Facility	Salem	Date of Exam: 01/10/00							Exam Level:	RO		
Tier	Group	K/A Category Points										Point Total
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	
1. Emergency & Abnormal Plant Evolutions	1	2	2	3				4	3			2	16
	2	4	2	4				2	3			2	17
	3			1				1	1				3
	Tier Totals	6	4	8				7	7			4	36
2. Plant Systems	1	4	1	2	3	2		2		5	3	1	23
	2	4		1	4		1	4	3	2	1		20
	3	1		1	1			1	1	1	1	1	8
	Tier Totals	9	1	4	8	2	1	7	4	8	5	2	51
3. Generic Knowledge and Abilities					Cat 1		Cat 2		Cat 3		Cat 4		
					3		4		3		3		13

Note:
- Attempt to distribute topics among all K/A Categories: select at least one topic from every K/A category within each tier.
- Actual point totals must match those specified in the table.
- Select topics from many systems: avoid selecting more than two or three K/A topics from a given system unless they relate to plant-specific priorities.
- Systems/evolutions within each group are identified on the associated outline.
- The shaded areas are not applicable to the category/tier.

Emergency and Abnormal Plant Evolutions - Tier 1/Group 1

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
005	Inoperable/Stuck Control Rod					X		AA2.01 Stuck or inoperable rod from in-core and ex-core NIS, in-core or loop temperature measurements	3.3	66
015	Reactor Coolant Pump (RCP) Malfunctions		X					AK2.10 RCP indicators and controls	2.8*	72
017	Reactor Coolant Pump (RCP) Malfunctions (Loss of RC Flow)									
024	Emergency Boration				X			AA1.20 Manual boration valve and indicators	3.2*	74
026	Loss of Component Cooling Water (CCW)					X		AA2.02 The cause of possible CCW loss	2.9	76
027	Pressurizer Pressure Control (PZR PCS) Malfunction		X					AK2.03 Controllers and positioners	2.6	77
040	Steam Line Rupture	X						AK1.06 High-energy steam line break considerations	3.7	82
051	Loss of Condenser Vacuum						X	2.1.32 Ability to explain and apply all system limits and precautions.	3.4	83
055	Loss of Offsite and Onsite Power (Station Blackout)						X	2.4.7 Knowledge of event based EOP mitigation strategies.	3.1	85
057	Loss of Vital AC Electrical Instrument Bus				X			AA1.06 Manual control of components for which automatic control is lost	3.5	87
062	Loss of Nuclear Service Water					X		AA2.01 Location of a leak in the SWS	2.9	89
067	Plant Fire on Site				X			AA1.06 Fire alarm	3.5	91
068	Control Room Evacuation				X			AA1.21 Transfer of controls from control room to shutdown panel or local control	3.9	92
069	Loss of Containment Integrity									
074	Inadequate Core Cooling			X				EK3.07 Starting up emergency feedwater and RCPs	4.0	93
076	High Reactor Coolant Activity			X				AK3.06 Actions contained in EOP for high reactor coolant activity	3.2	94
E06	Degraded Core Cooling									
E07	Saturated Core Cooling									
E08	Pressurized Thermal Shock									
E09	Natural Circulation Operations									
E10	Natural Circulation with Steam Void in Vessel with/without RVLIS	X						EK1.2 Normal, abnormal and emergency operating procedures associated with Natural Circulation with Steam Void in Vessel with/without RVLIS.	3.4	99
E12	Uncontrolled Depressurization of all Steam Generators			X				EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	3.5	100
E14	High Containment Pressure									
	K/A Category Point Totals:	2	2	3	4	3	2	Group Point Total:		16

ES-401 PWR RO Examination Outline ES-401-4

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp	Q#
001	Continuous Rod Withdrawal									
003	Dropped Control Rod		X					AK2.05 Control rod drive power supplies and logic circuits	2.5	65
007	Reactor Trip	X						EK1.05 Decay power as a function of time	3.3	67
008	Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)			X				AK3.02 Why PORV or code safety exit temperature is below RCS or PZR temperature	3.6	68
009	Small Break LOCA					X		EA2.01 Actions to be taken, based on RCS temperature and pressure, saturated and superheated	4.2	69
011	Large Break LOCA						X	2.1.31 Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup.	4.2	71
011	Large Break LOCA				X			EA1.11 Long-term cooling of core	4.2	70
022	Loss of Reactor Coolant Makeup				X			AA1.08 VCT level	3.4	73
025	Loss of Residual Heat Removal System (RHRS)			X				AK3.01 Shift to alternate flowpath	3.1	75
029	Anticipated Transient Without Scram (ATWS)	X						EK1.05 Definition of negative temperature coefficient as applied to large PWR coolant systems	2.8	79
032	Loss of Source Range Nuclear Instrumentation									
033	Loss of Intermediate Range Nuclear Instrumentation					X		AA2.02 Indications of unreliable intermediate-range channel operation	3.3	80
037	Steam Generator (S/G) Tube Leak									
038	Steam Generator Tube Rupture (SGTR)			X				EK3.06 Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures	4.2	81
054	Loss of Main Feedwater (MFW)	X						AK1.01 MFW line break depressurizes the S/G (similar to a steam line break)	4.1	84
058	Loss of DC Power					X		AA2.03 DC loads lost; impact on ability to operate and monitor plant systems	3.5	88
059	Accidental Liquid Radwaste Release									
060	Accidental Gaseous Radwaste Release									
061	Area Radiation Monitoring (ARM) System Alarms									
E02	SI Termination			X				EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	3.9	95
E03	LOCA Cooldown and Depressurization						X	2.4.14 Knowledge of general guidelines for EOP flowchart use.	3.0	96

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
E04	LOCA Outside Containment	X						EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the LOCA Outside Containment.	3.5	97
E05	Loss of Secondary Heat Sink		X					EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	3.9	98
E11	Loss of Emergency Coolant Recirculation									
E16	High Containment Radiation									
	K/A Category Point Totals:	4	2	4	2	3	2	Group Point Total:		17

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
028	Pressurizer (PZR) Level Control Malfunction					X		AA2.01 PZR level indicators and alarms	3.4	78
036	Fuel Handling Incidents									
056	Loss of Offsite Power				X			AA1.21 Reset of the ESF load sequencers	3.3*	86
065	Loss of Instrument Air			X				AK3.08 Actions contained in EOP for loss of instrument air	3.7	90
E13	Steam Generator Overpressure									
E15	Containment Flooding									
	K/A Category Point Totals:	0	0	1	1	1	0	Group Point Total:		3

ES-401	PWR RO Examination Outline Plant Systems - Tier 2/Group 1														ES-401-4
Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
001	Control Rod Drive System					X							K5.05 Interpretation of rod worth curves, including proper curve to use: all rods in (ARI), all rods out (ARO), hot zero power (HZP), hot full power (HFP)	3.5	15
001	Control Rod Drive System				X								K4.02 Control rod mode select control (movement control)	3.8	14
003	Reactor Coolant Pump System (RCPS)										X		A4.01 Seal injection	3.3	18
003	Reactor Coolant Pump System (RCPS)			X									K3.04 RPS	3.9	19
004	Chemical and Volume Control System (CVCS)									X			A3.10 PZR level and pressure	3.9	20
004	Chemical and Volume Control System (CVCS)	X											K1.18 CCWS	2.9	21
013	Engineered Safety Features Actuation System (ESFAS)									X			A3.02 Operation of actuated equipment	4.1	32
013	Engineered Safety Features Actuation System (ESFAS)										X		A4.03 ESFAS initiation	4.5	33
013	Engineered Safety Features Actuation System (ESFAS)				X								K4.03 Main Steam Isolation System	3.9	34
015	Nuclear Instrumentation System					X							K5.04 Factors affecting accuracy and reliability of calorimetric calibrations	2.6	37
015	Nuclear Instrumentation System			X									K3.01 RPS	3.9	36
017	In-Core Temperature Monitor (ITM) System							X					A1.01 Core exit temperature	3.7	39
022	Containment Cooling System (CCS)		X										K2.01 Containment cooling fans	3.0*	41
022	Containment Cooling System (CCS)	X											K1.01 SWS/cooling system	3.5	40
025	Ice Condenser System														
056	Condensate System	X											K1.03 MFW	2.6*	51
059	Main Feedwater (MFW) System									X			A3.07 ICS	3.4*	52
059	Main Feedwater (MFW) System				X								K4.19 Automatic feedwater isolation of MFW	3.2	53
061	Auxiliary / Emergency Feedwater (AFW) System							X					A1.01 S/G level	3.9	54
061	Auxiliary / Emergency Feedwater (AFW) System									X			A3.01 AFW startup and flows	4.2	55
068	Liquid Radwaste System (LRS)										X		A4.04 Automatic isolation	3.8	58
068	Liquid Radwaste System (LRS)	X											K1.07 Sources of liquid wastes for LRS	2.7	59

Plant Systems - Tier 2/Group 1

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
071	Waste Gas Disposal System (WGDS)									X			A3.02 Pressure-regulating system for waste gas vent header	2.8	60
072	Area Radiation Monitoring (ARM) System											X	2.1.32 Ability to explain and apply all system limits and precautions.	3.4	61
	K/A Category Point Totals:	4	1	2	3	2	0	2	0	5	3	1	Group Point Total:		23

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
002	Reactor Coolant System (RCS)	X											K1.07 Reactor vessel level indication system	3.5*	17
002	Reactor Coolant System (RCS)							X					A1.08 RCS average temperature	3.7	16
006	Emergency Core Cooling System (ECCS)	X											K1.08 CVCS	3.6	23
006	Emergency Core Cooling System (ECCS)				X								K4.05 Autostart of HPI/LPI/SIP.	4.3	24
010	Pressurizer Pressure Control System (PZR PCS)									X			A3.02 PZR pressure	3.6	28
010	Pressurizer Pressure Control System (PZR PCS)							X					A1.07 RCS pressure	3.7	27
011	Pressurizer Level Control System (PZR LCS)							X					A1.02 Charging and letdown flows	3.3	29
011	Pressurizer Level Control System (PZR LCS)						X						K6.04 Operation of PZR level controllers	3.1	30
012	Reactor Protection System								X				A2.04 Erratic power supply operation	3.1	31
014	Rod Position Indication System (RPIS)							X					A1.02 Control rod position indication on control room panels	3.2	35
016	Non-Nuclear Instrumentation System (NNIS)	X											K1.01 RCS	3.4*	38
026	Containment Spray System (CSS)	X											K1.01 ECCS	4.2	43
026	Containment Spray System (CSS)								X				A2.03 Failure of ESF	4.1	42
029	Containment Purge System (CPS)									X			A3.01 CPS isolation	3.8	46
033	Spent Fuel Pool Cooling System (SFPCS)														
035	Steam Generator System (S/GS)														
039	Main and Reheat Steam System (MRSS)								X				A2.04 Malfunctioning steam dump	3.4	48
055	Condenser Air Removal System (CARS)			X									K3.01 Main Condenser	2.5	50
062	A.C. Electrical Distribution System				X								K4.10 Uninterruptable ac power sources	3.1	56
063	D.C. Electrical Distribution System														
064	Emergency Diesel Generator (ED/G) System										X		A4.02 Adjustment of exciter voltage (using voltage control switch)	3.3	57
073	Process Radiation Monitoring (PRM) System														
075	Circulating Water System														

