System must not be attempted with the ROD CNTRL BANK SEL Switch in any position other than MAN. b. Manual rod control is required to establish equilibrium conditions, since C-5 blocks automatic rod withdrawal.			GOP-3
	RO	3.16. Maintain Tavg between 555°F and 559°F.	GOP-3
	CRS	3.17. Complete Attachment II.G, Operational Mode Change Plant Startup - Entering Mode 1, of GTP-702.	GOP-3
	CRS	3.18. Proceed to GOP-4A, Power Operation (Mode 1 - Ascending).	GOP-3
BOOTH OPERATOR:		IF called at any time to look at feedwater heaters because of the feedwater heater dump valve not being closed, wait 2 minutes and report back "no issues, everything is operating correctly".	GOP-3
EVALUATOR NOTE: The next event may be inserted following completion of the power ascension, or at any time per the discretion of the Lead Examiner.			GOP-3

Op Test No:	NRC-ILO-16-01	Scenario #	3	Event #	3	Page:	15	of	35
Event Description: Raise power to between 1% - 3%.									

Time	Position	Applicant's Actions or Behavior

GOP 3 REFERENCE PAGE

1. GENERAL NOTES

A. Procedure steps should normally be performed in sequence. However, it is acceptable to perform steps in advance after thorough evaluation of plant conditions and impact by the Shift Manager or Control Room Supervisor.

B. At least two licensed operators, one of whom is SRO licensed, must be present in the Control Room during Reactor Startup.

2. REACTOR CONTROL

A. Shutdown Bank Control:

1) The Shutdown Banks must be fully withdrawn whenever reactivity additions are being made by dilution, Xenon, T_{avg}, or control rods unless one of the following conditions exists:

a) The RCS is borated to Cold Shutdown concentration and verified by sample.

b) T_{avg} is 557°F and the RCS is borated to the hot, Xenon-free concentration and verified by sample. GOP-3

2) If the count rate on any source range channel increases by more than a factor of two during any increment of Shutdown Bank withdrawal, rod withdrawal shall be stopped and the Shutdown Bank reinserted. Until Reactor Engineering has made a satisfactory evaluation of the situation, rod withdrawal shall not resume.

B. Source Range Control:

1) Source Range Counts and Digital Rod Position indication should be monitored during any Shutdown and Control Bank withdrawal or insertion.

2) While in the Source Range, positive reactivity may be changed by only one controlled method.

C. Anticipate criticality anytime:

1) During rod motion.

2) Boron dilution is in progress.

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 4 Page: 16 of 35			
Event Description: High Temperature on XTF-31, transfer loads to alt power source.			
Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: XTF-31 will have elevated temperatures causing the crew to unload the transformer.			
BOOTH OPERATOR:		When directed - Initiate Event 4 (TRIGGER 4).	
Available Indications: XCP 633, 1-4, EMERG AUX XFMR XTF-31 TRBL Reports from the field			
	CRS	Enters XCP-633, 1-4, EMERG AUX XFMR XTF-31 TRBL	XCP-633, 1-4
	BOP	**Corrective Actions:** 1. Dispatch an Operator to XTF0031, EMERGENCY AUXILIARY Transformer #1, to determine the cause of the alarm. 2. If sudden pressure is the cause, refer to XFMR XTF31 LCKOUT 86T31 (XCP-639 4-2). 3. If necessary, contact PSE/Substation Maintenance for assistance.	XCP-633, 1-4
BOOTH OPERATOR:		When dispatched to XTF-31, wait 2 minutes and report "Multiple fans are off and oil temperatures are 93°C and slowly rising". If asked about winding temperatures, report "Winding temperatures are 115°C and slowly rising".	
	BOP	**Supplemental Actions:** 1. If oil temperatures exceed 92°C or winding temperatures exceed 125°C transfer loads from XTF0031, EMERGENCY AUXILIARY TRANSFORMER #1, per SOP-304, 115KV/7.2KV Operations.	XCP-633, 1-4
	CRS	Enters SOP-304, 115KV/7.2KV OPERATIONS, Section IV.C.	SOP-304
	CRS	2.1. Notify the System Controller of the applicable bus voltage limits from Enclosure B.	SOP-304
BOOTH OPERATOR:		If called again to report oil temperatures or winding temperatures, report: "Oil temperatures are 94°C and slowly rising" "Winding temperatures are 116°C and slowly rising".	
	BOP	2.2. If required, adjust the 115KV and/or 230KV alarm setpoints per Attachment VA and/or Attachment VB for the current lineup.	SOP-304
EVALUATOR NOTE: The crew may skip step 2.2. in order to transfer loads off of XTF-31 in a more timely manner.			

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 4 Page: 17 of 35
Event Description: High Temperature on XTF-31, transfer loads to alt power source.

Time	Position	Applicant's Actions or Behavior	
	BOP	2.4. Manually transfer BUS 1DB to alternate feed as follows: a. Ensure the following annunciators on Panel XCP-638 are clear: 1. 1-5, XTF4 OPC. 2. 1-6, XTF4 OPIS TRBL. 3. 2-5, XTF5 OPC. 4. 2-6, XTF5 OPIS TRBL. **CAUTION 2.4.b** **BUS 1DB XFER INIT Switch operation trips the Supplemental Instrument Air Compressor, due to an electrical perturbation caused by a momentary power interruption.** **NOTE 2.4.b** **If the Integrated Fire System computer is being powered from Train B, there will be a momentary power interruption to the computer.** b. Turn and hold BUS 1DB XFER INIT Switch to the N-E position. **(PEER ✓)** c. Verify the following: 1. BUS 1DB potential lights remain lit. 2. BUS 1DB ALT FEED breaker closes. 3. BUS 1DB NORM FEED breaker opens. d. Release BUS 1DB XFER INIT Switch and verify spring return to OFF. e. Match flags for the BUS 1DB ALT FEED and BUS 1DB NORM FEED breakers.	SOP-304
BOOTH OPERATOR:		If called again to report oil temperatures or winding temperatures after they have taken 1DB off the transformer, report: "Oil temperatures are 92°C and slowly lowering" "Winding temperatures are 114°C and slowly lowering".	

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 4 Page: 18 of 35		
Event Description: High Temperature on XTF-31, transfer loads to alt power source.		
Time	**Position**	**Applicant's Actions or Behavior**
	CRS	Enters Tech Spec 3.4.8.1, A.C. Sources Operating, Action a: a. With one offsite circuit of 3.8.1.1.a inoperable: 1. Demonstrate the OPERABILITY of the remaining offsite AC. sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter, and 2. If either EDG has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.3 separately for each such EDG within 24 hours unless the diesel is already operating, and 3. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

Tech Specs

EVALUATOR NOTE: The next event may be inserted following the CRS assessment of Tech Specs, or at any time per the discretion of the Lead Examiner.

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 5 Page: 19 of 35		
Event Description: Condensate Flow to Deaerator fails HIGH.		
Time	Position	Applicant's Actions or Behavior
EVALAUTOR NOTE: Flow controller for Condensate to the Deaerator will fail high in automatic. As flow rises, level in the Deaerator will rise.		
BOOTH OPERATOR:		When directed - Initiate Event 5 (TRIGGER 5).
Available Indications: XCP 632, 4-5, IPCS OPCRIT ALARM XCP 627, 4-1, DEAER STOR TK LVL HI/HI-HI Rising level on Deaerator Storage Tank Level Indicator LI-3135.		
	CRS	Enters XCP 627, 4-1, DEAER STOR TK LVL HI/HI-HI
	BOP	1. Place FLOW TO DEAERATOR in MAN and reduce flow to the DA Storage Tank as necessary. 2. Take manual control of LCV03235, DEAER START UP DRAIN CNTRL, and lower the level.
EVALUATOR NOTE: The BOP/CRS may take action as soon as they get the OPCRIT alarm in accordance with OAP-100.5 for equipment not responding properly in Automatic control.		
EVALUATOR NOTE: The next event may be initiated after the candidate has stabilized DA level or at any time at the discretion of the Lead Examiner.		

XCP 627, 4-1

XCP 627, 4-1

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 6 Page: 20 of 35		
Event Description: FCV-122 fails closed in AUTO.		
Time	Position	Applicant's Actions or Behavior
EVALUATOR NOTE: FCV-122 fails closed in AUTO, isolating charging flow.		
BOOTH OPERATOR:		When directed - Initiate Event 6 (TRIGGER 6).
Available Indications: XCP-614 5-1, CHG LINE FLO HI/LO. XCP-613 1-4, REGEN HX LTDN OUT TEMP HI. FI-122A, CHG FLOW GPM - no flow. PI-121, CHG PRESS PSIG - lowering value.		
	CRS	Enters XCP-614, 5-1, CHG LINE FLO HI/LO.
EVALUATOR NOTE: The following six steps are the "Corrective Actions" of the alarm response procedure.		
	CRS	1. If the running Charging Pump suction flowpath has become isolated, secure the Charging Pump and go to AOP-102.2, Loss of Charging.
	CRS	2. If the PUMP A(B) or PUMP C TRAIN A(B) ammeter indication is abnormal for the running Charging Pump and the pump must be tripped, go to AOP-102.2, Loss of Charging.
	RO	3. Monitor LT-112A and LT-115, % LEVEL, to verify proper VCT level.
	RO	4. Monitor FI-122A, CHG FLOW GPM.
	RO	5. Verify the Charging header valve lineup: a. Verify the following valves are open: 1. FCV-122, CHG FLOW. 2. MVG-8107, CHG LINE ISOL. 3. MVG-8108, CHG LINE ISOL. 4. Either of the following: a. PVT-8146, NORM CHG TO RCS LP B. b. PVT-8147, ALT CHG TO RCS LP A. b. If the Charging header has isolated go to AOP-102.2, Loss of Charging.

Right margin references: XCP-614 5-1 (beside the "Enters XCP-614" row and each of steps 1–5).

Op Test No: NRC-ILO-16-01	Scenario # 3 Event # 6	Page: 21 of 35
Event Description: FCV-122 fails closed in AUTO.		

Time	Position	Applicant's Actions or Behavior	
	RO	6. If Charging flow has NOT been lost but a loss of automatic control of FCV-122, CHG FLOW, is suspected perform the following: a. Place FCV-122, CHG FLOW, in MAN and adjust, as required, to maintain TI-140, REGEN HX OUT TEMP °F, between 250°F and 350°F while maintaining Pressurizer level. b. If FCV-122, CHG FLOW, fails to respond in MAN, perform SOP-102, Off Normal, Response To Malfunction Of FCV-122, to bypass FCV00122-CS, CHARGING HEADER FLOW CONTROL VALVE (AB-412 West Pen).	XCP-614 5-1

EVALUATOR NOTE: The next event may be initiated while the RO is re-establishing manual Pressurizer Level and Regenative Heat Exchanger Outlet temperature, or at any time at the discretion of the Lead Examiner.

Op Test No: NRC-ILO-16-01	Scenario # 3	Event # 7	Page: 22	of 35

Event Description: Feedwater Isolation Valve Accumulator low pressure alarm.

Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: The Feedwater Isolation Valve Accumulator low pressure alarm will come in and be evaluated for Tech Specs.			
BOOTH OPERATOR:		When directed - Initiate Event 7 (TRIGGER 7).	
Available Indications: XCP-625, 3-3 FIV A/B/C ACCUM PRESS LO			
	CRS	Enters XCP-625, 3-3 FIV A/B/C ACCUM PRESS LO**,** alarm response procedure.	XCP-625, 3-3
NOTE 1 a. If the affected valve is open, that valve will be inoperable if pressure decreases to less than 500 psi. b. If the affected valve is closed, that valve will be inoperable if pressure decreases to less than 75 psi.			XCP-625, 3-3
	CRS	1. Verify pressure on XPN 7301 (AB 436 West Penetration) and XPN 7302 (IB 436 East Penetration).	XCP-625, 3-3
BOOTH OPERATOR:		When called as Unit 6 to verify pressure on XPN 7301, wait 2 minutes and report back "accumulator pressure for PVG-1611A, A ISOL, is 0 psig". When called as Unit 7 to verify pressure on XPN 7302, wait 2 minutes and report back "accumulator pressure on "B" and "C" accumulators are both at 590 psig".	
	CRS	2. Refer to V.C. Summer Tech. Spec. 3.7.1.6.	XCP-625, 3-3
	CRS	Enters Tech Spec 3.7.1.6, Feedwater Isolation Valves, Action for MODE 2: With one feedwater isolation valve inoperable, subsequent and 3 operation in MODES 2 or 3 may proceed provided: a. The isolation valve is maintained closed. b. The provisions of Specification 3.0.4 are not applicable. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.	Tech Specs
EVALUATOR NOTE: The next event may be inserted following the CRS assessment of Tech Specs, or at any time per the discretion of the Lead Examiner.			

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 8 Page: 23 of 35			
Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.			
Time	Position	Applicant's Actions or Behavior	
EVALUATOR NOTE: An Earthquake will occur, this will cause the faulted Steam Generators and an ATWS.			
BOOTH OPERATOR:		When directed - Initiate Event 8 (TRIGGER 8).	
Available Indications: Both Reactor Trip breakers – RED light **lit** and GREEN light **dim**.			
	CRS	Directs RO and BOP to perform immediate actions of EOP-1.0, REACTOR TRIP OR SAFETY INJECTION.	
	CRS	Enters EOP-1.0, E-0 Reactor Trip or Safety Injection.	
IOA **Critical Task**	RO	1. Verify Reactor Trip: • Trip the Reactor using both Reactor Trip Switches. • Verify all Reactor Trip and Bypass Breakers are open. • Verify all Rod Bottom Lights are lit. • Verify Reactor Power level is decreasing.	EOP-1.0
IOA	BOP	2. Verify Turbine/Generator Trip: a. Verify all Turbine STM STOP VLVs are closed. b. Ensure Generator Trip (after 30 second delay): 1. Ensure the GEN BKR is open. 2. Ensure the GEN FIELD BKR is open. 3. Ensure the EXC FIELD CNTRL is tripped.	EOP-1.0
IOA	BOP	3. Verify both ESF buses are energized.	EOP-1.0
IOA	RO	4. Check if SI is actuated: a. Check if either: • SI ACT status light is bright on XCP-6107 1-1. OR • Any red first-out SI annunciator is lit on XCP-626 top row. b. Actuate SI using either SI ACTUATION Switch. c. GO TO Step 6.	EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 3	Event # 8	Page: 24 of 35
Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.			

Time	Position	Applicant's Actions or Behavior

REFERENCE PAGE FOR EOP-1.0

1 RCP TRIP CRITERIA

a. IF Phase B Containment Isolation has actuated (XCP-612 4-2), THEN trip all RCPs.

b. IF both of the following conditions occur, THEN trip all RCPs:

- SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

AND

- RCS Wide Range pressure is LESS THAN 1418 psig.

2 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.

3 MONITOR SPENT FUEL COOLING

Periodically check status of Spent Fuel Cooling by monitoring the following throughout event recovery:

- Spent Fuel Pool level.
- Spent Fuel Pool temperature.

4 RUPTURED STEAM GENERATOR

IF a RUPTURED Steam Generator has been positively identified, THEN throttle EFW to the RUPTURED Steam Generator WHEN its Narrow Range Level is GREATER THAN 26%[41%].

5 FAULTED STEAM GENERATOR

- IF a FAULTED Steam Generator has been positively identified, THEN isolate EFW to the faulted Steam Generator as soon as possible UNLESS all three Steam Generators are FAULTED.
- IF all three Steam Generators are FAULTED, THEN throttle EFW flow to all three Steam Generators to 50 gpm.

EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 3	Event # 8	Page: 25 of 35

Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.

Time	Position	Applicant's Actions or Behavior	
	BOP	6. Initiate ATTACHMENT 3, SI EQUIPMENT VERIFICATION.	EOP-1.0
	CRS	7. Announce plant conditions over the page system.	EOP-1.0
*	RO	8. Verify RB pressure has remained LESS THAN 12 psig on PR-951, RB PSIG (P-951), red pen.	EOP-1.0
	RO	9. Check RCS temperature: • With any RCP running, RCS Tavg is stable at OR trending to 557°F. **(NO)** OR • With no RCP running, RCS Tcold is stable at OR trending to 557°F.	EOP-1.0
* **Critical Task**	RO	**Alternative Action Step:** 9. IF RCS temperature is LESS THAN 557°F AND decreasing, THEN stabilize temperature by performing the following as required: a. Close IPV-2231, MS/PEGGING STM TO DEAERATOR. b. Perform one of the following: • IF Narrow Range SG level is LESS THAN 26% [41%] in all SGs, THEN reduce EFW flow as necessary to stop cooldown, while maintaining total EFW flow GREATER THAN 450 gpm. OR • WHEN Narrow Range SG level is GREATER THAN 26% [41%] in at least one SG, THEN control EFW flow as necessary to stabilize RCS temperature at 557°F. c. Initiate ATTACHMENT 6, STEAM VALVE ISOLATION, while continuing with this procedure. d. IF RCS cooldown continues, THEN close: • MS Isolation Valves, PVM-2801A(B)(C). • MS Isolation Bypass Valves, PVM-2869A(B)(C).	EOP-1.0

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 8 Page: 26 of 35

Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.

Time	Position	Applicant's Actions or Behavior	
	RO	10. Check PZR PORVs and Spray Valves: a. PZR PORVs are closed. b. PZR Spray Valves are closed. c. Verify power is available to at least one PZR PORV Block Valve: • MVG-8000A, RELIEF 445 A ISOL. • MVG-8000B, RELIEF 444 B ISOL. • MVG-8000C, RELIEF 445 B ISOL. d. Ensure one of the following Block Valves is open unless it was closed to isolate an open PZR PORV: • MVG-8000A, RELIEF 445 A ISOL. • MVG-8000B, RELIEF 444 B ISOL.	EOP-1.0
NOTE - Step 11 Seal Injection flow should be maintained to all RCPs.			EOP-1.0
	RO	11. Check if RCPs should be stopped: a. Check if either of the following criteria is met: • Annunciator XCP-612 4-2 is lit (PHASE B ISOL). OR • RCS pressure is LESS THAN 1418 psig AND SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. b. Stop all RCPs.	EOP-1.0
	RO	12. Verify no SG is FAULTED: • No SG pressure is decreasing in an uncontrolled manner. **(NO)** • No SG is completely depressurized.	EOP-1.0
	CRS	**Alternative Action Step:** 12. GO TO EOP-3.0, E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.	EOP-1.0

Op Test No: NRC-ILO-16-01	Scenario # 3	Event # 8 Page: 27 of 35
Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.		
Time	Position	Applicant's Actions or Behavior
EVALUATOR NOTE: The crew may have a red path on Heat Sink because IF they throttled EFW flow to <450 gpm. If so, they will transition to EOP-15.0, Response to Loss of Secondary Heat Sink. They will read a Caution at the beginning of the procedure and immediately transition out of EOP-15.0 because they throttled EFW on purpose.		
	CRS/CREW	Notices a Red Path on Heat Sink and transitions to EOP-15.0, Response to Loss of Secondary Heat Sink.
CAUTION • If total EFW flow is LESS THAN 450 gpm due to operator action, this procedure should NOT be performed, since these actions are NOT appropriate if 450 gpm EFW flow is available. • If a NON-FAULTED SG is available, feed flow should NOT be reestablished to any FAULTED SG, to prevent thermal shock to SG tubes.			EOP-15.0
	CRS	Transitions out of EOP-15.0 into EOP-3.0, E-2 Faulted Steam Generator Isolation
CAUTION • At least one SG must be maintained available for RCS cooldown. • Any FAULTED SG or secondary break should remain isolated during subsequent recovery actions unless needed for RCS cooldown, to prevent reinitiating the break.			EOP-3.0
	BOP	1. Ensure all the following are closed: • MS Isolation Valves, PVM-2801A(B)(C). • MS Isolation Bypass Valves, PVM-2869A(B)(C).	EOP-3.0
BOOTH OPERATOR:		When contacted to shut MSIVs, Wait 2 minutes and report back that you were unable to close the MSIVs.
	BOP	2. Check if any SG is NON-FAULTED: • Pressure in any SG is stable OR increasing. **(NO)** • Any SG is NOT completely depressurized.	EOP-3.0
	BOP	**Alternative Action Step:** 2. IF all SG pressures are decreasing in an uncontrolled manner OR completely depressurized, THEN GO TO EOP-3.1, ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, Step 1.	EOP-3.0
	CRS	Enters EOP-3.1, ECA-2.1 Uncontrolled Depressurization of all Steam Generators.

Op Test No:	NRC-ILO-16-01	Scenario #	3	Event #	8	Page:	28	of	35

Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.

Time	Position	Applicant's Actions or Behavior

REFERENCE PAGE FOR EOP-3.1

1 SI REINITIATION CRITERIA

IF either of the following conditions occurs, THEN start Charging Pumps and operate valves as necessary:

- RCS subcooling on TI-499A(B), A(B) TEMP °F, is LESS THAN 52.5°F [67.5°F].

OR

- PZR level can NOT be maintained GREATER THAN 10% [28%].

2 SECONDARY INTEGRITY TRANSITION CRITERION

IF any SG pressure increases at any time, except while performing SI Termination in Steps 13 through 18, THEN GO TO EOP-3.0, E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1. EOP-3.1

3 TUBE RUPTURE TRANSITION CRITERIA

IF any SG level increases in an uncontrolled manner OR if any SG has abnormal radiation, THEN start Charging Pumps and operate valves as necessary, and GO TO EOP-4.0, E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.

4 COLD LEG RECIRCULATION TRANSITION CRITERION

IF RWST level decreases to LESS THAN 18%, THEN GO TO EOP-2.2, ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1.

5 REDUCING CONTROL ROOM EMERGENCY VENTILATION

Reduce Control Room Emergency Ventilation to one train in operation within 30 minutes of actuation. REFER TO SOP-505, CONTROL BUILDING VENTILATION SYSTEM.

Op Test No: NRC-ILO-16-01	Scenario # 3	Event # 8	Page: 29 of 35
Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.			

Time	Position	Applicant's Actions or Behavior	
	BOP	1. Isolate secondary pressure boundaries for all SGs: a. Close all of the following valves: • MS Isolation, PVM-2801A(B)(C). • MS Isolation Bypass, PVM-2869A(B)(C). • FW Flow Control, FCV-478(488)(498). • FW Isolation, PVG-1611A(B)(C). • SG Blowdown, PVG-503A(B)(C). • FW Flow Control Bypass, FCV-3321(3331)(3341). **CAUTION - Step 1.b** **If the TD EFW Pump is the only available source of feed flow, the steam supply to the TD EFW Pump must be maintained from at least one SG, to maintain a secondary heat sink.** b. Complete isolation of all SGs: 1. Close all the following valves: • SG Chemical Feed Isolation, MVK-1633A(B)(C). • MS Drain Isolation, PVT-2843A(B)(C) PVT-2877A(B). 2. Locally open the following breakers: • XMC1DA2X 05EH, EF PUMP MAIN STEAM BLOCK VLV XVG2802A-MS (IB-463). • XMC1DB2Y 05EH, EMERG FEEDWATER PUMP MAIN STEAM BLOCK XVG2802B-MS (AB-463). 3. Locally close the following valves (IB-436 East Pen): • XVG02802A-MS, MS HEADER B EF PUMP TURBINE SUPPLY VLV. • XVG02802B-MS, MS HEADER C EF PUMP TURBINE SUPPLY VLV. 4. Place all Steamline PWR RELIEF A(B)(C) SETPT Controllers in MAN and closed. 5. Place all Steamline Power Relief A(B)(C) Mode Switches in PWR RLF.	EOP-3.1
BOOTH OPERATOR:		When contacted to locally open breakers and close 2802A/B, wait 3 minutes, use the LOAs on the LOA RESET PANEL to open the breakers and close the valves, Then report back "I've opened the breakers for 2802A/B and have closed both valves".	

Op Test No: NRC-ILO-16-01	Scenario # 3	Event # 8	Page: 30 of 35
Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.			

Time	Position	Applicant's Actions or Behavior	
		CAUTION - Step 2 A minimum EFW flow of 50 gpm must be maintained to each SG that has a Narrow Range level LESS THAN 26% [41%], to minimize thermal shock to SG components.	EOP-3.1
		NOTE - Step 2 Shutdown margin should be monitored during RCS cooldown.	EOP-3.1
	BOP	2. Ensure the RCS cooldown is minimized: a. Place MD EFP RESET to RESET. b. Place TD EFP RESET to RESET. c. Verify the cooldown rate in the RCS Cold Legs is LESS THAN 100°F/hr. d. Verify Narrow Range level in all SGs is LESS THAN 60%. e. Verify RCS Thot is stable OR decreasing.	EOP-3.1
		NOTE - Step 3 Seal Injection flow should be maintained to all RCPs.	EOP-3.1
	RO	3. Check if RCPs should be stopped: a. Verify annunciator XCP-612 4-2 is NOT lit (PHASE B ISOL). b. Check if RCS pressure is LESS THAN 1418 psig AND SI flow is indicated on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. **NOTE - Step 3.c** **RCPs should NOT be stopped if the RCS pressure decrease is due solely to the cooldown.** c. Stop all RCPs.	EOP-3.1
	RO	4. Check PZR PORVs and Block Valves: a. Verify power is available to the PZR PORV Block Valves: 1. MVG-8000A, RELIEF 445 A ISOL. 2. MVG-8000B, RELIEF 444 B ISOL. 3. MVG-8000C, RELIEF 445 B ISOL. **CAUTION - Step 4.b** **If any PZR PORV opens because of high PZR pressure, Step 4.b should be repeated after pressure decreases to LESS THAN 2330 psig, to ensure the PORV recloses.** b. Verify all PZR PORVs are closed. c. Verify at least one PZR PORV Block Valve is open.	EOP-3.1

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # 8 Page: 31 of 35		
Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.		
Time	**Position**	**Applicant's Actions or Behavior**
	RO	5. Reset both SI RESET TRAIN A(B) Switches.
	RO	6. Reset Containment Isolation: • RESET PHASE A - TRAIN A(B) CNTMT ISOL. • RESET PHASE B - TRAIN A(B) CNTMT ISOL.
NOTE - Step 7 Any high radiation level received on a radiation monitor that was unisolated at event initiation, may be considered a valid alarm.		
	BOP	7. Check if Secondary radiation levels are normal: a. Check radiation levels normal on all unisolated radiation monitors: • RM-G19A(B)(C), STMLN HI RNG GAMMA. • RM-L3, STEAM GENERATOR BLOWDOWN LIQUID MONITOR. • RM-L10, SG BLOWDOWN CW DISCHARGE LIQUID MONITOR. • RM-A9, CNDSR EXHAUST GAS ATMOS MONITOR. b. Place SVX-9398A(B)(C), SG A(B)(C) SMPL ISOL, in AUTO. c. Notify Chemistry to sample all SG secondary sides, and screen samples for abnormal activity using a frisker.
CAUTION - Step 8 RCS pressure should be monitored. If RCS pressure decreases in an uncontrolled manner to LESS THAN 325 psig, the RHR Pumps must be manually restarted to supply water to the RCS.		
	RO	8. Check if RHR Pumps should be stopped: a. Check if any RHR Pump is running with suction aligned to the RWST. b. Check RCS pressure: 1. RCS pressure is GREATER THAN 325 psig. 2. RCS pressure is stable OR increasing. c. Stop any RHR Pump which is running with suction aligned to the RWST and place in Standby.
	BOP	9. Verify RWST level is GREATER THAN 18%.

EOP-3.1 (margin references beside each row: steps 5, 6, Note, 7, Caution, 8, 9)

Op Test No: NRC-ILO-16-01	Scenario # 3	Event # 8	Page: 32 of 35
Event Description: Earthquake, followed by three faulted Steam Generators and an ATWS.			

Time	Position	Applicant's Actions or Behavior	
	RO	10. Establish Instrument Air to the RB: a. Start one Instrument Air Compressor and place the other in Standby. b. Verify PI-8342, INSTR AIR HDR PRESS PSIG, indicates GREATER THAN 60 psig. c. Open PVA-2659, INST AIR TO RB AIR SERV. d. Open PVT-2660, AIR SPLY TO RB.	EOP-3.1
	RO	11. Check if SI Accumulators should be isolated: a. Verify RCS pressure is LESS THAN 195 psig. **(NO)**	EOP-3.1
	CRS	**Alternative Action Step:** 11. Check if SI Accumulators should be isolated: a. WHEN RCS pressure is LESS THAN 195 psig, THEN COMPLETE Step 11. CONTINUE WITH Step 12.	EOP-3.1
*	RO	12. Check if SI flow should be reduced: a. Verify RCS subcooling on TI-499A(B), A(B) TEMP °F, is GREATER THAN 52.5°F [67.5°F]. b. Verify RCS pressure is stable OR increasing. **NOTE - Step 12.c** **If PZR level is LESS THAN 10% [28%], the PZR should refill from SI flow after pressure is stabilized.** c. Verify PZR level is GREATER THAN 10% [28%].	EOP-3.1
	RO	13. Stop all but one Charging Pump and place in Standby.	EOP-3.1
	RO	14. Verify RCS pressure is stable OR increasing.	EOP-3.1
Critical Task	RO	15. Establish Normal Charging: a. Close FCV-122, CHG FLOW. b. Open both MVG-8107 and MVG-8108, CHG LINE ISOL. c. Adjust FCV-122, CHG FLOW, to obtain 70 gpm Charging flow. d. Close both MVG-8801A(B), HI HEAD TO COLD LEG INJ.	EOP-3.1

EVALUTORE NOTE: The scenario may be terminated at any point after they have secured Safety Injection.

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # N/A Page: 33 of 35			
Event Description: EOP-1.0, Attachment 3			
Time	Position	Applicant's Actions or Behavior	
	BOP	1. Ensure EFW Pumps are running: a. Ensure both MD EFW Pumps are running. b. Verify the TD EFW Pump is running if necessary to maintain SG levels.	Attachment 3
	BOP	2. Ensure the following EFW valves are open: • FCV-3531(3541)(3551), MD EFP TO SG A(B)(C). • FCV-3536(3546)(3556), TD EFP TO SG A(B)(C). • MVG-2802A(B), MS LOOP B(C) TO TD EFP.	Attachment 3
	BOP	3. Verify total EFW flow is GREATER THAN 450 gpm.	Attachment 3
	BOP	4. Ensure FW Isolation: a. Ensure the following are closed: • FW Flow Control, FCV-478(488)(498). • FW Isolation, PVG-1611A(B)(C). • FW Flow Control Bypass, FCV-3321(3331)(3341). • SG Blowdown, PVG-503A(B)(C). • SG Sample, SVX-9398A(B)(C). b. Ensure all Main FW Pumps are tripped.	Attachment 3
	BOP	5. Ensure SI Pumps are running: • Two Charging Pumps are running. • Both RHR Pumps are running.	Attachment 3
	BOP	6. Ensure two RBCU Fans are running in slow speed (one per train).	Attachment 3

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # N/A Page: 34 of 35			
Event Description: EOP-1.0, Attachment 3			
Time	Position	Applicant's Actions or Behavior	
	BOP	7. Verify Service Water to the RBCUs: a. Ensure two Service Water Pumps are running. b. Verify Service Water Booster Pump A is stopped. **(NO)** **Alternative Action Step:** b. GO TO Step 7.e. 7e. Verify that Service Water Booster Pump B is stopped. **(NO)** **Alternative Action Step:** e. GO TO Step 7.h. 7h. Verify GREATER THAN 2000 gpm flow for each train on: • FI-4466, SWBP A DISCH FLOW GPM. • FI-4496, SWBP B DISCH FLOW GPM.	Attachment 3
	BOP	8. Verify two CCW Pumps are running.	Attachment 3
	BOP	9. Ensure two Chilled Water Pumps and Chillers are running.	Attachment 3
	BOP	10. Verify both trains of Control Room Ventilation are running in Emergency Mode.	Attachment 3
	BOP	11. Check if Main Steamlines should be isolated: a. Check if any of the following conditions are met: • RB pressure GREATER THAN 6.35 psig. OR • Steamline pressure LESS THAN 675 psig. OR • Steamline flow GREATER THAN 1.6 MPPH AND Tavg LESS THAN 552°F. b. Ensure all the following are closed: • MS Isolation Valves, PVM-2801A(B)(C). • MS Isolation Bypass Valves, PVM-2869A(B)(C).	Attachment 3

Op Test No: NRC-ILO-16-01 Scenario # 3 Event # N/A Page: 35 of 35

Event Description: EOP-1.0, Attachment 3

Time	Position	Applicant's Actions or Behavior	
	BOP	12. Ensure Excess Letdown Isolation Valves are closed: • PVT-8153, XS LTDN ISOL. • PVT-8154, XS LTDN ISOL.	Attachment 3
	BOP	13. Verify ESF monitor lights indicate Phase A AND Containment Ventilation Isolation on XCP-6103, 6104, and 6106. REFER TO ATTACHMENT 4, CONTAINMENT ISOLATION VALVE MCB STATUS LIGHT LOCATIONS, as needed.	Attachment 3
	BOP	14. Verify proper SI alignment: a. Verify SI valve alignment by verifying SAFETY INJECTION/PHASE A ISOL monitor lights are bright on XCP-6104. b. Verify all SAFETY INJECTION monitor lights are dim on XCP-6106. c. Verify SI flow on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM. d. Check if RCS pressure is LESS THAN 325 psig.	Attachment 3
	BOP	Report completion of Attachment 3.	

EVALUATOR NOTE: ATTACHMENT 3 is complete.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPS-001F-N18

Manual Safety Injection with Charging Pumps Fail to Start (Alternate Path)

(NRC JPM a)

CANDIDATE: ______________________________

EXAMINER: ______________________________

TASK: 006-013-01-01 Manually Initiate Safety Injection

TASK STANDARD:

The "B" Charging pump has been started and is injecting into the RCS.

TERMINATING CUE:

EOP-1.0 Attachment 3 is completed and CRS is notified.

PREFERRED EVALUATION LOCATION SIMULATOR

PREFERRED EVALUATION METHOD PERFORM

REFERENCES:

EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION

INDEX NO	***K/A NO.***		***RO***	***SRO***
006000A212	A2.12	Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions requiring actuation of ECCS.	4.5	4.8

TOOLS: Rack copy of EOP-1.0, Attachment 3, SI EQUIPMENT VERIFICATION

EVALUATION TIME 15 min ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)(7)

TIME START: ________ TIME FINISH: ________ PERFORMANCE TIME: ________

PERFORMANCE RATING: SAT: ____ UNSAT: ____

CANDIDATE: ____________________

EXAMINER: ____________________ ____________/________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

A small break LOCA has occurred. EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION has been entered.

Immediate actions 1 through 3 have been performed.

INITIATING CUES:

The CRS directs you to perform EOP-1.0, beginning at immediate action step 4.

A surrogate operator will acknowledge non-related alarms per your direction.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 4: Check if SI is actuated:

Step 4 a: Check if either:

- SI ACT status light is bright on XCP-6107 1-1.

OR

- Any red first-out SI annunciator is lit on XCP-626 top row.

Step 4 b: Actuate SI using either SI ACTUATION Switch.

STEP STANDARD:

Candidate observes XCP-626, 1-5 PZR SI lit indicating SI criteria met.

Candidate places either one of the 2 Safety Injection Manual actuation switches in the ACTUATE position.

CUES:

BOOTH OPERATOR CUE: Place the Simulator in RUN when the Evaluator indicates the JPM may begin.

EVALUATOR NOTE: Either switch will actuate the Train "A" equipment but Train "B" equipment will NOT start from an SI signal. The candidate must manually reposition all Train "B" equipment. Candidate may operate BOTH switches to assure themselves that Train "B" will not actuate.

EVALUATOR CUE: Once the candidate has manually actuated SI provide the following verbal cue "CRS Directs you to perform Attachment 3, SI EQUIPMENT VERIFICATION" then provide the rack copy of EOP-1.0, Attachment 3.

EVALUATOR CUE: IF the candidate notes RCS pressure has decreased to less than 1418 psig and indicates they are going to stop RCPs per the reference page guidance, inform them another operator will perform that action and they should continue with EOP-1.0 Attachment 3 actions.

EVALUATOR CUE: Acknowledge any communications as CRS.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Step 1: Ensure EFW Pumps are running:

Step 1a: Ensure both MD EFW Pumps are running.

Step 1b: Verify the TD EFW Pump is running if necessary to maintain SG levels.

STEP STANDARD:

Candidate locates MDEFP controls; verifies both the "A" and "B" pump breakers red lights ON, green lights OFF and normal running amps indicated on the ammeters.

Candidate locates TDEFP controls; checks turbine speed indicates normal.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 2: Ensure the following EFW valves are open:

- FCV-3531(3541)(3551), MD EFP TO SG A(B)(C).
- FCV-3536(3546)(3556), TD EFP TO SG A(B)(C).
- MVG-2802A(B),MS LOOP B(C) TO TD EFP.

STEP STANDARD:

Candidate verifies red lights ON, green lights OFF for MDEFP to SGs Control Valves, FCV-3531(3541)(3551).

Candidate verifies red lights ON, green lights OFF for TDEFP to SGs Control Valves, FCV-3536(3546)(3556).

Candidate verifies red lights ON, green lights OFF on TDEFP steam supply valves MVG-2802A(B).

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 3: Verify total EFW flow is GREATER THAN 450 gpm.

STEP STANDARD:

Candidate locates flow indicators and verifies total flow greater than 450 gpm.

CUES:

EVALUATOR NOTE: Flow indication is located in several places. One location is panel XCP-6111, flow meters FI-3561, FI-3571, and FI-3581 another is panel XCP-6112 flow meters FI-3561B, 3571B and 3581B are on Panel XCP-6112 or various SIPCS screens.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 5

Step 4: Ensure FW Isolation:

Step 4a: Ensure the following are closed:

- FW Flow Control, FCV-478(488)(498).
- FW Isolation, PVG-1611A(B)(C).
- FW Flow Control Bypass, FCV-3321(3331)(3341).
- SG Blowdown, PVG-503A(B)(C).
- SG Sample, SVX-9398A(B)(C).

Step 4b: Ensure all Main FW Pumps are tripped.

STEP STANDARD:

Candidate locates:

FW Flow Control Valve indications; verifies red lights OFF, green lights ON for FCV-478(488)(498).

FW Isolation Valve indications; verifies red lights OFF, green lights ON for PVG-1611A(B)(C).

FW Flow Control Bypass Valve indication; verifies red light OFF, green light ON for FCV-3321(3331)(3341).

SG Blowdown Valve indications; verifies red lights OFF, green lights ON for PVG-503A(B)(C).

SG Sample Valve indications; verifies red lights OFF, green lights ON for SVX-9398A(B)(C).

Main Feed Pump TRIP/RESET switches; observes all 3 amber RESET lights ON **or** observes red RESET status box for each main feed pump on the HMI screen at the Feedwater station. Places each TRIP/RESET switch to TRIP, observes green TRIP light ON, and amber RESET light OFF for each pump and green TRIPPED status box for each pump on the HMI screen.

CUES:

EVALUATOR NOTE: All indications are at the front of the Control Room in and around the Feedwater station except the Blowdown Sample valves which are located on Panel XCP-6104 near the RBCU controls.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 6

Step 5: Ensure SI Pumps are running:

- Two Charging Pumps are running.
- Both RHR Pumps are running.

STEP STANDARD:

Candidate locates Charging Pump Controls and notes;

"A" Charging Pump breaker indicates red light ON, green light OFF, pump amps very low and no Charging flow indicated. Places breaker in pull to lock.

"B" Charging Pump breaker indicates red light OFF, green light ON and no amps indicated on the ammeter. Places Control Switch to START, observes breaker indicates red light ON, green light OFF and amps at running amps.

Candidate locates RHR Pump Controls and notes;

"A" pump breaker indicates red light ON, green light OFF and normal running amps on the ammeter.

"B" pump breaker indicates red light OFF, green light ON and zero amps on the ammeter. Places control switch to START, observes breaker indicates red light ON, green light OFF and amps at running amps.

CUES:

EVALUATOR NOTE: The critical step is to start the "B" Charging pump to assure injection flow for the Small Break LOCA that is in progress. Candidate may start the "B" CCW pump prior to starting "B" Charging pump to assure it has cooling water. This is the point at which the JPM becomes Alternate Path.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 7

Step 6: Ensure two RBCU Fans are running in slow speed (one per train).

STEP STANDARD:

Candidate locates RBCU controls and notes;

Red light ON, green light OFF at the 1A SLOW switch and running amps on the 1A SLOW, RBCU Fan Ammeter.

Red light OFF, green light ON at the 1B SLOW switch and zero amps on the 1B SLOW, RBCU Fan Ammeter. Places control switch 1B NORM to STOP. Places control switch 1B SLOW to START, observes breaker indicates red light ON, green light OFF and running amps on 1B SLOW, RBCU Fan Ammeter.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 8

Step 7: Verify Service Water flow to RBCUs:

Step 7a: Ensure two Service Water Pumps are running.

STEP STANDARD:

Candidate locates Service Water Pump controls and notes; both the “A” and “B” pump breakers indicate red light ON, green light OFF and normal running amps on the ammeter.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 9

Step 7b: Verify that Service Water Booster Pump A is stopped. (NO)

Step 7b: ALTERNATIVE ACTION: GO TO Step 7e.