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
079	Station Air System (SAS)				X								K4.01 Cross-connect with IAS	2.9	63
086	Fire Protection System (FPS)				X								K4.06 CO2	3.0	64
	K/A Category Point Totals:	4	0	1	4	0	1	4	3	2	1	0	Group Point Total:		20

Plant Systems - Tier 2/Group 3

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
005	Residual Heat Removal System (RHRS)							X					A1.01 Heatup/cooldown rates	3.5	22
007	Pressurizer Relief Tank/Quench Tank System (PRTS)	X											K1.03 RCS	3.0	25
008	Component Cooling Water System (CCWS)									X			A3.08 Automatic actions associated with the CCWS that occur as a result of a safety injection signal	3.6*	26
027	Containment Iodine Removal System (CIRS)										X		A4.01 CIRS controls	3.3*	44
028	Hydrogen Recombiner and Purge Control System (HRPS)								X				A2.01 Hydrogen recombiner power setting, determined by using plant data book	3.4*	45
034	Fuel Handling Equipment System (FHES)				X								K4.02 Fuel movement	2.5	47
041	Steam Dump System (SDS) and Turbine Bypass Control			X									K3.02 RCS	3.8	49
045	Main Turbine Generator (MT/G) System														
076	Service Water System (SWS)											X	2.1.28 Knowledge of the purpose and function of major system components and controls	3.2	62
078	Instrument Air System (IAS)														
103	Containment System														
	K/A Category Point Totals:	1	0	1	1	0	0	1	1	1	1	1	Group Point Total:		8

Plant-Specific Priorities

System / Topic	Recommended Replacement for ...	Reason	Pts
Plant Specific Priority Total: (limit 10)			

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#

Facility Salem	Date: January 10, 2000		Exam Level:	RO
Category	KA #	KA Topic	Imp.	Q#
Conduct of Operations	2.1.29	Knowledge of how to conduct and verify valve lineups.	3.4	~~1~~ 2
	2.1.3	Knowledge of shift turnover practices.	3.0	~~2~~ 1
	~~2.1.33~~	~~Ability to recognize indications for system operating parameters which are entry level conditions for technical specifications.~~	~~3.4~~	~~3~~
	2.1.10	Knowledge of conditions and limitations in the facility license	2.7	3
			Total	3
Equipment Control	2.2.23	Ability to track limiting conditions for operations.	2.6	4
	2.2.24	Ability to analyze the affect of maintenance activities on LCO status.	2.6	5
	2.2.28	Knowledge of new and spent fuel movement procedures.	2.6	6
	2.2.34	Knowledge of the process for determining the internal and external effects on core reactivity.	2.8	7
			Total	4
Radiation Control	2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements.	2.6	8
	2.3.2	Knowledge of facility ALARA program.	2.5	9
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	2.9	10
			Total	3
Emergency Procedures and Plan	2.4.13	Knowledge of crew roles and responsibilities during EOP flowchart use.	3.3	11
	2.4.16	Knowledge of EOP implementation hierarchy and coordination with other support procedures.	3.0	12
	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control; 2. Core cooling and heat removal; 3. Reactor coolant system integrity; 4. Containment conditions; 5. Radioactivity release control.	3.7	13
			Total	3
		Tier 3 Target Point Total (RO)		13

11/23/99

Facility	Salem	Date of Exam: 01/10/00								Exam Level:	SRO		
Tier	Group	K/A Category Points											Point Total
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	
1. Emergency & Abnormal Plant Evolutions	1	3	2	4				5	7			3	24
	2	1	2	4				1	5			3	16
	3							1	2				3
	Tier Totals	4	4	8				7	14			6	43
2. Plant Systems	1	3	1	1		1		2	6	2	2	1	19
	2	2		1	5		1	1	5	1	1		17
	3	1		1				1				1	4
	Tier Totals	6	1	3	5	1	1	4	11	3	3	2	40
3. Generic Knowledge and Abilities			Cat 1		Cat 2		Cat 3		Cat 4				
			5		4		3		5				17

Note:
- Attempt to distribute topics among all K/A Categories: select at least one topic from every K/A category within each tier.
- Actual point totals must match those specified in the table.
- Select topics from many systems: avoid selecting more than two or three K/A topics from a given system unless they relate to plant-specific priorities.
- Systems/evolutions within each group are identified on the associated outline.
- The shaded areas are not applicable to the category/tier.

Emergency and Abnormal Plant Evolutions - Tier 1/Group 1

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
001	Continuous Rod Withdrawal					X		AA2.05 Uncontrolled rod withdrawal, from available indications	4.6	58
003	Dropped Control Rod		X					AK2.05 Control rod drive power supplies and logic circuits	2.8	59
005	Inoperable/Stuck Control Rod					X		AA2.01 Stuck or inoperable rod from in-core and ex-core NIS, in-core or loop temperature measurements	4.1	60
011	Large Break LOCA				X			EA1.11 Long-term cooling of core	4.2	64
011	Large Break LOCA						X	2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.0	65
015	Reactor Coolant Pump (RCP) Malfunctions		X					AK2.10 RCP indicators and controls	2.8	66
017	Reactor Coolant Pump (RCP) Malfunctions (Loss of RC Flow)									
024	Emergency Boration				X			AA1.20 Manual boration valve and indicators	3.3	68
026	Loss of Component Cooling Water (CCW)					X		AA2.02 The cause of possible CCW loss	3.6	70
029	Anticipated Transient Without Scram (ATWS)					X		EA2.04 CVCS Centrifugal Charging Pump operating indication	3.3*	73
029	Anticipated Transient Without Scram (ATWS)	X						EK1.05 Definition of negative temperature coefficient as applied to large PWR coolant systems	3.2	74
040	Steam Line Rupture									
051	Loss of Condenser Vacuum						X	2.1.32 Ability to explain and apply all system limits and precautions.	3.8	80
055	Loss of Offsite and Onsite Power (Station Blackout)						X	2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.3	82
057	Loss of Vital AC Electrical Instrument Bus				X			AA1.06 Manual control of components for which automatic control is lost	35	84
059	Accidental Liquid Radwaste Release					X		AA2.02 The permit for liquid radioactive-waste release	3.9	86
062	Loss of Nuclear Service Water					X		AA2.01 Location of a leak in the SWS	3.5	87
067	Plant Fire on Site				X			AA1.06 Fire alarm	3.7	89
068	Control Room Evacuation				X			AA1.21 Transfer of controls from control room to shutdown panel or local control	4.1	90
069	Loss of Containment Integrity									
074	Inadequate Core Cooling			X				EK3.07 Starting up emergency feedwater and RCPs	4.4	91
076	High Reactor Coolant Activity			X				AK3.06 Actions contained in EOP for high reactor coolant activity	3.8	92

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp	Q#
E02	SI Termination			X				EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	3.9	93
E04	LOCA Outside Containment	X						EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the LOCA Outside Containment.	3.9	95
E06	Degraded Core Cooling									
E07	Saturated Core Cooling									
E08	Pressurized Thermal Shock					X		EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	4.2	97
E09	Natural Circulation Operations									
E10	Natural Circulation with Steam Void in Vessel with/without RVLIS	X						EK1.2 Normal, abnormal and emergency operating procedures associated with Natural Circulation with Steam Void in Vessel with/without RVLIS.	3.6	98
E12	Uncontrolled Depressurization of all Steam Generators			X				EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	3.7	100
E14	High Containment Pressure									
	K/A Category Point Totals:	3	2	4	5	7	3	Group Point Total:		24

ES-401	PWR SRO Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2									ES-401-3
Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
007	Reactor Trip					X		EA2.03 Reactor trip breaker position	4.4	61
008	Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)			X				AK3.02 Why PORV or code safety exit temperature is below RCS or PZR temperature	4.1	62
009	Small Break LOCA					X		EA2.01 Actions to be taken, based on RCS temperature and pressure, saturated and superheated	4.8	63
022	Loss of Reactor Coolant Makeup				X			AA1.08 VCT level	3.3	67
025	Loss of Residual Heat Removal System (RHRS)			X				AK3.01 Shift to alternate flowpath	3.4	69
027	Pressurizer Pressure Control (PZR PCS) Malfunction		X					AK2.03 Controllers and positioners	2.8	71
032	Loss of Source Range Nuclear Instrumentation					X		AA2.05 Nature of abnormality, from rapid survey of control room data	3.2*	75
033	Loss of Intermediate Range Nuclear Instrumentation					X		AA2.02 Indications of unreliable intermediate-range channel operation	3.6	76
037	Steam Generator (S/G) Tube Leak						X	2.4.4 Ability to recognize abnormal indications - - - - which are entry level conditions for emergency and abnormal procedures	4.3	78
038	Steam Generator Tube Rupture (SGTR)			X				EK3.06 Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures	4.5	79
054	Loss of Main Feedwater (MFW)	X						AK1.01 MFW line break depressurizes the S/G (similar to a steam line break)	4.3	81
058	Loss of DC Power					X		AA2.03 DC loads lost; impact on ability to operate and monitor plant systems	3.9	85
060	Accidental Gaseous Radwaste Release									
061	Area Radiation Monitoring (ARM) System Alarms									
065	Loss of Instrument Air			X				AK3.08 Actions contained in EOP for loss of instrument air	3.9	88
E03	LOCA Cooldown and Depressurization						X	2.4.14 Knowledge of general guidelines for EOP flowchart use.	3.9	94
E05	Loss of Secondary Heat Sink		X					EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	4.2	96
E11	Loss of Emergency Coolant Recirculation						X	2.4.8 Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs.	3.7	99
E16	High Containment Radiation									
	K/A Category Point Totals:	1	2	4	1	5	3	Group Point Total:		16

ES-401	PWR SRO Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 3								ES-401-3
Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
028	Pressurizer (PZR) Level Control Malfunction					X		AA2.01 PZR level indicators and alarms	3.6	72
036	Fuel Handling Incidents					X		AA2.03 Magnitude of potential radioactive release	4.2*	77
056	Loss of Offsite Power				X			AA1.21 Reset of the ESF load sequencers	3.3*	83
E13	Steam Generator Overpressure									
E15	Containment Flooding									
	K/A Category Point Totals:	0	0	0	1	2	0	Group Point Total:		3