STEP STANDARD:

Candidate locates Service Water Booster Pump controls and notes the “A” Pump breaker indicates red light ON, green light OFF and normal running amps on the ammeter.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 10

Step 7e: Verify that Service Water Booster Pump B is stopped.

STEP STANDARD:

Candidate locates Service Water Booster Pump controls and notes the “B” Pump breaker indicates red light OFF, green light ON and zero amps on the ammeter.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 11

Step 7f: Verify both of the following:

- XVB-3107B, RBCU 64B/65B RTN TO SW PND is closed.

  AND

- Alarm XCP-605 1-5 XVB3107B-SW SLOW CLOSURE is NOT lit.

STEP STANDARD:

Candidate locates switch for XVB-3107B; verifies red lights OFF, green lights ON.

Candidate locates annunciator panel 605; verifies Alarm 1-5 is NOT lit.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 12

Step 7g: Start Service Water Booster Pump B.

STEP STANDARD:

Candidate locates “B” Service Water Booster Pump controls; places control switch to START, observes breaker indicates red light ON, green light OFF and amps at running amps and discharge valve XVB-3106B stroking open.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 13

Step 7h: Verify GREATER THAN 2000 gpm flow for each train on:

- FI-4466, SWBP A DISCH FLOW GPM.
- FI-4496, SWBP B DISCH FLOW GPM.

STEP STANDARD:

Candidate locates FI-4466, SWBP A DISCH FLOW GPM and FI-4496, SWBP B DISCH FLOW GPM and verifies each header indicates greater than 2000 gpm

CUES:

EVALUATOR NOTE: Flow indication on the Train “B” Service Water Booster pump will be elevated above the value displayed for Train “A” because MVG-3109D, RBCU 65B OUTLET ISOL valve did not recieve its close signal due to the Train “B” SI failure. Therefore Train “B” Service water booster pump has a parallel flow path allowing more flow.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 14

Step 8: Verify two CCW Pumps are running.

STEP STANDARD:

Candidate locates CCW Pump Controls and notes;

> “A” pump breaker indicates red light ON, green light OFF and normal running amps on the ammeter.
>
> “B” pump breaker indicates red light OFF, green light ON and zero amps on the ammeter. Places control switch to START, observes breaker indicates red light ON, green light OFF and amps at running amps

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 15

Step 9: Ensure two Chilled Water Pumps and Chillers are running.

STEP STANDARD:

Candidate locates safety related Chiller Controls and notes;

> “A” and “B” Pump breakers indicate red light ON, green light OFF.
>
> “A” and “B” Chiller Units breakers indicate red light ON, green light OFF.

CUES:

EVALUATOR NOTE: The Chiller controls are behind the main control boards at the HVAC panel.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 16

Step 10: Verify both trains of Control Room Ventilation are running in Emergency Mode. (NO)

Step 10 Alternative action: Start both trains of Control Room Emergency Ventilation as follows:

For Train B:

1) Ensure XFN-32B, SPLY FAN B, is running.

2) Start XFN-30B, EMERG FLTR FAN B.

STEP STANDARD:

Candidate locates Control Room Ventilation Controls and notes:

XFN-32A SPLY FAN A indicates; red light ON, green light OFF.

XFN-30A EMERG FLTR FAN A indicates; red light ON, green light OFF.

XFN-32B SPLY FAN B indicates; red light OFF, green light ON, places switch for XFN-32B to START and observes; red light ON, green light OFF.

XFN-30B EMERG FLTR FAN B indicates: red light OFF, green light ON, places switch for XFN-30B to START and observes red light ON, green light OFF.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 17

Step 11: Check if Main Steamlines should be isolated: (NO)

- RB pressure GREATER THAN 6.35 psig.

  OR

- Steamline pressure LESS THAN 675 psig.

  OR

- Steamline flow GREATER THAN 1.6 MPPH AND Tavg LESS THAN 552°F.

Step 11a: GO TO Step 12.

STEP STANDARD:

Candidate locates RB Pressure indications and verifies RB Pressure is less than 6.35 psig.

Candidate locates Steam Line Pressure indications and verifies Steam Line Pressure is less than 675 psig.

Candidate locates Steam Line Flow indications and verifies Steam Line flow is less than 1.6 MPPH.

CUES:

EVALUATOR NOTE: RB pressure indication may be found on panel XCP-6103 at indicators PI-950, 951, 952 and 953. Steam Line pressure indication may be found at panel XCP-6111at PI-475, 485, and 495. Steam line flow may be found at panel XCP-6111 at FI-474, 475, 484, 485, 494 and 495.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 18

Step 12: Ensure Excess Letdown Isolation Valves are closed:

- PVT-8153, XS LTDN ISOL.
- PVT-8154, XS LTDN ISOL.

STEP STANDARD:

Candidate locates Excess Letdown Isolation valve indications; verifies red lights OFF, green lights ON for PVT-8153 and PVT-8154.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 19

Step 13: Verify ESF monitor lights indicate Phase A and Containment Ventilation Isolation on XCP-6103, 6104, and 6106. REFER TO ATTACHMENT 4, CONTAINMENT ISOLATION VALVE MCB STATUS LIGHT LOCATIONS, as needed.

Step 13: Alternative Action: Perform the following:

a) Actuate Phase A/Containment Ventilation Isolation by placing either CS-SG02A(B), TRAIN A & B, Switch to ACTUATE.

STEP STANDARD:

Candidate locates ESF monitor lights on XCP-6103 and 6106 and verifies Phase A and Containment Ventilation Isolation are satisfied.

Candidate locates ESF monitor lights on XCP-6104 and notes required lights are DIM. Candidate locates TRAIN A&B CS-SG02A and TRAIN A&B CS-SG02B and places either or both switch to ACTUATE then observes; all specified XCP-6104 status lights are BRIGHT.

CUES:

EVALUATOR NOTE: Attachment 4 is located in the same folder as Attachment 3. The Candidate may flip back to Attachment 4 and perform place keeping.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 20

Step 14: Verify proper SI alignment:

Step 14a: Verify SI valve alignment by verifying SAFETY INJECTION/PHASE A ISOL monitor lights are bright on XCP-6104.

Step 14a: Alternative action: Ensure proper SI valve alignment:

1) Open MVG-8801A(B), HI HEAD TO COLD LEG INJ.
2) Close MVG-8107 and MVG-8108, CHG LINE ISOL.
3) Open LCV-115B(D), RWST TO CHG PP SUCT.
4) Close LCV-115C(E), VCT OUTLET ISOL.
5) Open MVG-8809A(B), RWST TO RHR PP A(B).
6) Open MVG-8888A(B), RHR LP A(B) TO COLD LEGS.

STEP STANDARD:

Candidate locates Safety Injection monitor lights on XCP-6104 and observes 6-8, HI HEAD TO COLD LEG INJ 8801B OPEN is DIM, 8-5, VCT TO CHG PP ISOL 115E CLSD is DIM, 8-6, CHG PP TO RCS ISOL 8108 CLSD is DIM and 8-12, RWST TO CHG PP 115D OPEN is DIM.

Candidate locates MVG-8801B, HI HEAD TO COLD LEG INJ switch and places it in OPEN, observes; red light ON, green light OFF.

Candidate locates MVG-8108, CHG LINE ISOL switch and places it in CLOSE, obseves red light OFF, green light ON.

Candidate locates LCV-115D, RWST TO CHG PP SUCT switch and places it in OPEN, observes red light ON. green light OFF.

Candidate locates LCV-115E, VCT OUTLET ISOL switch and places it in CLOSE, observes red light OFF. green light ON.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 21

Step 14b: Verify all SAFETY INJECTION monitor lights are dim on XCP-6106.

Step 14c: Verify SI flow on FI-943, CHG LOOP B CLD/HOT LG FLOW GPM.

STEP STANDARD:

Candidate locates Safety Injection monitor lights on XCP-6106 and verifies all lights are dim.

Candidate locates FI-943 and verifies SI flow is indicated.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 22

Step 14d: Check if RCS pressure is LESS THAN 325 psig. (NO)

Step 14d: Alternative Action: Notify the CRS that ATTACHMENT 3, SI EQUIPMENT VERIFICATION, is complete.

STEP STANDARD:

Candidate locates PI-403, WR PRES PSIG and verifies RCS pressure is greater than 325 psig.

Candidate notifies CRS that Attachment 3 is complete.

CUES:

EVALUATOR NOTE: Candidate may report a list of actions taken due to failure of “B” train SI to actuate.

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPS-001F-N18, Manual Safety Injection with Charging Pumps Fail to Start (Alternate Path) (NRC JPM a)

IC SET: 291

INSTRUCTIONS:

If IC-291 is designated for this JPM then reset to IC-291 leaving the simulator in FREEZE.

1. When Candidate is ready (on Evaluator cue) go to RUN.

If IC-291 is **not** designated for this JPM then initial conditions may be established by resetting to IC-10 and following the below directions:

1. With the simulator reset to IC-10 and in FREEZE, insert the following:

- **PRE-LOAD**
  - MAL-PCS005A
    SAFETY INJECTION FAILURE TRAIN A
    Fail To: FAIL TO AUTO INIT
  - MAL-PCS005B
    SAFETY INJECTION FAILURE TRAIN B
    Fail To: TOTAL FAILURE
- **AUTO-TRIGGER 1** X09i073a | X03i049a = = 1 (Either SI manual switch taken to actuate)
  - PMP-CS004S
    XPP0043A CHRG/SI PMP A SHEARED SHAFT
    Delay: 20 sec

2. Place the simulator in RUN and insert the following:
   - MAL-RCS006A
     REACTOR COOLANT SYSTEM LEAK COLD LEG (LOOP 1)
     Final Value 1700 GPM
3. When RCS Pressure is less than 1850 psig and greater than 1418 psig, place the Simulator in FREEZE and save to the desired IC.
4. When Candidate is ready (on Evaluator cue) go to RUN.

COMMENTS:

Provide spare operator to silence alarms.

BOOTH OPERATOR: Use LOA resets page to silence HVAC alarms when they come in.

Mark strip chart recorders with date and time at the completion of each performance of this JPM.

Roll strip chart recorders to show no traces from the just completed performance after marking them as noted above.

CRITICAL TASK METHODOLOGY:

Step 6 is critical because the plant is experiencing a Small Break LOCA and neither Charging/SI pump is injecting into the RCS until the Candidate starts the “B” Charging pump.

REVISION HISTORY:

This JPM is new for the 2018 NRC exam for ILO-16-01.
SAR 11/2017.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

A small break LOCA has occurred. EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION has been entered.

Immediate actions 1 through 3 have been performed.

INITIATING CUES:

The CRS directs you to perform EOP-1.0, beginning at immediate action step 4.

A surrogate operator will acknowledge non-related alarms per your direction.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPS-002F-N18

Pressurizer Pressure Control Malfunction (Alternate Path)
(NRC JPM b)

CANDIDATE: ______________________

EXAMINER: ______________________

TASK: 000-509-05-02 Recover From Reactor Trip per EOP-1.1.

TASK STANDARD:
RCPs are stopped and uncontrolled depressurization of the RCS is terminated.

TERMINATING CUE:
RCS depressurization is halted.

PREFERRED EVALUATION LOCATION SIMULATOR

PREFERRED EVALUATION METHOD PERFORM

REFERENCES:

EOP-1.1; ES-0.1 REACTOR TRIP RESPONSE

INDEX NO	***K/A NO.***		***RO***	***SRO***
000027A101	AA1.01	Actions to be taken if PZR pressure control malfunctions – PZR heaters, sprays and PORVs	4.0	3.9

TOOLS: Rack copy of EOP-1.1; ES-0.1, REACTOR TRIP RECOVERY with steps 1-7 marked as complete.

EVALUATION TIME 10 ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)(6)

TIME START: ______________ TIME FINISH: ______________ PERFORMANCE TIME: ____________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ____

EXAMINER: ______________________________________ ____________________/__________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The reactor was tripped from 100% power. The crew have transitioned to EOP-1.1, ES-0.1 REACTOR TRIP RESPONSE and have completed steps 1 through 7.

INITIATING CUES:

The CRS directs you to perform EOP-1.1 beginning with step 8.

A surrogate operator will acknowledge non-related alarms per your direction.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 8: Verify all Control Rods are fully inserted.

STEP STANDARD:

Candidate locates Control Rod Position Indication and verifies all rod bottom red lights ON and all rods indicate fully inserted, green column LIT for each rod.

CUES:

EVALUATOR CUE: Provide the marked up rack copy of EOP-1.1, ES-0.1, REACTOR TRIP RESPONSE once the candidate has been briefed on the initiating cue.

BOOTH OPERATOR CUE: Place the Simulator in RUN when the Evaluator indicates the JPM may begin.

BOOTH OPERATOR CUE: Once candidate commences EOP-1.1 Step 8 insert **TRIGGER 1** – PZR spray valves 444C and 444D fail partially open.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Step 9: Check DA level control:

Step 9 a: Open LCV-3235, DEAER START UP DRAIN CNTRL, as necessary to maintain DA level LESS THAN 10.5 ft as indicated on LI-3135, DEAER STOR TK WR LVL FEET.

Step 9 b: Locally adjust ITV-3062A(B)(C), BD COOLER A(B)(C) CDSTE OUT TEMP, to 90% (XPN-0029, NUCLEAR BLOWDOWN PROCESSING PANEL, AB-436).

STEP STANDARD:

Candidate locates DA Level Indication LI-3135 and verifies level is less than 10.5 ft.

Candidate contacts AO and directs adjustment of Condensate flow from the Blowdown heat exchangers.

CUES:

BOOTH OPERATOR CUE: When contacted as AO to adjust Blowdown Hx condensate flow acknowledge request.

BOOTH OPERATOR NOTE: the “Set TCV-3062A, B, C to 10% Open” button on the COMMON LOA/RESET PANEL will set the ITV to the requested value.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 10: Check PZR level control:

Step 10a: Verify PZR level is GREATER THAN 17%.

STEP STANDARD:

Candidate locates PZR LEVEL % LI-459A, 460 and 461 and verifies level is greater than 17%.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 10b: Verify Charging and Letdown are in service.

STEP STANDARD:

Candidate locates Charging controls and notes Charging and Letdown are in service.

CUES:

EVALUATOR NOTE: Charging and letdown indications may be found on panel XCP-6107. Indications include:

- "A" Charging pump breaker red light ON, green light OFF and normal running amps on the ammeter.
- CHG FLOW GPM, FI-122A.
- LO PRESS LTDN FLOW GPM, FI-150.
- PVT-459 and 460, LTDN LINE ISOL indicate red lights ON, green lights OFF.
- PVT-8149A and 8149B, LTDN ORIFCE A (B) ISOL indicate red lights ON, green lights OFF.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 5

Step 10c: Verify PZR level is trending to 25%.

Step 10c: Alternative action: Control Charging and Letdown to maintain PZR level at 25%.

STEP STANDARD:

Candidate locates indications and verifies level is trending to 25%.

Candidate locates controls and adjusts as necessary to maintain PZR level.

CUES:

EVALUATOR NOTE: Control of PZR level may be accomplihed by placing FCV-122 in manual. Candidate may chose to allow auto control to restore level to 25%. Pressurizer level indication may be found on panel XCP-6109L on meters LEVEL % LI-459A, 460 and 461.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 6

Step 11: Check PZR pressure control:

Step 11a: Verify PZR pressure is GREATER THAN 1850 psig.

STEP STANDARD:

Candidate locates indications and verifies pressure is greater than 1850 psig.

CUES:

EVALUATOR NOTE: Pressurizer pressure may be found on panel XCP-6109L on meters PRESS PSIG PI-455, 456, 457 and 444.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 7

Step 11b: Verify PZR pressure is stable at OR trending to 2230 psig (2220 psig to 2250 psig). (NO)

STEP STANDARD:

Candidate locates indications and verifies pressure is less than 2230 psig and trending downward.

CUES:

EVALUATOR NOTE: Candidate may refer to IPSC trend and or the WR pressure recorder (PR-402) in order to validate trend.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 8

Step 11b Alternative Action: IF PZR pressure is LESS THAN 2230 psig AND decreasing, THEN:

1) Ensure the PZR PORVs are closed. IF any PORV fails to close, THEN close its Block Valve.

STEP STANDARD:

Candidate locates PZR controls and notes; PCV-445A, 445B and 444B, PWR RELIEF indicate red lights OFF, green lights ON.

CUES:

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 9

Step 11b Alternative Action: IF PZR pressure is LESS THAN 2230 psig AND decreasing, THEN:

2) Ensure PZR Spray Valves are closed.

IF any valve fails to close, THEN perform the following:

a) Stop RCP A.

STEP STANDARD:

Candidate locates PZR controls and notes;

PCV-444C, PZR SPR CNTRL FR LOOP C indicates red light ON, green light ON and attempts to close PCV-444C are unsuccessful.

PCV-444D, PZR SPR CNTRL FR LOOP A indicates red light ON, green light ON and attempts to close PCV-444D are unsuccessful.

Candidate locates RCP controls and places switch for RCP “A” in STOP, observes red light off, green light on and zero amps on ammeter.

CUES:

EVALUATOR NOTE: The critical step is to stop the “A” RCP to assure spray flow is eliminated and RCS pressure is stabilized.

EVALUATOR NOTE: This is the point where the JPM becomes alternate path.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 10

Step 11b Alternative Action: IF PZR pressure is LESS THAN 2230 psig AND decreasing, THEN:

b) IF PZR pressure continues to decrease, THEN perform the following:

- IF PCV-444C, PZR SPR CNTRL FR LOOP C, will NOT close, THEN stop RCP C.
- IF PCV-444D, PZR SPR CNTRL FR LOOP A, will NOT close, THEN stop either RCP B or RCP C.

STEP STANDARD:

Candidate locates indications and verifies pressure is trending downward.

Candidate locates RCP controls and places switch for RCP “C” in STOP, observes red light off, green light on and zero amps on ammeter.

CUES:

EVALUATOR NOTE: The critical step is to stop the “C” RCP to assure spray flow is eliminated and RCS pressure is stabilized.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 11

Step 11b Alternative Action: IF PZR pressure is LESS THAN 2230 psig AND decreasing, THEN:

3) Ensure PZR Heaters are on.

STEP STANDARD:

Candidate locates PZR controls and notes;

BU GRP 1 breaker control indicates red light ON, green light OFF and BU GRP1 AMPS ammeter indicates amperage.

CNTRL GRP breaker comntrol indicates red light ON, green light OFF and CNTRL GRP AMPS ammeter indicates amperage.

BU GRP 2 breaker control indicates red light ON, green light OFF and BU GRP 2 AMPS ammeter indicates amperage.

CUES:

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPS-002F-N18, Pressurizer Pressure Control Malfunction (Alternate Path) (NRC JPM b)

IC SET: 292

INSTRUCTIONS:

If IC-292 is designated for this JPM then reset to IC-292 leaving the simulator in FREEZE.

1. Mark up copy of EOP-1.0 steps 1-5 and EOP-1.1 steps 1-7.

2. When Candidate is ready (on Evaluator cue) go to RUN.

If IC-292 is **not** designated for this JPM then initial conditions may be established by resetting to IC-10 and following the below directions:

1. With the simulator reset to IC-10 and in FREEZE, insert the following:

- **TRIGGER 1**
  - MAL-PRS003A
    PRESSURIZER SPRAY VALVE 444C FAILURE
    Ramp: 60 sec
    Final Value: 55%
  - MAL-PRS003B
    PRESSURIZER SPRAY VALVE 444D FAILURE
    Ramp: 60 sec
    Final Value: 55%

2. Place the Simulator in RUN.

3. Insert a manual Reactor Trip.

4. Perform Actions from EOP-1.0 and EOP-1.1 through step 7.

4. FREEZE and SAVE IC.

5. Mark up copy of EOP-1.0 steps 1-5 and EOP-1.1 steps 1-7.

6. When Candidate is ready (on Evaluator cue) go to RUN.

COMMENTS:

Provide spare operator to silence alarms.

Mark strip chart recorders with date and time at the completion of each performance of this JPM.

Roll strip chart recorders to show no traces from the just completed performance after marking them as noted above.

CRITICAL TASK METHODOLOGY:

Steps 9 and 10 are critical because spray valves PVT-444C and PVT-444D are failed in a partially open position and Pressurizer pressure will continue to degrade to an eventual SI actuation unless the Candidate takes action to stop the RCPs.

REVISION HISTORY:

This JPM is a modification of JPSF-011A, PRESSURIZER PRESSURE CONTROL MALFUNCTION
SAR 10/2017.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The reactor was tripped from 100% power. The crew have transitioned to EOP-1.1, ES-0.1 REACTOR TRIP RESPONSE and have completed steps 1 through 7.

INITIATING CUES:

The CRS directs you to perform EOP-1.1 beginning with step 8.

A surrogate operator will acknowledge non-related alarms per your direction.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPS-003F-N18

Respond to Inadequate Core Cooling (Attempt to Start RCPs and Depressurize Primary) (Alternate Path).

(NRC JPM c)

CANDIDATE: ______________________________

EXAMINER: ______________________________

TASK: 000-088-05-01 Response to Inadequate Core Cooling per SOP-122/EOP-12.0/EOP-2.0/EOP-14.0.

TASK STANDARD:

Completes the following until they are all completed:

1. Attempts start of "A" RCP. 2. Does not start "B" RCP. 3. Does not start "C" RCP. 4. Opens all pressurizer PORVs and Reactor Head Vent Valves.

TERMINATING CUE: Opens all pressurizer PORV's and Reactor Head Vent Valves to inject accumulators.

PREFERRED EVALUATION LOCATION SIMULATOR

PREFERRED EVALUATION METHOD PERFORM

REFERENCES:

EOP-2.0, LOSS OF REACTOR OR SECONDARY COOLANT

EOP-1.0, E-0, REACTOR TRIP OR SAFETY INJECTION

EOP-14.0, RESPONSE TO INADEQUATE CORE COOLING

INDEX NO	K/A NO.		RO	SRO
000074105	1.05	Ability to operate and monitor the following as they apply to Inadequate Core Cooling: PORV	3.9	4.1

TOOLS: Rack copy of EOP-14.0, FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, marked up through step 21.

Rack copy of EOP-2.0, LOSS OF REACTOR OR SECONDARY COOLANT, marked up through step 15.

EVALUATION TIME 10 **TIME CRITICAL** NO **10CFR55:** 45(a)6

TIME START: ____________ TIME FINISH: ____________ PERFORMANCE TIME: ____________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ______________________________

EXAMINER: ______________________________ ____________/__________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The plant has tripped with Safety Injection initiated due to a LOCA.

"C" Charging pump is tagged out for maintenance.

"A" charging pump failed a few minutes after the RCP's were secured.

"B" charging pump failed to start.

Reactor Building Spray actuated.

EOP-2.0 Loss of Reactor or Secondary Coolant has been completed through step 15.

A red path for Inadequate Core Cooling has been implemented in accordance with EOP- 14.0.

The crew has just attempted to depressurize SGs but were not able to.

INITIATING CUES:

CRS directs you as the NROATC to perform Step 22 of EOP-14.0, FR-C.1, RESPONSE TO INADEQUATE CORE COOLING.

A surrogate operator will acknowledge non-related alarms per your direction.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 22: Check if RCPs should be started:

a. Check if core exit TC temperatures are GREATER THAN 1200°F.

STEP STANDARD:

Candidate locates indication and notes; CETC temperatures are >1200°F and rising.

CUES:

BOOTH OPERATOR CUE: Place the Simulator in RUN when the Evaluator indicates the JPM may begin.

EVALUATOR CUE: Provide marked up copies of EOP-2.0 and EOP-14.0 once the candidate has been briefed on the initiating cue.

EVALUATOR NOTE: CETCs indicate on various SIPCS displays.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Procedure Notes:

- Normal RCP starting criteria are desired but are NOT required for starting an RCP.
- Preferred RCP starting sequence under Inadequate Core Cooling conditions is B,C,A, to preserve PZR spray capability during the recovery.

Step 22 b: Check if an idle RCS cooling loop is available:

- Verify SG Narrow Range level is GREATER THAN 26% [41%].
- Check if the RCP in the associated loop is available and NOT operating.

STEP STANDARD:

Candidate locates:

Level indication and determines “A” SG level is **greater** than 41%, “B” SG level is **less** than 41% and “C” SG level is **less** than 41%.

RCP controls and notes; RCP “A” breaker control indicates red light off, green light on and zero amps on ammeter with no abnormal annunciators standing on panel XCP-617.

CUES:

EVALUATOR NOTE: SG level indications may be found on LI-474, 475, 476, NR LEVEL % for SG “A”, LI-484, 485, 486, NR LEVEL % for SG “B”, and LI-494, 495, 496, NR LEVEL % for SG “C”.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 22c: Start XPP-87A(B)(C), A(B)(C) OIL LIFT PP.
Step 22d: Start an RCP in one idle RCS cooling loop.

STEP STANDARD:

Candidate attempts start of "A" RCP;

Places "A" oil lift pump to start, red light ON, green light OFF.

Places "A" RCP to start, red light OFF, green light ON, pump AMPs do NOT rise.

CUES:

EVALUATOR NOTE: JPM becomes alternate path at this point as no RCS loops are available any longer and Candidate must employ alternative action to achieve some means of core cooling.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 22a: Check if core exit TC temperatures are GREATER THAN 1200°F.

STEP STANDARD:

Candidate locates CETCs indication and notes; CETC temperatures are >1200°F and rising.

CUES:

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 5

Procedure Notes:

- Normal RCP starting criteria are desired but are NOT required for starting an RCP.
- Preferred RCP starting sequence under Inadequate Core Cooling conditions is B,C,A, to preserve PZR spray capability during the recovery.

Step 22 b: Check if an idle RCS cooling loop is available:

- Verify SG Narrow Range level is GREATER THAN 26% [41%].
- Check if the RCP in the associated loop is available and NOT operating.

STEP STANDARD:

Candidate locates:

Level indication and determines "B" SG level is **less** than 41%.

Level indication and determines "C" SG level is **less** than 41%.

Candidate determines that niether "B" or "C" RCP should be started due to level less than 41%.

CUES:

EVALUATOR NOTE: It is critical that the Candidate not start the "B" RCP or "C" RCP.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 6

Alternative Action 22 b; Perform the following:

1) Open all PZR PORV Block Valves.
2) Open all PZR PORVs.
3) If core exit TC temperature remains GREATER THAN 1200°F, THEN open all Reactor Vessel Head Vent Valves.

STEP STANDARD:

Candidate locates:

PZR PORV Block valve controls, MVG-8000A, 8000B and 8000C and notes; red lights ON, green lights OFF, all valves OPEN.

PZR PORVs PCV-445A, 445B and 444B and places control switches to OPEN, notes red lights ON, green lights OFF for all three valves.

CETCs indication and notes; CETC temperatures are >1200°F and rising, locates RX HEAD VENT VLV controls MVG-8095A, 8095B, 8096A and 8096B and places control switches to OPEN, notes red lights ON, green lights OFF for all four valves.

CUES:

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPS-003F-N18 Respond to Inadequate Core Cooling (Attempt to Start RCPs and depressurize primary) (Alternate Path) (NRC JPM c)

IC SET: 293

INSTRUCTIONS:

If IC-293 is designated for this JPM then reset to IC-293 leaving the simulator in FREEZE.

1. Place Danger Tag on 'C' Charging pump for Maintenance.
2. Mark up rack copies of EOP-2.0 steps 1-15 as complete and EOP-14.0 steps 1-21 as complete.
3. When Candidate is ready (on Evaluator cue) go to RUN.

If IC-293 is not designated for this JPM then initial conditions may be established by resetting to IC-10 and following the below directions:

1. With the simulator reset to IC-10 and in FREEZE, insert the following:

- **PRE-LOAD**
  - MAL-MSS007A
S/G A POWER OPERATED RELIEF VALVE FAILURE
Final Value = 0
  - MAL-MSS007B
S/G B POWER OPERATED RELIEF VALVE FAILURE
Final Value = 0
  - MAL-MSS007C
S/G C POWER OPERATED RELIEF VALVE FAILURE
Final Value = 0
  - MAL-MSS005
STEAM DUMP CONTROL FAILURE
Final Value = 0
  - LOA-RHR006
ACCUM A ISO VLV 8808A BKR
Position To = CLOSE
  - LOA-RHR007
ACCUM B ISO VLV 8808B BKR
Position To = CLOSE
  - LOA-RHR008
ACCUM C ISO VLV 8808C BKR
Position To = CLOSE
  - LOA-RCS009
RX HEAD VENT VLV 8095A BKR
Position To = CLOSE
  - LOA-RCS010
RX HEAD VENT VLV 8095B BKR
Position To = CLOSE
  - LOA-RCS011
RX HEAD VENT VLV 8096A BKR
Position To = CLOSE
  - LOA-RCS012
RX HEAD VENT VLV 8096B BKR
Position To = CLOSE
- **TRIGGER 1**
  - MAL-RCS006A
REACTOR COOLANT SYSTEM LEAK COLD LEG (LOOP 1)
Final Value = 10000
Delay =10 sec

- o MAL-CVC017A
  CHARGING PUMP A TRIP
  Delay = 120 sec
- o MAL-CVC017B
  CHARGING PUMP B TRIP
  Delay = 20 sec

2. Place the simulator in RUN then insert **TRIGGER 1**.
3. Manually trip RCPs when RCS pressure <1400 psig.
4. Perform the following actions >1 minute after SI is initiated:

   Trip the exciter field breaker
   Reset SI
   Reset Phase A
   Reset Phase B
   Reset the ESFLS
   Establish IA to the RB
   Secure RHR pumps
5. When RVLIS NR Level is <40%, reduce RCS leak to 500 GPM by modifying MAL-RCS006A, REACTOR COOLANT SYSTEM LEAK COLD LEG (LOOP 1).
6. Align EFW for normal operation and throttle to approximately 200 gpm per Steam Generator.
7. Set all Steam Dump Power Relief controls to PWR REL.
8. When Core Exit Thermocouples >1200°F: place the simulator in FREEZE
9. Insert:
   - o MAL-RCS003A
     REACTOR COOLANT PUMP 1 TRIP
     Fail To: NO RSTART
10. Ensure that "B" and "C" Steam Generator Narrow Range Levels are less than 41% and that "A" SG NR level is greater than 41%.
11. Ensure steps of EOP-1.0 and in particular EOP-1.0 attachment 3 have been fully and correctly implemented prior to saving setup for JPM.
12. Save IC.
13. Place Danger Tag on 'C' Charging pump for Maintenance.
14. Mark up rack copies of EOP-2.0 steps 1-15 as complete and EOP-14.0 steps 1-21 as complete.
16. When Candidate is ready (on Evaluator cue) go to RUN.

COMMENTS:

Provide spare operator to silence alarms.

BOOTH OPERATOR: Use LOA resets page to silence HVAC alarms when they come in.

Mark strip chart recorders with date and time at the completion of each performance of this JPM.

Roll strip chart recorders to show no traces from the just completed performance after marking them as noted above.

CRITICAL TASK METHODOLOGY:

Step 3 is critical because Loop “A” SG level is adequate to provide cooling and an attempt to start “A” RCP is required

Step 5 is critical because the “B” and “C” SGs do not contain adequate inventory and the associated RCPs must **not** be started.

Step 6 is critical as this is the lone remaining option to induce core cooling.

REVISION HISTORY:

This JPM is a minor revision of JPSF-044C, Respond to Inadequate Core Cooling (Attempt to Start RCPs and depressurize primary) which was a modification of JPSF-044B.
SAR 10/2017.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The plant has tripped with Safety Injection initiated due to a LOCA.

"C" Charging pump is tagged out for maintenance.

"A" charging pump failed a few minutes after the RCP's were secured.

"B" charging pump failed to start.

Reactor Building Spray actuated.

EOP-2.0 Loss of Reactor or Secondary Coolant has been completed through step 15.

A red path for Inadequate Core Cooling has been implemented in accordance with EOP- 14.0.

The crew has just attempted to depressurize SGs but were not able to.

INITIATING CUES:

CRS directs you as the NROATC to perform Step 22 of EOP-14.0, FR-C.1, RESPONSE TO INADEQUATE CORE COOLING.

A surrogate operator will acknowledge non-related alarms per your direction.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPS-004-N18

Respond To Steam Generator Overpressure

(NRC JPM d)

CANDIDATE: ______________________________

EXAMINER: ______________________________

TASK: 000-092-05-01 Respond To Steam Generator Overpressure per EOP-15.1/EOP-12.0

TASK STANDARD:
All feedwater flow to the "B" SG has been terminated and the "B" SG pressure is below 1230 psig.

TERMINATING CUE:
The "B" SG pressure has been reduced below 1230 psig in accordance with EOP-15.1 and RCS Tavg is stable at or trending to 557°F.

PREFERRED EVALUATION LOCATION
SIMULATOR

PREFERRED EVALUATION METHOD
PERFORM

REFERENCES:

EOP-15.1, FR-H.2 RESPONSE TO STEAM GENERATOR OVERPRESSURE

INDEX NO	K/A NO.		RO	SRO
WE013EA1.1	EA1.1	Ability to operate and/or monitor the following as they apply to the (Steam Generator Overpressure): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.1	3.3
WE013EA2.1	EA2.1	Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	2.9	3.4

TOOLS: Rack copy of EOP-15.1, FR-H.2, RESPONSE TO STEAM GENERATOR OVERPRESSURE
Rack copy of EOP-1.1 with steps 1-5 marked as completed and step 6 marked as Alternative Action for RCS temperature greater than 557°F through step c) 2).

EVALUATION TIME 10 **TIME CRITICAL** NO **10CFR55:** 45(a)13

TIME START: ______________ TIME FINISH: ______________ PERFORMANCE TIME: ______________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ________________________________

EXAMINER: ________________________________ ____________________/__________
SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The plant has tripped due to a turbine trip from 100% power.

The CRS is implementing EOP-1.1, ES-0.1 REACTOR TRIP RESPONSE and is currently directing actions from step 6.

The Shift Engineer has identified a Yellow Path on Heat Sink due to Steam Generator over pressurization on the "B" SG.

INITIATING CUES:

The CRS directs you to implement EOP-15.1, FR-H.2, RESPONSE TO STEAM GENERATOR OVERPRESSURE in response to the Yellow Path and then complete EOP-1.1 step 6.

A surrogate operator will acknowledge non-related alarms per your direction.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 1: Identify any SG with pressure GREATER THAN 1230 psig.

STEP STANDARD:

Candidate locates SG indications and identifies that the "B" SG pressure is greater than 1230 psig.

CUES:

BOOTH OPERATOR CUE: Place the Simulator in RUN when the Evaluator indicates the JPM may begin.

EVALUATOR CUE: Provide rack copy of EOP-15.1 and EOP-1.1 marked up to step 6 once the candidate has been briefed on the initiating cue.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Step 2: Ensure the following valves are closed to the AFFECTED SG(s):

- FW Flow Control, FCV-488
- FW Isolation, PVG-1611B
- FW Flow Control Bypass, FCV-3331

STEP STANDARD:

Candidate locates:

FW Flow Control Valve indications; notes red light ON, green light OFF for FCV-488, places control switch in CLOSE, verifies red light OFF, green light ON.

FW Isolation Valve indications; notes red light ON, green light OFF for PVG-1611B, places control switch in CLOSE, verifies red light OFF, green light ON.

FW Flow Control Bypass Valve indication; verifies red light OFF, green light ON for FCV-3331.

CUES:

EVALUATOR NOTE: FW Flow Control FCV-488, and FW Isolation PVG-1611B, are open due to a failure of the FW isolation to initiate. Closing either FCV-488 or PVG-1611B will accomplish the isolation of the Main Feedwater flowpath to prevent potential SG overfill.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 3: Check Narrow Range level in AFFECTED SG(s) is LESS THAN 90% [83%].

STEP STANDARD:

Candidate locates indications and determines “B” SG level is less than 90%.

CUES:

EVALUATOR NOTE: The “B” SG level indication may be found on LI-484, 485, 486, NR LEVEL %.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 4: Dump steam from each AFFECTED SG to the Condenser:

a. Verify PERMISV C-9 status light is bright on XCP-6114 1-3.

STEP STANDARD:

Candidate locates panel XCP-6114 and verifies that PERMISV C-9 is dim.

CUES:

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 5

Step 4: ALTERNATIVE ACTION:

Dump steam from each AFFECTED SG using the Steamline PORV:

a) Place the PWR RELIEF A(B)(C) SETPT Controller in MAN and closed.

b) Place the Steamline Power Relief A(B)(C) Mode Switch in PWR RLF.

c) Adjust the PWR RELIEF A(B)(C) SETPT Controller to reduce AFFECTED SG(s) pressure.

STEP STANDARD:

Candidate locates SG Power Relief controls;

Places PWR RELIEF B SETPT Controller in MAN and closed.

Verifies B SDPWR RELIEF in PWR RLF.

Adjusts PWR RELIEF B SETPT controller to reduce B SG pressure by pressing the OUTPUT button next to the “up arrow” thus raising controller output and opening “B” S/G PORV.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 6

Step 5: Check AFFECTED SG(s) pressures:

a. Verify each AFFECTED SG(s) pressure is decreasing.

b. Verify each AFFECTED SG(s) pressure is LESS THAN 1230 psig.

STEP STANDARD:

Candidate locates indication and notes “B” SG pressure is lowering to less than 1230 psig.

CUES:

EVALUATOR NOTE: The “B” SG pressure indication may be found on MS LINE PRESS, LINE B PSIG, PI-486 and PI-2010

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 7

Step 5: Check AFFECTED SG(s) pressures:

c. Control steam release to maintain SG pressures LESS THAN 1230 psig.

d. RETURN TO the Procedure and Step in effect.

STEP STANDARD:

Candidate adjusts PWR RELIEF B SETPT controller output to maintain “B” SG pressure less than 1230 psig and returns to EOP-1.1 step 6.

CUES:

EVALUATOR NOTE: The Initial Conditions indicated that the procedure and step in effect were EOP-1.1 step 6.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 8

Step 6: Check RCS temperature:

- With any RCP running, RCS Tavg is stable at OR trending to 557°F. **(NO)**

Step 6: Alternative Action: IF RCS temperature is GREATER THAN 557°F AND increasing, THEN:

a) Verify PERMISV C-9 status light is bright on XCP-6114 1-3.

b) IF the Condenser is available, THEN ensure Condenser Steam Dump Valves are open.

c) IF the Condenser is NOT available, THEN open the Steamline PORVs, PCV-2000(2010)(2020):

1) Place the Steamline Power Relief A(B)(C) Mode Switches in PWR RLF.

2) Adjust the PWR RELIEF A(B)(C) SETPT Controllers as necessary to reduce RCS temperature.

STEP STANDARD:

Candidate opens SG PORVs to lower SG pressure and RCS Tavg.

Candidate throttles SG PORVs to stabilize RCS Tavg at approximately 557°F

CUES:

EVALUATOR NOTE: The “A” and “C” SG PORVs may be controlled in auto. These PORVs are opened by lowering the setting on the SET PT potentiometer dial and these PORVs are closed by raising the setting on the SET PT potentiometer dial. The “B” SG PORV must NOT be placed in AUTO as its reference pressure input (MS LINE PRESS LINE B PSIG PI-2010) has failed low and the PORV will drive closed. The “B” SG PORV may only be adjusted using the OUTPUT up and down buttons.

EVALUATOR NOTE: The critical step is to **not** place “B” SG PORV in automatic since “B” SG PORV would close and SG pressure would rise to greater than 1230 psig.

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPS-004F-N18, Respond to Steam Generator Overpressure (NRC JPM d)

IC SET: 294

INSTRUCTIONS:

If IC-294 is designated for this JPM then reset to IC-294 leaving the simulator in FREEZE.

1. Mark rack copy of EOP-1.1 with steps 1-5 marked as complete and step 6 marked as having started the high level step only.
2. When Candidate is ready (on Evaluator cue) go to RUN.

If IC-294 is not designated for this JPM then initial conditions may be established by resetting to IC-10 and following the below directions:

1. With the simulator reset to IC-10 and in FREEZE, insert the following:

- **PRE-LOAD**
    - LOA-CND037
      CONDENSER AIR INLEAKAGE RATE (SCFM)
      Ramp: 60 sec
      Final Value: 1000 SCFM
    - MAL-MSS010B
      S/G B SAFETY VALVE FAILS
      Final Value: 0%
    - MAL-MSS007B
      S/G B POWER OPERATED RELIEF VALVE FAILURE
      Final Value: 0%
    - MAL- FWM015B
      FW CONTROL VALVE LV-488 POSITION FAILURE (SG B)
      Final Value: 76.3%
    - VLV- FW026P
      XVG01611B-FW FEEDWTR ISO VLV B FAIL POSITION
      Final Value: 100%
    - BST-MS054
      ILS02806A REL VLV MS
      Fail To: INHIBITED
    - BST-MS055
      ILS02806B REL VLV MS
      Fail To: INHIBITED
    - BST-MS056
      ILS02806C REL VLV MS
      Fail To: INHIBITED
    - BST-MS057
      ILS02806D REL VLV MS
      Fail To: INHIBITED
    - BST-MS058
      ILS02806E REL VLV MS
      Fail To: INHIBITED
    - BST-MS059
      ILS02806F REL VLV MS
      Fail To: INHIBITED
    - BST-MS060
      ILS02806G REL VLV MS
      Fail To: INHIBITED
    - BST-MS061
      ILS02806H REL VLV MS
      Fail To: INHIBITED

- BST-MS062
  ILS02806I REL VLV MS
  Fail To: INHIBITED
- BST-MS063
  ILS02806J REL VLV MS
  Fail To: INHIBITED
- BST-MS064
  ILS02806K REL VLV MS
  Fail To: INHIBITED
- BST-MS065
  ILS02806L REL VLV MS
  Fail To: INHIBITED
- BST-MS066
  ILS02806M REL VLV MS
  Fail To: INHIBITED
- BST-MS067
  ILS02806N REL VLV MS
  Fail To: INHIBITED
- BST-MS068
  ILS02806P REL VLV MS
  Fail To: INHIBITED

- **TRIGGER 1**
  - MAL-PCS014
    INADVERTENT MS ISOLATION
    Delay: 15 seconds
  - MAL-TUR001
    INADVERTENT TURBINE TRIP
- **AUTO TRIGGER 2** (x07i391c==1) FRV-488 placed in close.
  - MAL- FWM015B (NEW)
    FW CONTROL VALVE LV-488 POSITION FAILURE (SG B)
    Final Value: 76.3%
    Delete in: 1 sec
- **AUTO TRIGGER 3** (x07i091c==1) XVG-1611B placed in close.
  - VLV- FW026P
    XVG01611B-FW FEEDWTR ISO VLV B FAIL POSITION
    Final Value: 100%
    Delete in: 1 sec

2. Place the simulator in RUN then insert **TRIGGER 1**.
3. Perform all applicable actions from EOP-1.0 and EOP-1.1. Make certain to trip MFPs and TDEFP.
4. Control “B” SG level to maintain less than 90% NR level and adjust EFW and steam flow on “A” and “C” SG to achieve desired initial condition.
5. When “B” SG pressure is greater than 1230 psig stabilize “A” and “C” SG levels and pressure and place the simulator in FREEZE.
6. Save IC.
7. Mark rack copy of EOP-1.1 with steps 1-5 marked as complete and step 6 marked as Alternative Action for RCS temperature greater than 557°F complete through step c) 2).
8. When Candidate is ready (on Evaluator cue) go to RUN.

COMMENTS:

Provide spare operator to silence alarms.

Mark strip chart recorders with date and time at the completion of each performance of this JPM.

Roll strip chart recorders to show no traces from the just completed performance after marking them as noted above.

CRITICAL TASK METHODOLOGY:

Step 5 is critical because the Candidate must take action to remove energy from the affected SG to preclude excessive pressurization.

Step 8 is critical because the Candidate must control “B” SG pressure to preclude a return to greater than 1230 psig.

REVISION HISTORY:

This JPM is a minor revision of JPS-149, Respond to Steam Generator Overpressure.
SAR 12/2017.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The plant has tripped due to a turbine trip from 100% power.

The CRS is implementing EOP-1.1, ES-0.1 REACTOR TRIP RESPONSE and is currently directing actions from step 6.

The Shift Engineer has identified a Yellow Path on Heat Sink due to Steam Generator over pressurization on the "B" SG.

INITIATING CUES:

The CRS directs you to implement EOP-15.1, FR-H.2, RESPONSE TO STEAM GENERATOR OVERPRESSURE in response to the Yellow Path and then complete EOP-1.1 step 6.

A surrogate operator will acknowledge non-related alarms per your direction.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPS-005-N18

Loss of All ESF AC with Restoration via XTF-5052

(NRC JPM e)

CANDIDATE: ______________________________

EXAMINER: ______________________________

TASK: O-000-055-05-01 Respond To Loss of Off Site and On Site Power

TASK STANDARD:

ESF bus 1DA has been energized from the Alternate AC Source via XTF-5052 fed from the grid.

TERMINATING CUE: ESF bus 1DA is energized from the 115 KV line via XTF-5052.

PREFERRED EVALUATION LOCATION SIMULATOR

PREFERRED EVALUATION METHOD PERFORM

REFERENCES:

EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER

SOP-304, 115KV/7.2KV OPERATIONS

INDEX NO	*K/A NO.*		*RO*	*SRO*
00062A205	A2.05	Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Methods for energizing a dead bus.	2.9	3.3

TOOLS: Marked up copy of SOP-304, 115KV/7.2KV OPERATIONS, mark entry conditions complete for section V.A.

Marked up copy of EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION, mark steps 1-5 complete and step 6 in progress.

Marked up copy of EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER

EVALUATION TIME 20 ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)13

TIME START: ____________ TIME FINISH: ____________ PERFORMANCE TIME: ____________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ________________________________

EXAMINER: ________________________________ ________________/__________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The Unit Auxiliary Transformer (XTF-2) experienced a fault and caused a Turbine and Reactor Trip.

The Emergency Auxiliary transformer (XTF-32) is faulted.

ESF Transformer XTF-4 is faulted.

ESF Transformer XTF-5 was unavailable prior to the trip due to scheduled maintenance.

The 51BX-1DB relay is actuated.

The “A” DG failed to start automatically and will not start in manual.

The 115 KV Parr bus 2 and XTF-5052 have been determined to be available.

The crew is implementing EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER.

INITIATING CUES:

The CRS has directed you to restore offsite power to the 1DA ESF bus in accordance with EOP-6.0 step 6 a. Alternative Action. You are to refer to SOP-304, 115KV/7.2KV OPERATIONS, Section V.A.

A surrogate operator will acknowledge non-related alarms per your direction.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 2.1: Notify the System Controller of the situation.

STEP STANDARD:

Candidate uses system controller direct line and provides an update concerning VC Summer electrical plant status.

CUES:

BOOTH OPERATOR CUE: Place the Simulator in RUN when the Evaluator indicates the JPM may begin.

EVALUATOR CUE: Provide the marked up rack copy of SOP-304 and marked up copy of EOP-6.0 once the candidate has been briefed on the initiating cue.

BOOTH OPERATOR CUE: Acknowledge communications from candidate using the System Controller Direct line.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Step 2.2: At XPN6020, ESF LOADING SEQUENCE CONTROL PANEL - UNIT 1, deenergize the ESFLS by opening the switch labeled CIRCUIT BREAKER (CB-436).

STEP STANDARD:

Candidate contacts an AO to open the ESFLS breaker.

CUES:

BOOTH OPERATOR CUE: When contacted as AO to open the ESFLS breaker acknowledge the communication, wait briefly and then activate **TRIGGER 2.** Once the breaker LOA has inserted (15 seconds) contact the Candidate and report “ESFLS circuit breaker at XPN-6020 is open”.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 2.3: Unload and de-energize Bus 1DA as follows:
Step 2.3 a: Place SW PUMP A in PULL TO LOCK NON-A.

STEP STANDARD:

Candidate locates SW pump controls and places "A" Service Water pump switch in pull to lock.

CUES:

EVALUATOR CUE: If candidate requests a peer check, acknowledge the request by saying "Understand you request a peer check" This response should be provided as often as a peer check is requested.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 1): Component Cooling Water Pump A(C).

STEP STANDARD:

Candidate locates Component Cooling Water pump controls and places "A" CCW pump switch in pull to lock and then places "C" CCW pump Train "A" switch in pull to lock.

CUES:

EVALUATOR NOTE: If the candidate fails to place the "C" CCW pump switch in pull to lock, the pump will auto start upon power restoration to the 1DA bus due to a low header pressure auto start signal.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 5

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 2): Service Water Pump A(C).

STEP STANDARD:

Candidate locates SW pump controls and verifies “A” and “C” Service Water pump switches in pull to lock.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 6

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 3): Service Water Booster Pump A.

STEP STANDARD:

Candidate locates Service Water pump controls and places “A” Service Water Booster pump switch in pull to lock.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 7

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 4): RBCUs (Slow)

STEP STANDARD:

Candidate locates RBCU controls and places XFN-64A, 1A SLOW and XFN-65A, 2A SLOW switches in pull to lock.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 8

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 5): RB Spray Pump A

STEP STANDARD:

Candidate locates RB Spray pump controls and places “A” RB Spray pump switch in pull to lock.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 9

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 6): RHP Pump A.

STEP STANDARD:

Candidate locates RHR pump controls and places “A” RHR pump switch in pull to lock.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 10

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 7): Charging Pump A.

STEP STANDARD:

Candidate locates Charging pump controls and places “A” Charging pump switch in pull to lock.

CUES:

EVALUATOR NOTE: The “A” Charging pump will restart on re-energizing the bus if NOT placed in pull to lock.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 11

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 8): PZR Backup Heaters (Group 1)

STEP STANDARD:

Candidate locates PZR Heater controls and places Backup Group 1 switch in pull to lock.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 12

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 9): EFW Pump A.

STEP STANDARD:

Candidate locates EFW pump controls and places “A” EFW pump switch in pull to lock.

CUES:

EVALUATOR NOTE: The “A” EFW pump will restart on re-energizing the bus if NOT placed in pull to lock.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 13

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 10): HVAC Chiller Pump A(C).

STEP STANDARD:

Candidate locates HVAC controls and places “A” HVAC Chiller switch in pull to lock and verifies “C” HVAC Chiller switch in pull to lock.

CUES:

COMMENTS:

CRITICAL: No **SEQUENCED:** Yes **SAT** ☐ **UNSAT** ☐

JPM STEP: 14

Step 2.3 b: Ensure the following loads are in PULL TO LOCK:
Step 2.3 b 11): HVAC Chiller A(C).

STEP STANDARD:

Candidate locates HVAC controls and places "A" HVAC Chiller Pump switch in pull to lock and verifies "C" HVAC Chiller Pump switch in pull to lock.

CUES:

COMMENTS:

CRITICAL: No **SEQUENCED:** Yes **SAT** ☐ **UNSAT** ☐

JPM STEP: 15

Step 2.3 c: Open BUS 1DA2 FEED.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and places BUS 1DA2 FEED switch in TRIP and observes red light OFF, green light ON.

CUES:

COMMENTS:

CRITICAL: No **SEQUENCED:** Yes **SAT** ☐ **UNSAT** ☐

JPM STEP: 16

Step 2.3 d: Open BUS 1DA1 FEED.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and places BUS 1DA2 FEED switch in TRIP and observes red light OFF, green light ON.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 17

Step 2.3 e: Open XFMR 1DA1 & 1DA2 FEED.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and places XFMR 1DA1&1DA2 FEED switch in TRIP and observes red light OFF, green light ON.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 18

Step 2.3 f: Open BUS 1EA1 FEED.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and places BUS 1EA1 FEED switch in TRIP and observes red light OFF, green light ON.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 19

Step 2.3 g: Open XFMR 1EA1 FEED.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and places XFMR 1EA1 FEED switch in TRIP and observes red light OFF, green light ON.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 20

Step 2.3 h: Open BUS 1EA FEED.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and places BUS 1EA FEED switch in TRIP and observes red light OFF, green light ON.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 21

Step 2.3 i: Open BUS 1DA NORMAL FEED.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and places BUS 1DA NORMAL FEED switch in TRIP and observes red light OFF, amber light OFF, green light ON.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 22

Step 2.3 j: Open BUS 1DA ALT FEED.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and verifies BUS 1DA ALT FEED switch in indicates after TRIP (green flag) and observes red light OFF, green light ON.

CUES:

EVALUATOR NOTE: The Alternate feed breaker was never closed and was never given a close signal.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 23

Step 2.3 k: Verify Bus 1DA potential lights are not lit.

Step 2.3 l: Ensure the following Breakers are open:

1) BUS 1DA NORM FEED Breaker.
2) BUS 1DB ALT FEED Breaker.

STEP STANDARD:

Candidate locates Electrical Switchgear controls and verifies BUS 1DA white potential lights are dim and that BUS 1DA NORM and ALT FEED breakers indicate red lights OFF, green lights ON with green targets on breaker control switches.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 24

Step 2.4: De-energize XSW1DX as follows (TB-463):

a. Open XSW1DX 01, XTF-4 LOW SIDE BREAKER.
b. Open XSW1DX 04, XTF-5 LOW SIDE BREAKER.
c. Ensure the following potential lights indicate de-energized:
   1) BUS 1DA NORM FEED Breaker.
   2) BUS 1DB ALT FEED Breaker.

STEP STANDARD:

Candidate contacts an AO to open XSW1DX 01, XTF-4 LOW SIDE BREAKER and XSW1DX 04, XTF-5 LOW SIDE BREAKER.

Once breakers are reported open Candidate verifies BUS 1DA NORM FEED and BUS 1DA ALT FEED white potential lights are dim.

CUES:

BOOTH OPERATOR CUE: When contacted as AO to open XSW1DX 01, XTF LOW SIDE BREAKER and XSW1DX 04, XTF-5 LOW SIDE BREAKER acknowledge the communication, wait 1 minute and then activate **TRIGGER 3** Once the breaker(s) is(are) opened contact the Candidate and report “XSW1DX 01 (XSW1DX 04) is (are) open”.

EVALUATOR NOTE: The potential lights were dim prior to the breakers being opened due to events that occurred to create the JPM initial conditions.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 25

Step 2.5: Re-energize XSW1DX from XTF5052, ALTERNATE AC SOURCE TRANSFORMER as follows:

Step 2.5 a: Perform one of the following methods to verify XTF-5052 ALT AC PWR VOLTAGE, is ready for load (N/A step not performed).

1) Verify XTF-5052 ALT AC PWR VOLTAGE indicates between 6511 volts and 7920 volts on the MCB.
2) Have electrical maintenance verify XTF-5052 ALT AC PWR VOLTAGE indicates between 116 volts and 124 volts at XSW1DX 03 terminals C3 and C5 on device TX.

STEP STANDARD:

Candidate either locates XTF-5052 ALT AC PWR VOLTAGE meter and verifies 6511 – 7920 volts **or** contacts Electrical Maintenace to verify XTF-5052 ALT AC PWR VOLTAGE indicates 116-124 volts at XSW1DX 03 terminals C3 and C5 on device TX.

CUES:

BOOTH OPERATOR CUE:IF contacted as Electrical Maintenance acknowledge request and wait 1 minute then report "XTF-5052 ALT AC PWR VOLTAGE indicates 116.9 volts at XSW1DX 03 terminals C3 and C5 on device TX."

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 26

Step 2.5 b: At XSW1DX 03, close XSW1DX 03, ALTERNATE AC POWER SUPPLY BREAKER, by depressing the AUX 3 Pushbutton on SEL- 351S, RELAY METER CONTROL FAULT LOCATOR

Step 2.5 c: Ensure BUS 1DA NORM FEED bus potential lights are lit.

STEP STANDARD:

Candidate contacts an AO to close XSW1DX 03, ALTERNATE AC POWER SUPPLY BREAKER.
Once breaker is reported closed Candidate verifies BUS 1DA NORM FEED white potential lights are bright.

CUES:

BOOTH OPERATOR CUE: When contacted as AO to close XSW1DX 03, ALTERNATE AC POWER SUPPLY BREAKER acknowledge the communication, wait 1 minute and then activate **TRIGGER 4.** Once the breaker is closed contact the Candidate and report "XSW1DX 03 ALTERNATE AC POWER SUPPLY BREAKER is closed".

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 27

Step 2.6: To supply XSW1DA from XTF5052, ALTERNATE AC SOURCE TRANSFORMER, perform the following:

Step 2.6 a: Ensure BUS 1DA ALT FEED Breaker is open.

Step 2.6 b: Verify 1DA VOLTS reads zero volts across each phase.

Step 2.6 c: Close BUS 1DA NORM FEED Breaker.

Step 2.6 d: Verify Bus 1DA potential lights are lit.

STEP STANDARD:

Candidate locates:

BUS 1DA ALT FEED breaker switch and verifies it is in after TRIP and observes red light OFF, green light ON.

BUS 1DA VOLTMETER SEL and verifies zero volts across each phase.

BUS 1DA NORM FEED breaker control and places switch in CLOSE. Observes red light ON, green light OFF. Observes white potential lights bright for bus 1DA and voltage indicated on voltmeter 1DA VOLTS.

CUES:

EVALUATOR NOTE: The critical step is closing BUS 1DA NORM FEED breaker.

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPS-005-N18, Loss of All ESF AC with Restoration via XTF-5052 (NRC JPM e)

IC SET: 295

INSTRUCTIONS:

If IC-295 is designated for this JPM then reset to IC-295 leaving the simulator in FREEZE.

1. Mark up rack copy of SOP-304, 115KV/7.2KV OPERATIONS, mark entry conditions complete for section V.A.
2. Mark up rack copy of EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION, mark steps 1-5 complete and step 6 in progress.
3. Mark up rack copy of EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER mark steps 1 -6 a. as completed with 6 a. AA in progress.
4. Place red hold tag on EFS XFMR 5 FEED CKT SW 1838 (XES5)
5. When Candidate is ready (on Evaluator cue) go to RUN.

If IC-295 is **not** designated for this JPM then initial conditions may be established by resetting to IC-10 and following the below directions:

1. With the simulator reset to IC-10 and in FREEZE, insert the following:

- **PRE-LOAD**
  - MAL-EPS006A
    DIESEL GENERATOR A FAILURE
    Fail To: FAIL
- **TRIGGER 1**
  - LOA-EPS167
    RELAY 51BX-1DB – 7.2 KV BUS 1DB OVERCURRENT
    Fail To: TRIP
  - MAL-EPS009
    LOSS OF UNIT AUXILIARY TRANSFORMER
  - MAL-EPS003
    LOSS OF EMERGENCY AUXILIARY TRANSFORMER
  - MAL-EPS018A
    LOSS OF ESF TRANSFORMER XTF-4
- **TRIGGER 2**
  - ANN-SG012
    ESFLS PNL DOOR OPEN
    Fail To: ON
  - LOA-EPS136
    LOAD SEQUENCER A: CONTROL POWER SWITCH
    Position To: OPEN
    Delay: 15 seconds
- **TRIGGER 3**
  - LOA-EPS007
    ESF XFMR XTF-4 LOW SIDE BREAKER – DX1
    Position To: FALSE
  - LOA-EPS008
    ESF XFMR XTF-5 LOW SIDE BREAKER – DX4
    Position To: FALSE
- **TRIGGER 4**
  - LOA-EPS188
    XFMR XTF-5052 DISCONNECT BREAKER – DX3

Position To: CLOSED

2. Open breaker EFS XFMR 5 FEED CKT SW 1838 (XES5)
3. Place the simulator in RUN and insert **TRIGGER 1**.
4. Perform Steps 1 through 5 of EOP-6.0.
5. Place the Simulator in Freeze and save to the desired IC.
6. Mark up rack copy of SOP-304, 115KV/7.2KV OPERATIONS, mark entry conditions complete for section V.A.
7. Mark up rack copy of EOP-1.0, E-0 REACTOR TRIP OR SAFETY INJECTION, mark steps 1-5 complete and step 6 in progress.
8. Mark up rack copy of EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER mark steps 1 -6 a. as completed with 6 a. AA in progress.
9. Place red hold tag on EFS XFMR 5 FEED CKT SW 1838 (XES5)
10. When Candidate is ready (on Evaluator cue) go to RUN.

COMMENTS:

Provide spare operator to silence alarms.

BOOTH OPERATOR: Use LOA resets page to silence HVAC alarms when they come in.

Mark strip chart recorders with date and time at the completion of each performance of this JPM.

Roll strip chart recorders to show no traces from the just completed performance after marking them as noted above.

CRITICAL TASK METHODOLOGY:

Step 2 is critical because the Alternate AC source (5052 fed from the Grid) to ESF bus 1DA is limited in loading capacity. If the load sequencer is not defeated the Alternate AC source breaker will not close.

Step 26 is critical because the 1DX bus will not be energized until the local action is performed and that action will not be performed until Candidate asks for it. Bus 1DX must be energized in order to energize ESF bus 1DA.

Step 27 is critical because the ESF Bus 1DA will not be energized until the Candidate closes the BUS 1DA NORM FEED.

REVISION HISTORY:

This JPM is new for the 2018 NRC exam for ILO-16-01.
SAR 12/2017.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The Unit Auxiliary Transformer (XTF-2) experienced a fault and caused a Turbine and Reactor Trip.

The Emergency Auxiliary transformer (XTF-32) is faulted.

ESF Transformer XTF-4 is faulted.

ESF Transformer XTF-5 was unavailable prior to the trip due to scheduled maintenance.

The 51BX-1DB relay is actuated.

The "A" DG failed to start automatically and will not start in manual.

The 115 KV Parr bus 2 and XTF-5052 have been determined to be available.

The crew is implementing EOP-6.0, ECA-0.0 LOSS OF ALL ESF AC POWER.

INITIATING CUES:

The CRS has directed you to restore offsite power to the 1DA ESF bus in accordance with EOP-6.0 step 6 a. Alternative Action. You are to refer to SOP-304, 115KV/7.2KV OPERATIONS, Section V.A.

A surrogate operator will acknowledge non-related alarms per your direction.

.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPS-006-N18

Loss of Power Range Instrument N-44

(NRC JPM f)

CANDIDATE: ______________________________

EXAMINER: ______________________________

TASK: 000-034-05-01 Respond to Power Range Instrumentation Channel Failure

TASK STANDARD:
Channel N-44 has been removed from service. Control rod motion has been stopped; Bank selector switch placed in MAN.

TERMINATING CUE: Tech Spec Status lights for Channel IV have been verified in correct status.

PREFERRED EVALUATION LOCATION
SIMULATOR

PREFERRED EVALUATION METHOD
PERFORM

REFERENCES:

AOP-401.10 POWER RANGE CHANNEL FAILURE

INDEX NO	*K/A NO.*		*RO*	*SRO*
015000A201	A2.01	Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation.	3.5	3.9

TOOLS: Rack copy of AOP-401.10, POWER RANGE CHANNEL FAILURE

EVALUATION TIME 10 ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)4

TIME START: ________ TIME FINISH: ________ PERFORMANCE TIME: ________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ____________________

EXAMINER: ____________________ ____________/________
SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The plant is operating at 75% power with all controls in automatic.

INITIATING CUES:

Respond to developing plant conditions.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 1: Verify normal indication on Power Range Channel N-44.

STEP STANDARD:

Candidate locates % FULL POWER % NI-44B indicator and observes N-44 has failed low.

CUES:

BOOTH OPERATOR CUE: When the Evaluator gives the direction, insert **TRIGGER 1**.

EVALUATOR NOTE: This is an immediate operator action to be performed from memory.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Alternative Action Step 1; IF Power Range Channel N-44 has failed, THEN place the ROD CNTRL BANK SEL Switch in MAN.

STEP STANDARD:

Candidate locates ROD CNTRL BANK SEL and places it in the MAN position.

CUES:

EVALUATOR NOTE: This is an immediate operator action.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 2: Stabilize any plant transients in progress.

STEP STANDARD:

Verifies no load change is in progress.

CUES:

EVALUATOR NOTE: This is an immediate operator action to be performed from memory.

EVALUATOR NOTE: Provide the rack copy of AOP-401.10 once candidate has verified no transients in progress.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 3: Maintain stable plant conditions.

STEP STANDARD:

Pzr pressure and Tavg maintained stable.

CUES:

EVALUATOR NOTE: This is a continuous action step.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 5

Step 4: Verify no testing is in progress on the operable Power Range channels.

STEP STANDARD:

Candidate looks at NI panel and observes no testing and/or asks CRS if any testing is in progress.

CUES:

EVALUATOR CUE: If Candidate asks concerning testing in progress, as CRS reply “No testing is in progress”.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 6

Step 5: Place ROD STOP BYPASS Switch (on the MISCELLANEOUS CONTROL AND INDICATION PANEL) for the failed Power Range channel in BYPASS.

Step 6: Verify the appropriate Rod Stop Bypass status light is bright:

- For N-44, B2 OP ROD STOP BYP (XCP-6111 4-4).

STEP STANDARD:

Candidate locates the MISCELLANEOUS CONTROL AND INDICATION PANEL and the ROD STOP BYPASS SWITCH and places it in the BYPASS PR N44 position.

Candidate locates the Status Light display and observes XCP-6111 4-4, B2 OP ROD STOP BYP is bright.

CUES:

EVALUATOR NOTE: The status light panel XCP-6111 is located above the EFW pump controls and below annunciator panels XCP-622 and XCP-623.

EVALUATOR CUE: Provide the Candidate the following cue "The BOP will monitor Rods in manual."

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 7

Step 7: Adjust Control Rods to maintain Tavg within 1.0°F of Tref.

STEP STANDARD:

Candidate locates Tavg – Tref indication (TR-406 on panel XCP 6109R) and assesses mismatch. If Tavg is more than 1°F above Tref adjusts rod position by placing ROD CONTROL, ROD MOTION switch in the IN or OUT position as necessary to restore mismatch to within 1°F.

CUES:

EVALUATOR NOTE: Candidate may not make any adjustment if the mismatch is slight.

EVALUATOR CUE: If Candidate requests a control band provide the following cue "Control band is 195 to 185 steps on Control Bank D"

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 8

Step 8: Align the Power Range channel comparator circuits:

a. Place the following switches to the failed Power Range channel position:

1) COMPARATOR CHANNEL DEFEAT Switch (on the COMPARATOR AND RATE drawer).

2) UPPER SECTION Switch (on the DETECTOR CURRENT COMPARATOR drawer).

3) LOWER SECTION Switch (on the DETECTOR CURRENT COMPARATOR drawer).

STEP STANDARD:

Candidate locates:

COMPARATOR AND RATE drawer and places COMPARATOR CHANNEL DEFEAT switch to N44.

DETECTOR CURRENT COMPARATOR drawer and places UPPER SECTION switch to PRN44

DETECTOR CURRENT COMPARATOR drawer and places LOWER SECTION switch to PRN44

CUES:

EVALUATOR NOTE: The critical portion of this step is placing the UPPER SECTION and LOWER SECTION switches in defeat as these signals input to XCP-620 1-5 and 1-6 which are the TS Required QPTR monitoring alarms. Removal of the inoperable channel input restores the alarm to operable.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 9

Step 9: Ensure NR-45 is selected to the appropriate operable channels.

STEP STANDARD:

Candidate locates NIS RCDR PEN 1 (PEN 2) SELECT NR-45 and that the PEN 1 selector is NOT in the P4 position and that the PEN 2 selector is NOT in the P4 or △ FIV position.

CUES:

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 10

Step 10: Within one hour, verify that the following permissive status lights are in the required state for the existing plant condition.

- P-7.
- P-8.
- P-9.
- P-10.

STEP STANDARD:

Candidate locates XCP-6109 REACTOR PERMISSIVES and observes:

P7 light dim under REACTOR TRIP BLOCKED section.

P8 light dim under REACTOR TRIP BLOCKED section.

P9 light dim under REACTOR TRIP BLOCKED section.

P10 light bright under NIS PR section.

CUES:

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPS-006-N18 Loss of Power Range Instrument N-44 (NRC JPM f)

IC SET: 296

INSTRUCTIONS:

If IC-296 is designated for this JPM then reset to IC-296 leaving the simulator in FREEZE.

1. Place Simulator in RUN
2. Reset SIPCS screens to normal full power arrangement and clear all SIPCS and Bailey alarms.

If IC-296 is **not** designated for this JPM then initial conditions may be established by resetting to IC-11 and following the below directions:

1. With the simulator reset to IC-11 and in FREEZE, insert the following:
   - **TRIGGER 1**
     - MAL-NIS003D
POWER RANGE CHANNEL 44 FAILURE
Ramp: 5 sec
Final Value: 0%
2. Place the Simulator in FREEZE and save to the desired IC.
3. Place Simulator in RUN.
4. Reset SIPCS screens to normal full power arrangement and clear all SIPCS and Bailey alarms.

COMMENTS:

Mark strip chart recorders with date and time at the completion of each performance of this JPM.

Roll strip chart recorders to show no traces from the just completed performance after marking them as noted above.

CRITICAL TASK METHODOLOGY:

Step 2 is critical because rod control will eventually (3 min.) restore Tavg to Tref (power rate mismatch signals die off, rods control on Tavg/Tref). Leaving rods in AUTO constitutes failure, as further transients on the failed channel (e.g. trouble shooting) would produce more uncontrolled rod motion.

Step 8 is critical because the inoperable channel must be removed from the QPTR monitoring alarm required by TS 4.2.4.1.b. Failure to remove N-43 input results in a TS action requirement to perform manual monitoring for QPTR.

REVISION HISTORY:

This is a bank JPM that was selected for the 2018 NRC Exam for ILO-16-01 under safety function 7. The source JPM, JPS-008-A15 was revised to reflect current procedure guidance described in AOP-401.10.

SAR 1/2018.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The plant is operating at 75% power with all controls in automatic.

INITIATING CUES:

Respond to developing plant conditions.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPS-007-N18**

Swap Active CCW Loops

(NRC JPM g)

CANDIDATE: ______________________

EXAMINER: ______________________

TASK: 008-021-01-01 Switch Component Cooling Water Trains per SOP-117/SOP-118/SOP-501

TASK STANDARD:

CCW non-essential loads aligned to the "B" loop of CCW and the "A" CCW loop non-essentials are secured.

TERMINATING CUE: CCW flow to the RB is verified as not isolated.

PREFERRED EVALUATION LOCATION SIMULATOR

PREFERRED EVALUATION METHOD PERFORM

REFERENCES:

SOP-118, COMPONENT COOLING SYSTEM

INDEX NO	*K/A NO.*		*RO*	*SRO*
008000A401	A4.01	Ability to manually operate and/or monitor in the control room: CCW indications and controls	3.3	3.1

TOOLS: Rack copy of SOP-118, COMPONENT COOLING SYSTEM with Section III.B marked as complete up to step 2.5.

EVALUATION TIME 10 ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)6

TIME START: ______________ TIME FINISH: ______________ PERFORMANCE TIME: ____________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ________________________________

EXAMINER: ________________________________ ____________________/__________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The “A” CCW loop is currently the Active loop.

The “C” CCW pump has been aligned to Train “B” per SOP-118, COMPONENT COOLING WATER, Section III.B, step 2.2.

INITIATING CUES:

The CRS has directed you to establish CCW Train “B” as the active loop using SOP-118 Section III.B, step 2.5.

A surrogate operator will acknowledge non-related alarms per your direction.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 2.5 Establish Train B as the active loop as follows:

a: Ensure MVB-9503B, CC TO RHR HX B, is open.

STEP STANDARD:

Candidate locates control switch for MVB-9503B and notes red light ON, green light OFF.

CUES:

EVALUATOR NOTE: Provide marked up copy of SOP-118 once candidate has been briefed on the initiating cue.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Step 2.5b: Start one of the following in slow speed:

1) XPP-0001B, PUMP B.

STEP STANDARD:

Candidate locates the PUMP B XPP-0001B control switch and places it to START, verifies red light ON, green light OFF and pump AMPs indicated.

CUES:

EVALUATOR CUE: If Candidate requests a peer check, acknowledge request.

BOOTH OPERATOR CUE: When contacted as field operator concerning “B” CCW pump ready for start, acknowledge request, wait a briefly then report “The “B” CCW pump is ready for start”.

BOOTH OPERATOR CUE: Once Candidate starts the “B” CCW pump call back and report “The “B” CCW pump is running good after start and RML-2B indicates greater than 5 gpm flow.”

EVALUATOR NOTE: The “C” CCW pump breaker is not racked up thus the Candidate should start the “B” CCW Pump. The step is critical because a pump must be in service to provide flow to the oncoming active loop.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Procedure caution Step 2.5.c and 2.5.d: Failure to complete Step 2.5.d in a timely manner after reducing RHR Heat Exchanger flow will result in a loss of flow through the running CCW Pump or excessive flow perturbations in the CCW non-essential loop.

Step 2.5c: Start MVB-9503B, CC TO RHR HX B, stroking in the closed direction.

STEP STANDARD:

Candidate locates MVB-9503B control switch and places it to CLOSE, verifies red light ON, green light ON, proceeds to step 2.5d.

CUES:

EVALUATOR NOTE: This step is critical because the RHR Heat exchanger must be isolated from the active loop to assure adequate capacity for loads serviced on the non essential header and to preclude overload on the running CCW pump.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 2.5.d: When flow, as indicated on FI-7044, HX B FLOW GPM, is between 5000 gpm and 4000 gpm, perform the following in rapid succession:

1) Open MVB-9687B/9525B, LP B NON-ESSEN LOAD ISOL.

2) Open MVB-9524B/9526B, LP B NON-ESSEN LOAD ISOL.

3) Close MVB-9524A/9526A, LP A NON-ESSEN LOAD ISOL.

4) Close MVB-9687A/9525A, LP A NON-ESSEN LOAD ISOL.

5) Open MVB-9503A, CC TO RHR HX A.

STEP STANDARD:

Candidate locates:

FI-7044, HX B FLOW GPM and verifies between 5000 and 4000 gpm.

Control Switch for MVB-9687B/9525B and places it to OPEN, verifies red light ON, green light ON.

Control Switch for MVB-9524B/9526B and places it to OPEN, verifies red light ON, green light ON.

Control Switch for MVB-9524A/9526A and places it to CLOSE, verifies red light ON, green light ON.

Control Switch for MVB-9687A/9525A and places it to CLOSE, verifies red light ON, green light ON.

Control Switch for MVB-9503A and places it to OPEN, verifies red light ON, green light ON

CUES:

EVALUATOR NOTE: Steps 2.5.d 1), 2), 3) and 4) are critical to preserve the CCW pump function and to align Train “B” as the active loop

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPS-007-N18, Swap Active CCW Loops (NRC JPM g)

IC SET: 297

INSTRUCTIONS:

If IC-297 is designated for this JPM then reset to IC-297.

1. Place Simulator in RUN.
2. Mark up the rack copy of SOP-118 with Section IIIB steps 2.1 and 2.2 completed and steps 2.3 and 2.4 as NA.
3. Reset SIPCS screens to normal at power arrangement and clear all SIPCS and Bailey alarms.

If IC-297 is **not** designated for this JPM then initial conditions may be established by resetting to IC-10 and following the below directions:

1. With the simulator reset to IC-10 and in FREEZE, insert the following:
2. Go to RUN.
3. Perform SOP-118 steps 2.1 and 2.2.

   Swap "C" CCW pump to Train "B using the following:

   - LOA-CCW001
     CCW PUMP C DISCONNECT SWITCH
     Position to: TRAIN B
   - LOA-CCW009
     CC PP C SUCT LP B ISO VLV - 9519
     Final Value: 1 (OPEN)
   - LOA-CCW010
     CC PP C SUCT LP A ISO VLV – 9521
     Final Value: 0 (CLOSED)
   - LOA-CCW011
     CC PP C DISCH LP B ISO VLV - 9523C
     Final Value: 1 (OPEN)
   - LOA-CCW012
     CC PP C DISCH LP A ISO VLV - 9523D
     Final Value: 0 (CLOSED)
   - LOA-CCW044
     CELL SWITCH OF CCW PUMP C TRAIN A
     Position To: RACK OUT

4. Place the Simulator in FREEZE and save to the desired IC.
5. Mark up the rack copy of SOP-118 with steps 2.1 and 2.2 completed and steps 2.3 and 2.4 as NA.
6. Place Simulator in RUN, reset SIPCS screens to normal full power arrangement and clear all SIPCS and Bailey alarms.

COMMENTS:

Provide spare operator to silence alarms.

Mark strip chart recorders with date and time at the completion of each performance of this JPM.

Roll strip chart recorders to show no traces from the just completed performance after marking them as noted above.

CRITICAL TASK METHODOLOGY:

Step 2 is critical because a CCW pump must be in service to provide flow to the oncoming active loop.

Step 3 is critical because the RHR Heat exchanger must be isolated from the active loop to assure adequate capacity for loads serviced on the non essential header and to preclude overload on the running CCW pump.

Step 4 is critical because the valve alignment is required preserve the CCW pump function and to align Train “B” as the active loop.

REVISION HISTORY:

This is a modified bank JPM (JPS-070).
SAR 12/2017.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The “A” CCW loop is currently the Active loop.

The “C” CCW pump has been aligned to Train “B” per SOP-118, COMPONENT COOLING WATER, Section III.B, step 2.2.

INITIATING CUES:

The CRS has directed you to Establish CCW Train “B” as the active loop using SOP-118 Section III.B, step 2.5.

A surrogate operator will acknowledge non-related alarms per your direction.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPS-008F-N18

Respond to High RB Radiation Level (Alternate Path)

(NRC JPM h)

CANDIDATE: ____________________

EXAMINER: ____________________

TASK: 311-018-06-01 Response to High RB Radiation Level

TASK STANDARD:

Both RBCU Filter Trains are operating.

TERMINATING CUE: Both RBCU Filter Trains are operating.

PREFERRED EVALUATION LOCATION
SIMULATOR

PREFERRED EVALUATION METHOD
PERFORM

REFERENCES:

EOP-17.2, RESPONSE TO HIGH REACTOR BUILDING RADIATION LEVEL

INDEX NO	*K/A NO.*		*RO*	*SRO*
WE16EA11	EA1.1	Ability to operate and/or monitor the following as they apply to the (High Containment Radiation): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.1	3.2

TOOLS: Rack copy of EOP-17.2, RESPONSE TO HIGH REACTOR BUILDING RADIATION LEVEL

EVALUATION TIME 10 ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)8

TIME START: ____________ TIME FINISH: ____________ PERFORMANCE TIME: ____________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ______________________________

EXAMINER: ______________________________ ____________________/__________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

A LOCA is in progress.

A Yellow Path on Containment exists due to Containment Radiation > 2 R/hr

The Red Path indication on the Integrity Critical Safety Function has been evaluated and no action is required.

INITIATING CUES:

The CRS has directed you to implement EOP-17.2, RESPONSE TO HIGH REACTOR BUILDING RADIATION LEVEL.

A surrogate operator will acknowledge non-related alarms per your direction.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 1. Verify Containment Ventilation Isolation Valves closed by verifying the following SAFETY INJECTION monitor lights are dim:

a. XCP-6103 2-1 (POST ACCID HR EXH 6056/6066).

b. XCP-6103 3-4 (POST ACCID HR EXH 6057 & 6067).

STEP STANDARD:

Candidate locates XCP 6103 and verifies light 2-1 POST ACCID HR EXH 6056/6066 and 3-4 POST ACCID HR EXH 6057 & 6067 are dim.

CUES:

BOOTH OPERATOR CUE: Place the Simulator in RUN when the Evaluator indicates the JPM may begin.

EVALUATOR CUE: Provide the rack copy of EOP-17.2 once the candidate has been briefed on the initiating cue.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Step 2. Start both RBCU HEPA Filter Trains:

Step 2 a. Stop RBCU Normal Speed Fans:

- XFN 0064A-AH,1A NORM.
- XFN 0065A-AH, 2A NORM.
- XFN 0064B-AH,1B NORM.
- XFN 0065B-AH, 2B NORM.

STEP STANDARD:

Candidate locates RBCU controls and verifies RBCU Normal Speed Fans are stopped:

1A NORM, XFN-0064A-AH, observes red light OFF, green light ON with 0 amps indicated.

2A NORM, XFN-0065A-AH, observes red light OFF, green light ON with 0 amps indicated.

1B NORM, XFN-0064B-AH, observes red light OFF, green light ON with 0 amps indicated.

2B NORM, XFN-0065B-AH, observes red light OFF, green light ON with 0 amps indicated.

CUES:

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 2 b. Place RBCU HEPA Filters in service by placing the switches in FILTER:

- XDP-110A, RBCU 64A HEPA FLTR BYP DMPR.
- XDP-111A, RBCU 65A HEPA FLTR BYP DMPR.
- XDP-110B, RBCU 64B HEPA FLTR BYP DMPR.
- XDP-111B, RBCU 65B HEPA FLTR BYP DMPR.

STEP STANDARD:

Candidate locates RBCU damper controls and places:

XDP-110A switch in FILTER, observes red BYP light ON and green FILTER light OFF.

XDP-111A switch in FILTER, observes red BYP light OFF and green FILTER light ON.

XDP-110B switch in FILTER, observes red BYP light OFF and green FILTER light ON.

XDP-111B switch in FILTER, observes red BYP light ON and green FILTER light OFF.

CUES:

EVALUATOR NOTE: RBCU 64A HEPA and RBCU 65B HEPA Filter Bypass dampers failed to reposition to filter as shown by XDP-110A; RBCU 64A HEPA FLTR BYP DMPR, red light ON, green light OFF and XDP-111B; RBCU 65BA HEPA FLTR BYP DMPR, red light ON, green light OFF.

EVALUATOR NOTE: Alternate path begins at this step.

EVALUATOR NOTE: Placing XDP-111A in FILTER is critical because the damper will remain bypassed otherwise and there would be no Train "A" HEPA filter in service.

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Alternative Action Step 2 b. IF any HEPA Filter fails to position to FILTER, THEN ensure the RBCU TRAIN A(B) EMERG Switch is selected to the fan with its respective filter aligned.

STEP STANDARD:

Candidate locates RBCU TRAIN A EMERG switch and places it to the XFN65A position, observes XFN 64A white light OFF, XFN 65A white light ON and RBCU-65A does not start.

Candidate locates RBCU TRAIN B EMERG switch and verifies it is in the XFN64B position, observes XFN 64B white light ON, XFN 65B white light OFF.

CUES:

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 5

Step 2 c. Start the RBCU Slow Speed Fans selected on RBCU TRAIN A EMERG and RBCU TRAIN B EMERG.