Plant Systems - Tier 2/Group 1

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
001	Control Rod Drive System								X				A2.06 Effects of transient xenon on reactivity	3.7	18
001	Control Rod Drive System					X							K5.05 Interpretation of rod worth curves, including proper curve to use: all rods in (ARI), all rods out (ARO), hot zero power (HZP), hot full power (HFP)	3.9	19
003	Reactor Coolant Pump System (RCPS)										X		A4.01 Seal injection	3.2	22
003	Reactor Coolant Pump System (RCPS)			X									K3.04 RPS	4.2	23
004	Chemical and Volume Control System (CVCS)									X			A3.10 PZR level and pressure	3.9	24
004	Chemical and Volume Control System (CVCS)	X											K1.18 CCWS	3.2	25
013	Engineered Safety Features Actuation System (ESFAS)								X				A2.04 Loss of instrument bus	4.2	34
014	Rod Position Indication System (RPIS)							X					A1.02 Control rod position indication on control room panels	3.6	35
015	Nuclear Instrumentation System														
017	In-Core Temperature Monitor (ITM) System							X					A1.01 Core exit temperature	3.9	37
022	Containment Cooling System (CCS)	X											K1.01 SWS/cooling system	3.7	38
022	Containment Cooling System (CCS)		X										K2.01 Containment cooling fans	3.1	39
025	Ice Condenser System														
026	Containment Spray System (CSS)								X				A2.03 Failure of ESF	4.4	40
056	Condensate System	X											K1.03 MFW	2.6	46
059	Main Feedwater (MFW) System								X				A2.03 Overfeeding event	3.1*	47
061	Auxiliary / Emergency Feedwater (AFW) System								X				A2.04 pump failure or improper operation	3.8	48
063	D.C. Electrical Distribution System														
068	Liquid Radwaste System (LRS)										X		A4.04 Automatic isolation	3.7	51
071	Waste Gas Disposal System (WGDS)									X			A3.02 Pressure-regulating system for waste gas vent header	2.8	52
072	Area Radiation Monitoring (ARM) System											X	2.1.32 Ability to explain and apply all system limits and precautions.	3.8	54
072	Area Radiation Monitoring (ARM) System								X				A2.02 Detector failure	2.9	53
	K/A Category Point Totals:	3	1	1	0	1	0	2	6	2	2	1	Group Point Total:		19

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
002	Reactor Coolant System (RCS)	X											K1.07 Reactor vessel level indication system	3.7*	21
002	Reactor Coolant System (RCS)								X				A2.01 Loss of coolant inventory	4.4	20
006	Emergency Core Cooling System (ECCS)	X											K1.08 CVCS	3.9	27
006	Emergency Core Cooling System (ECCS)				X								K4.05 Autostart of HPI/LPI/SIP.	4.4	28
010	Pressurizer Pressure Control System (PZR PCS)									X			A3.02 PZR pressure	3.5	30
011	Pressurizer Level Control System (PZR LCS)							X					A1.02 Charging and letdown flows	3.5	31
011	Pressurizer Level Control System (PZR LCS)						X						K6.03 Relationship between PZR level and PZR heater control circuit	3.3	32
012	Reactor Protection System								X				A2.04 Erratic power supply operation	3.2	33
016	Non-Nuclear Instrumentation System (NNIS)								X				A2.01 Detector failure	3.1*	36
027	Containment Iodine Removal System (CIRS)										X		A4.01 CIRS controls	3.3*	41
028	Hydrogen Recombiner and Purge Control System (HRPS)								X				A2.01 Hydrogen recombiner power setting, determined by using plant data book	3.6*	42
029	Containment Purge System (CPS)														
033	Spent Fuel Pool Cooling System (SFPCS)														
034	Fuel Handling Equipment System (FHES)				X								K4.02 Fuel movement	3.3	43
035	Steam Generator System (S/GS)														
039	Main and Reheat Steam System (MRSS)														
055	Condenser Air Removal System (CARS)			X									K3.01 Main Condenser	2.7	45
062	A.C. Electrical Distribution System								X				A2.01 Types of loads that, if de-energized, would degrade or hinder plant operation	3.9	49
062	A.C. Electrical Distribution System				X								K4.10 Uninterruptable ac power sources	3.5	50
064	Emergency Diesel Generator (ED/G) System														
073	Process Radiation Monitoring (PRM) System														
075	Circulating Water System														

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
079	Station Air System (SAS)				X								K4.01 Cross-connect with IAS	3.2	56
086	Fire Protection System (FPS)				X								K4.06 CO2	3.3	57
103	Containment System														
	K/A Category Point Totals:	2	0	1	5	0	1	1	5	1	1	0	Group Point Total:		17

Plant Systems - Tier 2/Group 3

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Q#
005	Residual Heat Removal System (RHRS)							X					A1.01 Heatup/cooldown rates	3.6	26
007	Pressurizer Relief Tank/Quench Tank System (PRTS)	X											K1.03 RCS	3.2	29
008	Component Cooling Water System (CCWS)														
041	Steam Dump System (SDS) and Turbine Bypass Control			X									K3.02 RCS	3.9	44
045	Main Turbine Generator (MT/G) System														
076	Service Water System (SWS)											X	2.1.28 Knowledge of the purpose and function of major system components and controls	3.3	55
078	Instrument Air System (IAS)														
	K/A Category Point Totals:	1	0	1	0	0	0	1	0	0	0	1	Group Point Total:		4

Plant-Specifc Priorities

System / Topic	Recommended Replacement for ...	Reason	Pts
Plant Specific Priority Total: (limit 10)			

Facility: Salem	Date: January 10, 2000		Exam Level:	SRO
Category	**KA #**	**KA Topic**	**Imp.**	**Q#**
Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements.	3.8	1
	2.1.10	Knowledge of conditions and limitations in the facility license.	3.9	~~2~~ 5
	2.1.12	Ability to apply technical specifications for a system.	4.0	3
	2.1.29	Knowledge of how to conduct and verify valve lineups.	3.3	4
	2.1.33	Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.	4.0	2 ~~5~~
			Total	5
Equipment Control	2.2.20	Knowledge of the process for managing troubleshooting activities.	3.3	6
	2.2.23	Ability to track limiting conditions for operations.	3.8	7
	2.2.24	Ability to analyze the affect of maintenance activities on LCO status.	3.8	8
	2.2.28	Knowledge of new and spent fuel movement procedures.	3.5	9
			Total	4
Radiation Control	2.3.1	Knowledge of 10 CFR: 20 and related facility radiation control requirements.	3.0	10
	2.3.2	Knowledge of facility ALARA program.	2.9	11
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	3.3	12
			Total	3
Emergency Procedures and Plan	2.4.16	Knowledge of EOP implementation hierarchy and coordination with other support procedures.	4.0	13
	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control; 2. Core cooling and heat removal; 3. Reactor coolant system integrity; 4. Containment conditions; 5. Radioactivity release control.	4.3	14
	2.4.30	Knowledge of which events related to system operations/status should be reported to outside agencies.	3.6	15
	2.4.32	Knowledge of operator response to loss of all annunciators.	3.5	16
	2.4.38	Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting as emergency coordinator.	4.0	17
			Total	5
		Tier 3 Target Point Total (SRO)		17

Salem 1/10/00 NRC Exam Question Cognitive Level and Source Summary

RECORD #	RO	SRO	COGNITIVE LEVEL	SOURCE	NOTES
1		x	C	New(jkl)	
2	x		C	NRC(PI)	SM
3		x	A	New	
4		x	C	SalNRC 6/98	
5	x	x	M	New(jkl)	
6	x	x	M	New	
7		x	A	New(jkl)	
8	x	x	M	NRC(Braid)	
9	x	x	C	New(jkl)	
10	x	x	A	New	
11	x		C	New	
12	x	x	A	New	
13	x	x	A	New(jkl)	
14	x	x	A	New(jkl)	
15	x		C	New(jkl)	
16	x	x	A	BV Audit	SM
17	x	x	C	NRC(PI)	
18		x	A	New(jkl)	
19		x	M	New(jkl)	
20		x	M	SalNRC 2/98	SM
21		x	M	New(jkl)	
22	x		M	New(jkl)	
23	x	x	C	SalemNRC 2/99	Last two NRC-SM
24	x		C	NRC Bank	
25		x	C	NRC Bank	
26	x	x	M	NRC Bank	
27	x	x	M	NRC Bank	
28	x	x	M	NRC Bank	
29	x	x	C	NRC Bank	
30	x	x	A	New	
31	x	x	C	NRC Bank	
32	x	x	C	NRC Bank	
33	x	x	M	New(jkl)	
34	x	x	M	New(jkl)	

Salem 1/10/00 NRC Exam Question Cognitive Level and Source Summary

RECORD #	RO	SRO	COGNITIVE LEVEL	SOURCE	NOTES
35	x		C	New	
36	x		M	NRC Bank	
37	x	x	C	NRC Bank	
38	x	x	A	NRC Bank	
39		x	A	Salem NRC 9/98	Last two NRC
40	x		C	New(jkl)	
41	x	x	C	NRC Bank	SM
42		x	M	NRC Bank	
43	x		M	New	
44	x		M	NRC Bank	
45	x		C	SalNRC 6/98	
46	x	x	C	Fac Bk	
47	x		A	NRC Bank	
48	x		C	Fac Bk	SM
49		x	C	NRC Bank	
50	x		C	SalNRC 2/98	
51	x	x	M	New(jkl)	
52	x	x	M	Fac Bk	
53	x	x	A	New(jkl)	
54	x	x	M	New(jkl)	
55	x		C	Fac Bk	
56	x	x	M	SalNRC 6/98	
57	x	x	A	Fac Bk	
58	x		M	SalNRC 2/98	
59	x	x	M	SalNRC 2/98	
60	x		C	New	
61	x	x	C	SalNRC 6/98	
62	x	x	A	SalNRC 9/98	Last two NRC
63	x	x	M	SalNRC 9/98	
64		x	M	NRC Bank	
65	x		A	Fac Bk	SM
66	x		C	New	
67	x		C	SalNRC 6/98	
68		x	C	New(jkl)	

Salem 1/10/00 NRC Exam Question Cognitive Level and Source Summary

RECORD #	RO	SRO	COGNITIVE LEVEL	SOURCE	NOTES
69	x		C	NRC Bank	
70		x	C	New(jkl)	
71	x	x	M	Fac Bk	
72	x		M	New	
73	x	x	M	NRC(Braid)	SM
74	x		M	SalNRC 6/98	
75	x	x	M	Fac Bk	SM
76		x	M	Fac Bk	SM
77	x	x	M	SalNRC 2/98	
78	x	x	M	New(jkl)	
79	x	x	M	SalNRC 2/98	
80	x	x	M	SalNRC 6/98	
81		x	A	NRC(Braid)	SM
82	x	x	C	New(jkl)	
83	x	x	C	New(jkl)	
84		x	C	SalNRC 6/98	
85	x		M	NRC Bank	
86	x	x	A	NRC(PI)	
87	x	x	C	Fac Bk	
88	x	x	M	New(jkl)	
89	x		M	New(jkl)	
90		x	A	New(jkl)	
91	x	x	C	NRC(PI)	
92	x	x	C	PI Bank	
93	x	x	M	New	
94	x	x	C	New	
95	x	x	C	SalNRC 6/98	
96	x	x	M	New	
97	x	x	A	NRC(WC)	
98		x	M	New(jkl)	
99	x	x	M	SalNRC 6/98	
100		x	C	NRC Bank	
101	x	x	M	Fac Bk	SM
102		x	M	Fac Bk	

Salem 1/10/00 NRC Exam Question Cognitive Level and Source Summary

RECORD #	RO	SRO	COGNITIVE LEVEL	SOURCE	NOTES
103		x	M	New(jkl)	
104	x	x	M	NRC(Byron)	
105	x		C	NRC(Braid)	
106	x	x	M	New(jkl)	
107	x	x	C	New(jkl)	
108		x	M	NRC(PI)	
109	x		M	NRC(Braid)	
110	x	x	M	NRC(Braid)	SM
111	x	x	C	Fac Bk	
112	x	x	M	SalNRC 6/98	
113		x	C	New(jkl)	
114	x	x	C	Fac Bk	
115	x	x	M	Fac Bk	SM
116	x	x	M	SalNRC 2/97	
117	x	x	C	New	
118	x	x	M	PI Bank	
119	x	x	M	Fac Bk	SM
120	x	x	C	New(jkl)	
121	x	x	A	SalNRC 2/99	Last two NRC-SM
122	x	x	C	NRC(Braid)	SM
123	x	x	M	NRC(Braid)	
124		x	C	New(jkl)	
125	x	x	C	New(jkl)	
126		x	M	Fac Bk	SM
127	x	x	C	New(jkl)	

Salem January 2000 SRO Written Examination Question Source Statistic Summary

QUESTION SOURCE	Memory	Comp/Applic
Salem Bank	3	5
Salem Bank - Significantly Modified	6	None
Salem NRC Exams (not in Salem Bank)	10	4
Previous 2 Salem NRC Exams	None	4
NRC Bank	5	9
Other Facility (not in Salem Bank)	7	8
New	15	24
Total	46	54

Salem 1/10/00 SRO NRC Exam Question Cognitive Level and Source Summary

RECORD #	SRO	COGNITIVE LEVEL	SOURCE	NOTES
1	x	C	New(jkl)	
3	x	A	New	
4	x	C	SalNRC 6/98	
5	x	M	New(jkl)	
6	x	M	New	
7	x	A	New(jkl)	
8	x	M	NRC(Braid)	
9	x	C	New(jkl)	
10	x	A	New	
12	x	A	New	
13	x	A	New(jkl)	
14	x	A	New(jkl)	
16	x	A	BV Audit	SM
17	x	C	NRC(PI)	
18	x	A	New(jkl)	
19	x	M	New(jkl)	
20	x	M	SalNRC 2/98	SM
21	x	M	New(jkl)	
23	x	C	SalemNRC 2/99	Last two NRC-SM
25	x	C	NRC Bank	
26	x	M	NRC Bank	
27	x	M	NRC Bank	
28	x	M	NRC Bank	
29	x	C	NRC Bank	
30	x	A	New	
31	x	C	NRC Bank	
32	x	C	NRC Bank	
33	x	M	New(jkl)	
34	x	M	New(jkl)	
37	x	C	NRC Bank	
38	x	A	NRC Bank	
39	x	A	Salem NRC 9/98	Last two NRC
41	x	C	NRC Bank	SM
42	x	M	NRC Bank	

Salem 1/10/00 SRO NRC Exam Question Cognitive Level and Source Summary

RECORD #	SRO	COGNITIVE LEVEL	SOURCE	NOTES
46	x	C	Fac Bk	
49	x	C	NRC Bank	
51	x	M	New(jkl)	
52	x	M	Fac Bk	
53	x	A	New(jkl)	
54	x	M	New(jkl)	
56	x	M	SalNRC 6/98	
57	x	A	Fac Bk	
59	x	M	SalNRC 2/98	
61	x	C	SalNRC 6/98	
62	x	A	SalNRC 9/98	Last two NRC
63	x	M	SalNRC 9/98	
64	x	M	NRC Bank	
68	x	C	New(jkl)	
70	x	C	New(jkl)	
71	x	M	Fac Bk	
73	x	M	NRC(Braid)	SM
75	x	M	Fac Bk	SM
76	x	M	Fac Bk	SM
77	x	M	SalNRC 2/98	
78	x	M	New(jkl)	
79	x	M	SalNRC 2/98	
80	x	M	SalNRC 6/98	
81	x	A	NRC(Braid)	SM
82	x	C	New(jkl)	
83	x	C	New(jkl)	
84	x	C	SalNRC 6/98	
86	x	A	NRC(PI)	
87	x	C	Fac Bk	
88	x	M	New(jkl)	
90	x	A	New(jkl)	
91	x	C	NRC(PI)	
92	x	C	PI Bank	
93	x	M	New	

Salem 1/10/00 SRO NRC Exam Question Cognitive Level and Source Summary

RECORD #	SRO	COGNITIVE LEVEL	SOURCE	NOTES
94	x	C	New	
95	x	C	SalNRC 6/98	
96	x	M	New	
97	x	A	NRC(WC)	
98	x	M	New(jkl)	
99	x	M	SalNRC 6/98	
100	x	C	NRC Bank	
101	x	M	Fac Bk	SM
102	x	M	Fac Bk	
103	x	M	New(jkl)	
104	x	M	NRC(Byron)	
106	x	M	New(jkl)	
107	x	C	New(jkl)	
108	x	M	NRC(PI)	
110	x	M	NRC(Braid)	SM
111	x	C	Fac Bk	
112	x	M	SalNRC 6/98	
113	x	C	New(jkl)	
114	x	C	Fac Bk	
115	x	M	Fac Bk	SM
116	x	M	SalNRC 2/97	
117	x	C	New	
118	x	M	PI Bank	
119	x	M	Fac Bk	SM
120	x	C	New(jkl)	
121	x	A	SalNRC 2/99	Last two NRC-SM
122	x	C	NRC(Braid)	SM
123	x	M	NRC(Braid)	
124	x	C	New(jkl)	
125	x	C	New(jkl)	
126	x	M	Fac Bk	SM
127	x	C	New(jkl)	

Salem January 2000 RO Written ExaminationQuestion Source Statistic Summary

QUESTION SOURCE	Memory	Comp/Applic
Salem Bank	2	6
Salem Bank - Significantly Modified	5	1
Salem NRC Exams (not in Salem Bank)	11	5
Previous 2 Salem NRC Exams	None	3
NRC Bank	6	9
Other Facility (not in Salem Bank)	7	9
New	15	21
Total	46	54

Salem 1/10/00 RO NRC Exam Question Cognitive Level and Source Summary

RECORD #	RO	COGNITIVE LEVEL	SOURCE	NOTES
2	x	C	NRC(PI)	SM
5	x	M	New(jkl)	
6	x	M	New	
8	x	M	NRC(Braid)	
9	x	C	New(jkl)	
10	x	A	New	
11	x	C	New	
12	x	A	New	
13	x	A	New(jkl)	
14	x	A	New(jkl)	
15	x	C	New(jkl)	
16	x	A	BV Audit	SM
17	x	C	NRC(PI)	
22	x	M	New(jkl)	
23	x	C	SalemNRC 2/99	Last two NRC-SM
24	x	C	NRC Bank	
26	x	M	NRC Bank	
27	x	M	NRC Bank	
28	x	M	NRC Bank	
29	x	C	NRC Bank	
30	x	A	New	
31	x	C	NRC Bank	
32	x	C	NRC Bank	
33	x	M	New(jkl)	
34	x	M	New(jkl)	
35	x	C	New	
36	x	M	NRC Bank	
37	x	C	NRC Bank	
38	x	A	NRC Bank	
40	x	C	New(jkl)	
41	x	C	NRC Bank	SM
43	x	M	New	
44	x	M	NRC Bank	
45	x	C	SalNRC 6/98	

Salem 1/10/00 RO NRC Exam Question Cognitive Level and Source Summary

RECORD #	RO	COGNITIVE LEVEL	SOURCE	NOTES
46	x	C	Fac Bk	
47	x	A	NRC Bank	
48	x	C	Fac Bk	SM
50	x	C	SalNRC 2/98	
51	x	M	New(jkl)	
52	x	M	Fac Bk	
53	x	A	New(jkl)	
54	x	M	New(jkl)	
55	x	C	Fac Bk	
56	x	M	SalNRC 6/98	
57	x	A	Fac Bk	
58	x	M	SalNRC 2/98	
59	x	M	SalNRC 2/98	
60	x	C	New	
61	x	C	SalNRC 6/98	
62	x	A	SalNRC 9/98	Last two NRC
63	x	M	SalNRC 9/98	
65	x	A	Fac Bk	SM
66	x	C	New	
67	x	C	SalNRC 6/98	
69	x	C	NRC Bank	
71	x	M	Fac Bk	
72	x	M	New	
73	x	M	NRC(Braid)	SM
74	x	M	SalNRC 6/98	
75	x	M	Fac Bk	SM
77	x	M	SalNRC 2/98	
78	x	M	New(jkl)	
79	x	M	SalNRC 2/98	
80	x	M	SalNRC 6/98	
82	x	C	New(jkl)	
83	x	C	New(jkl)	
85	x	M	NRC Bank	
86	x	A	NRC(PI)	

Salem 1/10/00 RO NRC Exam Question Cognitive Level and Source Summary

RECORD #	RO	COGNITIVE LEVEL	SOURCE	NOTES
87	x	C	Fac Bk	
88	x	M	New(jkl)	
89	x	M	New(jkl)	
91	x	C	NRC(PI)	
92	x	C	PI Bank	
93	x	M	New	
94	x	C	New	
95	x	C	SalNRC 6/98	
96	x	M	New	
97	x	A	NRC(WC)	
99	x	M	SalNRC 6/98	
101	x	M	Fac Bk	SM
104	x	M	NRC(Byron)	
105	x	C	NRC(Braid)	
106	x	M	New(jkl)	
107	x	C	New(jkl)	
109	x	M	NRC(Braid)	
110	x	M	NRC(Braid)	SM
111	x	C	Fac Bk	
112	x	M	SalNRC 6/98	
114	x	C	Fac Bk	
115	x	M	Fac Bk	SM
116	x	M	SalNRC 2/97	
117	x	C	New	
118	x	M	PI Bank	
119	x	M	Fac Bk	SM
120	x	C	New(jkl)	
121	x	A	SalNRC 2/99	Last two NRC-SM
122	x	C	NRC(Braid)	SM
123	x	M	NRC(Braid)	
125	x	C	New(jkl)	
127	x	C	New(jkl)	

Question: Working hours

Which one of the following is an "hours of work" situation requiring approval by the Operations Manager?

a. A NEO works the 1900-0700 starting Sunday night, ending Thursday morning. He accepts overtime for Saturday, 0700-1900, and his normal shift will resume Sunday at 0700.

b. An OS on staff assignment works 0700-1600 doing procedure work. The next day she fills the OS position for the 0700-1900 shift and then stays until 2100 doing procedure work, prior to starting vacation.

c. Prior to starting days off, a CRS works the last 1900-0700 shift but is called in to work 1600-1900 that same day, to temporarily cover for a person who experienced a family emergency.

d. A RO works from 0700-2100 because his relief is involved in an automobile accident. He is scheduled to work the 0700-1900 shift on the following day.