STEP STANDARD:

Candidate places XFN 0065A-AH slow speed fan switch to START, observes; red light ON, green light OFF, and normal amps.

Candidate places XFN 0064B-AH slow speed fan switch to START, observes; red light ON, green light OFF, and normal amps.

CUES:

EVALUATOR NOTE: The Candidate may place XFN 0064A-AH to stop.

EVALUATOR NOTE: Placing the XFN-65A switch in START is critical because the fan must be started in order to align a Train “A” HEPA filter. Placing the XFN-64B switch in START is critical because the fan must be started in order to align a Train “B” HEPA filter.

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPS-008-N18, Respond to High RB Radiation Level (Alternate Path) (NRC JPM h)

IC SET: 298

INSTRUCTIONS:

If IC-298 is designated for this JPM then reset to IC-298 leaving the simulator in FREEZE.

1. When Candidate is ready (on Evaluator cue) go to RUN.

If IC-298 is **not** designated for this JPM then initial conditions may be established by resetting to IC-10 and following the below directions:

1. With the simulator reset to IC-10 and in FREEZE, insert the following:

- **PRE-LOAD**
  - MAL-RCS010
    GROSS ISOTOPIC CONCENTRATION IN THE RCS
    Final Value: 1.42e+008 uC/gm
  - PLP-RMS027
    AREA RAD MON RM-G7 BACKGROUND
    Final Value: 2.1R/hr
  - PLP-RMS038
    AREA RAD MON RM-G18 BACKGROUND
    Final Value: 2.1R/hr
  - OVR-AH025E
    SS-AH281 RBCU HEPA FILTER BYPASS DAMPER
    Override To: False
  - OVR-AH025A
    SS-AH281 RBCU HEPA FILTER BYPASS DAMPER
    Override To: Off
  - OVR-AH025B
    SS-AH281 RBCU HEPA FILTER BYPASS DAMPER
    Override To: On
  - OVR-AH018E
    SS-AH283 RBCU HEPA FILTER BYPASS DAMPER
    Override To: False
  - OVR-AH018A
    SS-AH283 RBCU HEPA FILTER BYPASS DAMPER
    Override To: Off
  - OVR-AH018B
    SS-AH283 RBCU HEPA FILTER BYPASS DAMPER
    Override To: On
  - OVR-AH026E
    SS-AH282 RBCU HEPA FILTER BYPASS DAMPER
    Override To: False
  - OVR-AH026A
    SS-AH282 RBCU HEPA FILTER BYPASS DAMPER
    Override To: Off
  - OVR-AH026B
    SS-AH282 RBCU HEPA FILTER BYPASS DAMPER
    Override To: On
  - OVR-AH024E
    SS-AH284 RBCU HEPA FILTER BYPASS DAMPER
    Override To: False
  - OVR-AH024A

SS-AH284 RBCU HEPA FILTER BYPASS DAMPER
Override To: Off
  - OVR-AH024B
SS-AH284 RBCU HEPA FILTER BYPASS DAMPER
Override To: On
  - PMP-AH046F
XFN0064BL RB CLG FAN LOW FAIL TO START

- **TRIGGER 1**
  - MAL-RCS019C
DBA LOCA (HOT LEG) LOOP 3
- **AUTO TRIGGER 2** (x02i082f==1) HEPA Filter Bypass damper XDP-111A placed in filter
  - OVR-AH018E (NEW)
SS-AH283 RBCU HEPA FILTER BYPASS DAMPER
Override To: False
Delete in: 1 sec
  - OVR-AH018A (NEW)
SS-AH283 RBCU HEPA FILTER BYPASS DAMPER
Override To: Off
Delete in: 1 sec
  - OVR-AH018B (NEW)
SS-AH283 RBCU HEPA FILTER BYPASS DAMPER
Override To: On
Delete in: 1 sec
- **AUTO TRIGGER 3** (x02i394f==1) HEPA Filter Bypass damper XDP-110B placed in filter
  - OVR-AH026E (NEW)
SS-AH284 RBCU HEPA FILTER BYPASS DAMPER
Override To: False
Delete in: 1 sec
  - OVR-AH026A (NEW)
SS-AH284 RBCU HEPA FILTER BYPASS DAMPER
Override To: Off
Delete in: 1 sec
  - OVR-AH026B (NEW)
SS-AH284 RBCU HEPA FILTER BYPASS DAMPER
Override To: On
Delete in: 1 sec

3. Place simulator in RUN and activate **TRIGGER 1.**
4. Perform EOP-1.0.
5. Perform steps of EOP-2.0 as necessary.
6. Run until RB Press < 11 psig and Containment CSFST is YELLOW due to high RB Rad Levels.
7. Place simulator in FREEZE and SAVE IC.
8. When Candidate is ready (on Evaluator cue) go to RUN

COMMENTS:

Provide spare operator to silence alarms.

BOOTH OPERATOR: Use LOA resets page to silence HVAC alarms when they come in.

Mark strip chart recorders with date and time at the completion of each performance of this JPM.

Roll strip chart recorders to show no traces from the just completed performance after marking them as noted above.

CRITICAL TASK METHODOLOGY:

Step 3 is critical because XDP-111B must be placed in filter since XDP-110B will not reposition to filter.

Step 5 is critical because XFN-65A is the only RBCU with a HEPA filter available on Train “A” likewise XFN-64B is the only RBCU with a HEPA filter available on Train “B”. The fans will not start unless the Candidate places the respective control switch to START.

REVISION HISTORY:

This JPM is a revision of JPSF-1003-N16, RESPOND TO HIGH RB RADIATION LEVEL which was developed as a new JPM on the 2016 NRC exam for ILO-14-0.

SAR 10/2017.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

A LOCA is in progress.

A Yellow Path on Containment exists due to Containment Radiation > 2 R/hr

The Red Path indication on the Integrity Critical Safety Function has been evaluated and no action is required.

INITIATING CUES:

The CRS has directed you to implement EOP-17.2, RESPONSE TO HIGH REACTOR BUILDING RADIATION LEVEL.

A surrogate operator will acknowledge non-related alarms per your direction.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPP-009F-N18

Locally Trip the Reactor (Open motor breakers on 1C1 and 1B1) (Alternate Path)

(NRC JPM i)

CANDIDATE: ______________________________

EXAMINER: ______________________________

TASK: 000-117-05-04 Respond to Abnormal Nuclear Power Generation

TASK STANDARD:

The Reactor has been tripped by opening Breaker XSW1C1 05D (TB-412) and Breaker XSW1B1 06C (TB-436).

TERMINATING CUE:

Breaker XSW1C1 05D (TB-412) and Breaker XSW1B1 06C (TB-436) are open.

PREFERRED EVALUATION LOCATION PLANT

PREFERRED EVALUATION METHOD SIMULATE

REFERENCES:

EOP-13.0, FR-S.1, RESPONSE TO ABNORMAL NUCLEAR POWER GENERATION

INDEX NO	***K/A NO.***		***RO***	***SRO***
000029EA112	EA1.12	Ability to operate and monitor the following as they apply to a ATWS: M/G set power supply and reactor trip breakers.	4.1	4.0
000029EK3.12	EK 3.12	Knowledge of the reasons for the following responses as they apply to the ATWS: Actions contained in EOP for ATWS.	4.4	4.7

TOOLS: Hard copy of EOP-13.0 ATT. 1, TRIPPING THE REACTOR LOCALLY
Picture 1 – Photo of Rx Trip Breaker A in the closed position.
Picture 2 – Photo of Rx Trip Breaker B in the closed position.
Picture 3 – Photo of Rx Trip Bypass Breaker A in the racked out and open position.
Picture 4 – Photo of Rx Trip Bypass Breaker B in the racked out and open position.

EVALUATION TIME 10 ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)8

TIME START: ____________ TIME FINISH: ____________ PERFORMANCE TIME: ____________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ________________________________

EXAMINER: ________________________________ ____________________/__________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: High Noise Area.

INITIAL CONDITION:

The RO inserted a manual Reactor trip but the Reactor trip breakers did not open. EOP-13.0 has been implemented for the ATWS.

INITIATING CUES:

Control Room Supervisor directs you to locally trip the reactor using EOP-13.0 Attachment 1, TRIPPING THE REACTOR LOCALLY.

AT NO TIME ARE YOU TO OPERATE ANY PLANT EQUIPMENT!

FOR ELECTRICAL MANIPULATIONS, AT NO TIME ARE YOU TO BREAK THE PLANE OF THE ELECTRICAL PANEL!

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 1; Open all the following breakers at XSW0001-CR, Reactor Trip Breaker Switchgear (IB-463):

- XSW0001-RT-A, REACTOR TRIP BREAKER A.
- XSW0001-RT-B, REACTOR TRIP BREAKER B.
- XSW0001-BY-A, REACTOR TRIP BYPASS BREAKER A.
- XSW0001-BY-B, REACTOR TRIP BYPASS BREAKER B.

STEP STANDARD:

Candidate locates reactor trip and bypass breakers and simulates an attempt to open the Reactor Trip Breakers by indicating they would depress the RX TRIP BKR A(B) MECH TRIP pushbuttons on the lower right of the reactor trip breaker cubicle door or that they would open the Trip breaker cubicle and depress the red Trip pushbutton.

CUES:

EVALUATOR NOTE: Provide a hard copy of EOP-1.0 Attachment 1once the Candidate has been briefed on the initiating cue.

EVALUATOR NOTE: If the Candidate indicates they would open the Reactor Trip Breaker cabinet to open the breaker provide a photograph of the Trip Breaker they ask for and have the Candidate identify the action they would take using the photograph.

Use Picture 1 for RT-A

Use Picture 2 for RT-B

EVALUATOR CUE: After the Candidate indicates whatever action they would perform on either Reactor Trip breaker, report “The Breaker did **not** open.” This is the point that the JPM becomes Alternate Path.

EVALUATOR NOTE: The Bypass breakers are racked out and no action is required to open them. The Candidate may indicate that they would depress the Shunt Trip Pushbutton for the Bypass breakers or the RX TRIP BYPASS BKR A(B) MECH TRIP trip button or they may not attempt any action.

EVALUATOR NOTE: If the Candidate indicates they would open the Reactor Trip Breaker Bypass Breaker cabinet to open the breaker provide the photograph of the Bypass Breaker they ask for and have the Candidate identify the action they would take using the photograph.

Use Picture 3 for BY-A

Use Picture 4 for BY-B

EVALUATOR CUE: After the Candidate indicates whatever action if any they would perform on either Bypass breaker, report “The Breaker is open.”

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Step 1, Alternative Action; Open all the following breakers at XCA0005-CR, Rod Drive MG Control Cabinet (IB-463):

- Generator No. 1 GENERATOR.
- Generator No. 1 MOTOR.
- Generator No. 2 GENERATOR.
- Generator No. 2 MOTOR.

STEP STANDARD:

Candidate locates Rod Drive MG Control Cabinet and simulates an attempt to open the Generator and Motor Breakers by indicating that they would turn the pistol grips counterclockwise to the TRIP position.

CUES:

EVALUATOR CUE: After the Candidate indicates the action they would perform, report "The red light is ON and the green light is OFF." Repeat the cue after each attempt to open a breaker.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 1, Alternative Action; IF the Reactor is NOT tripped, THEN open the following breakers:

- XSW1B1 06C, ROD DRIVE MG SET B XMG0001B-CR (TB-436).

STEP STANDARD:

Candidate locates Rod Drive MG Set "B" breaker and simulates an attempt to open it by indicating that they would push the TRIP pushbutton.

CUES:

EVALUATOR CUE: If the Candidate initially indicates they would depress the TRIP button on the right side of the breaker, report "The red light is ON, the green light is OFF and the red CLOSED Flag is showing."

EVALUATOR CUE: After the Candidate indicates the correct action, report "The red light is OFF, green light is ON and an OPEN flag is showing."

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 1, Alternative Action; IF the Reactor is NOT tripped, THEN open the following breakers:

- XSW1C1 05D, ROD DRIVE MG SET A XMG0001A-CR (TB-412).

STEP STANDARD:

Candidate locates Rod Drive MG Set "A" breaker and simulates an attempt to open it by indicating that they would push the TRIP pushbutton.

CUES:

EVALUATOR CUE: If the Candidate initially indicates they would depress the TRIP button on the right side of the breaker, report "The red light is ON, the green light is OFF and the red CLOSED Flag is showing."

EVALUATOR CUE: After the Candidate indicates the correct action they, report "The red light is OFF, green light is ON and an OPEN flag is showing."

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPP-009F-N18, Locally Trip the Reactor (Alternate Path) (NRC JPM i)

IC SET: NA

INSTRUCTIONS: NA

COMMENTS:

CRITICAL TASK METHODOLOGY:

Step 3 is critical because with neither the Reactor Tip Breaker nor the MG Set control breakers opening the 480 volt supply to the Motor must be opened on both MG sets in order to trip the reactor.

Step 4 is critical because with neither the Reactor Tip Breaker nor the MG Set control breakers opening the 480 volt supply to the Motor must be opened on both MG sets in order to trip the reactor.

REVISION HISTORY:

This JPM is a minor revision of JPPF-096A, LOCALLY TRIP THE REACTOR.
SAR 1/2018.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: None.

INITIAL CONDITION:

The RO inserted a manual Reactor trip but the Reactor trip breakers did not open. EOP-13.0 has been implemented for the ATWS.

INITIATING CUES:

The CRS directs you to locally trip the reactor using EOP-13.0 Attachment 1, TRIPPING THE REACTOR LOCALLY.

AT NO TIME ARE YOU TO OPERATE ANY PLANT EQUIPMENT!

FOR ELECTRICAL MANIPULATIONS, AT NO TIME ARE YOU TO BREAK THE PLANE OF THE ELECTRICAL PANEL!

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

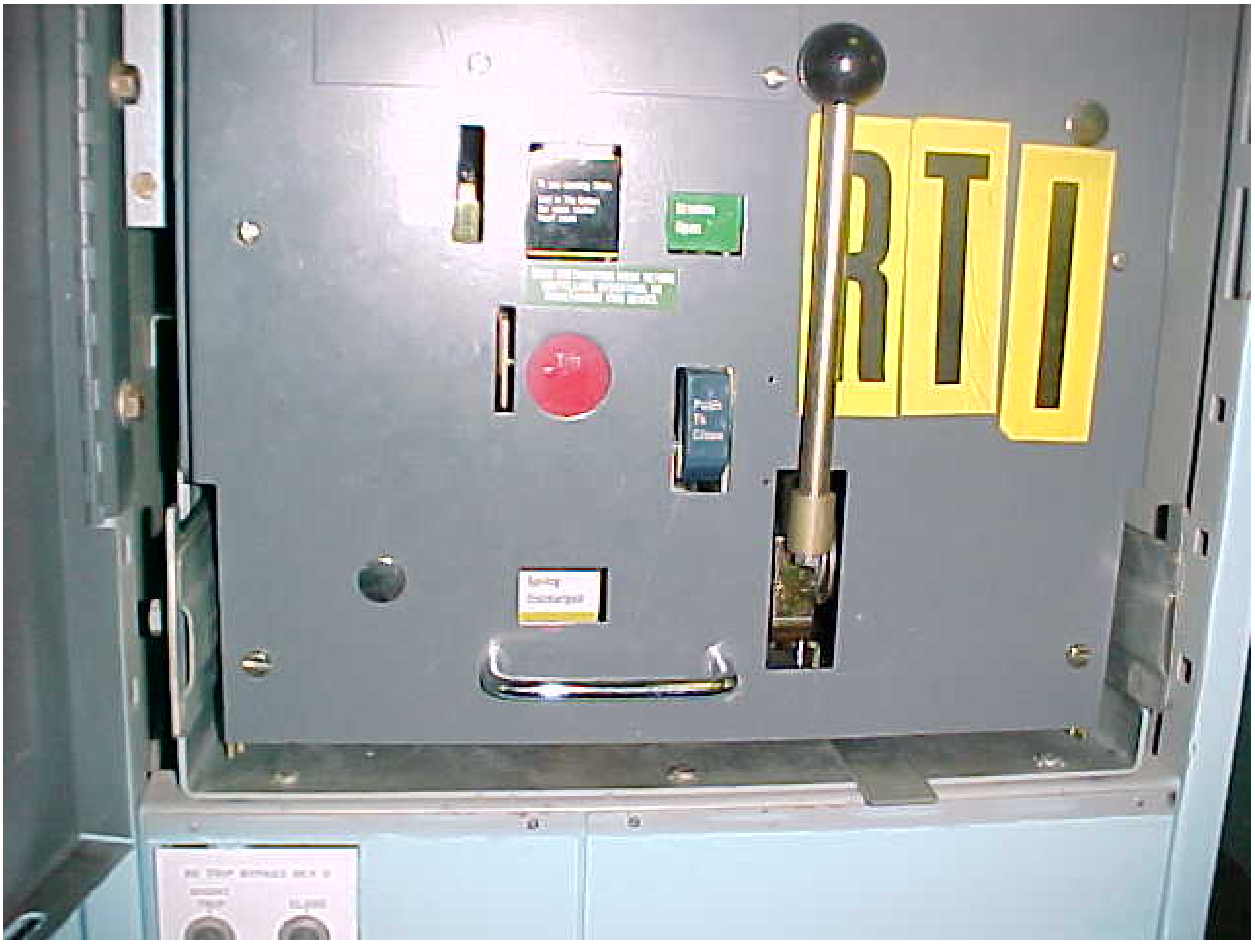
Picture 1


NO OVERCURRENT
PROTECTION PROVIDED
RT-1
Conn
Test
Disc
Remove
To use Levering Mech
Hold in Trip Button
Pull down shutter
Insert crank
Breaker
Closed
00049
WARNING
CAUTION
Trip
Push
To
Close
CIRCUIT BREAKER BUILT AT:
WESTINGHOUSE ELECTRIC COMPANY
Repair, Replacements & Automation Services
NEW STANTON, PA 15672-9606
SERIAL NUMBER: 208.039-1 IT.10
DATE COMPLETED: 05/16/2008
• Before doing dielectric testing, see caution note on left side of breaker.
• Read instruction book before installing, operating or maintaining this device.
Spring
Discharged

Picture 2

LOW-VOLTAGE AC POWER CIRCUIT BREAKER
RT-2
NO OVERCURRENT PROTECTION PROVIDED
To use Levering Mech't
Hold in Trip Button
Pull down shutter
Insert crank
Breaker Closed
WARNING
CAUTION
Trip
Push To Close
Spring Discharged
CIRCUIT BREAKER BUILT AT:
WESTINGHOUSE ELECTRIC COMPANY
Repair, Replacements & Automation Services
NEW STANTON, PA 15672-9606
SERIAL NUMBER: 208.039-2 IT.10
DATE COMPLETED: 05/16/2008
• Before doing dielectric testing, see caution note on left side of breaker.
• Read instruction book before installing, operating or maintaining this device.

Picture 3



Picture 4

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPP-010-N18

Locally De-Energize and Close MS Loop "B" and "C" To TDEFP

(NRC JPM j)

CANDIDATE: ____________________

EXAMINER: ____________________

TASK: 000-169-05-04 Locally Isolate a Faulted Steam Generator per EOP-3.0.

TASK STANDARD:

Turbine driven emergency feed pump main steam loop "B" and "C" supply valve are manually closed and power removed from 2802A per EOP-3.0.

TERMINATING CUE: MVG-2802A and MVG-2802B are closed and de-energized per EOP-3.0.

PREFERRED EVALUATION LOCATION PLANT

PREFERRED EVALUATION METHOD SIMULATE

REFERENCES:

EOP-3.0, E-2 FAULTED STEAM GENERATOR ISOLATION

INDEX NO	***K/A NO.***		***RO***	***SRO***
000040A110	AA1.10	Ability to operate and/or monitor the following as they apply to the Steam Line Rupture: AFW System	4.1	4.1

TOOLS: 2018 NRC JPM j handout (mark up of EOP-3.0, STEP 5.j (page 4 of 9))

EVALUATION TIME 20 ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)6

TIME START: ________ TIME FINISH: ________ PERFORMANCE TIME: ________

PERFORMANCE RATING: SAT: ______ UNSAT: ______

CANDIDATE: ____________________

EXAMINER: ____________________ ____________/________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: Thermal Burn Hazard.

INITIAL CONDITION:

- The plant was at 100% power when a steamline break occurred inside containment.
- "B" and "C" SGs are faulted as indicated by steam pressure dropping in an uncontrolled manner.
- Both motor driven emergency feedwater pumps have started.
- The CRS has implemented EOP-3.0, FAULTED STEAM GENERATOR ISOLATION, after exiting EOP-1.0, REACTOR TRIP/SAFETY INJECTION ACTUATION.
- The BOP has tried and failed to close MVG-2802A, MS LOOP B TO TD EFP and MVG-2802B, MS LOOP C TO TD EFP from the MCB.
- The Shift Manager has approved the waiving of ISP-027 requirements due to the emergency condition.
- The AB operator (Unit 6 upper) has already opened breaker 05EH on XMC1DB2Y in accordance with EOP-3.0 step 5 j Alternative Action.

INITIATING CUES:

The RO directs you as the IB operator (Unit 7) to complete the following actions in accordance with EOP-3.0 step 5 j Alternative Action:

- Open breaker 05EH on XMC1DA2X.
- Locally close valve XVG-2802A-MS.
- Locally close valve XVG-2802B-MS.

AT NO TIME ARE YOU TO OPERATE ANY PLANT EQUIPMENT!

FOR ELECTRICAL MANIPULATIONS,
AT NO TIME ARE YOU TO BREAK THE
PLANE OF THE ELECTRICAL PANEL!

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: Yes	***SEQUENCED:*** Yes		***SAT*** ☐	***UNSAT*** ☐

JPM STEP: 1

Step 5.j; Alternative Action; Locally deenergize and close the appropriate valve.

- For SG B:
  1) Open XMC1DA2X 05EH, EF PUMP MAIN STEAM BLOCK VLV XVG2802A-MS (IB-463).

STEP STANDARD:

Candidate locates Breaker 1DA2X 05EH for XVG-02802A and simulates opening the breaker by indicating that they would pull the breaker control lever down.

CUES:

EVALUATOR NOTE: Provide a hard copy EOP-3.0, E-2, FAULTED STEAM GENERATOR ISOLATION, page 4 which includes step 5.j once the Candidate has been briefed on the initiating cue.

EVALUATOR NOTE: The only sequence that is important is to open each breaker before closing each valve. If the examinee repositions a valve prior to opening the breaker the JPM is considered unsat.

EVALUATOR CUE: If the candidate simulates pulling the breaker control lever down, state "The lever moves and you hear a clunk from the breaker cubicle."

If the candidate simulates pushing the breaker control lever up, state "The lever does not move."

COMMENTS:

CRITICAL: No	***SEQUENCED:*** Yes		***SAT*** ☐	***UNSAT*** ☐

JPM STEP: 2

Step 5.j; Alternative Action; Locally deenergize and close the appropriate valve.

- For SG C:
  1) Open XMC1DB2Y 05EH, EF PUMP MAIN STEAM BLOCK VLV XVG2802B-MS (AB-463).

STEP STANDARD:

Determines that the AB operator has already opened XMC1DB2Y 05EH, EMERG FEEDWATER PUMP MAIN STEAM BLOCK XVG2802B-MS (AB-463).

CUES:

EVALUATOR NOTE: The initial conditions provided information indicating that the 1DB2Y breaker in the Aux Building had already been closed.

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 3

Step 5.j; Alternative Action; Locally deenergize and close the appropriate valve.

- For SG B:

  2) Close XVG02802A-MS, MS HEADER B EF PUMP TURBINE SUPPLY VLV (IB-436 East Pen).

STEP STANDARD:

Candidate locates XVG-2802A-MS and simulates manual closure by indicating that they would pull the manual engagement clutch lever down and then indicating they would turn the valve handwheel in the clockwise (CW) direction until the valve position arrow of the handwheel points to the right and the handwheel will no longer rotate.

CUES:

EVALUATOR CUE: If the Candidate simulates CW motion state: “The handwheel moves several turns then you feel resistance and the valve position arrow points to the right.”

If the Candidate initially simulates Counter Clockwise (CCW) motion state: “The handwheel does not move, and you feel resistance immediatly.”

If the Candidate subsequently simulates CW motion state: “The handwheel moves several turns then you feel resistance and the valve position arrow points to the right.”

If the Candidate does **not** indicate that they would operate the manual engagement lever **then** regardless of CW or CCW motion on the handwheel state: “The handwheel moves and you do NOT feel resistance”

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 4

Step 5.j; Alternative Action; Locally deenergize and close the appropriate valve.

- For SG C:

  2) Close XVG02802B-MS, MS HEADER C EF PUMP TURBINE SUPPLY VLV (IB-436 East Pen).

STEP STANDARD:

Candidate locates XVG-2802B-MS and simulates manual closure by indicating that they would pull the manual engagement clutch lever down and then indicating they would turn the valve handwheel in the clockwise (CW) direction until the valve position arrow of the handwheel points to the right and the handwheel will no longer rotate.

CUES:

EVALUATOR CUE: If the Candidate simulates CW motion state: "The handwheel moves several turns then you feel resistance and the valve position arrow points to the right."

If the Candidate initially simulates Counter Clockwise (CCW) motion state: "The handwheel does not move, and you feel resistance immediatly."

If the Candidate subsequently simulates CW motion state: "The handwheel moves several turns then you feel resistance and the valve position arrow points to the right."

If the Candidate does **not** indicate that they would operate the manual engagement lever **then** regardless of CW or CCW motion on the handwheel state: "The handwheel moves and you do NOT feel resistance"

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPP-010-N18, Locally De-Energize and Close MS Loop "B" and "C" to TDEFP (NRC JPM j)

IC SET: NA

INSTRUCTIONS: NA

COMMENTS:

CRITICAL TASK METHODOLOGY:

Step 1 is critical because the breaker must be open to assure that the valve motor cannot be operated to re-open the valve.

Step 3 is critical because the valve is normally open and must be closed to assure isolation of the faulted SG.

Step 3 is critical because the valve is normally open and must be closed to assure isolation of the faulted SG.

REVISION HISTORY:

This JPM is a minor revision of JPP-107A-N16, LOCALLY DE-ENERGIZE AND CLOSE MS LOOP "B" AND "C" TO TDEFP which was last used on the 2016 NRC exam for ILO-14-01 for safety function 4S.
SAR 1/2018.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: Thermal Burn Hazard.

INITIAL CONDITION:

- The plant was at 100% power when a steamline break occurred inside containment.
- "B" and "C" SGs are faulted as indicated by steam pressure dropping in an uncontrolled manner.
- Both motor driven emergency feedwater pumps have started.
- The CRS has implemented EOP-3.0, FAULTED STEAM GENERATOR ISOLATION, after exiting EOP-1.0, REACTOR TRIP/SAFETY INJECTION ACTUATION.
- The BOP has tried and failed to close MVG-2802A, MS LOOP B TO TD EFP and MVG-2802B, MS LOOP C TO TD EFP from the MCB.
- The Shift Manager has approved the waiving of ISP-027 requirements due to the emergency condition.
- The AB operator (Unit 6 upper) has already opened breaker 05EH on XMC1DB2Y in accordance with EOP-3.0 step 5 j Alternative Action.

INITIATING CUES:

The RO directs you as the IB operator (Unit 7) to complete the following actions in accordance with EOP-3.0 step 5 j Alternative Action:

- Open breaker 05EH on XMC1DA2X.
- Locally close valves XVG02802A-MS.
- Locally close valvesXVG02802B-MS.

AT NO TIME ARE YOU TO OPERATE ANY PLANT EQUIPMENT!

FOR ELECTRICAL MANIPULATIONS, AT NO TIME ARE YOU TO BREAK THE PLANE OF THE ELECTRICAL PANEL!

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: JPP-011-N18

Fill Spent Fuel Pool from Rx M/U Water

(NRC JPM k)

CANDIDATE: ____________________

EXAMINER: ____________________

TASK: 033-002-01-04 MAKEUP TO THE SPENT FUEL POOL PER SOP-123

TASK STANDARD:

Spent fuel pool level is increased to 461.8 ft using Reactor Makeup water and system is returned to standby; XVA-16729 is closed and locked with the Reactor Makeup Water pump back in service.

TERMINATING CUE:

"B" Reactor Makeup Water pump has been restarted.

PREFERRED EVALUATION LOCATION
PLANT

PREFERRED EVALUATION METHOD
SIMULATE

REFERENCES:

SOP-123, SPENT FUEL COOLING SYSTEM

INDEX NO	*K/A NO.*		*RO*	*SRO*
000033A1.01	A1.01	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including: Spent Fuel Pool Water Level	2.7	3.3

TOOLS: Hard copy of SOP-123, Section IV.Z, Raising the level in the Spent Fuel Pool Using the Reactor Makeup Water System.

JPM k Handout – OAP-106.3 Attachment II, Locked Component Operating Sheet.

EVALUATION TIME 10 ***TIME CRITICAL*** NO ***10CFR55:*** 45(a)6

TIME START: ______________ TIME FINISH: ______________ PERFORMANCE TIME: __________

PERFORMANCE RATING: SAT: _____ UNSAT: _____

CANDIDATE: ____

EXAMINER: ____ ____________________/________

SIGNATURE DATE

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: Radiological Control Area.

INITIAL CONDITION:

Plant is at 100% power.

AB Upper reports from the Fuel Handling building that Spent Fuel Pool level is 461.1ft.

The "B" Reactor Makeup water pump is in service.

Chemistry reports that the Reactor Makeup Water Tank contents are within specification for addition to the Spent Fuel Pool.

Spent Fuel Cooling Loop "B" is cooling the Spent Fuel Pool.

INITIATING CUES:

CRS directs you to raise level in the Spent Fuel Pool to 461.8 ft using the guidance found in SOP-123, Spent Fuel Cooling System, Section IV.Z, Raising the Level in the Spent Fuel Pool Using the Reactor Makeup Water System.

AT NO TIME ARE YOU TO OPERATE ANY PLANT EQUIPMENT!

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 1

Step 2.1; Minimize the DP across and open XVA16729-SF, RMU SYSTEM TO SF SYSTEM SUPPLY ISOLATION VALVE, as follows:

a. Secure any operating Reactor Makeup Water Pump.

STEP STANDARD:

Candidate contacts control room and requests that the “B” Reactor Makeup water pump be secured. Candidate may request ALL Reactor Makeup water pump be secured.

CUES:

EVALUATOR NOTE: Provide a hard copy of SOP-123 Section IV.Z and JPM k handout, OAP-106.3 Attachment II, once the Candidate has been briefed on the initiating cue.

EVALUATOR CUE: When handing the Candidate the the procedures state “The CRS would also provide you with a copy of the Locked Valve Program Key Number 27”

EVALUATOR CUE: When contacted as Control Room to secure the “B” Reactor Makeup Water pump, acknowledge the request and then report “All Reactor Makeup Water Pumps are OFF.”

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 2

Step 2.1.b; Close XVT06726-SF, REACTOR MU WTR SF SYS POOL SUPPLY VALVE (AB-388).

STEP STANDARD:

Candidate locates XVT-6726 and simulates closing it by indicating they would turn the handwheel in the Clockwise (CW) direction.

CUES:

EVALUATOR CUE: If the Candidate simulates CW motion state: “The Handwheel moves one half turn, the stem lowers and then you feel resistance.”

If the Candidate initially simulates Counter Clockwise (CCW) motion state: “The handwheel operates several turns, the stem rises then you feel resistance.”

If the Candidate subsequently simulates CW motion state: “The handwheel operates several turns, the stem lowers then you feel resistance.”

COMMENTS:

CRITICAL: Yes **SEQUENCED:** Yes **SAT** ☐ **UNSAT** ☐

JPM STEP: 3

Step 2.1.c; Unlock XVA16729-SF, RMU SYSTEM TO SF SYSTEM SUPPLY ISOLATION VALVE (AB-388).

STEP STANDARD:

Candidate locates XVA-16729 and simulates unlocking the valve by indicating they would place the LVP key number 27 into the lock and turn the key to open the lock.

CUES:

EVALUATOR NOTE: Valve XVA-16729 is approximately 7 feet above the floor. Candidate may require use of a ladder to positively identify and/or operate the valve.

EVALUATOR NOTE: The LVP key 27 would be furnished with the Locked Component Operating Sheet.

EVALUATOR CUE: Once the Candidate locates the valve and simulates unlocking it report: “The valve is unlocked.”

COMMENTS:

CRITICAL: Yes **SEQUENCED:** Yes **SAT** ☐ **UNSAT** ☐

JPM STEP: 4

Step 2.1.d; Open XVA16729-SF, RMU SYSTEM TO SF SYSTEM SUPPLY ISOLATION VALVE (AB-388).

STEP STANDARD:

Candidate locates XVA-16729 and simulates opening it by indicating they would turn the operating handle in the CCW (as viewed from above the valve) direction until parallel with the pipe.

CUES:

EVALUATOR CUE: If the Candidate simulates CCW motion after removing the lock state: “The handle operates until it is parallel with the piping, then you feel resistance.”

If the Candidate initially simulates CW motion or did not remove the lock state: “The handle does not operate and you feel resistance immediately.”

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 5

Step 2.2; Start at least one Reactor Makeup Water Pump.

STEP STANDARD:

Candidate contacts control room and requests that a Reactor Makeup water pump be started.

CUES:

EVALUATOR CUE: When contacted as Control Room to start a Reactor Makeup Water pump, acknowledge the request and then report "The "B" Reactor Makeup Water Pump is ON."

COMMENTS:

CRITICAL: Yes ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 6

Step 2.3; To prevent RMUW Pump runout, throttle XVT06726-SF, REACTOR MU WTR SF SYS POOL SUPPLY VALVE (AB-388), to less than or equal to ½ turn open.

STEP STANDARD:

Candidate locates XVT-6726 and simulates opening it by indicating they would turn the handwheel in the CCW direction one half turn.

CUES:

EVALUATOR CUE: If the Candidate simulates CCW motion state: "The handwheel operates one half turn."

If the Candidate initially simulates CW motion state: "The handwheel does not operate and you feel resistance immediately."

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 7

Step 2.4 Verify Spent Fuel Pool level increases.

STEP STANDARD:

Candidate contacts the Control Room or AB Upper to verify Spent Fuel Pool Level increase.

CUES:

EVALUATOR CUE: When contacted as Control Room or AB Upper to verifiy level rise, ackowledge the request.

If Candidate opened XVA-16729 and XVT-6726 and requested a Reactor Makeup Water pump start then report "Spent Fuel Pool Level is rising"

If candidate did not open either XVA-16729 or XVT-6726 or did not request a Reactor Makeup Water pump start then report "Spent Fuel Pool Level is NOT rising."

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 8

Step 2.5 When desired level has been reached, secure any operating Reactor Makeup Water Pump.

STEP STANDARD:

CUES:

EVALUATOR NOTE: This step is a control room function. The Control room will monitor the level increase and secure the Reactor Mkeup Water pump at the desired level.

EVALUATOR CUE: If the Candidate has lined up flow by opening the valves and requesting the pump start report "Spent Fuel Pool level is 461.8 ft. The "B" Reactor Makeup Water Pump is OFF. You may proceed with step 2.6 of SOP-123 Section IV.Z."

COMMENTS:

CRITICAL: No ***SEQUENCED:*** Yes ***SAT*** ☐ ***UNSAT*** ☐

JPM STEP: 9

Step 2.6; Close XVA16729-SF, RMU SYSTEM TO SF SYSTEM SUPPLY ISOLATION VALVE (AB-388).

STEP STANDARD:

Candidate locates XVA-16729 and simulates closing it by indicating they would turn the handle in the CW (as viewed from above the valve) direction until it is perpindicular to the piping.

CUES:

EVALUATOR CUE: If the Candidate simulates CW motion state: “The handwheel operates until it is perpindicular to the piping and then you feel resistance.”

If the Candidate initially simulates CCW motion state: “The handle does not operate and you feel resistance immediately.”

COMMENTS:

Examiner ends JPM at this point.

JPM SETUP SHEET

JPM: JPP-011-N18, Fill Spent Fuel Pool from Rx M/U Water (NRC JPM k)

IC SET: NA

INSTRUCTIONS:

Determine the date and time of performance for this JPM. Contact the ALARA group in advance of the scheduled date of performing this JPM. Inform the ALARA group that the JPM should not result in transit through radiation areas and all aspects of the task will be performed in low dose areas.

COMMENTS:

Survey Maps can be accessed from the Company Intranet by typing VSDS (Virtual Survey Display System) in the Explorer command line.

CRITICAL TASK METHODOLOGY:

Step 3 is critical because the lock would prevent opening of XVA-16729.

Step 4 is critical because without opening XVA-16729 no flow path to the Spent Fuel Pool is available for Reactor Makeup Water.

Step 5 is critical because without starting a Reactor Makeup Water Pump no flow to the Spent Fuel Pool is available in order to raise Spent Fuel Pool level.

Step 6 is critical because without opening XVT-6726 no flow path to the Spent Fuel Pool is available for Reactor Makeup Water.

REVISION HISTORY:

This is a new JPM that was created for the 2018 NRC Exam for ILO-16-01. It is assigned to Safety Function 8 for that exam.
SAR 1/2018.

JPM BRIEFING SHEET

OPERATOR INSTRUCTIONS:

SAFETY CONSIDERATIONS: Radiological Control Area.

INITIAL CONDITION:

Plant is at 100% power.

AB Upper reports from the Fuel Handling building that Spent Fuel Pool level is 461.1ft.

The “B” Reactor Makeup water pump is in service.

Chemistry reports that the Reactor Makeup Water Tank contents are within specification for addition to the Spent Fuel Pool.

Spent Fuel Cooling Loop “B” is cooling the Spent Fuel Pool.

INITIATING CUES:

CRS directs you to raise level in the Spent Fuel Pool to 461.8 ft using the guidance found in SOP-123, Spent Fuel Cooling System, Section IV.Z, Raising the Level in the Spent Fuel Pool Using the Reactor Makeup Water System.

AT NO TIME ARE YOU TO OPERATE ANY PLANT EQUIPMENT!

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-102-(R)N18**

2018 NRC A1-a (RO)

CANDIDATE: ______________________________

EXAMINER: ______________________________

DESCRIPTION: Verification of Operator Watchstanding Certification

TASK:

TASK STANDARD: Candidate determines that 24 hours must be stood by September 30th to maintain an active license.

TERMINATING CUE: The candidate enters answers on the handout provided and returns it to the examiner.

PREFERRED LOCATION: CLASSROOM

PREFERRED METHOD: PERFORM

REFERENCES:

OAP-110.2, OPERATOR WATCHSTANDING CERTIFICATION AND TRACKING

K/A 2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc. (RO 3.3)

10CFR55: 41 b(10)

TOOLS: Calculator
Access to paper or electronic copies of V.C. Summer procedures

EVALUATION TIME: 15 Minutes. ***TIME CRITICAL:*** NO

TIME START: ____________ ***TIME FINISH:*** ____________ ***PERFORMANCE TIME:*** ____________

CANDIDATE: ______________________________________

EXAMINER: ______________________________________

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- Today is **September 16**.
- Your REACTOR OPERATOR license was issued on **June 29** of this year.
- **After** June 29, you stood watches on the dates indicated on the handout that has been provided to you.
- You have been working a special assignment and it is anticipated that you will be permanently assigned to a shift on October 15.

INITIATING CUES:

You are directed to determine the minimum number of watch hours that you must stand to maintain your license continuously active through October 15th and that latest date by which the hours must be stood.

Enter your answers on the Handout.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL:	NO	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 1

Refers to OAP-110.2, OPERATOR WATCHSTANDING CERTIFICATION AND TRACKING.

STEP STANDARD:

Refers to either a hard-copy or electronic controlled copy of OAP-110.2.

CUES:

Note(s) to examiner:

The candidate should have the JPM Briefing Sheet and the Handout.

It is not critical for the candidate to refer to the procedure if the candidate can make the correct determination in step 4 from memory.

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 2

Determines the minimum watchstanding requirement for the quarter.

STEP STANDARD:

Determines that the minimum requirement is five 12-hour shifts by the end of the current quarter.

CUES:

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 3

Determines the hours previously stood that count toward the minimum requirement.

STEP STANDARD:

Determines that the AB UPPER watch on August 8 and the partial watches stood on August 20 and September 9 do **not** count toward the minimum requirement.

Determines that the three 12-hour shifts in an RO-licensed position (NROATC or BOP) stood on July 10, July 11 and September 15 do count toward the requirement.

CUES:

COMMENTS:

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 4

Determines the minimum hours that must be stood to maintain an active license.

STEP STANDARD:

Determines that 24 hours are left to be stood to maintain an active license.

60 hours - 36 hours (three 12-hour shifts) = 24 hours in two 12 hour shifts left to be stood to maintain the license. These watched must be stood by the last day in September.

CUES:

COMMENTS:

JPM SETUP SHEET

JPM: JPM: JPA-102-(R)N18

IC SET: N/A

INSTRUCTIONS:

Ensure that current procedures and curve book are available in hard copy or electronically.