Answer d **Exam Level** S **Cognitive Level** Comprehension
Record Number: 1 **RO Number:** **SRO Number:** 1
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.1 Conduct of Operations
2.1.1 Knowledge of conduct of operations requirements. 3.7 3.8

Explanation: Limits are 16 straight hours, 16 in any 24 hour period, 24 in any 48 hour period, 72 hours in any 7 day period, 8 hour break between shifts. d. – Correct, >24 in 48; a. - <72 in 7 days; b. - <16 consecutive, <24 In 48; c. - <16 in 24, >8 hours off between shifts.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
STATION OPERATING PRACTICES	NC.NA-AP.ZZ-0005(Q), Sect. 5.10, pg. 11				
LESSON PLAN	300-000.00S-CONDOP, Obj. 9				

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: Log review

A licensed reactor operator (RO) is currently assigned to administrative duties reviewing proposed abnormal procedure revisions. On Wednesday, the operator was required to cover the position of Unit 1 RO for the 0700-1900 due to illness of the assigned person. At 1900 on the following Sunday evening, the operator again assumed the position of Unit 1 RO at turnover.

Which one of the choices correctly completes the following sentence regarding review of the Unit 1 Control Room Narrative Log, following shift turnover on Sunday?

The operator must review the Unit 1 Control Room Narrative Log(s) back to . . . .

a. 1900, Wednesday

b. 1900, Thursday

c. 1900, Friday

d. 1900, Saturday

Answer a **Exam Level** R **Cognitive Level** Comprehension
Record Number: 2 **RO Number:** 1 **SRO Number:**
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.3 Knowledge of shift turnover practices. 3.0 3.4

Explanation: a. – Correct. The on-coming operator will review the balance of unreviewed material generated within the previous 5 days after turnover is complete. The balance of unreviewed material exists from the end of the operator's shift on Wednesday; b. - Must review the narrative logs for the previous 72 hours or last time on shift, whichever time is shorter, prior to turnover. c.&d. – 48 and 24 hrs. prior is incorrect for post-turnover review

Reference Title	Facility Reference Number	SectionPage	Revision	L. O.
SHIFT TURNOVER RESPONSIBILITIES	SH.OP-AP.ZZ-0107(Q)	5.3.1	5	
SHIFT TURNOVER AND LOGKEEPING	0300-000.00S-TNOVER-01	II.D.2.d	17	4, 5b

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Prairie Island 5/1999

Question: TS ACTIONS evaluation

Given the following conditions for Unit 2:

- S/G tube leak has been identified at 0.25 gpm
- Chemistry has confirmed rising SG activity levels
- The unit was tripped from 10% power and transition has been made to 2-EOP-TRIP-2 "REACTOR TRIP RESPONSE"
- RCS Tave is 547°F and a cooldown is being initiated as directed by S2.OP-AB.SG-0001(Q) "STEAM GENERATOR TUBE LEAK" (AB.SG-1)
- Source Range (SR) channel N-32 failed earlier and, while I&C is checking the failed channel, misoperation results in failure of SR Channel N-31

Which one of the following describes the action(s) to be taken with respect to the Unit 2 Technical Specifications?

a. Stabilize RCS temperature. Within ONE hour perform a shutdown margin surveillance, then continue with RCS cooldown.

b. Invoke 10CFR50.54(x) to continue the cooldown with both SR channels out-of-service. Perform a shutdown margin surveillance prior to reaching cold shutdown.

c. Make a one hour report to NRC. TS 3.0.3 will be violated when Mode 5 is entered with both SR channels out-of-service

d. Continue the cooldown as directed by AB.SG-1. Abnormal Procedures take precedence over TS actions.

Answer a **Exam Level** S **Cognitive Level** Application

Record Number: 3 **RO Number:** **SRO Number:** 2

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. 3.4 4.0

Explanation: a. – Correct. Apply the ONE hour TSAS; b. - 10CFR50.54(x) is NOT applicable since departure from TS conditions is NOT required. c. - LCO 3.0.3 is NOT applicable; d. - Procedures do NOT take precedence over T.S.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Salem - Unit 2 Technical Specifications		TABLE 3.3-1, FU 6	3/4 3-2, 3-6	113/121	
TECHNICAL SPECIFICATIONS		0300-000.00S-TECHSP-01	II.A.6, III.C.7.c, g	3-14, 22-23,13, 26	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: Surveillance time interval applicability

Given the following conditions for Unit 1:

- Today is June 23 @ 1115
- Unit is at normal operating temperature and pressure in preparation for reactor startup
- 1A Emergency Diesel Generator was declared inoperable today @1100.
- The weekly surveillance, S1.OP-ST.500-0001(Q), "ELECTRICAL POWER SYSTEMS AC SOURCES ALIGNMENT" was last performed June 16 at 1300.

Which one of the following correctly identifies the latest time for completion of this surveillance?

a. 1200 hours today.

b. 1215 hours today.

c. 1300 hours today.

d. 0700 hours on June 25.

Answer a **Exam Level** S **Cognitive Level** Comprehension
Record Number: 4 **RO Number:** **SRO Number:** 3
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC
2.1 Conduct of Operations
2.1.12 Ability to apply technical specifications for a system. 2.9 4.0

Explanation: Normal TS surveillance interval is every 7 days. Also, T.S ACTION a. requires surveillance be performed within ONE hour. a. – Correct. This meets the 1-hour requirement of ACTION statement; b. - Incorrectly applies TS 4.0.2, 1.25 extension allowance to a compensatory 1-hour ACTION statement; c. - Identifies the 7 day surveillance interval; d. – 1.25 x 7 day surveillance interval.

eference Title	Facility Reference Number	Section	Page	Revision	L. O.
ELECTRICAL POWER SYSTEMS AC SOURCES ALIGNMENT	S1.OP-ST.500-0001(Q)	1.3	2	7	
Salem - Unit 2 Technical Specifications		3.8.1.1 ACTION b	3/4 8-1	170	
Surveillances and Testing	0300-000.00S-SURV00-00	III.C.6	12		3

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: PAM NRC exam 6/98. Times modified.

Question: Independent Verification

Motor-operated valve (MOV) 2CV175, Rapid Borate Stop Valve, is being closed as part of a tagging operation.

Which one of the following describes a correct method for performing independent position verification for t valve?

a. Check local valve stem position

b. After power is removed, attempt to manually close the valve

c. Check the bezel position lights after removing electrical power from the motor operator

d. Prior to removing power, have the verifier attempt to close the valve from the control room

Answer a **Exam Level** B **Cognitive Level** Memory
Record Number: 5 **RO Number:** 2 **SRO Number:** 4

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.1 Conduct of Operations

2.1.29 Knowledge of how to conduct and verify valve lineups. 3.4 3.3

Explanation: a. - Correct. The valve stem position indication is checked locally. b. - For a MOV, the valve is NOT normally operated by handwheel for closure (prevent binding). c. - There will be no indication after power is removed; d. - This just verifies that the same operation was performed.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
STATION OPERATING PRACTICES	NC.NA-AP.ZZ-0005(Q)	Attach 6, 2.2	2	9	
CONDUCT OF OPERATIONS	0300-000.00S-CONDOP-00	II.J.3	25		10

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: Licensed power level

Which one of the following describes the requirements for maintaining the maximum allowable steady-state full power level in accordance with IOP-4 "Power Operations"?

a. Average power may exceed 100% for a 12-hour shift, but at NO time shall it exceed 102%.

b. Average power for a 12-hour shift is to be <101%. If it exceeds 102%, then power shall be reduced to ≤100% within the next hour.

c. Due to operator actions, power may exceed 100% for a short duration but at NO time shall it exceed 102%. The average power for a 12-hour shift is to be ≤100%.

d. Due to load fluctuation, power may exceed 100% for a short duration but at NO time shall it exceed 102%. The average power for a 12-hour shift is to be ≤100%.

Answer d **Exam Level** B **Cognitive Level** Memory
Record Number: 6 **RO Number:** 3 **SRO Number:** 5
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
Conduct of Operations
2.1.10 Knowledge of conditions and limitations in the facility license. 2.7 3.9

Explanation: The average power level over a 12-hour shift should not exceed 100%. Intentional power excursions greater that 100 % are not allowed. These guidelines are applicable to unintentional power excursions and are reportable if exceeded: 1) Power excursions to 100.5% for up to one hour; 2) Power excursions to 101.0% for up to 30 minutes 3) Power excursions to 102.0% for up to 15 minutes. a. – NOT allowed to exceed 100% for the shift; b. – power cannot exceed 102%; c. – Intentional power excursions > 100% are NOT allowed; d. – correct answer, unintentional power level changes may occur up to 102% power within given time limits.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
IOP–4, POWER OPERATIONS	0300-000.00S-IOP004-01	II.C.6.b	12		2
~2>OP-IO.ZZ-0004, Power Operations, Precaution 3.7					

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: Responsibility in troubleshooting

Given the following conditions on Unit 2:

- Abnormal Service Water system pressure indications have been observed
- The System Manager directs that troubleshooting be initiated in accordance with SH.OP-AP.ZZ-0008(Q), "Operations Troubleshooting And Evolutions Plan Development"

Which one of the following is a responsibility of the CRS prior to initiating troubleshooting activities?

a. Approving the troubleshooting plan but only if it is evaluated by the system manager as a HIGH RISK or VERY HIGH RISK evolution

b. Independent verification of the proper installation of test equipment specified by the maintenance supervisor and/or the system manager

c. Approval of any system manager waiver of a 10CFR50.59 review requirement

d. Determining the risk level for the troubleshooting evolution

Answer d **Exam Level** S **Cognitive Level** Application

Record Number: 7 **RO Number:** **SRO Number:** 6

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.2 Equipment Control

2.2.20 Knowledge of the process for managing troubleshooting activities. 2.2 3.3

Explanation: d.-Correct. CRS determines risk level for all troubleshooting plans; a.-System Manager reviews all plans at this risk level but does not assign risk level; b.-Independent verification is performed by qualified individuals. The CRS is not qualified to evaluate all installations; c.-A 10CFR50.59 review requirement cannot be waived.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
OPERATIONS TROUBLESHOOTING AND EVOLUTIONS PLAN DEVELOPMENT	SH.OP-AP.ZZ-0008(Q)	3.3.1	3	0	
CONDUCT OF OPERATIONS	0300-000.00S-CONDOP-00	III.A.3.b.3)	11		2
WORK CONTROL AND DCR PROCESS	0300-000.00S-WORK00-00				8

Material Required for Examination

Question Source: New(jkl) **Question Modification Method:**

Question Source Comments:

Question: Out of Spec values

The control room readings are being logged by the NCO. The NCO has made a red circle around the reading for 1PI-936A, 11 Accumulator Pressure.