Provide Handout containing the following:

Name: ______________________________________ Date: ________________________

Your Watchstanding History

NROATC	0730 - 1930	July 10
NROATC	0730 - 1930	July 11
BOP	0730 - 1030	August 1
AB UPPER	1930 - 0730	August 8
BOP	0730 - 1430	August 20
BOP	1230 – 1930	September 9
NROATC	1930 – 0730	September 15

1. What is the **minimum** number of hours of watch that you must stand to maintain your license continuously active through October 15?

2.When is the **latest** date in this year by which those hours must be stood?

COMMENTS:

JPM BRIEFING SHEET

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- Today is **September 16**.
- Your REACTOR OPERATOR license was issued on **June 29** of this year.
- **After** June 29, you stood watches on the dates indicated on the handout that has been provided to you.
- You have been working a special assignment and it is anticipated that you will be permanently assigned to a shift on October 15.

INITIATING CUES:

You are directed to determine the minimum number of watch hours that you must stand to maintain your license continuously active through October 15th and that latest date by which the hours must be stood.

Enter your answers on the Handout.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

2018 NRC A1-a RO Handout 1

Name: ______________________________________ Date: ________________________

Your Watchstanding History

NROATC	0730 - 1930	July 10
NROATC	0730 - 1930	July 11
BOP	0730 - 1030	August 20
AB UPPER	1930 - 0730	August 20
BOP	0730 - 1430	August 20
BOP	1230 – 1930	September 9
NROATC	1930 – 0730	September 15

1. What is the **minimum** number of hours of watch that you must stand to maintain your license continuously active through October 15?

2. When is the **latest** date in this year by which those hours must be stood?

******2018 NRC A1-a RO KEY******

Name: ______________________________________ Date: ________________________

Your Watchstanding History

NROATC	0730 - 1930	July 10
NROATC	0730 - 1930	July 11
BOP	0730 - 1030	August 1
AB UPPER	1930 - 0730	August 8
BOP	0730 - 1430	August 20
BOP	1230 – 1930	September 9
NROATC	1930 – 0730	September 15

1. What is the **minimum** number of hours of watch that you must stand to maintain your license continuously active through October 15?

Answer: ***24 HOURS***

Justification: ***OAP-110.2 section 4.5.b states that an active License holder must stand a minimum of five 12-hour shifts per calendar quarter in one of the licensed positions as applicable to the licensed operator. In accordance with attachment 1 of that procedure, Either RO or BOP meets the requirement for an RO License. The AB UPPER watch on August 8 and the partial watches stood on August 20 and September 9 do not count.***

Three 12-hour shifts in a RO-licensed position were stood on July 10, July 11 and September 15.

The minimum requirement is 5 x 12 = 60 hours per quarter ->

60 minimum – 36 stood = 24 hours.

2. When is the **latest** date in this year by which those hours must be stood?

Answer: ***September 30 is the final day for standing the watches. Acceptable answers are also the last day in September or the last day of the third quarter.***

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-81E-(R)N18**

2018 NRC A1-b (RO)

CANDIDATE: ____________________

EXAMINER: ____________________

DESCRIPTION: Operational Leak Rate Test without IPCS available

TASK: 02-002-02-01 PERFORM AN RCS LEAKRATE CALCULATION WITH THE IPCS UNAVAILABLE IAW STP-114.002

TASK STANDARD: Candidate correctly calculates the leak rate to within the specified tolerance contained in the answer key.

TERMINATING CUE: The candidate completes the STP-114.002, ATTACHMENT I contained in a handout and returns it and the briefing sheet to the examiner.

PREFERRED LOCATION:	***PREFERRED METHOD:***
CLASSROOM	PERFORM

REFERENCES:

STP-114.002 OPERATIONAL LEAKAGE CALCULATION.

STATION CURVE BOOK

K/A 2.1.20 Ability to interpret and execute procedure steps. (RO 4.6)

10CFR55: 45 b(1)(12)

TOOLS:

Calculator
Handout 1 containing STP-114.002, OPERATIONAL LEAKAGE CALCULATION
Handout 2 containing plant data
Access to paper or electronic copies of V.C. Summer procedures

EVALUATION TIME: 25 Minutes. ***TIME CRITICAL:*** NO

TIME START: ____________ ***TIME FINISH:*** ___________ ***PERFORMANCE TIME:*** __________

CANDIDATE: ____

EXAMINER: ____

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is in Mode 1.
- The IPCS Leak Rate Program unavailable.
- IPCS can be used to obtain plant data.

INITIATING CUES:

You are directed to perform the following:

USING PLANT DATA ON THE HANDOUT PROVIDED, perform STP-114.002, OPERATIONAL LEAKAGE CALCULATION starting at STEP 6.4.b and **complete through** STEP 6.4.j.

- You are to assume that another operator operates valves as required in STEP 6.4.c.
- No RCS makeup or diversion to the recycle holdup tanks will occur.
- No Chemical additions or primary samples will be performed.
- RCS pressure will be maintained at 2235 psig.
- Primary to secondary leakage is 0 gpm.
- There is no recorded leakage to atmosphere in the RB.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 1

Inputs data into STP-114.002. ATTACHMENT I, TEST DATA SHEET.

STEP STANDARD:

Student inputs data, initial and final.

CUES:

Note to examiner: The candidate should have the JPM Briefing Sheet, Handouts #1 (Includes a complete copy of STP-114.002, OPERATIONAL LEAKAGE CALCULATION) and Handout 2.

COMMENTS:

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 2

Calculates change in test data.

STEP STANDARD:

Student subtracts initial data from final to determine changes in the following parameters as indicated on answer key.

- TAVG
- PZR LEVEL
- VCT LEVEL
- PRT LEVEL
- RCDT LEVEL

CUES:

NOTE TO EXAMINER:

If the candidate requests operation of valves PVD-7170 and PVD-7136, refer the candidate to the briefing sheet.

It is not critical that the candidate follow the sequence to determine all changes as long as Attachment I is filled in correctly at the completion of the JPM.

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 3

Determine the RCS inventory change due to change in TAVG.

STEP STANDARD:

Change/deviation calculated accurately with correct polarity as shown on answer key.

CUES:

NOTE TO EXAMINER: It is not critical that the candidate follow the JPM sequence to determine all inventory deviations as long as Attachment I is filled in correctly at the completion of the JPM.

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 4

Determine the RCS inventory deviation due to Pressurizer level change.

STEP STANDARD:

Change/deviation calculated accurately with correct polarity as shown on answer key.

CUES:

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 5

Determine the RCS inventory deviation due to VCT level change.

STEP STANDARD:

Change/deviation calculated accurately as shown on answer key.

CUES:

COMMENTS:

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 6

Determine the RCS inventory change due to change in TAVG.

STEP STANDARD:

Change/deviation calculated accurately with correct polarity as shown on answer key.

CUES:

COMMENTS:

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 7

Determine the RCS inventory deviation due to PRT level change.

STEP STANDARD:

Recorded as zero.

CUES:

COMMENTS:

CRITICAL: YES	SEQUENCED: YES	SAT		UNSAT	

JPM STEP 8

Determine the RCS inventory deviation due to RCDT level change.

STEP STANDARD:

Change/deviation calculated accurately with correct polarity as shown on answer key.

CUES:

COMMENTS:

CRITICAL:	YES	SEQUENCED:	YES	SAT		UNSAT	

JPM STEP 9

Perform calculations and enters results for final leakrates as indicated on Attachment 1, Part 2.

STEP STANDARD:

The final leakrates are required to be within the tolerances indicated on the answer key.

CUES:

COMMENTS:

JPM SETUP SHEET

JPM: JPM: JPA-81E-(R)N18

IC SET: N/A

INSTRUCTIONS:

Ensure that current procedures and curve book are available in hard copy or electronically.

Provide Handout 1 - STP-114.002, Attachment I

Provide Handout 2 - sheet Containing the following:

Name: ______________________________ Date: ____________________

INITIAL DATA Start time 0800:
T0499A, RCL MEDIAN TAVG = 586.2°F
L0480A, PRESSURIZER LEVEL-LT459 = 60.5%
L0112A, VOLUME CONTROL TANK LEVEL-LT-115 = 38.0%
L0485A, PRESSURIZER RELIEF TANK L-LT470 = 72.5%
L1028, REACTOR COOL DR TNK LEV = 4%

FINAL DATA Stop time 0915:
T0499A, RCL MEDIAN TAVG = 586.8°F
L0480A, PRESSURIZER LEVEL-LT459 = 60.0%
L0112A, VOLUME CONTROL TANK LEVEL-LT-115 = 22.4%
L0485A, PRESSURIZER RELIEF TANK L-LT470 = 72.5%
L1028, REACTOR COOL DR TNK LEV = 55%

COMMENTS:

JPM BRIEFING SHEET

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is in Mode 1.
- The IPCS Leak Rate Program unavailable.
- IPCS can be used to obtain plant data.

INITIATING CUES:

You are directed to perform the following:

USING PLANT DATA ON THE HANDOUT PROVIDED, perform STP-114.002, OPERATIONAL LEAKAGE CALCULATION starting at STEP 6.4.b and **complete through** STEP 6.4.j.

- You are to assume that another operator operates valves as required in STEP 6.4.c.
- No RCS makeup or diversion to the recycle holdup tanks will occur.
- No Chemical additions or primary samples will be performed.
- RCS pressure will be maintained at 2235 psig.
- Primary to secondary leakage is 0 gpm.
- There is no recorded leakage to atmosphere in the RB.

Hand this paper back to your Evaluator when_you feel that you have satisfactorily completed the assigned task.

2018 NRC A1-b   RO Handout 2

Name: _______________________________________   Date: ______________________

INITIAL DATA

Start time 0800:

T0499A, RCL MEDIAN TAVG = 586.2°F

L0480A, PRESSURIZER LEVEL-LT459 = 60.5%

L0112A, VOLUME CONTROL TANK LEVEL-LT-115 = 38.0%

L0485A, PRESSURIZER RELIEF TANK L-LT470 = 72.5%

L1028, REACTOR COOL DR TNK LEV = 4%

FINAL DATA

Stop time 0915:

T0499A, RCL MEDIAN TAVG = 586.8°F

L0480A, PRESSURIZER LEVEL-LT459 = 60.0%

L0112A, VOLUME CONTROL TANK LEVEL-LT-115 = 22.4%

L0485A, PRESSURIZER RELIEF TANK L-LT470 = 72.5%

L1028, REACTOR COOL DR TNK LEV = 55%

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

SURVEILLANCE TEST PROCEDURE

STP-114.002

OPERATIONAL LEAKAGE CALCULATION

REVISION 12

SAFETY RELATED

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	07/01/09		E	P	09/29/11	
B	P	06/17/10		F	P	10/17/12	
C	P	05/29/11		G	P	10/28/15	
D	P	09/09/11					

REFERENCE USE

Procedure Segments May be Performed From Memory.
Must Verify Work Following Each Segment.

TABLE OF CONTENTS

1.0 PURPOSE/SCOPE

1.1 The purpose of this procedure is to perform a Reactor Coolant System Water inventory balance per Technical Specification Surveillance Requirement 4.4.6.2.1.d to calculate RCS Leakage rate.

1.2 The scope of this procedure is governed by Technical Specifications 3.4.6.2, 10CFR50.65a(4), and 10CFR50, Appendix B. A 10CFR50.59 and/or 10CFR72.48 Review is not required for this procedure. FSAR Section 5.2.7 describes the various RCS Pressure Boundary leakage limits and methods for leakage detection but does not describe the method for an RCS leakage calculation. CHG G

> NOTE 2.0 through 8.0
>
> An asterisk (*) preceding a step or section indicates that data or a signoff is required on the attachment identified within that step or section.

* 2.0 PRECAUTIONS

2.1 Operation of the Reactor Coolant Makeup System or diverting of primary water to the Recycle Holdup Tanks during the performance of this test will invalidate test results.

2.2 Chemical additions or primary samples should not be performed for the duration of this test.

2.3 In order to ensure an accurate leakage rate calculation, it may be necessary to plot the level change in the associated tanks for an extended time and then average the level change to obtain accurate results. These plots should be attached to the required Attachments.

2.4 If a “CALL PSE” flag is set on the LRATE program monitoring tool, notify PSE within 12 hours and refer to ES-161, RCS LEAKAGE MANAGEMENT PROGRAM, Attachment III for the appropriate Action Level.

2.5 Changes in RCS temperature in excess of 1°F should be avoided.

3.0 TEST EQUIPMENT

3.1 None.

4.0 TEST FREQUENCY

4.1 At least once per 72 hours when the Plant is in Mode 1, 2, 3, or 4.

* 5.0 INITIAL CONDITIONS

☐ 5.1 RCS pressure is being maintained between 350 psig and 400 psig or between 2220 psig and 2250 psig.

☐ 5.2 Reactor Power is being maintained within a 1% band.

☐ 5.3 VCT level less than 65%

NOTE 6.0
1) Optimum Test Data is obtained if the test period is extended for several hours.
2) The recommended test period is two hours.
3) During EOL conditions, where Tave droop is a concern, the recommended test period is one hour.
4) During transient conditions, for example startup, shutdown, or during times of increased leakage, a test period of less than one hour is acceptable.

CHG C

6.0 PROCEDURE

☐ *6.1 Perform an AS FOUND valve lineup per Attachment II.

6.2 Close the following vaves:

☐ a. PVD-7136, RCDT TO WPS.

☐ b. PVD-7170, RCDT PP TO WPS (IRB).

CHG E

<u>NOTE 6.3 and 6.4</u>

Either Step 6.3 or Step 6.4 should be performed:

a. Step 6.3, which automatically calculates the RCS Leakrate using the IPCS Leakrate Program, is the preferred method while in Mode 1 through Mode 3.

b. If the IPCS Leakrate Program is not available or the plant is in Mode 4, the RCS leakrate must be manually calculated per Step 6.4.

6.3 If the IPCS Leakrate Program is available, perform an RCS leakrate as follows:

☐ a. At an IPCS terminal, activate LRATE.

☐ b. Using the F1 function onscreen prompt, enter the latest primary to secondary leakage from the Chemistry Department per CHP-307, Primary to Secondary Leakage Rate Determination.

☐ c. If it is desired to change the duration of the leak rate calculation to a length other than the default 120 minutes, use the onscreen prompts as necessary to enter the desired duration from 15 minutes to 240 minutes, otherwise proceed to Step 6.3.d.

Step 6.3 Continued

CAUTION 6.3 d

1) If any of the following messages are displayed during the leakrate and valve position cannot be verified by other means, the data should be considered invalid, a Test Deficiency written and a new leakrate performed:

   a) DETECTED BAD QUAL OR NOT CLOSED FOR PVD7170.
   b) DETECTED BAD QUAL OR NOT CLOSED FOR PVG7136.

2) If any of the following messages are displayed during the leakrate, the data should be considered invalid, a Test Deficiency written and a new leakrate performed:

   a) DETECTED BAD QUAL OR > +/- 1 DEGF CHG FOR TAVG.
   b) DETECTED BAD QUAL OR > 2220-2250 PSIG FOR RCS PRESS.
   c) DETECTED BAD QUAL OR > +/- 1 PCT CHG FOR RX POWER.
   d) DETECTED BAD QUAL OR POSITIVE RMW FLOW.
   e) DETECTED BAD QUAL OR POSITIVE BORIC ACID FLOW.
   f) DETECTED PLANT MODE OTHER THAN 1-3 (ALL CALCS ARE INVALID).

CHG
B

NOTE 6.3.d

The IPCS Leakrate Program automatically stops after the selected duration has elapsed. Depressing the F2 onscreen prompt again anytime after the program has started will stop the leakrate determination.

☐ d. To start the leakrate, depress the F2 onscreen prompt.

Step 6.3 Continued

e. Upon completion of the test period, perform the following as applicable:

☐ 1) Observe the data for the "Least-Squares Leakrate Calculation". If all data is acceptable (i.e., all fields green with "GOOD" or "GOO*" quality), obtain a printout of this display.

☐ 2) If the "Least-Squares Leakrate Calculation" display is not acceptable due to any field being red ("UNK" will be displayed for the applicable leakrate value), then depress the TOGGLE DISPLAY onscreen prompt for the "Point-To-Point Leakrate Calculation" Screen and obtain a printout of this display.

NOTE 6.3.f

Although negative values for the RCDT and PRT may appear on the Leak Rate print out, the IPCS Leakrate Program automatically changes these values to zero during the calculation.

☐ f. Record the STTS Number and the procedure number on the IPCS printout and attach to the STTS.

☐ g. Record the appropriate data from the IPCS printout on the leakage trend spreadsheets using the guidelines listed on Enclosure A.

☐ h. If Attachment III was used to quantify identified RCS to atmospheric leakage, the adjusted IDENTIFIED and UNIDENTIFIED LEAKAGE value from Attachment III should be recorded on the leakage trend spreadsheets. CHG E

Step 6.3 Continued

☐ i. Perform one of the following to restore the RCDT valve lineup:

1) If RCDT level is less than 60%, open the following valves:

☐ a) PVD-7170, RCDT PP TO WPS (IRB).

☐ b) PVD-7136, RCDT TO WPS.

2) If RCDT level is greater than 60%, perform the following:

☐ a) Place LCV01003-WL, REACTOR COOLANT DRAIN TANK LEVEL CONTROLLER, in Manual (AB-412).

☐ b) Close LCV01003-WL, REACTOR COOLANT DRAIN TANK LEVEL CONTROLLER, by taking valve position to 0%. (AB-412). CHG F

☐ c) Open PVD-7170, RCDT PP TO WPS (IRB).

☐ d) Open PVD-7136, RCDT TO WPS.

☐ e) Adjust RCDT level to between 58% and 60% using LCV01003-WL, REACTOR COOLANT DRAIN TANK LEVEL CONTROLLER (AB-412). CHG F

☐ f) Place LCV01003-WL, REACTOR COOLANT DRAIN TANK LEVEL CONTROLLER, in AUTO (AB-412).

☐ * j. Perform a RETURN AS FOUND valve lineup per Attachment II.

☐ * k. Perform a RETURN AS FOUND valve lineup independent verification per Attachment II.

NOTE 6.4

When RCS pressure is between 350 psig and 400 psig, computer point U0091, AVERAGE T/C TEMPERATURE, or TR-413, HOT LEG °F WIDE RNG, should be used, as applicable.

6.4 If the IPCS Leakrate Program is not available or the plant is in Mode 4, perform the following:

a. Perform **one** of the following as applicable:

* 1) If the IPCS is available for data gathering, use Group Trend S114.002 or GRPDIS S114.002 to record the following information in Part 1of Attachment I:

		POINT ID	**DESCRIPTION**
☐	a)	N/A	Test Start Time
☐	b)	T0499A	RCL MEDIAN TAVG
		OR	
☐		U0091	AVERAGE T/C TEMPERATURE
☐	c)	L0480A	PRESSURIZER LEVEL-LT459
☐	d)	L0112A	VOLUME CONTROL TANK LEVEL-LT115
☐	e)	L0485A	PRESSURIZER RELIEF TANK L-LT470
☐	f)	L1028	REACTOR COOL DR TNK LEV

Step 6.4.a continued

* 2) If the IPCS is not available in any capacity, record the following parameters in the appropriate column in Part 1 of Attachment I along with the instrument number where applicable:

☐ a) LOOP A(B)(C) T-AVG, TI-412D(422D)(432D).

☐ b) PZR LEVEL %, LI-459A(460)(461).

☐ c) VCT LEVEL %, LI-115(112A).

☐ d) PRT LEVEL %, LI-470.

☐ e) ILI01003, RC DRAIN TANK LEVEL INDICATOR WL.

☐ f) Test Start Time.

☐ * b. Upon completion of the test period, record the monitored parameters from Step 6.4.a.1) or Step 6.4.a.2), as applicable, as well as the test stop time in the appropriate column in Part 1 of Attachment I.

c. Perform **one** of the following:

1) If RCDT level is less than 60%, open the following:

☐ a) PVD-7170, RCDT PP TO WPS (IRB).

☐ b) PVD-7136, RCDT TO WPS.

2) If RCDT level is greater than 60%, perform the following:

☐ a) Place LCV01003-WL, REACTOR COOLANT DRAIN TANK LEVEL CONTROLLER, in Manual/Closed (AB-412).

☐ b) Open PVD-7170, RCDT PP TO WPS (IRB).

☐ c) Open PVD-7136, RCDT TO WPS.

☐ d) Adjust RCDT level to between 58% and 60% using LCV01003-WL, REACTOR COOLANT DRAIN TANK LEVEL CONTROLLER, then place it in AUTO (AB-412).

Step 6.4 Continued

NOTE 6.4.d through 6.4.i
Steps 6.4.d through 6.4.i may be performed in any sequence.

NOTE 6.4.d
A net T_{avg} decrease over the performance of this test will result in negative gallons, while a net T_{avg} increase will result in positive gallons.

* d. To determine the RCS inventory deviation due to changes in T_{avg}, subtract the Initial T_{avg} data from the Final T_{avg} data (Attachment I, Part 1) and perform the following calculation;

☐ 1) Total T_{avg} change: ± 0.6 °F (Attachment I, Part 1).

☐ 2) Determine the RCS density factor by referring to VCS Station Curve Book Figure V-6 if RCS pressure is between 350 psig and 400 psig, or Figure V-7 if RCS pressure is between 2220 psig and 2250 psig: 81 (81-83.5) gallons/°F. see attached

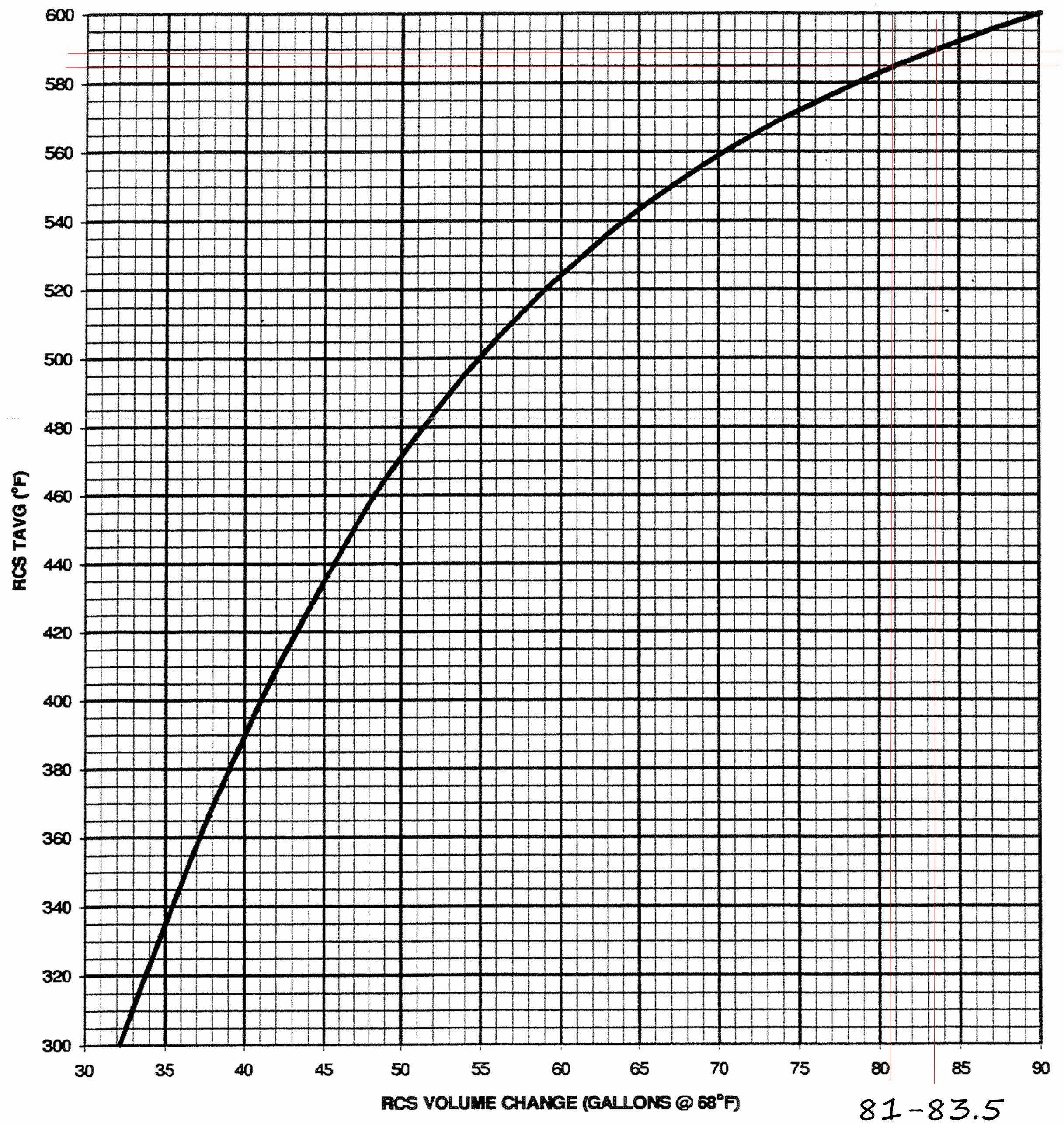
☐ 3) Multiply:

± 0.6 (6.4.d.1) °F x (81-83.5) 81 (6.4.d.2) gallons/°F =

± 48.6 (48.6 - 50.1) gallons (Attachment I, Part 2a).

Figure V-7
Revision Date: 1-16-95
Prepared By: LR Carter
Verified By: [signature]
Approved By: Will Haltiwanger

RCS VOLUME CHANGE FOR 1°F CHANGE IN TAVG @ 2250 PSIG



Step 6.4 Continued

NOTE 6.4.e
A net Pressurizer level decrease over the performance of this test will result in positive gallons, while a net Pressurizer level increase will result in negative gallons.

* e. To determine the RCS inventory deviation due to Pressurizer level change, subtract the Initial PZR Level data from the Final PZR Level data (Attachment I, Part 1) and perform the following calculation:

☐ 1) Total PZR Level change: ± -0.5% (Attachment I, Part 1).

2) Determine the density factor utilizing **one** of the following as applicable:

☐ a) If PZR pressure is between 350 psig and 400 psig, use -87.6 gal/%.

☐ b) If PZR pressure is between 2220 psig and 2250 psig, use -56.57 gal/%.

☐ 3) Multiply:

± -0.5 % x -56.57 gal/% =
(6.4.e.1)) (6.4.e.2)a) **or** (6.4.e.2)b)

± 28.285 gallons (Attachment I, Part 2a).
(28-29)

*f. To determine the RCS inventory deviation due to VCT level change, subtract the Initial VCT Level data from the Final VCT Level data (Attachment I, Part 1) and perform the following calculation:

☐ 1) Total VCT level change: ± -15.6 %
(Attachment I, Part 1).

☐ 2) Multiply:

± -15.6 % x (-) 14.00 gal/% =
(6.4.f.1)

± 218.4 gallons (Attachment I, Part 2a).
(218.4 - 224)

Step 6.4 continued

NOTE 6.4.g and h.

1. When the PRT or RCDT level changes indicate a negative value, the Total Deviation value should be recorded as zero on Attachment I.

2. Changes in RB temperature may affect PRT reference leg, which may impact the leak rate results. If results are suspect, a second leakrate should be performed, but the first must be documented unless invalid (see Caution 6.3.d and Initial Conditions).

3. If PRT is to be vented during the calculations, contact Operations Management for evaluation. CHG C

☐ * g. To determine the RCS inventory deviation due to PRT level change, subtract the Initial PRT Level data from the Final PRT Level data (Attachment I, Part 1; Interpolate as necessary using the Table of PERCENT vs. GALLON of VCS Station Curve Book Figure VI-21).

☐ * h. To determine the RCS inventory deviation due to RCDT level change, subtract the Initial RCDT Level data from the Final RCDT Level data (Attachment I, Part 1; Interpolate as necessary using the Table of % LEVEL vs. GALLONS of VCS Station Curve Book Figure VI-22). see attached

☐ * i. Obtain the latest primary to secondary leakage from the Chemistry Department per CHP-307, Primary To Secondary Leakage Rate Determination (Attachment I, Part 2b).

☐ * j. Perform calculations as indicated on Attachment I, Part 2.

☐ * k. Record the appropriate data from Attachment I, Part 2 on the leakage trend spreadsheets using the guidelines listed on Enclosure A.

☐ l. If Attachment III was used to quantify identified RCS to atmospheric leakage, the adjusted IDENTIFIED and UNIDENTIFIED LEAKAGE value from Attachment III should be recorded on the leakage trend spreadsheets. CHG E

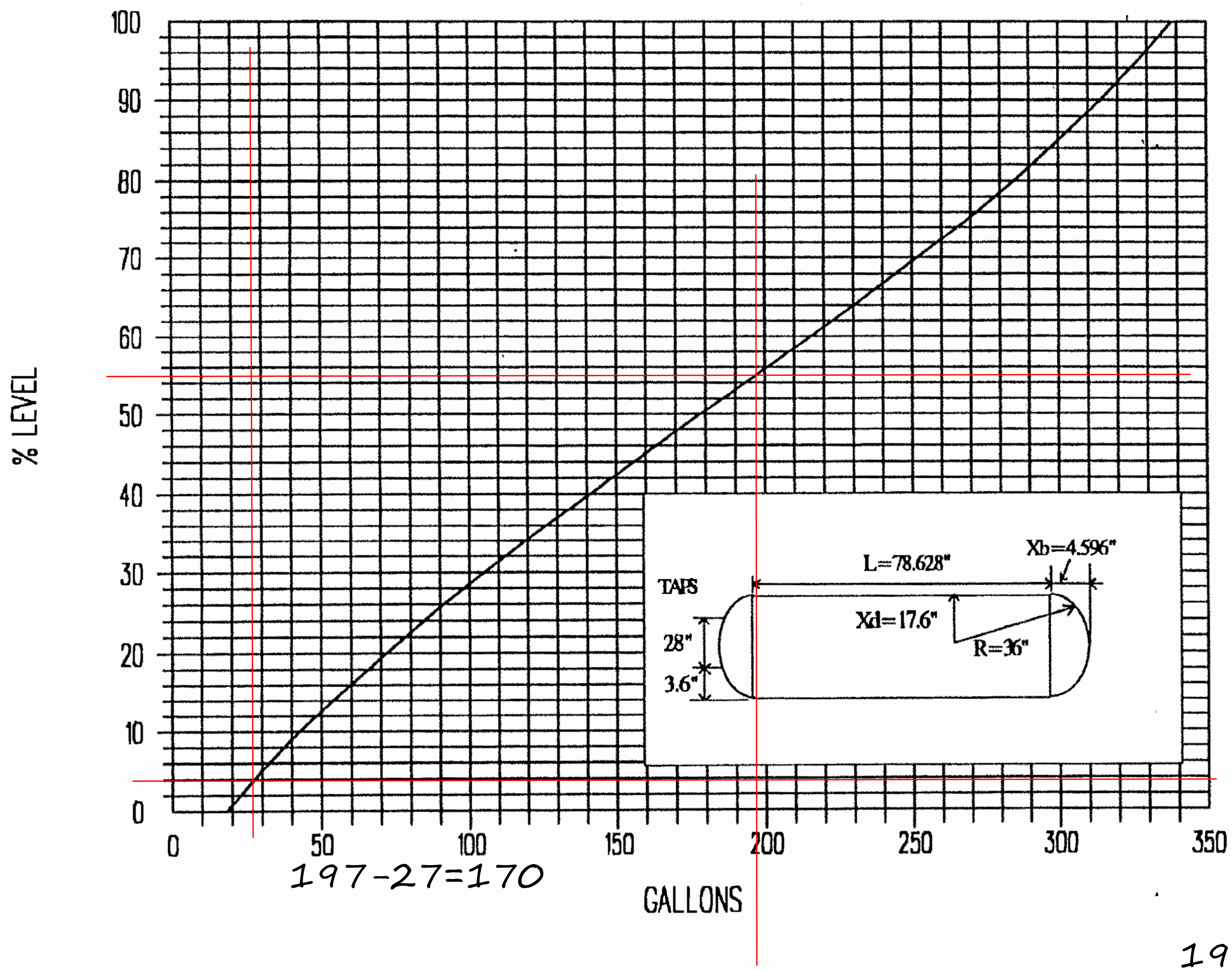
☐ * m. Perform a RETURN AS FOUND valve lineup per Attachment II.

☐ *n. Perform a RETURN AS FOUND valve lineup independent verification per Attachment II.

Figure VI-22
RevisionDate: 4-21-92
Prepared By: [signature]
Verified By: [signature]
Approved By: [signature]

29.5(5%) - 2.44gal/%=27.06
Interpolation -> 27.22

REACTOR COOLANT DRAIN TANK



% LEVEL	GALLONS	GALLONS/%
0	18.1	2.09
5	29.5	2.44
10	42.4	2.71
15	56.5	2.94
20	71.7	3.13
25	87.9	3.34
30	105	3.5
35	122.8	3.61
40	141	3.69
45	159.6	3.73
50	178.3	3.75
55	197	3.73
60	215.6	3.69
65	233.8	3.61
70	251.6	3.5
75	268.7	3.34
80	284.9	3.13
85	300.1	2.94
90	314.2	2.71
95	327.1	2.44
100	338.5	2.09

27-28

197-27.22
=169.78
(169-170)

197 x 51/55=182

Tech Spec Ref: N/A

Procedure Ref: N/A

Figure Ref: TWR,WT WOOD,"RCDT VOL",TAB 92-3,4/8/92, SERIAL 251394263

7.0 DATA REQUIREMENTS

7.1 IPCS printout and Attachment III (if used) will be attached to Attachment II, as applicable. CHG E

7.2 Required data will be entered on the applicable attachments.

> NOTE 7.3
>
> A negative value for identified and/or unidentified leakage does not in itself result in an Invalid or Suspect test and does not require re-performance of the test.

CHG C

7.3 If a test violates the test requirements of Caution 6.3.d, it is invalid and a Test Deficiency must be written and a new test performed. The invalid test results do not need to be documented in the Current Cycle Trends spreadsheet.

* 8.0 ACCEPTANCE CRITERIA

8.1 RCS leakage shall be limited to criteria specified in Technical Specification 3.4.6.2.

8.2 RCS leakage in excess of the limits established in Technical Specification 3.4.6.2 requires implementation of the associated Technical Specification Action Statement.

9.0 REFERENCES

9.1 CHP-307, Primary to Secondary Leakage Rate Determination.

9.2 FSAR Section 5.2.7.

9.3 PTP-175.001, RCS Supplementary Leakage Assessment.

9.4 V.C. Summer Station Curve Book.

9.5 V.C. Summer Technical Specification 3.4.6.2.

9.6 ES-161, RCS Leakage Management Program

RCS LEAKAGE TREND DATA GUIDELINES

NOTE

The information given here is to be used as a guideline and is not to be considered the only correct method of documenting the necessary data.

1. Each time a valid STP114.002 is performed, no matter how many are done in a day, the calculated data from Attachment II is to be entered on the RCS Lkge, STP114.002 tab of the Current Cycle Trends spreadsheet.

    a. Access the Current Cycle Trends spreadsheet by following the path Operations website, VCS Links, Operations Trending Folder, Cycle Trends History folder.

    b. Add new rows to the RCS Lkge, STP114.002 tab of the Current Cycle Trends spreadsheet as required.

    c. Enter each data point from Attachment II into the appropriate spreadsheet cell.

    d. If a test cannot be performed, leave the cells blank, and make a notation of why in the comments column.

    e. Examples of additional information that could be placed in the "Comments" column are;

        1) Method of determination (Manual, Least squares, Point to Point).

        2) Duration of the test, if it is not the normal 2 hours.

2. On the Unidentified Leakage tab of the Current Cycle Trends spreadsheet, update the AVERAGE unidentified leakage using data from each valid test performed on this date. There will only be one entry on this spreadsheet per calendar day and that entry must be updated after each completed test. Never add another date row.

RCS LEAKAGE TREND DATA GUIDELINES (Cont'd)

3. Review the action level columns (presently D through M), if any of them say "Call PSE" then PSE must be notified within the next 12 hours (RCS System Engineer during normal working hours, if possible; otherwise the Duty ES Supervisor).

   a. Standard Action Level Criteria requiring Action by PSE per ES161 are:

      1) The Unidentified Leak Rate Action Levels in gpm are:

         a) One seven (7) day rolling average of daily Unidentified RCS leak rates > 0.1 gpm.

         b) Two consecutive daily Unidentified RCS leak rates > 0.15 gpm.

         c) One daily Unidentified RCS leak rate > 0.3 gpm.

         d) The Action Level of 0.1 gpm is 1/10th of the current TS Limit for unidentified leakage.

      2) Deviation from the baseline mean in gpm:

         a) Nine (9) consecutive daily Unidentified RCS leak rates > baseline mean [μ].

         b) Two (2) of three (3) consecutive daily Unidentified RCS leak rates > [μ + 2σ].

         c) One (1) daily Unidentified RCS leak rate > [μ +3σ].

      3) Total integrated unidentified Leakage in gallons:

         a) Short Term (30 Day) Total Integrated Unidentified Leakage > 5,000 gallons.

         b) Long Term (Operating Cycle) Total Integrated Unidentified RCS Leakage > 50,000 gallons.

   b. Additionally a "suspect" value is one which is more than 2.5 times the standard deviation from the mean. A suspect test must be fully documented, entered in the Current Cycle Trends spreadsheet, a Test Deficiency written, new test performed, and then generate a CR to document the deficiency. Only one retest is required per surveillance. CHG G

TEST DATA SHEET

PART 1

	TIME	TAVG		PZR LEVEL		VCT LEVEL		PRT LEVEL		RCDT LEVEL	
		MCB TI-___	COMPUTER T0499A/ U0091	MCB LI-___	COMPUTER L0480A	MCB LI-___	COMPUTER L0112A	MCB LI-470	COMPUTER L0485A	XPN-0007 ILI01003	COMPUTER L1028
FINAL	0915		586.8°F		60.0%		22.4%		72.5%		55%
INITIAL	0800		586.2°F		60.5%		38.0%		72.5%		4%
CHANGE	75 min	*	* 0.6°F	**	** -0.5%		-15.6%		0.0%		51%

CHG C

PART 2

() (48.6-50.1) (28-29) (218.4-224) (295-303.1)

2a: 48.6 gallons + 28.285 gallons + 218.4 gallons = 295.285 gallons ÷ 75 minutes
(6.4.d.3), Tavg) (6.4.e.3), PZR Level) (6.4.f.2), VCT Level) (Test Time)

= 3.937 gallons/minute TOTAL LEAKAGE
(3.93-4.04)

(165-185) (165-185)

2b: (1) 0 gallons + (1) 169.78 gallons = 169.78 gallons ÷ 75 minutes
(6.4.g), PRT Level) (6.4.h), RCDT Level) (Test Time)

= 2.263 gallons/minute + (1) 0 gallons/minute = 2.263 gallons/minute IDENTIFIED LEAKAGE
(2.2-2.46) (6.4.i, Primary to Secondary leakage) (2.2-2.46)

2c: 3.937 gallons/minute - 2.263 gallons/minute = 1.674 gallons/minute UNIDENTIFIED LEAKAGE
(2a, Total Leakage) (2b, Identified Leakage)
(3.93-4.04) (2.2-2.46)

(1.47-1.84) <-CRITICAL RANGE FOR CALCULATION

* Tavg decrease = negative gal./Tavg increase = positive gal. ** Pzr Level decrease = positive gal./Pzr Level increase = negative gal.

CHG C

(1) Record value as zero for negative changes.

TEST SIGNOFF/VALVE LINEUP

TEST SIGNOFF

2.0 PRECAUTIONS REVIEWED: ________ / _________
INITIAL / DATE

5.0 INITIAL CONDITIONS MET: ________ / _________
INITIAL / DATE

NOTE 7.0
A negative value for identified and/or unidentified leakage does not in itself result in an Invalid or Suspect test and does not require re-performance of the test.

CHG C

7.0 DATA REQUIREMENTS MET: ________ / _________
INITIAL / DATE

8.0 ACCEPTANCE CRITERIA MET: ________ / _________
INITIAL / DATE

(Check Yes/No as applicable)

RCS Leakage Trend Spreadsheets updated @ ________________
(time)

”Call PSE” Flag set? Yes___ No___
If Yes, contact PSE within 12 hours.
Person notified: _________________________ @ _____________
(time)

Suspect Flag set? Yes___ No___
If Yes, perform one additional Leak Rate per surveillance.
Completed @ __________
(time)

VALVE LINEUP

COMPONENT	COMPONENT DESCRIPTION	AS FOUND	REQUIRED OPERABLE POSITION	RESTORERS INITIALS	VERIFIERS INITIALS
PVD-7136	RCDT TO WPS		AUTO/ OPEN		
PVD-7170	RCDT PP TO WPS (IRB)		AUTO/ OPEN		
412 AUXILIARY BUILDING					
LCV01003-WL	REACTOR COOLANT DRAIN TK LEVEL CONT VLV		AUTO		

CHG A

CHG F

COMMENTS:____

____

____

____

RCS TO ATMOSPHERIC LEAKAGE

NOTE

1. Use of this option should require an Engineering Evaluation to verify that the Technical Specification definition of the leakage to be transitioned to IDENTIFIED LEAKAGE is met.

2. General Regulatory Requirements associated with RCS Leakage:

    a. Atmospheric leakage is inside containment.

    b. Leak rate has been actually quantified and identified.

    c. An Engineering evaluation has been performed and has confirmed that the leakage does not “interfere with the operation of leakage detection systems.

    d. The known source has been confirmed as not being pressure boundary leakage.