'hich one of the following is indicated by the circled value?

a. The indicator is fluctuating within the log limits but may be failing

b. The reading must be independently verified by the Shift Technical Advisor (STA)

c. The data falls outside the limits specified in the LCO statement

d. Accumulator pressure has changed by >±5% since the previous reading but is still within specifications

Answer c **Exam Level** B **Cognitive Level** Memory
Record Number: 8 **RO Number:** 4 **SRO Number:** 7
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.2 Equipment Control
2.2.23 Ability to track limiting conditions for operations. 2.6 3.8

Explanation: c. Correct. Out of specification readings should be circled in red ink. a. – This is a maintenance issue and should be addressed via maintenance notification; b. – Independent verification of log readings is not required; d. – Readings could change by 5% and, while requiring investigation/explanation, still be within the band.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
OPERATIONS STANDARDS	SH.OP-DD.ZZ-0004 (Z)	5.4.4.D.	39	3	
SHIFT TURNOVER AND LOGKEEPING	0300-000.00S-TNOVER-01	V.B.6.e	20		8

'laterial Required for Examination
.uestion Source: NRC Exam Bank **Question Modification Method:** Editorially Modified
Question Source Comments: Braidwood 1999 NRC exam

Question: Breaker re-closure

Unit 1 is currently in hot shutdown (HSD), heating up to hot standby (HSB) when the RO attempts to open 11SJ54, SI Accumulator Stop Valve, in accordance with S1.OP-IO.ZZ-0002, CSD to HSB. The valve fails to ſ ke open and the NEO sent to investigate reports that the breaker is tripped.

Which one of the following describes the correct action for the crew?

a. Unseat the valve manually then reset and re-close the breaker. Under these conditions, two more attempts to stroke the valve are permitted.

b. Unseat the valve manually then reset and re-close the breaker. Under these conditions, one additional attempt to stroke the valve is permitted.

c. Refer to technical specifications and initiate a Notification to have maintenance investigate the problem.

d. Dispatch a NEO to open the valve manually. Then reset the breaker but red tag it open and inform the Shift Electrician of the valve operation problem.

Answer c **Exam Level** B **Cognitive Level** Comprehension
Record Number: 9 **RO Number:** 5 **SRO Number:** 8
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.2 Equipment Control
2.2.24 Ability to analyze the affect of maintenance activities on LCO status 2.6/3.8
Explanation: c. – Correct, no abnormal plant condition or situation exists; a – Three attempts is a misuse of the limitations for stroking a MOV in any one hour period; b. – is for conditions where an abnormal plant condition or situation exists; d – while the valve may ultimately be de-energized and opened, the procedure requires an AR (now a Notification).

eference Title	Facility Reference Number	Section	Page	Revision	L. O.
PROTECTIVE CIRCUIT/BREAKER RESET AND RECLOSURE POLICY	SC.OP-DD.ZZ-0006(Z)	5.1.1.a	3	0	
MISCELLANEOUS DIRECTIVES	0300-000.00S-MISCOD-01	II.C.4	8-9		3

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: Fuel Movement

Unit 2 has been shutdown for refueling in accordance with the following schedule:

- 1/3/00, 0600 hours - Unit entered MODE 3
- 1/5/00, 2000 hours - Unit entered MODE 4
- 1/7/00, 1000 hours - Unit entered MODE 5
- 1/9/00, 1600 hours - Unit entered MODE 6

Which one of the following is the earliest date and time that spent fuel movement in the reactor vessel is permissible?

a. 1/7/00, 1001 hours

b. 1/10/00, 0601 hours

c. 1/11/00, 1401 hours

d. 1/14/00, 1001 hours

Answer b **Exam Level** B **Cognitive Level** Application
Record Number: 10 **RO Number:** 6 **SRO Number:** 9
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.2 Equipment Control
2.2.28 Knowledge of new and spent fuel movement procedures. 2.6 3.5

Explanation: b. – Correct. TS requires the reactor to be sub-critical for at least 168 hours (7 days); a. - Based on 100 hours (4 days + 4 hours) from MODE 4 entry. Several surveillances must be performed within 100 hours of fuel movement. and 100 hrs. is the old T.S. requirement; c.&d. – 100 and 168 hours from Mode 5 entry.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
ƆLD SHUTDOWN TO REFUELING	S2.OP-IO.ZZ-0007(Q)	5.1.9F		2C	
Salem Unit 2 Technical Specifications		LCO 3.9.3	3/4 9-3	131	
REFUELING SYSTEM	0300-000.00S-REFUEL-00	VIII.A.3	48		10, 12

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: Core reactivity

In accordance with SH.OP-DD.ZZ-0004, Operations Standards, which one of the following is an activity that directly affects core reactivity during power operations?

a. Starting a second charging pump

b. Initiating flow to the S/Gs from a motor-driven AFW pump

c. Tripping one running S/G Feed Pump at 50% power

d. Placing Reactor Makeup controls in AUTO following a boration

Answer b **Exam Level** R **Cognitive Level** Comprehension
Record Number: 11 **RO Number:** 7 **SRO Number:**

Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

GENERIC

2.2 Equipment Control

2.2.34 Knowledge of the process for determining the internal and external effects on core reactivity. 2.8 3.2

Explanation: b. correct - Initiating flow from AFW adds relatively cold water (90°F) to S/Gs. This removes additional heat from the primary (Tcold), resulting in a positive reactivity addition to the core. a. - Starting a charging pump in equilibrium conditions will NOT change reactivity. c. - Tripping one SGFP at <65% power has no measurable effect on reactivity d. - Placing Makeup Controls in AUTO at the proper boron concentration has no affect on reactivity.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
OPERATIONS STANDARDS	SH.OP-DD.ZZ-0004 (Z)	4.1.1.E, 4.1.4	12, 14	3	
CONDUCT OF OPERATIONS	0300-000.00S-CONDOP-00				7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: Radiation exposure limits for PSE (Emergency)

Given the following information for an operator:

- Age is 47years
- Total lifetime exposure is 9200 mRem TEDE
- Current year exposure is 900 mRem TEDE

A Site Area Emergency has been declared due to a LOCA outside containment, with limited makeup to the RWST available. The operator volunteers to make an emergency entry into the penetration area to attempt to isolate the leak. This action would result in a significant reduction in offsite dose. The action has been properly approved.

Which one of the following is the maximum allowed exposure (TEDE) the operator may receive while performing this action?

a. 2100 mRem TEDE.

b. 3600 mRem TEDE.

c. 24,100 mRem TEDE.

d. 25,000 mRem TEDE.

Answer d **Exam Level** B **Cognitive Level** Application
Record Number: 12 **RO Number:** 8 **SRO Number:** 10
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1

2.3 Radiation Control
2.3.1 Knowledge of 10 CFR: 20 and related facility radiation control requirements. 2.6 3.0

(planation: d. - correct, Emergency Exposure Limit for accident mitigation is 25 rem TEDE. This limit is for the event in progress. a. - PSE&G provides for normal routine administrative exposure control level of 3000 mRem TEDE/year. If the admin. limit is incorrectly applied, then 2100 mRem TEDE (3000 - 900) would take the NEO to his annual limit. b. - In the event of an emergency declaration of ALERT or higher, the annual limit is automatically extended to 4500 mRem, leaving 3600 mRem TEDE (4500 - 900) to reach the annual limit. c. - Incorrectly applying the current yearly exposure to the PSE limit gives 24,100 mRem (25, 000-900).

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RADIATION PROTECTION PROGRAM	0300-000.00S-RADCON-01	IV.E.3.j	17		3
NC.NA-AP.ZZ-0024, Radiation Protection Program					

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: ALARA Determination

Unit 2 is in cold leg recirculation due to a LOCA and the Operations Superintendent has declared a Site Area Emergency. Four manual valves need to be operated in the Auxiliary Building in order to enhance C 'W cooling capability. While important to the overall mitigation strategy, operation of the four valves is not a plant equipment or life-saving operation. Radiation levels in the work area are much higher than normal and airborne radiation levels have escalated due to pump seal leakage. The general area radiation level is 2 R/hr and isotopic analysis is such that, without a respirator, personnel would eventually receive 100 mR (TEDE) for each ten minutes spent in the area. The OSC Coordinator estimates that two operators will each spend 20 minutes performing their part of the job without a respirator and 30 minutes with a respirator.

Which one of the following describes the correct application of ALARA principles?

a. The operators should perform the job wearing a respirator

b. The operators should perform the job without a respirator

c. Do NOT dispatch the operators since operation of these valves is not a plant equipment or life saving action

d. Process a dose limit extension in case the operators exceed their first level administrative dose limit

Answer b **Exam Level** B **Cognitive Level** Application
Record Number: 13 **RO Number:** 9 **SRO Number:** 11
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.3 Radiation Control
2.3.2 Knowledge of facility ALARA program. 2.5 2.9

Explanation: b – Correct. 20/60(2000) + 20/10(100) = ~867 mR; a – 30/60(2000) = 1000 mR; c – Since it is important to the overall mitigation strategy and < than even normal administrative limits, the work can be performed; d – The dose limit is automatically extended to 4500 MR for ERO personnel.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RADIATION PROTECTION PROGRAM	0300-000.00S-RADCON-01		XVII.B	28	4
NC.NA-AP.ZZ-0024, Radiation Protection Program					

Material Required for Examination
Question Source: New(jkl) **Question Modification Method:**

Question Source Comments:

Question: Radiation exposure and IV

A procedure requires independent verification (IV) on a group of valves located inside a radiation area. The dose rate is 50 mR/hr and it is projected that the two operators will each have to spend 20 minutes in the area in order to perform the task.

Which one of the following describes the correct process for performing this IV?

a. Two operators who have sufficient margin to perform the task and yet still remain below administrative dose limits shall be assigned to do a "hands on" IV

b. The IV is not required if the Unit CRS and the WCC SRO verify that none of the valves have been re-positioned since the last IV

c. Based on the ALARA concept, the Operations Superintendent has the authority to waive any IV requirement when entry into a defined radiation area is necessary to perform the task

d. The "hands on IV" can be waived by the Unit CRS. However, an alternative means of IV via observation of process parameters/indications is required

Answer d **Exam Level** B **Cognitive Level** Application
Record Number: 14 **RO Number:** 10 **SRO Number:** 12
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.3 Radiation Control
2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. 2.9 3.3

Explanation: The threshold for not requiring "hands on IV" is 10 mR. d. – Correct, IV shall be accomplished by observation of process parameters, etc.; a. – 10 mR will be exceeded. "Hands on IV" should not be performed; b., c. – IV is not waived.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
STATION OPERATING PRACTICES	NC.NA-AP.ZZ-0005(Q)	Attach 6, 1.4	1	9	
CONDUCT OF OPERATIONS	0300-000.00S-CONDOP-00	III.J	24		10

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: RO responsibilities

A LOCA has occurred on Unit 2. The crew initiated a manual reactor trip/safety injection and has entered EOP-TRIP-1. RCS pressure is steadily trending down and has dropped below 1500 psig. While in route to t control room, the shift technical advisor (STA) went into labor and is unable to report and fill the position.

Which one of the following correctly identifies both the position responsible for monitoring the continuous action summaries (CAS) and when 2CV139 and 2CV140 can be closed?

a. Reactor Operator (RO) and Plant Operator (PO) monitor and report parameter values as the CAS's are read by the CRS, when each EOP page is turned. The valves can be closed any time after the functional restoration procedure implementation step (20) in EOP-TRIP-1.

b. RO and PO monitor parameters associated with the CAS's. The valves can be closed any time after the immediate actions of EOP-TRIP-1 have been verified.

c. RO monitors parameters associated with the CAS's. The valves can be closed any time after the functional restoration procedure implementation Step (20) in EOP-TRIP-1.

d. RO and PO monitor and report values when the CAS's are read by the CRS, at page 2 of EOP-TRIP-1 and at each procedure transition. The valves can be closed any time after the immediate actions of EOP-TRIP-1 have been verified.

Answer b **Exam Level** R **Cognitive Level** Comprehension
Record Number: 15 **RO Number:** 11 **SRO Number:**
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.4 Emergency Procedures / Plan
2.4.13 Knowledge of crew roles and responsibilities during EOP flowchart use. 3.3 3.9
Explanation: b. – Correct; a – Former practice; c. – Incorrect implementation point; d. – CAS's are not read by CRS.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Use of Procedures, SC.OP-AP.ZZ-0102	Sect. 5.3.5.f				

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: FR implementation

Given the following conditions for Unit 2:

- A break has occurred on 22 main steam line and 22MS167 cannot be closed
- The other S/Gs were initially overfed with current levels approximately 30% NR
- A reactor trip and SI have occurred
- Pressurizer pressure is 1300 psig and slowly lowering
- Pressurizer level is off-scale low
- 2-EOP-TRIP-1 Step 21, RCS Temperature Control, is being performed
- The RO reports the lowest loop Tcold is 260°F and slowly lowering

Which one of the following identifies the correct process of EOP implementation?

a. Immediately transition to FRTS-1 "Response To Imminent Pressurized Thermal Shock Conditions".

b. Immediately transition to FRTS-2 "Response To Anticipated Pressurized Thermal Shock Conditions".

c. Complete actions of TRIP-1 through Faulted S/G Evaluation, transition to LOSC-1 "Loss of Secondary Coolant" and then immediately transition to FRTS-1 "Response To Imminent Pressurized Thermal Shock Conditions".

d. Complete actions of TRIP-1 through Faulted S/G Evaluation, transition to LOSC-1 "Loss of Secondary Coolant" and then immediately transition to FRTS-2 "Response To Anticipated Pressurized Thermal Shock Conditions".

Answer c **Exam Level** B **Cognitive Level** Application
cord Number: 16 **RO Number:** 12 **SRO Number:** 13
ier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.4 Emergency Procedures / Plan
2.4.16 Knowledge of EOP implementation hierarchy and coordination with other support procedures. 3.0 4.0

Explanation: This is a Thermal Shock RED Path c.- Correct. The CSFTs are NOT implemented until transition from TRIP-1 occurs. This will occur when Faulted S/G is determined in TRIP-1 and crew is directed to LOSC-1. a.&b. – Functional restorations are not implemented until directed or after transition from TRIP-1. d. – Transition is to FRTS-1.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
USE OF PROCEDURES	SC.OP-AP.ZZ-0102(Q)	5.3.12.C; E.2	19-20	7	
CRITICAL SAFETY FUNCTION STATUS TREES	2-EOP-CFST-1	1.0, Figs. 4, 4A	2	23	
USE AND CONTROL OF PROCEDURES	0300-000.00S-PROCED-02	III.E.14.c & e.2	28-29		3

Material Required for Examination 2-EOP-CSFST-1 Figs 4 & 4A
Question Source: Other Facility **Question Modification Method:** Significantly Modified
Question Source Comments: Beaver Valley Audit exam

Question: Loss of AC path

Given the following conditions on Unit 2:

- A LOCA has occurred
- Safety Injection is actuated
- Actions of 2-EOP-TRIP-1, "REACTOR TRIP OR SAFETY INJECTION" were initiated
- When the main turbine tripped, all AC power was lost for the Site
- The crew has initiated actions of 2-EOP-LOPA-1, "LOSS OF ALL AC POWER"
- The crew notes the following for the Critical Safety Function Status Trees:
  (a) PURPLE path condition exists for the Core Cooling Status Tree
  (b) RED path condition exists for the Containment Environment Status Tree

Which one of the following is the correct action for these conditions?

a. Continue the actions of 2-EOP-LOPA-1"LOSS OF ALL AC POWER"

b. Transition to 2-EOP-LOPA-3 "LOSS OF ALL AC POWER RECOVERY/SI REQUIRED"

c. Transition to 2-EOP-FRCE-1, "RESPONSE TO EXCESSIVE CONTAINMENT PRESSURE"

d. Transition to 2-EOP-FRCC-2, "RESPONSE TO DEGRADED CORE COOLING"

Answer a **Exam Level** B **Cognitive Level** Comprehension
Record Number: 17 **RO Number:** 13 **SRO Number:** 14
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.4 Emergency Procedures / Plan
2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control; 2. Core cooling and heat removal; 3. Reactor coolant system integrity; 4. Containment conditions; 5. Radioactivity release control. 3.7 4.3

Explanation: a. – Correct. During a loss of all AC power, the Critical Safety Functions are monitored for information only. The availability of at least one train of safeguards AC power is assumed on entry into the Function Restoration procedures. b. – This is a recovery procedure which may be implemented after power is restored c. - If AC power had not been lost and transition had been made from TRIP-1, or if power had been restored and the appropriate actions of LOPA-1 completed, then transition to the highest-ranked Critical Safety Function, the RED path on Containment Environment. d. - When evaluating and choosing appropriate Critical Safety function actions, the (highest-ranked) RED path is addressed first, and once these actions are completed, then the PURPLE Path is considered.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF ALL AC POWER	2-EOP-LOPA-1	Step 1 NOTE	1	22	
EOP-LOPA-1, 2, 3; LOSS OF ALL AC POWER AND RECOVERY	0300-000.00S-LOPA00-02	4.3.1	29		7, 8
USE AND CONTROL OF PROCEDURES	0300-000.00S-PROCED-02	III.E.14	28-30		

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Prairie Island 1998 NRC RO Exam.

Question: Use of ECG

Unit 2 is at 100% power when a loss of all overhead annunciators occurs for greater than 15 minutes. The Emergency Coordinator, in this case the Operations Superintendent, properly classified the problem as an I` 'JSUAL EVENT (UE) and provided the Initial Contact Message Form to the Primary Communicator. However, before the Primary Communicator makes any of the "within 15 minute notifications", the problem is corrected and the annunciators are restored.

Which one of the following describes the correct course of action for the Emergency Coordinator?

a. Complete the actions for declaration of a UE and then terminate IAW the proper attachments

b. Complete the actions for declaration of a UE and then issue a retraction IAW the ECG

c. Make a 4 Hour Report in accordance with ECG Sect. 11.10 "Voluntary Notifications"

d. Make a 1 Hour Report in accordance with ECG Sect. 11.6 "After the Fact"

Answer a **Exam Level** S **Cognitive Level** Application
Record Number: 18 **RO Number:** **SRO Number:** 15
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.4 Emergency Procedures / Plan
2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies. 2.2 3.6

Explanation: a. – Correct. The EAL has been exceeded, regardless of the current status; b. – retractions are for incorrect/improper reports; c. – This category is for voluntary/courtesy reports not clearly meeting existing EAL's or RAL's; d. – After the Fact reports are for events that are not ongoing but a report should have been made when it was discovered or ongoing.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Event Classification Guide	SGS ECG	Applicable Sections			
Lesson Plan	ON SHIFT EMERGENCY RESPONSE, Obj. 1.0				

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: Loss of Annunciator

During implementation of abnormal procedure S2.OP-AB.ANN-0001, Loss of Overhead Annunciators, operators are directed to perform the following step:

.2 PERFORM System Lamp Test and VERIFY at least two of the following occur:

- OHA Window A-9, alarms or reflashes
- OHA CRT displays 11 logic error alarms or reflashes
- OHA local printer cabinet, ANN115-2, displays incoming alarms

Which one of the following correctly describes the basis for performing that step?

a. Any 2/3 of those validates that the system is not functional and will not actuate on a valid alarm

b. 2/3 of those indicates that a single component may have failed but the system is responding and will actuate on a valid alarm

c. The step identifies the specific source of the OHA problem

d. Failure of any one of those to actuate indicates that SER B is in command and system capability is limited

Answer b **Exam Level** S **Cognitive Level** Memory
Record Number: 19 **RO Number:** **SRO Number:** 16
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.4 Emergency Procedures / Plan
2.4.32 Knowledge of operator response to loss of all annunciators. 3.3 3.5

Explanation: b – Correct, per AB.ANN-1 Basis; a. – Reciprocal of b.; c – The step is part of the diagnoses for the operator to determine if the system will actuate, not to isolate the problem; d – SER operation is indicated by the status of the LED's.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
OVERHEAD ANNUNCIATOR SYSTEM	0300-000.00S-OHA000-00	IX.C.1	33		14
S2.OP-AB.ANN-0001, Basis section					

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: Emergency on one unit

Given the following conditions:

- A Unit 1 reactor startup is in progress with Control Bank A presently being withdrawn
- The third NCO has started 12 SI Pump to refill 11 SI Accumulator due a low level alarm
- Reactor Engineering is using the fuel handling crane bridge to verify serial numbers for several spent fuel assemblies in Unit 1 Spent Fuel Pool
- Unit 2 is at 100% power
- Unit 3 is running and is synchronized to the grid for peak load support

Which one of the following actions is required if Unit 2 experiences a reactor trip/safety injection and an ALERT is declared by the Operations Superintendent?

a. Unload and shutdown Unit 3.

b. Insert all control rods on Unit 1.

c. Dispatch a NEO to evacuate all personnel from Unit 1 FHB and disable the bridge.

d. Secure the lineup to fill 11 SI Accumulator and stop the 12 SI Pump.

Answer b **Exam Level** S **Cognitive Level** Memory
Record Number: 20 **RO Number:** **SRO Number:** 17
Tier: Generic Knowledge and Abilities **RO Group:** 1 **SRO Group:** 1
GENERIC
2.4 Emergency Procedures / Plan
2.4.38 Ability to take actions called for in the facility emergency plan, including (if required) supporting or acting 2.2 4.0

Explanation: b. – Correct. When an emergency event of ALERT or higher is declared on a unit and the unaffected unit is being started up and the turbine is not synchronized to the grid, return the unit to Hot Standby with all control rod banks fully inserted. a. - Operation of Unit 3 is NOT addressed, by the procedure. c. - Fuel handling operations are addressed, but only to determine whether fuel movement should be continued. Specific actions address fuel movement only. d. - Specific unaffected unit system operations are NOT addressed. Operations required to meet Technical Specification LCOs should continue.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
OPERATING WITH AN EMERGENCY ON OPPOSITE UNIT	SC.OP-DD.ZZ-0039(Z)	5.8	3	2	
MISCELLANEOUS DIRECTIVES	0300-000.00S-MISCOD-01	III.B.4.f	7		2

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: CGroup NRC Q97. Alter to change correct answer. Conditions, including accident event, modified.

Question: Xenon transient

Abnormal procedure S2.OP-AB.ROD-0002, Dropped Rod, states that a dropped rod shall be recovered at a rate of withdrawal determined and approved by Reactor Engineering.