3. Both Identified and Unidentified leak rates are impacted by this information.

4. Identified Leakage must go up as much as Unidentified Leakage goes down.

5. Leakage must be measured each time the STP is performed.

6. RB Sump inleakage may NOT be used to measure leaks for the purposes of this test.

RCS TO ATMOSPHERIC LEAKAGE(Continued)

1 Record the UNIDENTIFIED LEAKAGE value from Attachment I or Attachment II: __________

2 Record the measured RCS to atmospheric leakage value (use date and time leakage was measured): __________gpm
Location(s) of RCS TO ATMOSPHERIC LEAKAGE

______/______
DATE / TIME

LOCATION	LEAKAGE	DATE/TIME
		/
		/
		/
		/
		/

3 Subtract the measured RCS to atmospheric leakage value from the UNIDENTIFIED LEAKAGE value: __________

4 Record this new UNIDENTIFIED LEAKAGE value on the leakage trend spreadsheets:

______/______
DATE / TIME

5 Record IDENTIFIED LEAKAGE from Attachment I or II: __________

6 Add the measured RCS to atmospheric leakage value to the IDENTIFIED LEAKAGE from Attachment I or II: __________

7 Record this new IDENTIFIED LEAKAGE value on the leakage trend spreadsheets:

______/______
DATE / TIME

Document Review Form (DRF)

Section I	Document Identification	Page 1 of 2

Preparer Name: Anthony Williams Ext: 54315 Mail Code P40

Date: 10/19/15 Document #: STP-114.002 Revision: 12 Change G

Title: OPERATIONAL LEAKAGE CALCULATION ☒ SR ☐ QR ☐ NNS

Development Process: ☐ New ☒ Revision/Change ☐ Editorial Correction ☐ Temporary Approval

Description: See attched.

ISFSI Related? ☐ Yes ☒ No
Has scope changed? ☒ Yes ☐ No [If YES, attach 50.59 and/or 72.48 documentation]

Reason/Basis for Revision/Change: See attached.

Temporary Approval – if final approval is not completed within 30 days; initiate CR #________

Qualified Reviewer DCRM person notified Shift Manager Date

Section II List Required Reviewers including All Impacted Groups
Additional Reviewers – identify with an *

Position	Type/Print Name	Comments	Position	Type/Print Name	Comments
QR	DANNY RHYMER	☐ Yes ☒ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No

A

[signature] 10/20/15
Designated Supervisor Concurrence Date

Comment Due Date 11/3/15
GM concurrence______for expedited review

Section III Pre- implementation Actions

All Comments Resolved? ☒ NA ☐ YES [signature] 10/22/15
Preparer Sign Date

50.59 and/or 72.48 Review Requirements Addressed?	☐ NA ☑ YES	Attached? YES ☑ No ☐	
50.59/Part 52 Review Requirements Addressed?	☑ NA ☐ YES	Attached? YES ☐ No ☐	
Commitments (PCAP and MLSA) Addressed?	☑ NA ☐ YES	PCAP # ____	____ NL Initial/Date
QR Qualification Verified?	☑ YES		
Security Compliance Review Completed?	☑ NA ☐ YES		
Pre-Implementation Training Completed?	☑ NA ☐ YES		
Training required after implementation?	☑ NA ☐ YES	CR#____	
PSRC Review Completed?	☑ NA ☐ YES	Mtg. #____	
NSRC Review Completed?	☑ NA ☐ YES	Mtg. #____	
CMMS Update Required? [Unit 1]	☑ NA ☐ YES	Planner Notified YES ☐____	

[signature] 10/24/15
Designated Supervisor Approval Date

[signature] 10/28/15
Approval Authority Approval Date
Effective Date:________

DRF Form (Continued)

Page 2 of 2

DOCUMENT # STP-114.002 Rev. 12 Chg. G

DESCRIPTION CONTINUED:

1. Page 1, Section 1.0, performed Applicability Determination for 10CFR50.59 and 72.48.
2. Enclosure A, page 2 Step 3.b:
   a. Changed "Test Deviation" to "Test Deficiency".
   b. Enhanced step by informing user that a CR should be written to document the test deficiency.

REASON/BASIS CONTINUED:

1. Updated procedure to determine whether 72.48 is applicable.

2.a PF 130039/140010 (Galloway/Crawford) - The use of the term "Test Deviation" is in correct, changed to "Term Deficiency".

2.b PF 130039 (Galloway) - Clarification for whether suspect data during the first test requires a CR for a test deficiency.

REQUIRED REVIEWERS CONTINUED:

Position	Type/Print Name	Comments	Position	Type/Print Name	Comments
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No

DOCUMENT REVIEW FORM

Document Identification

Originators Name: Anthony Williams **Ext:** 54047 **Mail Code:** 410

Date: 09/15/12 **Document No.:** STP-114.002 **Revision No.:** 12 **Change Letter:** F

Title: OPERATIONAL LEAKAGE CALCULATION ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction ☐ Temporary Approval

Description: See attached.

Reason/Basis for Change: See attached.

Is the SCOPE of the procedure affected by this change? NO ☒ YES ☐

Temporary Approval **Final approval required by: (30 days)**

QR	DC&R (Person Notified)	SS	Date
		/	

Document Reviewers (Enclosure C)

Required: Position	Type/Print Name	Comments Yes	Comments No	*Additional: Position	Type/Print Name	Comments Yes	Comments No
QR	B. SEO	☐	☑			☐	☐
		☐	☐			☐	☐
		☐	☐			☐	☐
		☐	☐			☐	☐
		☐	☐			☐	☐

QR Qualification Verified? ☒ Yes
Concurrence by Designated Supervisor 10/13/12
Supervisor/Date **or** enter CR # ________ (per 6.4.8.C)

ASAP Comment Due Date
Standard review period is 21 days
GM concurrence _______ for expedited review period

B

Pre- implementation Actions

All Comments Resolved ☒ None Received ☐ Yes 10-13-12
Originator/Date

Commitments Addressed per SAP-0630 ☒ NA ☐ Yes P/CAP # ______ ☐ MLSA ______ Initial/Date

50.59 Applicability/Review Completed (SAP-0107) ☒ NA ☐ Yes, Attached
Security Compliance Review Completed (SAP-0163) ☒ NA ☐ Yes (Security review required)
Pre-implementation Training Completed ☒ NA ☐ Yes
Training required after implementation ☒ NA ☐ Yes, CR # ___________
PSRC Review Completed ☒ NA ☐ Yes, Mtg. No. _______
NSRC Review Completed ☒ NA ☐ Yes, Mtg. No. _______
CMMS Update Required ☒ NA ☐ Yes Planner Notified ________ Initial/Date

10/13/12 Supervisor/Date	10/17/12 Approval Authority/Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document No.: STP-114.002 Rev. No. 12 Chg. Ltr. F

DESCRIPTION CONTINUED:

1. Page 6, Step 6.3.i:
   a. Divided Step 6.3.i.2)a) into two separate steps, Steps 6.3.i.2)a) and 6.3.i.2)b). One to instruct user to place LCV01003-WL in Manual and a second step to take the close.
   b. Divided Step 6.3.i.2)d in two separate steps, Steps 6.3.i.2)e) and 6.3.i.2)f). One to instruct the user to adjust the RCDT level to between 58% and 60%. Another step to place the level controller into AUTO.
2. Attachment II, page 2: added LCV01003-WL, REACTOR COOLANT DRAIN TK LEVEL CONTROL VLV.

REASON/BASIS FOR CHANGE CONTINUED:

1. PF 120462 (Moon) – removed combination steps and replaced with separate one steps actions. To reduce possibility of error and coincide with OAP-101.5 Step 6.3.a.
2. PF 120462 (Moon) – added LCV01003-WL to Valve Lineup to aid in component control.

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document Identification

Originators Name: R Perrill **Ext:** 5524 **Mail Code:** 410

Date: 9/28/2011 **Document No.:** STP-114.002 **Revision No.:** 12 **Change Letter:** E

Title: OPERATIONAL LEAKAGE CALCULATION ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction ☐ Temporary Approval

Description: See page 2.

Reason/Basis for Change: See page 2.

Is the SCOPE of the procedure affected by this change? NO ☒ **YES** ☐

Temporary Approval **Final approval required by: (30 days)**

/

QR **DC&R** (Person Notified) **SS** **Date**

Document Reviewers (Enclosure C)

Required Position	Type/Print Name	Comments Yes/No	*Additional Position	Type/Print Name	Comments Yes/No
QR	D Gilliam	☑ ☐			☐ ☐
		☐ ☐			☐ ☐
		☐ ☐			☐ ☐
		☐ ☐			☐ ☐
		☐ ☐			☐ ☐

QR Qualification Verified? ☒ Yes
Concurrence by: 9/29/11
Supervisor/Date or Enter CR # (per 6.4.8.C)

Comment Due Date ASAP

Pre- implementation Actions

All Comments Resolved ☐ None Received ☑ Yes 9/29/11
Originator/Date

Commitments Addressed per SAP-0630 ☒ NA ☐ Yes P/CAP # ______ ☐ MLSA ______ Initial/Date

50.59 Applicability/Review Completed (SAP-0107) ☒ NA ☐ Yes, Attached
Security Compliance Review Completed (SAP-0163) ☒ NA ☐ Yes (Security review required)
Pre-implementation Training Completed ☒ NA ☐ Yes
Training required after implementation ☐ NA ☒ Yes, CR # CR-10-00097-141
PSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______
NSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______
CMMS Update Required ☒ NA ☐ Yes Planner Notified ______ Initial/Date

9/29/11
Supervisor/Date

9/29/11
Approval Authority/Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document No.: STP-114.002 Rev. No. 12 Chg. Ltr. E

DESCRIPTION CONTINUED:

1) Removed Steps for closure of LCV01003-WL, REACTOR COOLANT DRAIN TANK LEVEL CONTROLLER, to perform leak rate.

2) Added new Steps 6.3.h and 6.4.l and new Attachment III, RCS To Atmospheric Leakage, to adjust the Trend Sheet recorded IDENTIFIED LEAKAGE for RCS leakage to the RB atmosphere. Changed Step 7.1 to require that Attachment III, if used, to be attached to Attachment II.

REASON/BASIS FOR CHANGE CONTINUED:

Per Management direction.

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document Identification

Originators Name: R. Perrill **Ext:** 55524 **Mail Code:** 410

Date: 09/09/11 **Document No.:** STP-114.002 **Revision No.:** 12 **Change Letter:** D

Title: OPERATIONAL LEAKAGE CALCULATION ☒ SR ☐ QR ☐ NNS

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction ☐ Temporary Approval

Description: See attached.

Reason/Basis for Change: See attached.

Is the SCOPE of the procedure affected by this change? NO ☒ YES ☐

Temporary Approval **Final approval required by: (30 days)**

QR ______ DC&R (Person Notified) ______ SS ______ / ______ Date

Document Reviewers (Enclosure C)

	Position	Type/Print Name	Comments Yes/No		Position	Type/Print Name	Comments Yes/No
Required	QR	A. Williams	☐ ☒	*Additional			☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐

QR Qualification Verified? ☒ Yes
Concurrence by: 9/9/11
Supervisor/Date or Enter CR # ______ (per 6.4.8.C)

Comment Due Date ASAP

B

Pre- implementation Actions

All Comments Resolved ☒ None Received ☐ Yes ______ Originator/Date

Commitments Addressed per SAP-0630 ☒ NA ☐ Yes P/CAP # ______ ☐ MLSA ______ Initial/Date

50.59 Applicability/Review Completed (SAP-0107) ☒ NA ☐ Yes, Attached
Security Compliance Review Completed (SAP-0163) ☒ NA ☐ Yes (Security review required)
Pre-implementation Training Completed ☒ NA ☐ Yes
Training required after implementation ☒ NA ☐ Yes, CR # ______
PSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______
NSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______
CMMS Update Required ☒ NA ☐ Yes Planner Notified ______ Initial/Date

9/9/11
Supervisor/Date

9/9/11
Approval Authority/Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document No.: STP-114.002 Rev. No. 12 Chg. Ltr. D

DESCRIPTION CONTINUED:

Added new Steps 6.2.c and 6.3.h for controlling the position of LCV01003-WL, REACTOR COOLANT DRAIN TANK LEVEL CONTROLLER.

REASON/BASIS FOR CHANGE CONTINUED:

Management direction.

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document Identification

Originators Name: MD Johnson **Ext:** 54300 **Mail Code:** 410

Date: 5/14/2011 **Document No.:** STP-114.002 **Revision No.:** 12 **Change Letter:** C

Title: OPERATIONAL LEAKAGE CALCULATION ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction ☐ Temporary Approval

Description: See page 2.

Reason/Basis for Change: See page 2.

Is the SCOPE of the procedure affected by this change? NO ☒ **YES** ☐

Temporary Approval **Final approval required by: (30 days)**

QR	DC&R (Person Notified)	SS	Date
		/	

Document Reviewers (Enclosure C)

Required Position	Type/Print Name	Comments Yes/No	*Additional Position	Type/Print Name	Comments Yes/No
QR	R Perrill	☐ ☒			☐ ☐
		☐ ☐			☐ ☐
		☐ ☐			☐ ☐
		☐ ☐			☐ ☐
		☐ ☐			☐ ☐

Concurrence [signature] 5/24/11
Supervisor/Date or Enter CR # __________ (per 6.4.8.C)

Comment Due Date ASAP

Pre- implementation Actions

All Comments Resolved ☒ None Received ☐ Yes [signature] 5/25/11
Originator/Date

Commitments Addressed per SAP-0630 ☒ NA ☐ Yes P/CAP # ______ ☐ MLSA ______ Initial/Date

50.59 Applicability/Review Completed (SAP-0107) ☒ NA ☐ Yes, Attached

Security Compliance Review Completed (SAP-0163) ☒ NA ☐ Yes (Security review required)

Pre-implementation Training Completed ☒ NA ☐ Yes

Training required after implementation ☐ NA ☒ Yes, CR # CR-10-00097-115

PSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______

NSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______

CMMS Update Required ☒ NA ☐ Yes Planner Notified ________ Initial/Date

[signature] 5/25/11
Supervisor/Date

[signature] 5/25/11
Approval Authority/Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document No.: STP-114.002 Rev. No. 12 Chg. Ltr. C

DESCRIPTION CONTINUED:

1. Page 2, NOTE 6.0, Modified wording and broke out into steps acceptable test time periods (several hours, two hours, 1 hour, or less than 1 hour) for different scenarios.
2. Page 10, NOTE 6.4.g and .h, Added step 3, If PRT is to be vented during the calculations, contact Operations Management for evaluation.
3. Page 11, NOTE 7.3, Added: A negative value for identified and/or unidentified leakage does not in itself result in an Invalid or Suspect test and does not require re-performance of the test.
4. Attachment I, Deleted Factor and Deviation Blocks and associated Notes. Old Note (6) is new Note (1).
5. Attachment II, Page 1, Added new NOTE 7.0 and step 7.0 for Data Requirements Met.

REASON/BASIS FOR CHANGE CONTINUED:

1. CR-08-01267-001, Tave droop during two hour leak rates caused slightly elevated unidentified leakage. Ops should consider adding some guidelines to leak rate stp for when to perform 2 hour, 1 hour and less leak rates, there are also times (outages) where you may only be able to get a 20 or 30 min leak rate.
2. PF090840 (Himel), STP0114.002 – RCS leakrate. Feedback driven by: OE 28650 –RCS Leakage Calc Affected by [PRT] venting. Feedback: Fred, would this (yellow/quotations) capture what you mentioned ? I think Rob is the "OPUS" I am supposed to send this too? "If PRT is to be vented during the calculation contact operations management for evaluation."
3. PF100011 (Ray), Some confusion exists as to whether a negative RCS leak rate is a valid test. A negative leak rate value does not necessarily make the test results are invalid. Running additional, unnecessary tests is neither required nor desired due to the impact on the RCS leak rate trending data. 1) Under Section 7.0 "Data Requirements" add "Note 7.3: A negative value for identified and/or unidentified leakage does not in itself result in an Invalid or Suspect test and does not require re-performance of the test.
4. PF090974 (Bruner), Within Step 6.4 and Attachment I, there is nothing that directs the operator to fill out the "FACTOR" and "DEVIATION" blocks on Attachment I. The procedure has the operator perform calculations within the STP (steps 6.4.d thru 6.4.h) and transfer the results to PART 2 of Attachment I. Remove the DEVIATION and FACTOR blocks from Attachment I or direct the use of these blocks within the STP. PW note: The data needed for the blocks were in the notes. The notes came out of the body of the procedure. The attachment has gone through numerous iterations in an attempt to use the attachment as a worksheet with all the required information on it. Subsequently the attachment and the body did not integrate well. With removal of the two lines and the notes, primary reliance for performance of the STP is shifted back to the body of the procedure.
5. PF100011 (Ray), Some confusion exists as to whether a negative RCS leak rate is a valid test. A negative leak rate value does not necessarily make the test results are invalid. Running additional, unnecessary tests is neither required nor desired due to the impact on the RCS leak rate trending data. 2) On Attachment II (pg. 1) Add a sign off for Section 7.0 "Data Requirements met and add a note on Att. II similar to Note 7.3 above.

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Page 1 of 1

Document Identification

Originators Name: Barry Stroup **Ext:** 55547 **Mail Code:** 410

Date: 06/17/10 **Document No.:** STP-114.002 **Revision No.:** 12 **Change Letter:** B

Title: OPERATIONAL LEAKAGE CALCULATION ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction ☐ Temporary Approval

Description: Change Caution 6.3.d to allow alternate valve position verification if IPCS indication is erroneous.

Reason/Basis for Change: Management request (basis CR-10-02437).

Is the SCOPE of the procedure affected by this change? NO ☒ YES ☐

Temporary Approval **Final approval required by: (30 days)**

QR	DC&R (Person Notified)	SS	Date
		/	

Document Reviewers (Enclosure C)

	Position	Type/Print Name	Comments Yes/No		Position	Type/Print Name	Comments Yes/No
Required	QR	TODD PRICE	☐ ☒	*Additional			☐ ☐
			☐ ☐				☐ ☐
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			☐ ☐				☐ ☐
			☐ ☐				☐ ☐

Concurrence [signature] 6/17/10
Supervisor/Date or Enter CR # (per 6.4.8.C)

Comment Due Date ASAP

Pre- implementation Actions

All Comments Resolved ☒ None Received ☐ Yes [signature] 6/17/10
Originator/Date

Commitments Addressed per SAP-0630 ☒ NA ☐ Yes P/CAP # ☐ MLSA Initial/Date

50.59 Applicability/Review Completed (SAP-0107) ☒ NA ☐ Yes, Attached
Security Compliance Review Completed (SAP-0163) ☒ NA ☐ Yes (Security review required)
Pre-implementation Training Completed ☒ NA ☐ Yes
Training required after implementation ☒ NA ☐ Yes, CR #
PSRC Review Completed ☒ NA ☐ Yes, Mtg. No.
NSRC Review Completed ☒ NA ☐ Yes, Mtg. No.
CMMS Update Required ☒ NA ☐ Yes Planner Notified Initial/Date

[signature] 6/17/10
Supervisor/Date

[signature] 6/17/10
Approval Authority/Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document Identification

Originators Name: MD Johnson **Ext:** 54300 **Mail Code:** 410

Date: 7/1/2009 **Document No.:** STP-114.002 **Revision No.:** 12 **Change Letter:** A

Title: OPERATIONAL LEAKAGE CALCULATION ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction ☐ Temporary Approval

Description: See page 2.

Reason/Basis for Change: See page 2.

Is the SCOPE of the procedure affected by this change? NO ☒ YES ☐

Temporary Approval **Final approval required by: (30 days)**

QR	DC&R (Person Notified)	SS	Date
		/	

Document Reviewers (Enclosure C)

	Position	Type/Print Name	Comments Yes/No		*Additional Position	Type/Print Name	Comments Yes/No
Required	QR	DUHAIME	☑ Yes ☑ No				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐

Concurrence [signature] 7/1/09
Supervisor/Date or enter CR # ______

Comment Due Date 7/2/09

Pre- implementation Actions

All Comments Resolved	☐ None Received ☑ Yes	[signature] 7/1/09 Originator/Date
Commitments Addressed per SAP-0630	☒ NA	☐ Yes, P/CAP # ______ ☐ MLSA ______ Initial/Date
50.59 Applicability/Review Completed (SAP-0107)	☒ NA	☐ Yes, Attached
Pre-implementation Training Completed	☒ NA	☐ Yes
Training required after implementation	☒ NA	☐ Yes, CR # ______
PSRC Review Completed	☒ NA	☐ Yes, Mtg. No. ______
NSRC Review Completed	☒ NA	☐ Yes, Mtg. No. ______
Champs Update Required	☒ NA	☐ Yes Planner Notified ______ Initial/Date

[signature] 7/1/2009
Supervisor/Date

[signature] 7/1/2009
Approval Authority/Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document No.: STP-114.002 Rev. No. 12 Chg. Ltr. A

DESCRIPTION CONTINUED:

1. Att II, page 1,
   a. Rearranged order of checkoffs at bottom.
   b. Added Person notified and time signoff.
   c. Added to Suspect flag set to perform one additional leak rate and time complete.
2. Att II, page 2, Added AUTO/OPEN to Required Operable position.
3. Encl A, page 2, Step 3.b. Added "Only one retest is required per surveillance."

REASON/BASIS FOR CHANGE CONTINUED:

1. PF090899 (Goldston), We want to only perform one additional leak rate, when the suspect flag is set.
   We want a confirmation from PSE that they have been notified within 12 hours. So a phone call is required, not an email.
   PF090772 (Crawford), Reorder sequence of step and clarify expectations.
2. PF090434-005 (Generic-Beckham), Add required Operable Position.
3. Reviewer comment (Duhaime) to clarify only one retest is required per surveillance and not a continuous "do-loop" of retests.

DOCUMENT REVIEWERS CONTINUED:

	Position	Type/Print Name	Comments Yes/No		Position	Type/Print Name	Comments Yes/No
Required	____	____________	☐ ☐	*Additional	____	____________	☐ ☐
	____	____________	☐ ☐		____	____________	☐ ☐
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* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document Identification

Originators Name: Riley R. Johnson **Ext:** 55021 **Mail Code:** 410

Date: 10/28/08 **Document No.:** STP-114.002 **Revision No.:** 12 **Change Letter:** N/A

Title: OPERATIONAL LEAKAGE CALCULATION ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction
☐ Temporary Approval

Description: SEE PAGE 2

Reason/Basis for Change: SEE PAGE 2

Is the SCOPE of the procedure affected by this change? NO ☒ **YES** ☐

Temporary Approval **Final approval required by: (30 days)**

QR	**DC&R** (Person Notified)	**SS**	**Date**
		/	

Document Reviewers (Enclosure C)

Required

Position	Type/Print Name	Comments Yes	Comments No
QR	DAN GOLDSTON	☒	☐
		☐	☐
		☐	☐
		☐	☐
		☐	☐

***Additional**

Position	Type/Print Name	Comments Yes	Comments No
		☐	☐
		☐	☐
		☐	☐
		☐	☐
		☐	☐

Concurrence [signature] 12/10/8
Supervisor/Date

Comment Due Date 12/17/8

Pre- implementation Actions

All Comments Resolved ☒ Yes [signature] / 11/23/09
Originator/Date

Commitments Addressed per SAP-0630 ☒ NA ☐ Yes, P/CAP # ______ ☐ MLSA ______ Initial/Date

50.59 Applicability/Review Completed (SAP-0107) ☒ NA ☐ Yes, Attached

Pre-implementation Training Completed ☐ NA ☒ Yes

Training required after implementation ☐ NA ☒ Yes, CR # CR-08-00440-103

PSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______

NSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______

CHAMPS Update Required ☒ NA ☐ Yes Planner Notified ______ Initial/Date

[signature] 3/9/9
Supervisor/Date

[signature] 3/23/09
Approval Authority/Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document No.: STP-114.002 Rev. No. 12 Chg. Ltr. N/A

DESCRIPTION CONTINUED:

1) Updated step 2.4.

2) Updated note 6.0.

3) Updated 6.3.c.

4) Updated caution 6.3.d.

5) Updated note 6.4.g and h.

6) Added data requirement 7.3.

7) Created Enclosure A.

8) Updated step 6.4.a.2.f to remove requirement to record TR413.

9) Removed the Revision Summary Section.

REASON/BASIS FOR CHANGE CONTINUED:

1) To incorporate the requirement to notify PSE.

2) To ensure test period of two hours.

3) To use 120 minutes for test.

4) To ensure a test deficiency is written.

5) To ensure proper action taken if results are suspect.

6) To ensure a test deficiency is written; **NOTE:** actions 1 thru 6 are incorporating comments from Dan Goldston.

7) Feedback 080801, procedure improvement to support ES-161 (e-mail from Dan Goldston).

8) Feedback 08620, data not used; 9) Revision Summary no longer used in Ops procedures.

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-220-(R)N18**

2018 NRC A2 (RO)

CANDIDATE: ______________________________

EXAMINER: ______________________________

DESCRIPTION: Determine allowed use of valve extension devices and required documentation.

TASK: O-119-015-03-01 - Apply Tech Spec requirements.

TASK STANDARD: Candidate determines that the longest wrench that can be used is 18” and that the valve must be recorded in an R&R as inoperable.

TERMINATING CUE: The candidate returns the handout and briefing sheet to examiner.

PREFERRED LOCATION: CLASSROOM

PREFERRED METHOD: PERFORM

REFERENCES:

OAP-100.5, GUIDELINES FOR CONFIGURATION CONTROL AND OPERATION OF PLANT EQUIPMENT

K/A 2.2.37 Ability to determine operability and/or availability of safety related equipment. (RO 3.6)

10CFR55: 45 b(12)

TOOLS: Access to paper or electronic copies of V.C. Summer procedures.

EVALUATION TIME: 15 Minutes. ***TIME CRITICAL:*** NO

TIME START: ____________ ***TIME FINISH:*** __________ ***PERFORMANCE TIME:*** __________

CANDIDATE: ______________________________________

EXAMINER: ______________________________________

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is in Mode 5 with an outage in progress.
- A tagout that provided mechanical isolation for piping work is being **cleared**.
- XVG08485C-CS, CHARGING/SI PUMP C DISCHARGE VALVE cannot be opened locally by hand using only the 14” handwheel.
- The Shift Manager has concurred with the use of a valve wrench to provide additional force to open the valve.
- CMMS is out of service.

INITIATING CUES:

You are directed to determine the following:

- The largest valve wrench that can be used to open XVG08485C-CS locally.
- The Operability status of the valve after it is opened with the valve wrench.
- The tracking tool that will be used to log the status of the valve as a result of this operation.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 1

Refers to OAP-100.5, GUIDELINES FOR CONFIGURATION CONTROL AND OPERATION OF PLANT EQUIPMENT

STEP STANDARD:

Refers to either a hard-copy or electronic controlled copy of OAP-100.5.

CUES:

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 2

Determines the largest size wrench allowed to provide mechanical advantage.

STEP STANDARD:

Refers to table on page 32 of OAP-100.5 and determines up to a 18" wrench can be used for a 14" handwheel.

CUES:

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 3

Determines that the valve is a Safety Related valve.

STEP STANDARD:

Consults drawing 302-675 and determines that the valve is within a Code class 2A safety-related piping boundary or uses other means to determine the safety class.

CUES:

Notes to evaluator:

The common method for this identification is CMMS. Declaration that this valve is safety-related by reasoning its position in the charging system is acceptable.

The QList Information in CMMS lists the valve as "SR" (Safety Related).

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 4

Determine the Operability status of the valve.

STEP STANDARD:

Determines that the valve must be declared inoperable.

CUES:

Note to evaluator: Requirement in accordance with OAP-100.5, step 10.3.

COMMENTS:

CRITICAL:	YES	SEQUENCED:	NO	SAT		UNSAT	

JPM STEP 5

Determine the documentation requirement for the valve status.

STEP STANDARD:

Determine that the valve status must be entered in the R&R log .

CUES:

Note to evaluator: Requirement in accordance with OAP-100.5, step 10.3.

COMMENTS:

JPM SETUP SHEET

JPM: JPM: JPA-220-(R)N18

IC SET: N/A

INSTRUCTIONS:

Provide a handout containing the following:

Name: ____ Date: _______________________

Record the following:

1. The largest valve wrench that can be used to open XVG08485C-CS locally.

2. The Operability status of the valve after it is opened with the valve wrench.

3. The tracking tool that will be used to log the status of the valve as a result of this operation.

COMMENTS:

JPM BRIEFING SHEET

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is in Mode 5 with an outage in progress.
- A tagout that provided mechanical isolation for piping work is being **cleared**.
- XVG08485C-CS, CHARGING/SI PUMP C DISCHARGE VALVE cannot be opened locally by hand using only the 14” handwheel.
- The Shift Manager has concurred with the use of a valve wrench to provide additional force to open the valve.
- CMMS is out of service.

INITIATING CUES:

You are directed to determine the following:

- The largest valve wrench that can be used to open XVG08485C-CS locally.
- The Operability status of the valve after it is opened with the valve wrench.
- The tracking tool that will be used to log the status of the valve as a result of this operation.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

2018 NRC A2   RO Handout

Name: ______________________________________   Date: ________________________

Record the following:

1. The largest valve wrench that can be used to open XVG08485C-CS locally.

2. The Operability status of the valve after it is opened with the valve wrench.

3. The tracking tool that will be used to log the status of the valve as a result of this operation

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-815-(R)N18**

2018 NRC A3 (RO)

CANDIDATE: ____________________

EXAMINER: ____________________

DESCRIPTION: Locate a component on a survey map and calculate worker stay times.

TASK: O-119-010-03-01 Apply radiation and contamination safety procedures

TASK STANDARD: Candidate determines with work shown for calculations that only AO 2 can perform the work described in the initial conditions.

TERMINATING CUE: The candidate returns the handout and briefing sheet to examiner.

PREFERRED LOCATION:	***PREFERRED METHOD:***
CLASSROOM	PERFORM

REFERENCES:

VCS-HPP-0153, ADMINISTRATIVE EXPOSURE LIMITS

K/A 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (RO 3.2)

10CFR55: 45 b(10)

TOOLS: Access to paper or electronic copies of V.C. Summer procedures.
Handout 1 containing survey maps of RHR/Spray pump rooms and “A” RHR HX room.
Handout 2 answer sheet.

EVALUATION TIME: 30 Minutes. ***TIME CRITICAL:*** NO

TIME START: ____________ ***TIME FINISH:*** ___________ ***PERFORMANCE TIME:*** __________

CANDIDATE: ____

EXAMINER: ____

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- Today is June 16.
- The plant is in mode 5 with an outage in progress.
- The plant was shut down to address a fuel failure and high RCS activity.
- To support work, an operator must be stationed for **30 minutes** at the following location:
  - Approximately 1 foot from the bend of the RHR system piping directly upstream of XVG08728A-RH, RESIDUAL HEAT REMOVAL PUMP A DISCH VLV.
- There are 2 AOs who are available.
  - AO 1 is female who has formally declared pregnancy and is due to deliver in September.
She has worked at V.C. Summer as an AO for 10 years.
She received 45 mrem yesterday and this is her ONLY exposure for the current year.
  - AO 2 is a male who has worked as an AO for a two years.
He has received 990 mrem at V.C. Summer this year.

INITIATING CUES:

You are to calculate and report which of the two operators, if any, can be used WITHOUT additional approval for an increase in administrative exposure limits.

Assume the following:

- ALL dose received in transit is negligible.
- ANY extensions that may have been previously granted this year were the minimum allowed by procedure.

SHOW ALL WORK ON THE HANDOUT PROVIDED TO YOU.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 1

Determines initial administrative exposure limits for each operator.

STEP STANDARD:

Determines or applies the limits contained in VCS-HPP-153, ADMINISTRATIVE EXPOSURE LIMITS.

Operator 1: 500 mr per gestation period and 50 mr per calendar month.

Operator 3: 2000 mr for first extension after Form 4 is complete.

CUES:

Note to Evaluator: These limits may be applied from memory as long as work is shown for the required calculation.

COMMENTS:

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 2

Determines current V.C. Summer exposure of each operator. (Given in initial conditions.)

STEP STANDARD:

AO 1: 45 mr

AO 2: 990 mr

CUES:

Note to Evaluator: This information will be shown in the required calculation. The dose rate is obtained by referring to the survey map in the handout provided and determining that the dose rate at 1 foot is 18 mrem/hr. The 30 mrem/hr number indicated is the contact reading and is not the correct value to use according to the assumptions provided. Using this value is NOT a critical error but should be documented as a knowledge weakness.

COMMENTS:

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 3

Determines expected exposure for the work to be performed.

STEP STANDARD:

18 mr/hr x 0.5 hours = 9 mrem

CUES:

Note to Evaluator: This information will be shown in the required calculation.

COMMENTS:

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 4

Determines workers that can perform the work.

STEP STANDARD:

AO 1: 50 mrem monthly limit - 45 VCS exposure this month = 5 mrem to reach limit.

AO 2: 2000 mrem limit - 990 mrem VCS exposure = 1010 mrem to reach limit.

Identifies that only Operators 2 has the available exposure to work the required time.

CUES:

Evaluator Note: All work for the required calculation will be shown in the Handout provided to support this conclusion.

COMMENTS:

JPM SETUP SHEET

JPM: JPM: JPA-815-(R)N18

IC SET: N/A

INSTRUCTIONS:

Provide a handout containing survey maps for the RHR/Spray pump rooms and the “A” RHR Heat Exchanger room. The 1 ft doserate for the pipe bend upstream of XVG08728A must indicate 18 mr/hr.

Provide Handout 2 containing the following:

Name: ________________ Date: _______________________

SHOW ALL WORK TO SUPPORT YOUR DETERMINATION BELOW:

Provide access to hard-copy or electronic station procedures.

COMMENTS:

JPM BRIEFING SHEET

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- Today is June 16.
- The plant is in mode 5 with an outage in progress.
- The plant was shut down to address a fuel failure and high RCS activity.
- To support work, an operator must be stationed for **30 minutes** at the following location:
  - Approximately 1 foot from the bend of the RHR system piping directly upstream of XVG08728A-RH, RESIDUAL HEAT REMOVAL PUMP A DISCH VLV.
- There are 2 AOs who are available.
  - AO 1 is female who has formally declared pregnancy and is due to deliver in September. She has worked at V.C. Summer as an AO for 10 years. She received 45 mrem yesterday and this is her ONLY exposure for the current year.
  - AO 2 is a male who has worked as an AO for a two years. He has received 990 mrem at V.C. Summer this year.

INITIATING CUES:

You are to calculate and report which of the two operators, if any, can be used WITHOUT additional approval for an increase in administrative exposure limits.

Assume the following:

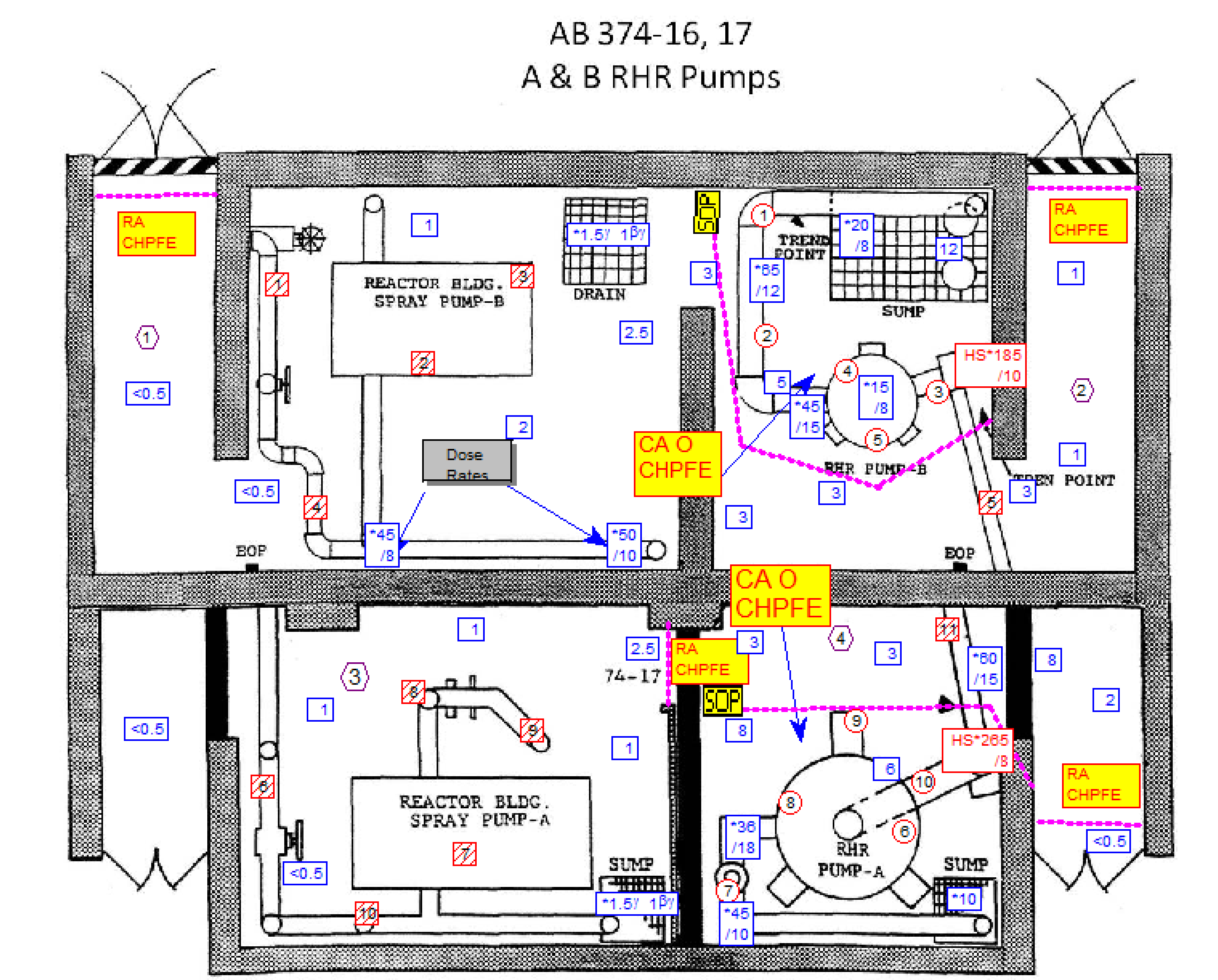
- ALL dose received in transit is negligible.
- ANY extensions that may have been previously granted this year were the minimum allowed by procedure.

SHOW ALL WORK ON THE HANDOUT PROVIDED TO YOU.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

VSDS Standard Map Survey Report

AB374-16,17	Survey #: VCS1-M-20170530-19	Date/Time: 5/30/2018 08:50



Comments:

Summary of Highest Readings (All available values may not be listed)	
Smears	**Air Samples & Wipes**
10) 2000 dpm/100 cm2 β/γ	Wipe 11) =BKG NCPM β/γ

Type: Quarterly

Symbol Legend (for example only)

Dose Rate: *150 Contact Reading; /75 30 cm Reading; 20 General Area
HS-50 Hot Spot; RCA Posting; Drip Bag
15 Smear; 15 Air Sample; 1 Sweep; 15 Wipe

RWP #: 17-04006
Reactor Power = 0%

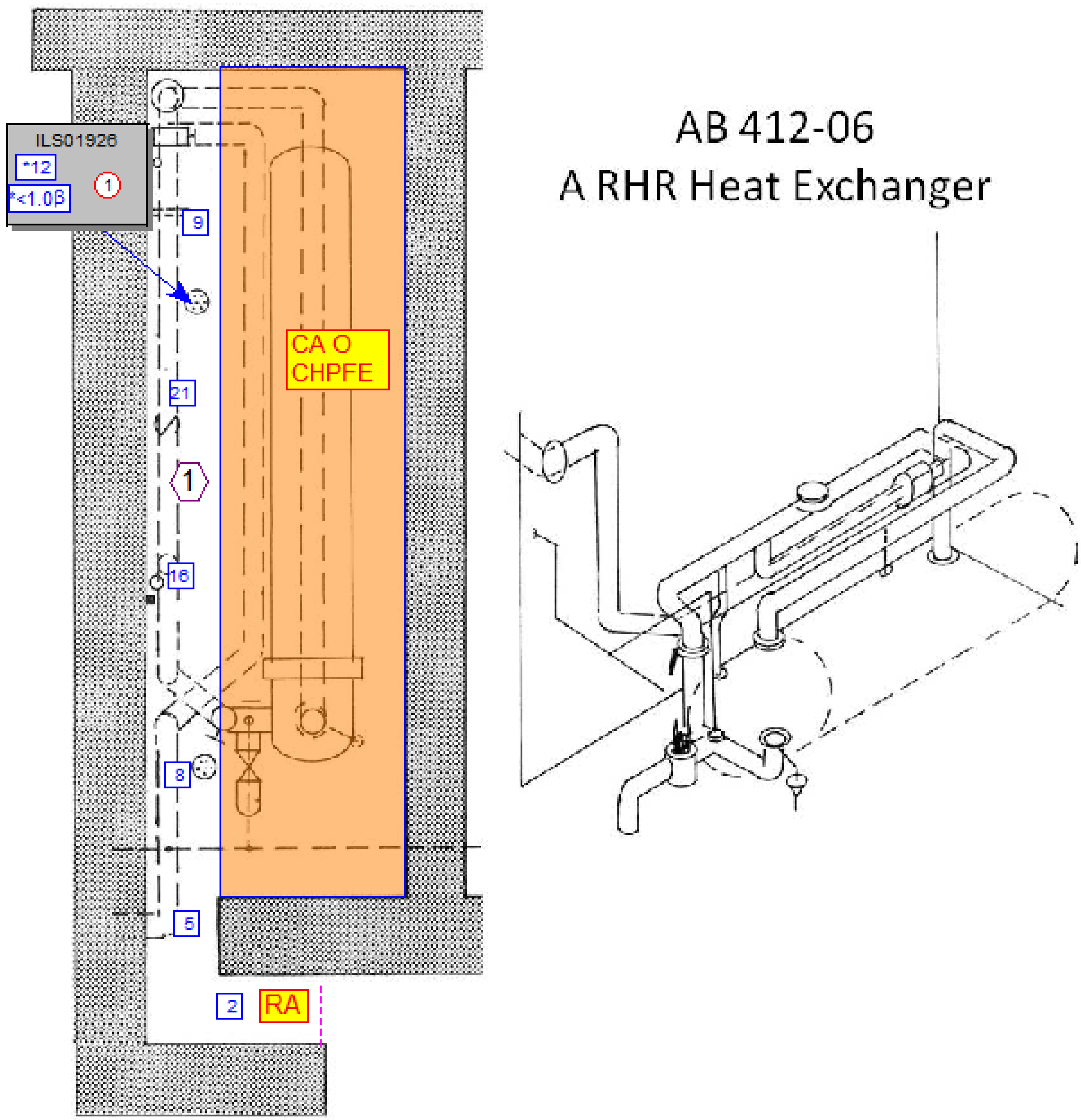
Unless otherwise noted, dose rates in mrem/hr.

Lead Surveyor: Ashley Bates	Status: Approved by: Timothy Raucci, 5/30/2017 15:32:36
Location Code: AB	Bldg/Area Name: 374
Location Description: AB374-16,17	

Survey ID#: 08039

VSDS Standard Map Survey Report

AB412-06-A-RHR	Survey #: VCS1-M-20160330-17	Date/Time: 5/30/2018 08:40



Comments: I/C entered AB412-06 to test the leak detection switch ILS01926 in drain in rear of room.

Summary of Highest Readings (All available values may not be listed)	
Smears	**Air Samples & Wipes**
1) 23000 dpm/100 cm2 β/γ	

Type: Job Coverage

Symbol Legend (for example only)

Dose Rate: *150 Contact Reading; /75 30 cm Reading; 20 General Area

HS-50 Hot Spot; RCA Posting; Drip Bag

15 Smear; 15 Air Sample; 1 Sweep; 15 Wipe

RWP #: 16-01080
Reactor Power = 100%

Unless otherwise noted, dose rates in mrem/hr.

Lead Surveyor: Matthew Kelley	Status: Approved by: Julius Burkett, 3/30/2016 15:39:58
Location Code: AB	Bldg/Area Name: 412
Location Description: AB412-06-A-RHR	

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-66D-(S)N18**

2018 NRC A1a (SRO)

CANDIDATE: ______________________________

EXAMINER: ______________________________

DESCRIPTION: Review a power range heat balance

TASK: O-342-026-03-02 Review results of surveillance tests (SAP-134, GTP-301, AND GTP-302)

TASK STANDARD: Candidate identifies errors in blowdown heat loss, steam generator power, the CVCS Letdown Heat Loss constant and the calculation of reactor power and determines that acceptance criteria are not met.

TERMINATING CUE: The candidate returns JPM briefing sheet and Handout materials provided to the examiner.

PREFERRED LOCATION:	***PREFERRED METHOD:***
CLASSROOM	PERFORM

REFERENCES:

STP-102.002, NIS POWER RANGE HEAT BALANCE

Station Curve Book

K/A 2.1.45 Ability to identify and interpret diverse indications to validate the response of another indicator (RO 4.3)

10CFR55: 45 b(1)(4)

TOOLS: Copies of 2018 NRC A1-a SRO Only Handout
Electronic or Hardcopy versions of station procedures

EVALUATION TIME: 45 Minutes. ***TIME CRITICAL:*** NO

TIME START: ____________ ***TIME FINISH:*** ___________ ***PERFORMANCE TIME:*** __________

CANDIDATE: ____

EXAMINER: ____

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is being started up after an outage.
- GOP-4A, POWER OPERATION (MODE 1 - ASCENDING), requires that STP-102.002, NIS POWER RANGE HEAT BALANCE be completed.
- The CALM function on IPCS is not available.
- An RO has completed STP-102.002, Attachment III, REACTOR POWER CALCULATION to determine Reactor Power in percent.

INITIATING CUES:

You are to perform the following:

- Review STP-102.002, Attachment III that is provided to you in the handout.
- Correct any deficiencies in the calculation.
- Determine if the Nuclear Instrumentation meets the acceptance criteria of STP-102.002.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: YES SEQUENCED: YES SAT UNSAT

JPM STEP 1

Reviews the completed SOP-102.002, Attachment III to verify correct performance.

STEP STANDARD:

Reviews the completed SOP-102.002, Attachment III contained in the handout provided and determines the following discrepancies:

1) Loop Steam Pressures and Loop Steam Enthalpies are transposed incorrectly.

2) Loop Delta Hs in step 9 are incorrect.

3) Wrong values carried forward resulting in incorrect power calculation in step 11.

4) Wrong value for K2011 used in step 12 resulting in an incorrect Letdown Heat Loss calculation.

5) Errors carried forward resulted in an incorrect calculation of Percent Power in step 13.

CUES:

Evaluator note(s):

Provide candidate with the 2018 NRC A1-a RO Handout.

Justifications for the incorrect items are contained in the A1-a SRO only Justification document.

COMMENTS:

CRITICAL: YES SEQUENCED: YES SAT UNSAT

JPM STEP 2

Corrects calculations.

STEP STANDARD:

Corrects calculations for deficiencies identified in previous step and recalculates a value for Percent Power of 25.7% to 26.7%.

CUES:

Evaluator note(s):

Ranges for expected values are included in the A1-a SRO Only Key.

The critical value for this step is for the final calculation to fall within the expected range. Any minor errors in correcting the calculation may be written up as knowledge or ability deficiencies that are weighted according to severity.

COMMENTS:

CRITICAL: YES	SEQUENCED: YES	SAT		UNSAT	

JPM STEP 3

Determines that Acceptance Criteria are not met.

STEP STANDARD:

Determines that Power Range indications are not within 1% of the calculated value for power as required by STP-102.002, Step 8.3.

CUES:

Evaluator note(s): The candidate may report that instrumentation adjustment is required in accordance with step 6.2.g. This is acceptable as a final answer since that adjustment will ensure that power range NI readings are within the 1% criteria.

COMMENTS:

JPM SETUP SHEET

JPM: JPA-66D-(S)N18

IC SET: N/A

INSTRUCTIONS:

Ensure that current procedures are available in hard copy or electronically.

COMMENTS:

JPM BRIEFING SHEET

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is being started up after an outage.
- GOP-4A, POWER OPERATION (MODE 1 - ASCENDING), requires that STP-102.002, NIS POWER RANGE HEAT BALANCE be completed.
- The CALM function on IPCS is not available.
- An RO has completed STP-102.002, Attachment III to determine Reactor Power in percent.

INITIATING CUES:

You are to perform the following:

- Review STP-102.002, Attachment III, REACTOR POWER CALCULATION provided to you in the handout.
- Correct any deficiencies in the calculation.
- Determine if the Nuclear Instrumentation meets the acceptance criteria of STP-102.002.

Hand this paper back to your Evaluator when_you feel that you have satisfactorily completed the assigned task.

******2018 NRC A1-a SRO Only KEY***********

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

NUCLEAR OPERATIONS

NUCLEAR OPERATIONS

COPY NO.____________

SURVEILLANCE TEST PROCEDURE

STP-102.002

NIS POWER RANGE HEAT BALANCE

REVISION 8

SAFETY RELATED

CHG D

RECORD OF CHANGES

CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE	CHANGE LETTER	TYPE CHANGE	APPROVAL DATE	CANCELLATION DATE
A	P	02/22/05		E	P	08/31/15	
B	P	06/06/05		F	P	05/10/16	
C	P	06/16/05					
D	P	02/13/12					

REFERENCE USE

Procedure Segments May Be Performed From Memory.
Must Verify Work Following Each Segment.

******2018 NRC A1-a SRO Only KEY***********

TABLE OF CONTENTS

1.0 PURPOSE/SCOPE

1.1 The purpose of this procedure is to perform a daily calibration of the Nuclear Instrumentation System power range channels by performing a heat balance in accordance with Technical Specification Table 4.3.1, item 2. CHG A

1.2 The provisions of 10CFR50.65a(4) and 10CFR50 Appendix B are applicable.

1.3 10CFR50.59 and 10CFR72.48 do not apply to this procedure. CHG E

2.0 PRECAUTIONS

2.1 The Rod Control System should be placed in manual prior to adjustment of any power range channel gain.

2.2 When adjusting channel gains, only unlock/adjust gain pot on one NI channel at a time. CHG E

2.3 Operation at other than the program T_{avg} will directly affect the Nuclear Instrumentation System indication.

3.0 TEST EQUIPMENT

3.1 Steam Tables if performing Step 6.2. CHG A

4.0 TEST FREQUENCY

4.1 This test shall be performed daily when reactor power is greater than 15%.

NOTE 5.0 through 8.0
An asterisk (*) preceding a step indicates that data or a signoff is required on the attachment identified within the step.

CHG A

5.0 INITIAL CONDITIONS

5.1 The following plant conditions have been established:

☐ a. Reactor power and turbine load are constant.

☐ b. No boron concentration changes are in progress.

Step 5.1 continued

☐ c. Steam Generator Blowdown is either secured (preferred condition) or held constant. CHG D

☐ d. Tavg is constant and preferably within ±1ºF of program Tavg.

5.2 The Shift Manager/Control Room Supervisor has signed on Attachment I, signifying the following: CHG E

☐ *a. The Precautions of Section 2.0 have been discussed with the necessary personnel involved in the performance of this test.

☐ *b. All required Initial Conditions of Section 5.0 for this test have been met.

NOTE 6.0

Either Step 6.1 or Step 6.2 should be completed. The preferred method for calculating the heat balance is by use of the IPCS per Step 6.1 when reactor power is greater than 20%, at which FIVCALS activates. CHG A

6.0 PROCEDURE

6.1 Perform a heat balance using the Integrated Plant Computer System as follows:

☐ a. Activate CALM. NA SMh 6/5/18

☐ b. Under the REAL-TIME FUNCTIONS, select GENERATE CURRENT CALORIMETRIC REPORT (report generated, not displayed). CHG B

NOTE 6.1.c

The display provides the most current average of QCORE1 and reactor power by quadrant.

c. Under the REAL-TIME FUNCTIONS:

☐ 1) Select DISPLAY QCORE1 & NIS HISTORY. CHG C

☐ *2) Under QCORE1 (U8605) enter the AVERAGE on Attachment I.

Step 6.1 continued

☐ *d. Record the indicated reactor power from N41A through N44A on Attachment I. CHG D

☐ *e. Record the Main Control Board indications from N41B through N44B on Attachment I. CHG E

☐ f. Use Attachment I to determine the meter correction factor and document any adjustments, if required. CHG C

g. If meter adjustments are required/desired proceed as follows: CHG F

NOTE 6.1.g.1)

NA [signature] 6/5/18

a. The ROD CONTROL BANK SEL Switch should be placed in MAN prior to adjustment of any power range channel gain.

b. If NIS METER CORRECTION cannot be adjusted with the GAIN potentiometers on the front of the NIS power range drawers, Enclosure A should be used for Coarse Level Correction of affected power range channels.

c. Adjustment of indicated Reactor Power is required if any absolute value is greater than 0.5%.

CHG D

d. Difference in MCB reading minus calculated value or greater than or equal to 1%, I&C should be contacted to perform mechanical zero adjustments. CHG E

☐ 1) If adjustment of the GAIN potentiometers is being performed, place the ROD CONTROL BANK SEL Switch in MAN.

☐ 2) If adjustment is required, slowly adjust the GAIN potentiometer on the front of the affected NIS power range channel drawer one at a time while closely monitoring power range channels.

☐ *3) If adjustment is required, adjust all four channels by their individual Meter Correction Factor and record meter correction in Attachment I.

☐ 4) When adjustment of GAIN potentiometers is complete, place the ROD CONTROL BANK SEL Switch in AUTO.

CHG D

☐ *5) If MCB indications adjustment is required, record N41B through N44B indication on Attachment I. CHG E

Step 6.1.g continued

SML 6/5/18
NA

☐ h. ~~Attach a printout of the CALORIMETRIC REPORT, to the STTS package identifying the STP number and STTS number in the upper right hand corner of each page.~~

6.2 Perform a heat balance using Main Control Board readings as follows:

☐ *a. Record the indicated reactor power from N41A through N44A on Attachment I. CHG A

☐ *b. Record the Main Control Board indications required on Attachment II.

☐ *c. If blowdown flow is not isolated, read flow on the Integrated Plant Computer System or the Blowdown Panel (XPN0029) and record on Attachment II.

☐ d. Using the data from Attachment II, determine Main Steam, Feedwater, and Blowdown enthalpies using the steam tables and enter the data on Attachment III. CHG D

NOTE 6.2.e
Calorimetric (K) constants for Attachment III are obtained from the Station Curve Book Figure V-16.

*e. Calculate reactor thermal power using Attachment III. CHG A

☐ f. Use Attachment I to determine the meter correction factor and document any adjustments, if required.

Step 6.2 continued

g. If meter adjustments are required/desired proceed as follows: CHG F

NOTE 6.2.g.1)

a. The ROD CONTROL BANK SEL Switch should be placed in MAN prior to adjustment of any power range channel gain.

b. If NIS METER CORRECTION cannot be adjusted with the GAIN potentiometers on the front of the NIS power range drawers, Enclosure A should be used for Coarse Level Correction of affected power range channels.

c. Adjustment of indicated Reactor Power is required if any absolute value is greater than 1%.

SMl 6/5/18

NA

CHG D

☐ 1) If adjustment of the GAIN potentiometers is being performed, place the ROD CONTROL BANK SEL Switch in MAN.

☐ 2) If adjustment is required, slowly adjust the GAIN potentiometer on the front of the affected NIS power range channel drawer one at a time while closely monitoring power range channels.

☐ *3) If adjustment is required, adjust all four channels by their individual Meter Correction Factor and record meter correction in Attachment I.

☐ 4) When adjustment of GAIN potentiometers is complete, place the ROD CONTROL BANK SEL Switch in AUTO.

7.0 DATA REQUIREMENTS

7.1 Data obtained in the performance of this test shall be recorded on the appropriate Attachments and computer outputs should be attached to the STTS sheet.

8.0 ACCEPTANCE CRITERIA

8.1 Each Nuclear Instrumentation System Power Range channel shall be within ±0.5% of the reactor power level calculated per Step 6.1 of this procedure.

8.2 Each Nuclear Instrumentation System Power Range channel shall be within ±1% of the reactor power level calculated per Step 6.2 of this procedure. CHG A

*8.3 Verify Section 8.1 or 8.2 Acceptance Criteria are met and record on Attachment I.

9.0 REFERENCES

9.1 V.C. Summer Tech Specs.

9.2 V.C. Summer Station Curve Book.

CHG D

NIS METER COARSE LEVEL CORRECTION

1.0 INITIAL CONDITIONS

1.1 This enclosure is used when the Gain Potentiometer on the front of the NIS Power Range drawer is out of range for adjustment during an NIS Power Range Heat Balance.

1.2 No other Reactor Trip System instrumentation testing should be in progress while Power Range adjustments are being made.

1.3 All power range bistables must be in their normal state for the present power level.

1.4 The Rod Control System must be in manual prior to adjustment of any power range channel.

CAUTION 2.0

Rate trip bistables may actuate while adjusting R-312, COARSE LEVEL ADJ.

NOTE 2.0

Power range level can be changed ±6% by adjusting the GAIN potentiometers on the front of the NIS Power Range drawers.

2.0 PROCEDURE

2.1 Have I&C perform NIS METER COARSE LEVEL CORRECTION per STP-310.005(6)(7)(8), NIS Power Range N41(42)(43)(44) Calibration.

2.2 Complete the NIS METER CORRECTION adjustments per Attachment I for the remaining Power Range channels.

2.3 Return the Rod Control System to Automatic when NIS METER CORRECTION is complete.

2.4 Forward copies of the completed data sheets and STTS to the following:

a. Reactor Engineering.

b. The Nuclear Instrumentation System Engineer.

******** **NOTE TO EXAMINER: VALUES THAT ARE CIRCLED IN THE ATTACHMENTS INDICATE A CALCULATION ERROR THAT THE CANDIDATE MUST FIND.**

NIS METER CORRECTION

5.2.a Precautions of Section 2.0 have been reviewed. [signature] SM/CRS signature CHG E

5.2.b All Initial Conditions of Section 5.0 have been met. [signature] SM/CRS signature

CALCULATION METHOD (CHECK ONE)
( ) Step 6.1 - IPCS Method
(X) Step 6.2 - MCB Meter Method

1. CALCULATED REACTOR POWER or QCORE1 = 23.9 %

2. INDICATED REACTOR POWER

N-41A	N-42A	N-43A	N-44A
24.6 %	24.0 %	24.6 %	24.3 %

3. Record the following reading from the MCB: CHG E

INSTRUMENT	MCB READING
INI00041B	24.6
INI00042B	24.0
INI00043B	24.6
INI00044B	24.3

4. METER CORRECTION FACTOR (Step 1 minus Step 2)

(INSERT + or - IN PARENTHESIS)

N-41A	N-42A	N-43A	N-44A
(–) 0.7 %	(–) 0.1 %	(–) 0.7 %	(–) 0.4 %

GAIN POTENTIOMETERS (CHECK ONE)
( ) ADJUSTMENT REQUIRED/DESIRED
(X) ADJUSTMENT NOT REQUIRED CHG F

5. NIS METER CORRECTION CHG D

	N-41A	N-42A	N-43A	N-44A
PRESENT INDICATION	%	%	%	%
METER CORRECTION	+ ( ) %	+ ( ) %	+ ( ) %	+ ( ) %
CORRECTED POWER	= %	= %	= %	= %

6 Subtract the reading from Step 1 from the MCB readings recorded in Step 3.

Instrument	MCB Reading from Step 3	Calculated REACTOR POWER or QCORE1 from Step 1	MCB - Calculated =
INI00041B			
INI00042B			
INI00043B			
INI00044B			

CHG E

7. If the difference in the MCB reading minus the calculated value is greater than or equal to 1%, contact I&C to make mechanical zero adjustments.
(CHECK ONE) ( ) ADJUSTMENT REQUIRED/DESIRED
( ) ADJUSTMENT NOT REQUIRED

CHG F

INSTRUMENT	MCB AS LEFT READING
INI00041B	
INI00042B	
INI00043B	
INI00044B	

CHG E

CHG D

8.3 8.1 or 8.2 Acceptance Criteria have been met. ______________ Initials

DATA SHEET

(For Data collection purposes only)

1. NIS POWER RANGE INDICATION

N-41A	N-42A	N-43A	N-44A	Median Tavg
24.6 %	24.0 %	24.6 %	24.3 %	564.6 °F

2. STEAM PRESSURE

PI-474 1055 psig	PI-484 1058 psig	PI-494 1057 psig
PI-475 1060 psig	PI-485 1058 psig	PI-495 1054 psig
PI-476 1060 psig	PI-486 1056 psig	PI-496 1055 psig

3. FEEDWATER TEMPERATURE

SG A (TI-3322)	SG B (TI-3332)	SG C (TI-3342)
322	323	322

4. FEEDWATER FLOW

SG A		SG B		SG C	
FI-477	1.00	FI-487	1.00	FI-497	0.99
FI-476	1.02	FI-486	0.98	FI-496	1.01

5. STEAM GENERATOR BLOWDOWN FLOW

(Integrated Plant Computer System or XPN0029)

(F0407A or FI-4702A)	(F0427A or FI-4702B)	(F0447A or FI-4702C)
25.0 gpm	22.0 gpm	27.0 gpm

6. CVCS LETDOWN FLOW RATE

F0134A1M (IPCS) (preferred) or FI-150 LO PRESS LTDN FLOW GPM	103 gpm

REACTOR POWER CALCULATION

1. NIS Power Range Indication and Median Tavg

N-41A 24.6 % N-42A 24.0 % N-43A 24.6 % N-44A 24.3 % Median Tavg 564.6 °F

2. Feedwater Flow Rate

	FI (MPPH)	FI (MPPH)	Average (MPPH)		K2002		FW Flow Rate (LB/HR)
A Loop	1.02 (FI476)	1.00 (FI477)	1.01	x 1E+03 x	981.868	=	9.917E5 (A Loop FW Flow Rate)
B Loop	0.98 (FI486)	1.00 (FI487)	0.99	x 1E+03 x	981.868	=	9.720E5 (B Loop FW Flow Rate)
C Loop	1.01 (FI496)	0.99 (FI497)	1.00	x 1E+03 x	981.868	=	9.819E5 (C Loop FW Flow Rate)

3. Blowdown Flow Rate

	GPM		K2001 (LB/HR/GPM)		BD Flow Rate (LB/HR)
A Loop	25.0 (FI4702A)	x	370.680	=	9267 (A Loop BD Flow Rate)
B Loop	22.0 (FI4702B)	x	370.680	=	8155 (B Loop BD Flow Rate)
C Loop	27.0 (FI4702C)	x	370.680	=	10008 (C Loop BD Flow Rate)

4. Calculate Main Steam Flow Rate

	FW Flow Rate (LB/HR)		BD Flow Rate (LB/HR)		Steam Flow Rate (LB/HR)
A Loop	9.917E5 (A Loop FW Flow Rate)	-	9267 (A loop BD Flow Rate)	=	9.824E5 (A Loop Steam Flow Rate)
B Loop	9.720E5 (B Loop FW Flow Rate)	-	8155 (B loop BD Flow Rate)	=	9.638E5 (B Loop Steam Flow Rate)
C Loop	9.819E5 (C Loop FW Flow Rate)	-	10008 (C loop BD Flow Rate)	=	9.719E5 (C Loop Steam Flow Rate)

REACTOR POWER CALCULATION

5. Steam Pressure

	PI		PI		PI		Average Pressure		K		Loop Steam Pressure	
A Loop	1055 (PI474)	PSIG	1060 (PI475)	PSIG	1060 (PI476)	PSIG	1058.33 (Average Pressure)	PSIG + 15 +	2.470 (K2040)	=	1075.8 (A Loop Steam Pressure)	PSIA
B Loop	1058 (PI484)	PSIG	1058 (PI485)	PSIG	1056 (PI486)	PSIG	1057.33 (Average Pressure)	PSIG + 15 +	2.700 (K2057)	=	1075.0 (B Loop Steam Pressure)	PSIA
C Loop	1057 (PI494)	PSIG	1054 (PI495)	PSIG	1055 (PI496)	PSIG	1055.33 (Average Pressure)	PSIG + 15 +	2.270 (K2000)	=	1072.6 (C Loop Steam Pressure)	PSIA

TI3342

CHG D

6. Steam Enthalpy (From Step 5)

CHG D

A Loop 1075.8 PSIA (A Loop Steam Pressure) — Enthalpy of Saturated Steam 1189.6 BTU/LB (A Loop Steam Enthalpy)

B Loop 1075.0 PSIA (B Loop Steam Pressure) — Enthalpy of Saturated Steam 1189.6 BTU/LB (B Loop Steam Enthalpy)

C Loop 1072.6 PSIA (C Loop Steam Pressure) — Enthalpy of Saturated Steam 1189.7 BTU/LB (C Loop Steam Enthalpy)

7. Feedwater Temperature

A Loop 322 °F (TI3322)

B Loop 323 °F (TI3332)

C Loop 322 °F (TI3342)

Corrected transposition errors

REACTOR POWER CALCULATION

8. Feedwater Enthalpy

A Loop 322 °F (A Loop Feedwater Temp) 1075.8 PSIA (A Loop Steam Pressure) Enthalpy of Feedwater 292.52 BTU/LB (A Loop Feedwater Enthalpy)

290.4 - 294.6

B Loop 323 °F (B Loop Feedwater Temp) 1075.0 PSIA (B Loop Steam Pressure) Enthalpy of Feedwater 293.56 BTU/LB (B Loop Feedwater Enthalpy)

290.4 - 294.6

C Loop 322 °F (C Loop Feedwater Temp) 1072.6 PSIA (C Loop Steam Pressure) Enthalpy of Feedwater 292.52 BTU/LB (C Loop Feedwater Enthalpy)

9. Steam Generator Enthalpy Rise

A Loop 1189.6 BTU/LB (A Loop Steam Enthalpy) - 292.52 BTU/LB (A Loop Feedwater Enthalpy) = 897.08 BTU/LB (A Loop Delta H) **(897-898)**

B Loop 1189.6 BTU/LB (B Loop Steam Enthalpy) - 293.56 BTU/LB (B Loop Feedwater Enthalpy) = 896.04 BTU/LB (B Loop Delta H) **(896-897)**

C Loop 1189.7 BTU/LB (C Loop Steam Enthalpy) - 292.52 BTU/LB (C Loop Feedwater Enthalpy) = 897.18 BTU/LB (C Loop Delta H) **(897-898)**

10. Blowdown Heat Loss

Total Blowdown Flow Rate = 9267 LB/HR (A Loop BD Flow Rate) + 8155 LB/HR (B Loop BD Flow Rate) + 10008 LB/HR (C Loop BD Flow Rate) = 27430 LB/HR (Total Blowdown Flow Rate)

Average Feedwater Enthalpy =(292.52 BTU/LB (A Loop Feedwater Enthalpy) + 293.56 BTU/LB (B Loop Feedwater Enthalpy) + 292.52 BTU/LB (C Loop Feedwater Enthalpy))÷ 3 = 292.86 BTU/LB (Average FW Enthalpy) **(292-293)**

Q_{BD} = 27430 LB/HR (Total Blowdwon Flow Rate) x (544.670 BTU/LB (K2025) - 292.86 BTU/LB (Average FW Enthalpy)) x 2.931 E-07 = 2.024 MWT (Blowdown Heat Loss)

Corrections carried forward from steps 9 and 10.

REACTOR POWER CALCULATION

11. Steam Generator Power

A Loop (9.824E5 LB/HR x 897.08 BTU/LB x 2.931 E-07) + (2.024 ÷ 3) = 258.98 MWT
(A Loop Steam Flow Rate; A Loop Delta H **(897-898)**; Blowdown Heat Loss; A Loop SG Power **(258.5 - 259.5)**)

B Loop (9.638E5 LB/HR x 896.04 BTU/LB x 2.931 E-07) + (2.024 ÷ 3) = 253.79 MWT
(B Loop Steam Flow Rate; B Loop Delta H **(896-897)**; Blowdown Heat Loss; B Loop SG Power **(253.25 - 254.25)**)

C Loop (9.719E5 LB/HR x 897.18 BTU/LB x 2.931 E-07) + (2.024 ÷ 3) = 2.56.25 MWT
(C Loop Steam Flow Rate; C Loop Delta H **(897-898)**; Blowdown Heat Loss; C Loop SG Power **(255.75 - 256.75)**)

12. Adjustments To Reactor Power

NSSS Insulation Heat Loss = 3.189 MWT (K2010)

Correct value for K2011.

CVCS Letdown Heat Loss = 0.027 (K2011) x 103 (FI150) GPM = 2.781 (Letdown Heat Loss) MWT

RCP Heat Into RCS = 14.757 MWT (K2008)

Pressurizer Heater Heat Into RCS = 0.477 MWT (K2009)

13. Calculation of Reactor Power, Qcore1

Total Loop Power = 258.98 MWT + 253.79 MWT + 256.25 MWT = 769.02 MWT
(A Loop SG Power (258.5 to 259.5); B Loop SG Power (253.25 to 254.25); C Loop SG Power (255.75 to 256.75); Total Loop Power (768 to 770.5))

Reactor Power, Qcore1 = 769.02 MWT + 3.819 MWT + 2.781 MWT - 14.757 MWT - 0.477 MWT
(Total Loop Power **(768 - 770.5)**; K2010; Letdown Heat Loss; K2008; K2009)

= 760.36 MWT ÷ 29 = 26.22 %
(Reactor Power, Qcore1 **(759 - 761)**)

CRITICAL Acceptable range for recalculated value is 25.7% to 26.7%.

Document Review Form (DRF)

Section I	**Document Identification**	**Page 1 of 2**

Preparer Name: Larry Haley **Ext:** 54047 **Mail Code** 410

Date: 2/25/2016 **Document #:** STP-102.002 **Revision:** 8 **Change** F

Title: NIS Power Range heat Balance ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process: ☐ New ☒ Revision/Change ☐ Editorial Correction ☐ Temporary Approval

Description: See Attached

ISFSI Related? ☐ Yes ☒ No
Has scope changed? ☐ Yes ☒ No [If YES, attach 50.59 and/or 72.48 documentation]

Reason/Basis for Revision/Change: See Attached

Temporary Approval – if final approval is not completed within 30 days; initiate CR #______________

Qualified Reviewer ______ **DCRM person notified** ______ **Shift Manager** ______ **Date** ______

Section II **List Required Reviewers** including All Impacted Groups
Additional Reviewers – identify with an *

Position	Type/Print Name	Comments	Position	Type/Print Name	Comments
QR	ALLEN WILLIAMS	☐ Yes ☒ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No

A

[signature] 2/25/16
Designated Supervisor Concurrence **Date**

Comment Due Date 3/17/16
GM concurrence________For expedited review on documents that require PSRC review.

D

Section III **Pre- implementation Actions**

All Comments Resolved? ☒ **NA** ☐ **YES** [signature] 3/21/2016
Preparer Sign **Date**

50.59 and/or 72.48 Review Requirements Addressed?	☒ NA ☐ YES	Attached? YES ☐ No ☐
50.59/Part 52 Review Requirements Addressed?	☒ NA ☐ YES	Attached? YES ☐ No ☐
Commitments (PCAP and MLSA) Addressed?	☒ NA ☐ YES	PCAP # ______ ______ NL Initial/Date
QR Qualification Verified?	☒ YES	
Security Compliance Review Completed?	☒ NA ☐ YES	
Pre-Implementation Training Completed?	☒ NA ☐ YES	
Training required after implementation?	☒ NA ☐ YES	CR#______
PSRC Review Completed?	☒ NA ☐ YES	Mtg. #______
NSRC Review Completed?	☒ NA ☐ YES	Mtg. #______
CMMS Update Required? [Unit 1]	☒ NA ☐ YES	Planner Notified YES ☐______

[signature] 5/9/16
Designated Supervisor Approval **Date**

[signature] 5-10-16
Approval Authority Approval **Date**
Effective Date:______________

DRF Form (Continued)

Page 2 of 2

DOCUMENT # STP-102.002 Rev. 8 Chg. F

DESCRIPTION CONTINUED:

Attachment I and Steps 6.1.g. and 6.2.g, added desired for clarification that ajustments may not be required but may be performed if desired.

REASON/BASIS CONTINUED:

FB 150808 (J Geib) Identified there are times when NIS adjustments are not required but are performed. Change was to clarify the requirement that adjustments could be made at discretion of Control Room Personnel.

REQUIRED REVIEWERS CONTINUED:

Position	Type/Print Name	Comments	Position	Type/Print Name	Comments
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No

Document Review Form (DRF)

Section I	**Document Identification**	**Page 1 of _____**
Preparer Name: Larry Haley	**Ext:** 54047	**Mail Code** 410
Date: 2/26/2015 **Document #:** STP-102.002	**Revision:** 8	**Change** E
Title: NIS Power Range Heat Balance		☒ **SR** ☐ **QR** ☐ **NNS**

Development Process: ☐ New ☒ Revision/Change ☐ Editorial Correction ☐ Temporary Approval

Description: See Attached

ISFSI Related? ☐ Yes ☒ No
Has scope changed? ☒ Yes ☐ No [If YES, attach 50.59 and/or 72.48 documentation]

Reason/Basis for Revision/Change: See Attached

Temporary Approval – if final approval is not completed within 30 days; initiate CR #_______________

Qualified Reviewer	**DCRM person notified**	**Shift Supervisor**	**Date**

Section II **List Required Reviewers** including All Impacted Groups
Additional Reviewers – identify with an *

Position	**Type/Print Name**	**Comments**	**Position**	**Type/Print Name**	**Comments**
QR	ELVIS ZIMMERMAN	☐ Yes ☑ No			☐ Yes ☐ No
I+C	J. Kennerly	☑ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No

[signature] 7/24/15
Designated Supervisor Concurrence **Date**

Comment Due Date 8/14/15
GM concurrence________for expedited review

Section III **Pre- implementation Actions**

All Comments Resolved? ☐ **NA** ☒ **YES** Larry Haley 8-13-2015
Preparer Sign **Date**

50.59 and/or 72.48 Review Requirements Addressed?	☐ NA ☑ YES	Attached? YES ☑ No ☐
50.59/Part 52 Review Requirements Addressed?	☑ NA ☐ YES	Attached? YES ☐ No ☐
Commitments (PCAP and MLSA) Addressed?	☑ NA ☐ YES	PCAP # _______ __________ NL Initial/Date
QR Qualification Verified?	☑ YES	
Security Compliance Review Completed?	☑ NA ☐ YES	
Pre-Implementation Training Completed?	☑ NA ☐ YES	
Training required after implementation?	☑ NA ☐ YES	CR#______________
PSRC Review Completed?	☑ NA ☐ YES	Mtg. #______________
NSRC Review Completed?	☑ NA ☐ YES	Mtg. #______________
CMMS Update Required? [Unit 1]	☑ NA ☐ YES	Planner Notified YES ☐________

[signature] 8/20/15
Designated Supervisor Approval **Date**

[signature] 8/31/15
Approval Authority Approval **Date**
Effective Date:__________________

DRF Form (Continued)

Page 2 of 2

DOCUMENT # STP-102.002 Rev. 8 Chg. E

DESCRIPTION CONTINUED:

1. Added steps to Att I to record MCB NIS readings and to calculate error from the calculated power reading. If there is a difference of greater than or equal to 1%, I&C is to make adjustments. Added Step 6.1.e to record MCB reading on ATT 1, and added step d to note 6.1.g.1/ step 6.1.g.5) to identify actions if MCB is out of tolerance.

2. – Made clarification of precaution 2.2 for adjusting NI

3. Added 10CFR72.48 requirement to section 1.0, step 1.3

REASON/BASIS CONTINUED:

1. - CR-13-04956, CR-14-01170, Feedback 130510, are issues with drift of MCB NI reading from cabinet readings. Steps are provided to adjust the meters to keep them within Spec

2. - Feedback 130524- Clarification of step 2.2.

3. Identified 10CFR72.48 applicability as it applies to this procedure.

REQUIRED REVIEWERS CONTINUED:

Position	Type/Print Name	Comments	Position	Type/Print Name	Comments
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No
		☐ Yes ☐ No			☐ Yes ☐ No

DOCUMENT REVIEW FORM

Document Identification

Originators Name: Patrick Shields **Ext:** 54699 **Mail Code:** 410

Date: 5-31-11 **Document No.:** STP-102.002 **Revision No.:** 8 **Change Letter:** D

Title: NIS POWER RANGE HEAT BALANCE ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction ☐ Temporary Approval

Description: See page 2.

Reason/Basis for Change: See page 2.

Is the SCOPE of the procedure affected by this change? NO ☒ **YES** ☐

Temporary Approval **Final approval required by: (30 days)**

QR	**DC&R** (Person Notified)	**SS**	**Date**
			/

Document Reviewers (Enclosure C)

	Position	Type/Print Name	Comments Yes/No		Position	Type/Print Name	Comments Yes/No
Required	QR	Purcell	☐ ☑	***Additional**		W H Bell	☐ ☑
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐

Concurrence 6-15-11
Supervisor/Date or Enter CR # ________ (per 6.4.8.C)

Comment Due Date 6/20/2011

Pre- implementation Actions

All Comments Resolved ☑ None Received ☐ Yes 6/21/11
Originator/Date

Commitments Addressed per SAP-0630 ☒ NA ☐ Yes P/CAP # ______ ☐ MLSA ______ Initial/Date

50.59 Applicability/Review Completed (SAP-0107) ☒ NA ☐ Yes, Attached
Security Compliance Review Completed (SAP-0163) ☒ NA ☐ Yes (Security review required)
Pre-implementation Training Completed ☒ NA ☐ Yes
Training required after implementation ☒ NA ☐ Yes, CR # ________
PSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______
NSRC Review Completed ☒ NA ☐ Yes, Mtg. No. ______
CMMS Update Required ☒ NA ☐ Yes Planner Notified ________ Initial/Date

Supervisor/Date 2/10/12

Approval Authority/Date 2/13/12

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document No.: STP-102.002 Rev. No. 8 Chg. Ltr. D

DESCRIPTION CONTINUED:

1. Removed signature lines from cover page.
2. Added placekeeping boxes to sections 5 & 6.
3. Add step continued lines.
4. Page 3, Step 6.1.f Added steps for adjusting NIs from Att I to body of procedure. Split out Note 6.1.f.1) c to identify the 0.5% difference and adjustment for IPCS calc.
5. Deleted old step 6.2.d and modified new step 6.2.d to read : Using the data from Attachment II....enter data on Attachment III.
6. Page 5, Step 6.2.g Added steps for adjusting NIs from Att I to body of procedure. Split out Note 6.2.h.1) c to identify the 1.0% difference and adjustment for manual calc.
7. Removed revision summary section 10 and in TOC.
8. Attachment I, Moved steps for NI gain adjust and placed in body of procedure.
9. Attachment II, Deleted entries requiring conversions to enthalpies. Attachment II is for data collection only.
10. Att III, page 2:
    a. Step 5. Steam Press calc, changed 14.6 to 15.
    b. Swapped steps 6 and 7, New step 6.(Steam enthalpy) now refers to using corrected Steam pressures from step 5.

REASON/BASIS FOR CHANGE CONTINUED:

1. FB90401 and Writers Guide.
2. FB 90401
3. FB90401
4. PF07093 (Edwards), Should consider placing NI adjustment procedure in body of procedure to ensure more positive control of the procedure.
5. PF07023 (Crawford), Lessen confusion about purpose of Attachment II. (Per B Bell LEFM engineer) and how it relates to Attachment III.
6. PF07093 (Edwards), Should consider placing NI adjustment procedure in body of procedure to ensure more positive control of the procedure.
7. Writers guide.
8. PF07093 (Edwards), Should consider placing NI adjustment procedure in body of procedure to ensure more positive control of the procedure.
9. PF07023 (Crawford), Lessen confusion about purpose of Attachment II. (Per B Bell LEFM engineer) and how it relates to Attachment III.
10. a. FB 5164 (Stokes), For consistency and inability to read MCB indicators to the 10^{th} use 15 for conversion factor.
    b. PF07023 (Crawford) Clarify on step 7 (new 6) which steam enthalpy to use, and if using corrected steam pressure from step 5

* Failure by the “Additional Reviewers” to provide comments within 5 working days following the comment due date may be considered as “No Comment”.

NUCLEAR OPERATIONS
COPY NO

DOCUMENT REVIEW FORM

Page 1 of 1

Document Identification

Originators Name: Eric L. Erickson **Ext:** 55666 **Mail Code:** 410

Date: 06/16/05 **Document No.:** STP-~~012~~ 102 .002 **Revision No.:** 8 **Change Letter:** C

Title: NIS Power Range Heat Balance ☒ SR ☐ QR ☐ NNS

Development Process:
Permanent: (check one) ☐ Normal Rev/Chg or ☒ Editorial Correction
☐ Temporary Approval ☐ Restricted Chg (expires: ______)

Description: Corrected step numbering typos on Pages 2 and 3.
Corrected typo on Att I

Procedure SCOPE is Unaffected.

Reason/ Basis for Change: Feedbacks

Temporary Approval **Final approval required by: (30 days)**

QR	DC&R (Person Notified)	SS	Date

Document Reviewers (Enclosure C)

	Position	Type/Print Name	Comments Yes/No		Position	Type/Print Name	Comments Yes/No
Required	QR	Fenstermacher	☐ ☒	Additional			☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐

Concurrence DAB 6-16-05
Discipline Supervisor/ Date

Comment Due Date N/A

Pre- implementation Actions

All Comments Resolved ☒ Yes DAB 6-16-05
Originator/ Date

Commitments Addressed per SAP-0630	☒ NA	☐ Yes, P/CAP # ______ ☐ MLSA
50.59 Applicability/ Review Completed (SAP0107)	☒ NA	☐ Yes, Attached
Pre-implementation Training Completed	☒ NA	☐ Yes
Training required after implementation	☒ NA	☐ Yes, CER # ______
PSRC Review Completed	☒ NA	☐ Yes, Mtg. No. ______
NSRC Review Completed	☒ NA	☐ Yes, Mtg. No. ______
Other: ______	☒ NA	☐ Yes

DAB 6-16-05
Discipline Supervisor / Date

N/A
Approval Authority / Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Page 1 of 1

Document Identification

Originators Name: MD Johnson **Ext.:** 54300 **Mail Code:** 410

Date: 6/6/2005 **Document No.:** STP-102.002 **Revision No.:** 8 **Change Letter:** B

Title: NIS POWER RANGE HEAT BALANCE ☒ **SR** ☐ **QR** ☐ **NNS**

Development Process:
Permanent: (check one) ☒ Normal Rev/Chg or ☐ Editorial Correction
☐ Temporary Approval ☐ Restricted Chg (expires: ______________)

Description: Change B Summary, ECR50518 (IPCS Update) necessitated procedure enhancements to collect data.

Reason/ Basis for Change: ECR50518
SCOPE NOT AFFECTED.

Temporary Approval **Final approval required by:**
QR ______________ **DC&R** ______________ **SS** __________/_____ **(30 days)** ___/___/___
Person Notified Date

Document Reviewers (Enclosure C)

	Position	Type/ Print Name	Comments Yes/No		Position	Type/ Print Name	Comments Yes/No
Required	QR	H Anderson	☒ ☐	Additional			☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐

Concurrence
DAB 6-6-05
Discipline Supervisor/ Date

Comment Due Date
ASAP

Pre- implementation Actions

All Comments Resolved	☒ Yes	[signature] Originator/ Date
Commitments Addressed per SAP-0630	☒ NA	☐ Yes, PICAP # ______ ☐ MLSA
50.59 Applicability/ Review Completed (SAP0107)	☒ NA	☐ Yes, Attached
Pre-implementation Training Completed	☒ NA	☐ Yes
Training required after implementation	☒ NA	☐ Yes, CER # ________
PSRC Review Completed	☒ NA	☐ Yes, Mtg. No. ________
NSRC Review Completed	☒ NA	☐ Yes, Mtg. No. ________
Other: ____________	☒ NA	☐ Yes

[signature] 6/6/05
Discipline Supervisor / Date

[signature] 6/6/05
Approval Authority / Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

DOCUMENT REVIEW FORM

Document Identification

Originators Name: LATHREN **Ext.:** 55547 **Mail Code:** 410

Date: 9/2/4 **Document No.:** STP-102.002 **Revision No.:** 8 **Change Letter:** A

Title: NIS POWER RANGE HEAT BALANCE

X SR/ QR
☐ NNS

Development Process:
X Rev/Chg ☐ Temporary Approval ☐ Restricted Chg (expires: ________) ☐ Editorial Correction

Description: UPDATED FORMAT, ENHANCED ATTACHMENTS, REVISED POWER ADJUSTMENT CRITERIA, AND ADDED SCOPE.

Reason/ Basis for Change: PROCEDURE FEEDBACKS FOR BETTER REACTIVITY MANAGEMENT.

Temporary Approval QR ________ **DC&R** ________ QA ________ person notified **person notified** SS ________ / ________ Date **Final approval required by: (30 days)** __/__/__

Document Reviewers (Enclosure C)

	Position	Type/ Print Name	Comments Yes/No		Position	Type/ Print Name	Comments Yes/No
Required	QR	T. Phillips ~~B. Ervin~~ 12-7-04	☒ ☐	Additional			☐ ☐
	QA	Hicks	☒ ☐				☐ ☐
	RE	B. Bell	☒ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐
			☐ ☐				☐ ☐

Concurrence
DAB 12.7-04
Discipline Supervisor/ Date

Comment Due Date *
12-~~13~~-04 15
(8 days)

Pre- implementation Actions

All Comments Resolved		☒ Yes
Commitments Addressed per SAP-0630	☒ NA	☐ Yes, P/CAP # ______ ☐ MLSA
50.59 Applicability/ Review Completed (SAP0107)	☐ NA	☒ Yes, Attached
Pre-implementation Training Completed	☐ NA	☒ Yes
Training required after implementation	☒ NA	☐ Yes, CER #________
PSRC Review Completed	☒ NA	☐ Yes, Mtg. No.______
NSRC Review Completed	☒ NA	☐ Yes, Mtg. No.______
Other: ____________	☒ NA	☐ Yes

DAB 1-10-05
Discipline Supervisor / Date

2/22/05
Approval Authority / Date

* Failure by the "Additional Reviewers" to provide comments within 5 working days following the comment due date may be considered as "No Comment".

PROCEDURE CONVERSION FORM

DATE: 11-4-98 PROCEDURE # STP-102.002 REVISION # 8

SAFETY CLASSIFICATION: Safety Related

TITLE: NIS Power Range Heat Balance

This document has been converted from the Xerox Globalview Format to the Microsoft Word for Windows Format. **No changes of content have been made (format only).**

[signature], 11-4-98
Converter Date

This procedure has been reviewed for accuracy and completeness against the Master Control Copy.

[signature], 11-5-98
Qualified Reviewer Date

[signature], 11-6-98
Discipline Supervisor Date

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-135-(S)N18**

2018 NRC A1-b (SRO)

CANDIDATE: ______________________________

EXAMINER: ______________________________

DESCRIPTION:	Determine reportability requirements.
TASK:	O-341-013-03-02 Report safety limit violations and Reportable occurrences per NL-122
TASK STANDARD:	Candidate determines that a reporting requirement in accordance with item NL-122, Enclosure A, item F-1 is required.
TERMINATING CUE:	The candidate records the reportability requirement determined for the event on the briefing sheet and returns the briefing sheet to examiner.

PREFERRED LOCATION:	***PREFERRED METHOD:***
CLASSROOM	PERFORM

REFERENCES:

NL-122, REGULATORY NOTIFICATION AND REPORTING

K/A 2.1.18 Ability to make accurate, clear, and concise logs, records, status boards, and reports.(SRO 3.8)

10CFR55: 45 b(13)

TOOLS: Access to paper or electronic copies of V.C. Summer procedures

EVALUATION TIME: 15 Minutes. ***TIME CRITICAL:*** NO

TIME START: __________ ***TIME FINISH:*** __________ ***PERFORMANCE TIME:*** __________

CANDIDATE: ______________________________

EXAMINER: ______________________________

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- On June 4th at 0900, “A” and “C” Service Water Pumps were tagged out due to common mode mechanical problems.
- The arrival of parts which will allow repair has been significantly delayed.
- The time is now 1300 on June 7th.
- The crew is now reducing power in accordance with GOP-4B, POWER OPERATION (MODE 1 - DESCENDING).
- Power is 30% and decreasing.

INITIATING CUES:

You are directed to determine the **soonest** that a regulating agency must be notified to satisfy station reporting requirements and record that requirement on your briefing sheet.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: YES	SEQUENCED: YES	SAT		UNSAT	

JPM STEP 1

Refers to NL-122, REGULATORY NOTIFICATION AND REPORTING.

STEP STANDARD:

Refers to either a hard-copy or electronic controlled copy of NL-122.

CUES:

Note(s) to examiner:

Ensure that candidates record their names on their briefing sheets.

COMMENTS:

CRITICAL: YES	SEQUENCED: YES	SAT		UNSAT	

JPM STEP 2

Determines the reporting item that is required.

STEP STANDARD:

Determines that reporting is required in accordance with NL-122, Enclosure A, item F-1 due to a shutdown reactivity insertion required by Technical Specifications.

CUES:

The determination from memory that a shutdown is required by Technical Specifications is acceptable and **reference to Technical Specifications in not required**.

If the candidate states from memory that a 4 hour reporting requirement is required, use follow up questioning to ensure the proper rationale. Stating the 4 hour requirement for notification without identifying the correct reporting requirement in accordance with NL-122 does **not** meet the JPM standard.

COMMENTS:

JPM SETUP SHEET

JPM: JPM: JPA-135-(S)N18

IC SET: N/A

INSTRUCTIONS:

Ensure candidates have access to hard-copy or electronic procedures.

COMMENTS:

JPM BRIEFING SHEET

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues. I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task. Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- On June 4th at 0900, “A” and “C” Service Water Pumps were tagged out due to common mode mechanical problems.
- The arrival of parts which will allow repair has been significantly delayed.
- The time is now 1300 on June 7th.
- The crew is now reducing power in accordance with GOP-4B, POWER OPERATION (MODE 1 - DESCENDING).
- Power is 30% and decreasing.

INITIATING CUES:

You are directed to determine the **soonest** that a regulating agency must be notified to satisfy station reporting requirements and record that requirement on your briefing sheet.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-040A-(S)N18**

2018 NRC A2 (SRO)

CANDIDATE: ______________________________

EXAMINER: ______________________________

DESCRIPTION: Review a tagout prior to approval.

TASK: O-342-005-03-02 Authorize hanging of tags on plant equipment (SAP-201)

TASK STANDARD: Indentifies errors on the proposed tagout including the omission of a power supply breaker, the omission of venting and an improper sequence of tag installation.

TERMINATING CUE: The candidate returns JPM briefing sheet and Handout materials provided to the examiner.

PREFERRED LOCATION:	***PREFERRED METHOD:***
CLASSROOM	PERFORM

REFERENCES:
SAP-0201, EQUIPMENT TAGGING AND LOCKOUT-TAGOUT.
OAP-100.5, GUIDELINES FOR CONFIGURATION CONTROL AND OPERATION OF PLANT EQUIPMENT
SOP-118, COMPONENT COOLING WATER
DRAWING 302-611

K/A 2.2.13 Knowledge of tagging and clearance procedures.(SRO 4.3)

10CFR55: 45 b(1)(13)

TOOLS:
Copies of 2018 NRC A2 SRO Only Handouts 1 and 2
Marked up excerpt of 302-611 with proposed mechanical isolation.
Electronic or Hardcopy versions of station procedures

EVALUATION TIME: 20 Minutes. ***TIME CRITICAL:*** NO

TIME START: ____________ ***TIME FINISH:*** ___________ ***PERFORMANCE TIME:*** __________

CANDIDATE: ____

EXAMINER: ____

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is in Mode 1.
- Mechanical Maintenance has requested a tagout to repair the outboard bearing and pump shaft on “C” CCW Pump.
- “A” and “B” CCW pumps will remain in service.
- A proposed tagout to perform this work is in “PREPARED” status awaiting review.

INITIATING CUES:

You are to review the tagout sheet provided in Handout 1 and record any and all discrepancies that will require correction to advance the tagout enclosure to the “REVIEWED” state.

If discrepancies do exist, circle the discrepancy on Handout 1 and fully explain the nature of the discrepancy on Handout 2.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: NO	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 1

Reviews SAP-0201, EQUIPMENT TAGGING AND LOCKOUT-TAGOUT and OAP-105, GUIDELINES FOR CONFIGURATION CONTROL AND OPERATION OF PLANT EQUIPMENT.

STEP STANDARD:

Reviews the guidance contained in SAP-201, step 6.1.3 and OAP-100.5, step 9.1.

CUES:

Evaluator note: Provide candidate with the 2018 NRC A2 RO Handouts 1 and 2.

SAP-201 and OAP-100.5 are classified as "INFORMATION USE". These procedures may be performed from memory; however, the user retains accountability for proper performance.

NOTE: The steps in this JPM do not need to be performed in order.

COMMENTS:

CRITICAL: YES	SEQUENCED: NO	SAT		UNSAT	

JPM STEP 2

Examinee reviews the tagout contained in handout 1.

STEP STANDARD:

Identifies the following errors:

- Power supply breaker from 7.2 KV Bus 1DB has not been included.
- There are no vent valves included within the proposed tagout boundary.
- The sequence of valve tagging is not correct.

CUES:

Evaluator notes:

The following technically specific items are not required to satisfy the task standard. They are, however, normally specified for a tagout of “C” CCW pump and are included here for reference if required during follow-up questioning:

- Breaker XSW1DB 11 Breaker Open/Racked down.
- At least one Vent valve included (XVT(s) 19549, 19550 and 19551) OPEN and UNCAPPED.
- Suction valve XVB09518C-CC closed after the discharge valve XVB09501C-CC and before venting and draining.

COMMENTS:

JPM SETUP SHEET

JPM: JPA-040A-(S)N18

IC SET: N/A

INSTRUCTIONS:

Provide:

Handout 1 – LOTO Tagout enclosure for “C” CCW Pump without 1DB power supply breaker or vent valves included and the drain valves specified open before suction valve is tagged closed.

Handout 2 that includes the following:

Name: ______________________________________ Date: ____________________
IF DISCREPANCIES WERE FOUND, FULLY EXPLAIN THEN BELOW:

Marked up excerpt of drawing 302-611 with suction valve, discharge valve and drain valves identified as part of isolation boundary.

Ensure that current procedures are available in hard copy or electronically.

COMMENTS:

JPM BRIEFING SHEET

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is in Mode 1.
- Mechanical Maintenance has requested a tagout to repair the outboard bearing and pump shaft on "C" CCW Pump.
- "A" and "B" CCW pumps will remain in service.
- A proposed tagout to perform this work is in "PREPARED" status awaiting review.

INITIATING CUES:

You are to review the tagout sheet provided in Handout 1 and record any and all discrepancies that will require correction to advance the tagout enclosure to the "REVIEWED" state.

If discrepancies do exist, circle the discrepancy on Handout 1 and fully explain the nature of the discrepancy on Handout 2.

Hand this paper back to your Evaluator when you feel that you have satisfactorily completed the assigned task.

VC Summer Nuclear Station	TAGOUT	Tagout ID: 18-1264
Hang 18 -1264-1	Status: **Prepared**	Page 1 of 1
Authorized By: FOR INFORMATION ONLY		

Enclosure Name:	XPP0001C (COMPONENT COOLING PUMP C)
Enclosure Purpose:	REPAIR / REPLACE OB BEARING AND PUMP SHAFT AS REQUIRED.
Enclosure Comments:	ENSURE XPP0001C IS REMOVED FROM SERVICE PER SOP-118 PRIOR TO PERFORMING THIS ENCLOSURE. ALIGN XPP0001C TO THE MAINTENANCE TRAIN ('A' TRAIN) FOR PROMPT RESTORATION AND RETEST WHEN TAGS ARE CLEARED.
Enclosure Start Date/Time:	____________________ / _________

Seq / Tag	Component	Position	Sign-off
Seq: 1.0 Red Tag 1805416	XSW1DA 07 CC PUMP C XPP0001C-CC IB-463-G-09	Breaker Open/ Racked Down	Hung By: Verified By: Hold Tag Inst:
Seq: 2.0 Red Tag 1805417	XVB09501C-CC COMPONENT COOLING PUMP C DISCHARGE VLV IB-412-H-05	Closed	Hung By: Verified By: Hold Tag Inst:
THIS VALVE REQUIRES A LOCKED VALVE TRACKING SHEET.			
Seq: 3.0 Procedure Step	See comment Perform guidance specified in the comment below.	Complete	Performed By: Verified By:
DRAIN THE 'C' COMPONENT COOLING PUMP PER SOP-118 SECTION IV.A.			
Seq: 4.0 Red Tag 1805418	XVT19556-CC CC PUMP C DISCHARGE HEADER DRAIN VALVE IB-412-H-05	Open/ Uncapped	Hung By: Verified By: Hold Tag Inst:
Seq: 4.0 Red Tag 1805419	XVT19560-CC CC PUMP C SUCTION HEADER DRAIN VALVE IB-412-H-05	Open/ Uncapped	Hung By: Verified By: Hold Tag Inst:
Seq: 5.0 Red Tag 1805420	XVB09518C-CC COMPONENT COOLING PUMP C SUCTION VALVE IB-412-H-05	Closed	Hung By: Verified By: Hold Tag Inst:
THIS VALVE REQUIRES A LOCKED VALVE TRACKING SHEET.			

Enclosure Completion Date/Time: ____________________ / _________ Recorded in LOTO: _____

Enclosure Completion Notes:

Enclosure Performed By:	
User ID	Initials

VC Summer Nuclear Station	TAGOUT	Tagout ID: 18-1264
Tagout Details	Status: **Prepared**	Page 1 of 1

System:	System Name:	Train:	Outage:
CC	COMPONENT COOLING	X	

Reason For Tagout:
REPAIR / REPLACE OB BEARING AND PUMP SHAFT AS REQUIRED.

Tagout Comments:
DWGS: D-302-611

Impact Section

Safety Impacts:	Safety Impact Comments:	Reviewed By:
Caution/Notes	Recommend aligning 'C' CCW pump to 'A' train in preparation for testing.	
EOOS	This task impacts EOOS Risk: GREEN. This task impacts FEP Equipment Availability Risk: YELLOW (30-day RMA's.) RxM is not impacted.	
Fire Protection	30-day RMA's	
FME	STANDARD, as defined by SAP-363	
Operations Retest	STP-222.002	
Removal & Restoration	R&R # 170646	
S/RWP	Required	
Security Review	SAP-163 Security Compliance Review: Exclude	
Technical Specifications	3.7.3, OAP-100.5 Encl. Q	
WPO	WPO # 10-35	

Review Section

Tagout Prepared By:	Tagout Reviewed By:	Tagout Approved By:
DENNIS SMITH 6/3/2018 10:15		

Out Of Service Section

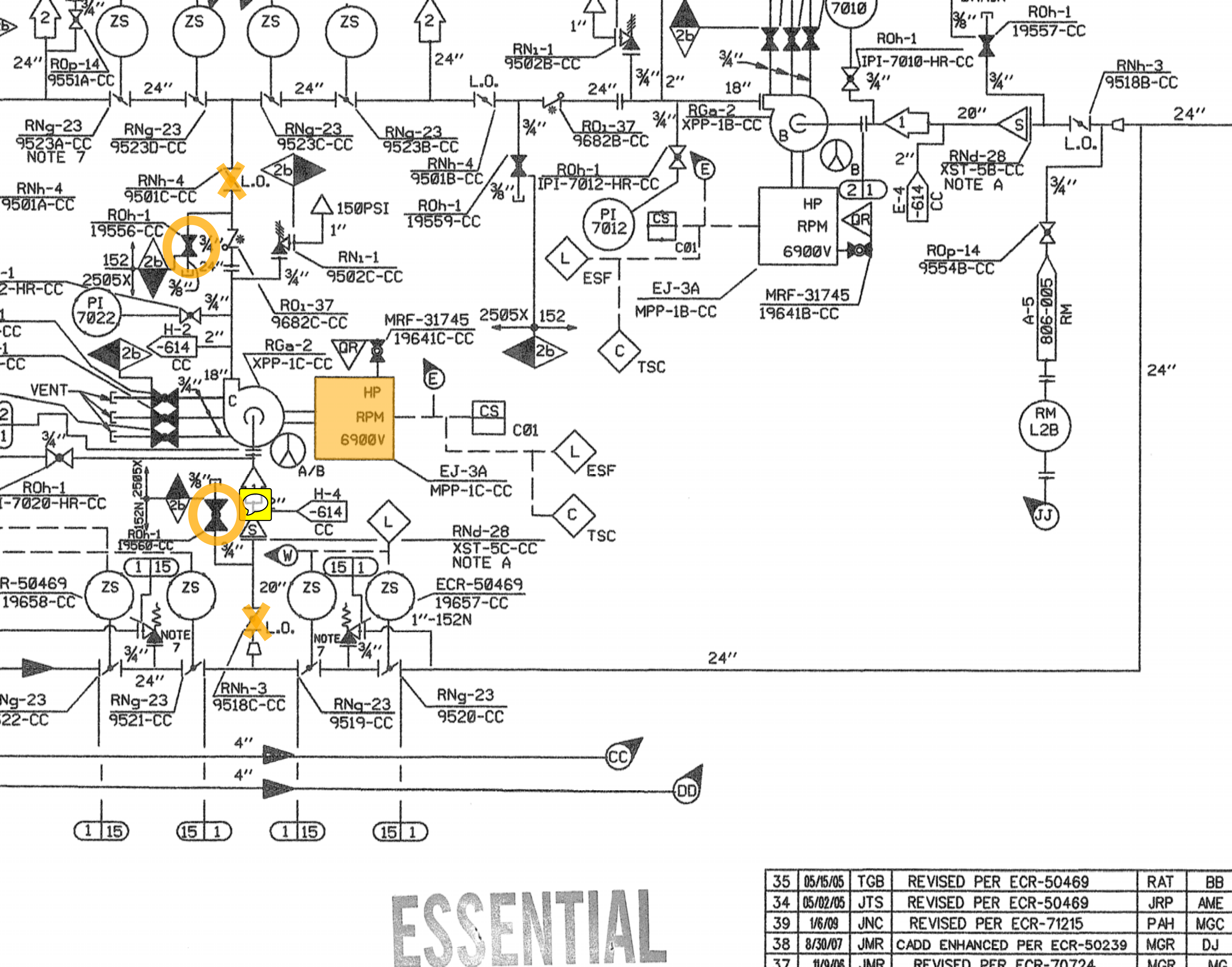
Tagout Hung Date/Time	Tagout Cleared Date/Time	Total Time Out Of Service

Enclosure Section

Enclosure Type	ID	Enclosure Name	Authorized By
Hang	1	XPP0001C (COMPONENT COOLING PUMP C)	
Clear	999	XPP0001C (COMPONENT COOLING PUMP C)	

Work Order Section

Work Order #	Step #	EQ/Device ID	Crew	Start Date/Time	Clearance Authorized By
1812563	001	XPP0001C	MM		
1812563	003	XPP0001C	MI		
1812563	008	XPP0001C	MI		
1812563	001	ITE07024	MI		



G

NOTES:

1. ALL NUC. SAFETY CLASS PIPING TO BE LINE SPEC 152N.
2. ALL NON-NUC. SAFETY CLASS PIPING EXCEPT THAT SUPPLIED BY CHEM. INJ. SYSTEM VENDOR, OR AS INDICATED OTHERWISE, TO BE LINE SPEC. 151X.
3. * DENOTES MISSION "DUO-CHEK" OR EQUAL.
4. ALL PIPING TO BE ANS. SAFETY CLASS 2b EXCEPT AS NOTED.
5. ITEMS DESIGNATED WITH (f) ARE FURNISHED BY EQUIP. SUPPLIER.
6. ALL ◇ STATUS LIGHTS INDICATE BOTH OPEN AND CLOSED UNLESS NOTED OTHERWISE.
7. BENCH SET PRESSURE OF 120 PSIG + 30 PSIG BACK PRESSURE = DESIGN SET PRESSURE OF 150 PSIG.

H

NOTE:

A. TEMPORARY STRAINER FOR SYSTEM CLEAN UP, TO BE REMOVED AFTER FINAL FLUSH.

J

THIS IS A
NUCLEAR SAFETY RELATED
DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL

FSAR FIGURE 9.2-4

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM
COMPONENT COOLING

DESIGN ENGINEERING
SCE&G A SCANA COMPANY — V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

K

MADE	CHECKED	LE APPROVAL
1. SRM	2. MGR	3. RLJ

SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	D-302-611		39

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
35	05/15/05	TGB	REVISED PER ECR-50469	RAT	BB
34	05/02/05	JTS	REVISED PER ECR-50469	JRP	AME
39	1/6/09	JNC	REVISED PER ECR-71215	PAH	MGC
38	8/30/07	JMR	CADD ENHANCED PER ECR-50239	MGR	DJ
37	11/9/06	JMR	REVISED PER ECR-70724	MGR	MG
36	06/08/05	JTS	REVISED PER ECR-50469	MGR	AME

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-190-(S)N18**

2018 NRC A3 (SRO)

CANDIDATE: ______________________________

EXAMINER: ______________________________

DESCRIPTION: Review a release permit prior to approval/ release.

TASK: 341-012-03-02 Approve radioactive waste discharge/release permits (HPP-709 and HPP-710)

TASK STANDARD: Errors identified in the Waste Gas Release Worksheet and Waste Gas Release Permit that will prevent the release of the Waste Gas Decay Tank.

TERMINATING CUE: The candidate returns JPM briefing sheet and Handout materials provided to the examiner.

PREFERRED LOCATION:	***PREFERRED METHOD:***
CLASSROOM	PERFORM

REFERENCES:

HPP-709 Sampling and Release of Radioactive Gaseous Effluents

SOP-119 WASTE GAS PROCESSING

K/A 2.3.11 Ability to control radiation releases. (SRO 4.3)

10CFR55: 45 b(1)(12)

TOOLS: Copies of 2018 NRC A3 SRO Only Handouts 1 and 2
Electronic or Hardcopy versions of station procedures

EVALUATION TIME: 20 Minutes. ***TIME CRITICAL:*** NO

TIME START: __________ ***TIME FINISH:*** __________ ***PERFORMANCE TIME:*** __________

CANDIDATE: ______________________________

EXAMINER: ______________________________

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is in an extended outage in Mode 6.
- A release of Gas Decay Tank G7 is to occur during the next shift.
- Wind is out of the northwest at 5 miles per hour.
- The DATE and TIME is now 0830, 6/2/18.

INITIATING CUES:

You are to review the partially completed SOP-119, ATTACHMENT VA, GASEOUS WASTE RELEASE WORKSHEET-CONTROL ROOM and the HPP-709, ATTACHMENT I, GASEOUS WASTE RELEASE PERMIT (GWRP) and approve and sign the GWRP IF conditions allow the release at the current time.

If discrepancies are found that will not allow approval, circle them and fully explain the nature of the discrepancy on the worksheet that is also provided.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: YES	SEQUENCED: YES	SAT		UNSAT	

JPM STEP 1

Reviews the SOP-119, ATTACHMENT VA, GASEOUS WASTE RELEASE WORKSHEET-CONTROL ROOM and the HPP-709, ATTACHMENT I, GASEOUS WASTE RELEASE PERMIT (GWRP).

STEP STANDARD:

Notes and records the following discrepancies that will not allow approval of the GWRP:

- More than 24 hours have passed since the Gas Decay Tank was sampled.
- Notes that BYPASS is incorrecty entered on step 3.b for RM-A3 and RM-A10.
- Notes that wind speed is not sufficient for release due to the value for differential temperature in the current condition.

CUES:

Evaluator note: Provide candidate with the 2018 NRC A3 RO Handouts 1 and 2.

COMMENTS:

JPM SETUP SHEET

JPM: JPA-190-(S)N18

IC SET: N/A

INSTRUCTIONS:

Ensure that current procedures are available in hard copy or electronically.

COMMENTS:

JPM BRIEFING SHEET

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- The plant is in an extended outage in Mode 6.
- A release of Gas Decay Tank G7 is to occur during the next shift.
- Wind is out of the northwest at 5 miles per hour.
- The DATE and TIME is now 0830, 6/2/18.

INITIATING CUES:

You are to review the partially completed SOP-119, ATTACHMENT VA, GASEOUS WASTE RELEASE WORKSHEET-CONTROL ROOM and the HPP-709, ATTACHMENT I, GASEOUS WASTE RELEASE PERMIT (GWRP) and approve and sign the GWRP IF conditions allow the release at the current time.

If discrepancies are found that will not allow approval, circle them and fully explain the nature of the discrepancy on the worksheet that is also provided.

Hand this paper back to your Evaluator when_you feel that you have satisfactorily completed the assigned task.

2018 NRC A3 SRO Only Handout 1

GASEOUS WASTE RELEASE WORKSHEET-CONTROL ROOM

GWRP #: WG-18-02	WGDT #: G7	DATE: 6/1/18

CAUTION 1

During the release of gases, the conditions specified in the Gaseous Waste Release Permit (GWRP) must be adhered to (flow rate, radiation monitor setpoints, meteorological conditions, etc.).

INITIALS

1. Gaseous Waste Release Permit (GWRP) is returned from Health Physics with PART I completed and approved. EC

2. Ensure the following:

   a. Gas Decay Tank G(7) or H(8) is not in service. EC

   b. The AB Ventilation System is operating per SOP-502 with at least one AB Charcoal Exhaust Fan verified running (XFN-19A or XFN-19B). EC

   c. Wind direction is not from the East Southeast per HPP-709 to prevent activity from being drawn into the AB Ventilation System. EC

INITIALS

3. Radiation Monitors:

NOTE 3.a
1) The channel check should include a comparison of any local indication.
2) With the number of inoperable meteorological monitoring channels less than that which is required by Technical Specification 3.3.3.4, all gaseous releases must be stopped until the inoperable channel is restored.

a. Perform a channel and source check of RM-A3 Gas Channel and RM-A10: EC

RM-A3 Gas Channel	Channel Check: SAT/UNSAT Source Check: SAT/UNSAT	COMMENTS:
RM-A10	Channel Check: SAT/UNSAT Source Check: SAT/UNSAT	COMMENTS:

NOTE 3.b

If RM-A10 or RM-A3 is not operable, refer to offsite Dose Calculation Manual 1.2.1.1.

INITIALS

b. Verify RM-A10 or RM-A3 is operable and the Interlock Switch(s) on the Radiation Monitoring Panel is (are) in the NORMAL position: EC

RM-A3	NORMAL/BYPASS	COMMENTS:
RM-A10	NORMAL/BYPASS	COMMENTS:

c. Adjust Radiation Monitor setpoints per the Gaseous Waste Release Permit (GWRP). EC

4. Verify meteorological instrumentation is operable and meteorological conditions are satisfactory for the release per TABLE A (next page). EC

5. Mark the chart recorders for RM-A3G and RM-A10 in the Control Room with the following:

a. Tank Name/Number. EC

b. Date/Time. EC

NOTE 6

RM-A11 Point 3 may alarm during Waste Gas Release due to the inability to bypass point 3 locally.

6. Direct the building operator to commence Attachment VB. ________

INITIALS

7. At least once per hour, monitor Control Room meteorological indicators to verify conditions specified in TABLE A (next page) are acceptable for continued release. ________

8. When notified by the local operator that the required volume of gas and nitrogen has been released, verify Radiation Monitors return to normal background. ________

9. If Radiation Monitors do not return to normal, notify the Count Room. ________

10. Reset the RM-A10 alarm setting as specified in the Gaseous Waste Relase Permit (GWRP), (RM-A3 alarm setting to remain at 300 cpm). ________

11. Shift Manager review package and attach worksheet and applicable attachments. ________

Release conducted by: ___________________________ Date: __________

Shift Manager review:___________________________ Date:__________

REMARKS: ____
____
____
____
____
____
____
____
____
____

TABLE A

ACCEPTABLE METEOROLOGY FOR PLANNED WGDT RELEASES

DIFFERENTIAL TEMPERATURE (ΔT) °F (NOTE 1)		STABILITY CLASS	MINIMUM WIND SPEED (mph) (NOTE 2)
61m - 10m	40m - 10m		
$\Delta T \leq -1.74$	$\Delta T \leq -1.03$	A	*
$-1.74 < \Delta T \leq -1.56$	$-1.03 < \Delta T \leq -0.92$	B	*
$-1.56 < \Delta T \leq -1.38$	$-0.92 < \Delta T \leq -0.81$	C	1.6
$-1.38 < \Delta T \leq -0.46$	$-0.81 < \Delta T \leq -0.27$	D	4.1
$-0.46 < \Delta T \leq 1.38$	$-0.27 < \Delta T \leq 0.81$	E	6.6
$1.38 < \Delta T \leq 3.67$	$0.81 < \Delta T \leq 2.16$	F	14.0
$3.67 < \Delta T$	$2.16 < \Delta T$	G	18.9

NOTES:

1. The ΔT values for 61m - 10m are considered as primary indicators for determination of stability class. The 40m - 10m ΔT values are used only when 61m - 10m values are not available. All ΔT values are listed in °F and are based on values in USNRC Regulatory Guide 1.23.

2. The 10m wind speed is considered the primary indication for wind speed. The 61m wind speed indication should only be used if the 10m indicator is not available.

* No wind is required for planned releases.

GWRP No. WG-18-02

GASEOUS WASTE RELEASE PERMIT (GWRP)

☒ G TANK ☐ H TANK

I. RELEASE AUTHORIZATION (COUNT ROOM)

Date/time Sampled: 6/1/18 0739 Total Noble Gas, (μCi/cc): 1.34 E -03

Maximum WGDT Release Rate, (cfm): 15.0 Initial Tank Pressure, (psig): 22.0

	Background (cpm)	Alarm Set point (cpm)
RM-A3 Gas Channel	100	300
RM-A10 Gas Channel	65	210

Additional Requirements: ______

Count Room: [signature] Date/Time: 6/1/18 0900

II. ACTUAL RELEASE DATA (Operations)

Release Approved, SS/CRS: ______ Date/Time: ______

Meteorology Acceptable ☒ or, Unacceptable ☐ Wind Speed, (mph): 5.0 ΔT: (61m-10m) -0.25

	RM-A3 (CPM)	RM-A10 (CPM)	INITIALS
Alarm Set Point (cpm)	300	210	[initials]
Source Check		Sat/Unsat	[initials]
Reading @ Release Start (cpm)			
Reading @ 10 mins into Release (cpm)			
Reading @ End of Release (cpm)			
Reading After Purge (cpm)			
Alarm Set Point returned to 2 x ni			

	Start	Finish	Net
Release Date/Time			hours
Flow, (cfm)			
Pressure, (psig)			psig

COMMENTS ______

Release Conducted by: ______ Date/Time: ______

Operations Review: ______ Date/Time: ______

Updated by: ______ Date/Time: ______
(Count Room)

V.C. SUMMER NUCLEAR STATION
JOB PERFORMANCE MEASURE

JPM NO: **JPA-1008-(S)N18**

2018 NRC A4 (SRO)

TIME CRITICAL JPM

CANDIDATE: ____________________

EXAMINER: ____________________

DESCRIPTION: Determine Protectove Action Recommendations during after a declaration of General Emergency.

TASK: O-344-057-03-02 Make recommendations based on protective action guidelines in the emerg plan (EPP-001.4 AND EPP-005)

TASK STANDARD: The candidate completes the VCS-EPP-0002 Attachment 1 notification form in conformance with the JPM answer key within 15 minutes.

TERMINATING CUE: The candidate returns JPM briefing sheet and Handout materials to the examiner.

PREFERRED LOCATION: CLASSROOM

PREFERRED METHOD: PERFORM

REFERENCES:

VCS-EPP-0001.4 GENERAL EMERGENCY

K/A 2.4.40 Knowledge of the SRO's responsibilities in emergency plan implementation. (SRO 4.5)

10CFR55: 45 b(1)(12)

TOOLS: Electronic or Hardcopy versions of station procedures

EVALUATION TIME: 15 Minutes. ***TIME CRITICAL:*** YES

TIME START: ____________ ***TIME FINISH:*** ____________ ***PERFORMANCE TIME:*** ____________

CANDIDATE: ______________________________

EXAMINER: ______________________________

INSTRUCTIONS TO OPERATOR

READ TO OPERATOR:

When I tell you to begin, you are to perform the actions as directed in the initiating cues.

I will describe the general conditions under which this task is to be performed and provide the necessary tools with which to perform this task.

Before starting, I will explain the initial conditions, which steps to simulate or discuss, and provide initiating cues. When you complete the task successfully, this Job Performance Measure will be satisfied.

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- Hostile intruders entered the protected area and have damaged several ESF components.
- You have declared a General Emergency for condition HG1.1. HOSTILE ACTION resulting in a loss of physical control of the facility.
- Reactor Power was initially 100%.
- The reactor tripped and Safety Injection actuated 15 minutes ago.
- RM-G7 and RM-G18, CNTMT HI RNG GAMMA both read 2000 R/hr.
- RM-G19A, MAIN STEAM LINE reads 75 mr/hr.
- All other Radiation Monitors read normal.
- Wind direction is from 105°.
- Wind speed is 15 mph.
- Stability class is E.
- The duty ED is in route to the facility.
- The confirmation phone number is 803-334-1234.
- The time at which the classification was made will be given by the evaluator just prior to beginning this JPM.

INITIATING CUES:

You are directed to complete Attachment I of VCS-EPP-0002 in the handout provided to you.

Assume that this notification is the first General Emergency declaration.

THIS IS A TIME CRITICAL JPM.

HAND JPM BRIEFING SHEET TO OPERATOR AT THIS TIME!

CRITICAL: YES	SEQUENCED: YES	SAT		UNSAT	

JPM STEP 1

Complete VCS-EPP-002, Attachment 1, Nuclear Power Plant Emergency Notification Form.

STEP STANDARD:

Correctly completes VCS-EPP-002 Attachment 1 within 15 minutes after declaration time given as a cue below. See key for correct manner of completing the attachment.

CUES:

Evaluator cues: Announce that time of declaration will be the current time by classroom clock.

Evaluator note: The VCS-EPP-002 Attachment must be approved within a span of 15 minutes of time after their declaration time. Refer to 2017 NRC A4 SRO Only Key.

COMMENTS:

JPM SETUP SHEET

JPM: JPA-1008-(S)N18

IC SET: N/A

INSTRUCTIONS:

Ensure that current procedures are available in hard copy or electronically.

COMMENTS:

Revised Steps to align with current procedure.

JPM BRIEFING SHEET

SAFETY CONSIDERATIONS: NONE

INITIAL CONDITIONS:

- Hostile intruders entered the protected area and have damaged several ESF components.
- You have declared a General Emergency for condition HG1.1. HOSTILE ACTION resulting in a loss of physical control of the facility.
- Reactor Power was initially 100%.
- The reactor tripped and Safety Injection actuated 15 minutes ago.
- RM-G7 and RM-G18, CNTMT HI RNG GAMMA both read 2000 R/hr.
- RM-G19A, MAIN STEAM LINE reads 75 mr/hr.
- All other Radiation Monitors read normal.
- Wind direction is from 105°.
- Wind speed is 15 mph.
- Stability class is E.
- The duty ED is in route to the facility.
- The confirmation phone number is 803-334-1234.
- The time at which the classification was made will be given by the evaluator just prior to beginning this JPM.

INITIATING CUES:

You are directed to complete Attachment I of VCS-EPP-0002 in the handout provided to you.

Assume that this notification is the first General Emergency declaration.

*****THIS IS A TIME CRITICAL JPM.*****

Hand this paper back to your Evaluator when _you feel that you have satisfactorily completed the assigned task.

*************2018 NRC A4 SRO Only KEY**************

NUCLEAR POWER PLANT EMERGENCY NOTIFICATION FORM

MESSAGE# 1 **Confirmation Phone#** 803-334-1234 **AUTHENTICATION#**______

Lines 1-6 are required for INITIAL Notification

1. **EVENT:** [X] A DRILL [X] B ACTUAL DECLARATION C TERMINATION (*ONLY Lines 1, 2, & 4 required*)
2. **AFFECTED SITE:** V.C. SUMMER

Declaration Time given in initial cue.

3. **EMERGENCY CLASSIFICATION:**
   A UNUSUAL EVENT B ALERT C SITE AREA EMERGENCY *** [X] D GENERAL EMERGENCY
4. **EAL#** HG1.1*** **Declaration Date:** Today*** __/__/__ Time:______
   Termination Date:___/___/___ Time:______ (mark "N/A" for EAL# & Description)

EAL DESCRIPTION: HOSTILE ACTION resulting in a loss of physical control of the facility.***

5. **RELEASE TO THE ENVIRONMENT** (caused by the emergency): *** [X] A None B Is Occurring C Has Occurred
6. **PROTECTIVE ACTION RECOMMENDATIONS:**
   A NONE

   Items marked with *** are CRITICAL

   B EVACUATE: ______
   *** [X] C SHELTER: A-0, A-1, F-1
   *** [X] D Consider the use of KI (potassium iodide) in accordance with ORO plans and policies.
   E 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.** [X] A Yes [X] B No
8. **SITE UNIT(S) STATUS**:
   AFFECTED UNIT
   [X] A Yes Unit 1 - 100 % Power Shutdown: Date Today __/__ Time: T-0 minus 15 minutes
   B Yes Unit 2 - ______% Power Shutdown: Date___/___/___Time:______
   C Yes Unit 3 - ______% Power Shutdown: Date___/___/___Time:______

Either Yes or No is acceptable.

9. **METEOROLOGICAL DATA**:
   Wind direction from: 105 degrees Wind Speed: 15 mph Precipitation:______inches
   Stability Class: A B C D [X] E F G

Note: Lines 7 - 11 are not procedurally required and are not CRITICAL steps.

Lines 10 - 11 are completed for Follow-Up notifications, IF Line 5 IS OCCURING or HAS OCCURRED is selected

10. **AIRBORNE RELEASE CHARACTERIZATION:** A Ground B Mixed C Elevated
    MAGNITUDE UNITS: ☐ Ci ☐ Ci/sec ☐ µCi/sec
    Noble Gases:______ Iodines:______ Particulates:______
11. **DOSE PROJECTION:** Projection period: ______Hours Estimated Release Duration ______Hours

Performed:	**DISTANCE**	**TEDE (mrem)**	**Thyroid CDE (mrem)**
	Site Boundary		
Date: ___/___/___	2 Miles		
Time: ______	5 Miles		
	10 Miles		

Approved time must be within a span of 15 minutes after declaration time given as a cue.

12. **REMARKS (As Applicable)**: ______
13. **APPROVED BY:** Candidate signature*** Title Shift Manager/IED*** Date Today*** ___/___/___ Time:______ ***
14. **NOTIFIED BY:** ______ Date ___/___/___ Time:______
15. **RECEIVED BY** (ORO use only) ______ Date ___/___/___ Time:______

Either is acceptable..