Which one of the following correctly describes the basis for that statement?

a. Reactor Engineering must ensure that the rate of rod withdrawal does not result in a reactivity addition in excess of the value of the delayed neutron fraction, for that point in core life

b. Continuous withdrawal to the bank position may cause a positive rate reactor trip

c. Dependent on how long the rod has been dropped, continuous withdrawal to the bank position could cause fuel damage

d. Withdrawing the rod at 48 steps per minute would require the initiation of rapid boration to offset the rate of reactivity change

Answer c **Exam Level** S **Cognitive Level** Memory
Record Number: 21 **RO Number:** **SRO Number:** 18

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

A2. Ability to (a) predict the impacts of the following on the CRDS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.06 Effects of transient xenon on reactivity 3.4 3.7

Explanation: c. – Correct, per AB.ROD-2 Basis Document; a. – Rod speeds of entire banks are not adjusted over core life; b. – Withdrawal of a single rod will only affect one quadrant. The trip is 2/4; d. – Turbine load can be adjusted as necessary to maintain Tave on target.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
S2.OP-AB.ROD-0002, Dropped Rod, Basis Document					
Lesson Plan: 300.00S-ABROD2, Obj. 2					

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: Rod Control Mode selection

When a dropped rod in Control Bank D is being recovered, the CBD position is selected on the Rod Bank Selector Switch (RBSS).

Which one of the following correctly describes a reason for that switch position selection?

a. Using the CBD position prevents actuation of an URGENT FAILURE alarm during the rod withdrawal

b. Using the CBD position ensures the Bank Overlap Unit is not tracking the affected rod motion while withdrawal is in progress

c. Using the MANUAL position would require operators to open the lift coil disconnects for the non-affected rods in all control banks

d. Using the MANUAL position would result in dropping all the rods in the opposite Control Bank D group

Answer b **Exam Level** R **Cognitive Level** Memory
Record Number: 22 **RO Number:** 14 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

K4. Knowledge of CRDS design feature(s) and/or interlock(s) which provide for the following:

K4.02 Control rod mode select control (movement control) 3.8 3.8

Explanation: b. - Correct, When Individual Bank positions are used, the Bank Overlap Unit is bypassed (GO pulses are not counted); a. – An URGENT FAILURE alarm still actuates from the opposite group; c.&d. – An URGENT FAILURE comes in from the opposite group and rod motion would be inhibited in MANUAL.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Rod Control and Position Indication Systems	RODS00-00	IV.B.8.f.7)	39		7d
S2.OP-AB.ROD-0002, Dropped Rod Basis Document					

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: Improper rod worth curve use

The Unit 2 reactor is at BOL and was manually tripped due to a feedwater problem. An Estimated Critical Position (ECP) calculation has been performed and boron concentration was adjusted for a critical rod height of Control Bank D at 128 steps. However, when determining control bank worth, personnel performing the ECP incorrectly used the EOL HZP Integral Rod Worth Curve instead of the BOL HZP curve.

Which one of the following correctly describes this how error affects critical rod height?

a. Criticality would occur below the rod insertion limit (C-58 steps).

b. Criticality would occur below the +/-300 pcm administrative limit but above the rod insertion limit.

c. Criticality would occur above the +/-300 pcm administrative limit.

d. The magnitude of the error is such that criticality cannot be achieved on rods alone.

Answer d **Exam Level** B **Cognitive Level** Comprehension
Record Number: 23 **RO Number:** 15 **SRO Number:** 19

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

001 Control Rod Drive System

K5. Knowledge of the operational implications of the following concepts as they apply to the CRDS:

K5.05 Interpretation of rod worth curves, including proper curve to use: all rods in (ARI), all rods out (ARO), hot zero power (HZP), hot full power (HFP) 3.5 3.9

Explanation: This improper curve use introduces an approx. 600 pcm error. Since the rods appear to be worth more, operators will maintain the boron concentration at a higher value and the actual reactivity addition during rod withdrawal is less. d. – Correct. There is insufficient reactivity left @D-128 on the BOL-HZP Curve to overcome the error; a. – This error is in the opposite direction but a curve reading error (C-30 instead of D-30) makes it a valid distractor; b. – This the reciprocal of the correct answer; c. – This is an error in reading from the EOL HZP to the EOL HFP Curves.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
ESTIMATED CRITICAL POSITION	0300-000.00S-ECP000-00	I.C.1; II.B.9; IV.C.1.b	10; 14;20-21		2
ESTIMATED CRITICAL POSITION	S2.RE-RA.ZZ-0001(Q)	Attachment 1, 5.1 & 6	10-11	6	
FIGURES	S2.RE-RA.ZZ-0012(Q)	Figure 4	8	38	

Material Required for Examination Rod Worth curve - S2.RE-RA.ZZ-0012(Q), Figure 4

Question Source: Previous 2 NRC Exams

Question Modification Method: Significantly modified (jkl). Changed rod position, curve reading error, choices a. and d.

Question Source Comments: DGroup

Question: Th RTD failure

Which one of the following will cause the Rod Control System to insert control rods at 72 steps per minute?

a. T-hot wide range RTD shorted.

b. T-cold wide range RTD is open.

c. T-hot narrow range RTD is open.

d. T-cold narrow range RTD is shorted.

Answer c **Exam Level** R **Cognitive Level** Comprehension
Record Number: 24 **RO Number:** 16 **SRO Number:**

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

002 Reactor Coolant System (RCS)

A1. Ability to predict and/or monitor changes in parameters associated with operating the RCS controls including:

A1.08 RCS average temperature 3.7 3.8

Explanation: c. Correct - An open RTD will be seen as maximum resistance & therefore maximum temperature. Tavg will fail high causing Rod Control to insert rods. a&b. - Tavg uses narrow range RTD for Rod Control input. d. - A shorted RTD will be seen as minimum resistance & therefore minimum temperature. Tave will decrease, but since Rod Control uses auctioneered HIGH Tave, this will have NO effect on rod motion.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Instrument Failure	0300-000.00S-ICFAIL-00	III.B.1.d.1)a)	12		1

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: Evaluation of RCS leak

Given the following conditions on Unit 2:

- Reactor power is 75%
- A leak rate surveillance indicates the following:
  - Total RCS leakage rate is 5.2 gpm
  - Leakage to PRT is 2.0 gpm
  - Leakage to the Reactor Coolant Drain Tank is 1.3 gpm
  - Total primary to secondary leakage is 0.08 gpm

Which one, if any, of the following Technical Specification leakage limits has been exceeded?

a. Identified

b. Unidentified

c. Primary to Secondary

d. Pressure Boundary

Answer b **Exam Level** S **Cognitive Level** Comprehension
Record Number: 25 **RO Number:** **SRO Number:** 20
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

002 Reactor Coolant System (RCS)

A2. Ability to (a) predict the impacts of the following on the RCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.01 Loss of coolant inventory 4.3 4.4

Explanation: b. - Correct, Unidentified leakage is TOTAL-IDENTIFIED. Identified leakage includes PRT & S/Gs. RCDT is considered unidentified. Unidentified leakage is (5.2 - (2.0+0.08)) = 3.12 gpm. The leakage limits are: Identified - 10 gpm, Unidentified - 1 gpm, each S/G - 500 gpd & total S/G - 1gpm. Therefore, Unidentified leakage limit is exceeded. a. - Identified leakage is 2.08 gpm. c. - S/G leakage is 0.08 gpm and 115.2 gpd. d. – None of this would be pressure boundary leakage.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Salem Unit 2 Technical Specifications		3.4.7.2	3/4 4-17	159	
REACTOR COOLANT SYSTEM WATER INVENTORY BALANCE	S2.OP-ST.RC-0008(Q)	Attach 3	1-3	16	

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: RVLIS

A loss of coolant accident has occurred. The RVLIS Summary Display Page is displaying dynamic range. During a cooldown and depressurization, void content indication remains constant at 80%.

'hich one of the following describes actual void content response during the cooldown and depressurization?

a. Actual void content decreased due to change in density as pressure and temperature decreased.

b. Actual void content increased due to change in density as pressure and temperature decreased.

c. Actual void content remained constant; indicated void content is compensated using pressure and temperature signals.

d. Actual void content remained constant; differential pressure is an accurate indication of void content.

Answer c **Exam Level** B **Cognitive Level** Memory
Record Number: 26 **RO Number:** 17 **SRO Number:** 21
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

002 Reactor Coolant System (RCS)

K1. Knowledge of the physical connections and/or cause-effect relationships between the RCS and the following:

K1.07 Reactor vessel level indication system 3.5 3.7

Explanation: c. - Correct, Hot leg temp and wide range press are used for density compensation in all ranges. a&b. - Indication will be constant due to compensation from temp and press. d. - dP is not accurate w/o density compensation.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM	0300-000.00S-RVLIS0-00	IV.B.8	22		4

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: Seal return alignment basis

RCS pressure is 50 psig, VCT pressure is 18 psig.

Which one of the following describes both the proper alignment of and the basis for the RCP seal 'ection/seal return alignment?

a. All No. 1 Seal return valves are closed to prevent VCT water from backflushing through the seals.

b. Seal injection is isolated to prevent excessive seal leakoff flow.

c. Seal leakoff is fully open to prevent boric acid from crystallizing and accumulating on the seal surfaces.

d. Seal injection is isolated to prevent VCT water from backfilling the RCS.

Answer a **Exam Level** B **Cognitive Level** Memory
Record Number: 27 **RO Number:** 18 **SRO Number:** 22
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1
003 Reactor Coolant Pump System (RCPS)
A4. Ability to manually operate and/or monitor in the control room:
A4.01 Seal injection 3.3 3.2

Explanation: a. - Correct, RCP seals are isolated when RCS is less than 100 psig. b. - Excessive leakoff flow will not exist at this pressure. c. - Boric acid plate out is not a problem under these conditions. d. - Not a reason for isolating seal injection.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RCP Operation	S1/S2.OP-SO.RC-0001	3.6	4	16	
RCP	0300-000.OOS-RCPUMP-01				12

Material Required for Examination
Question Source: NRC Exam Bank **Question Modification Method:**
Question Source Comments:

Question: RCP/RPS trips

With reactor power at 30%, the power supply breaker to 24 RCP trips.

Which one of the following is correct concerning the plant response with NO operator action?

a. The plant will continue at 30% power unless a SG water level trip setpoint is exceeded

b. A reactor trip will occur on low RCS flow

c. A SI will occur on high steam flow (from 21/22/23 SG's) coincident with LO-LO Tave or low steam pressure

d. A reactor trip will occur on 1/4 RCP under voltage

Answer a **Exam Level** B **Cognitive Level** Memory
Record Number: 28 **RO Number:** 19 **SRO Number:** 23

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

003 Reactor Coolant Pump System (RCPS)

K3. Knowledge of the effect that a loss or malfunction of the RCPS will have on the following:

K3.04 RPS 3.9 4.2

Explanation: a. - Correct, Power is less than P-8 (36%) but there is a feedwater transient.. b. - If power is < 36% but > 10% power, TWO RCPs must trip to automatically initiate a reactor trip. c. – The resultant increase in steam flow will still be below the SI setpoint and the coincidental setpoints will not be reached. d. - UV trips are sensed on the Group Bus, not the RCP breaker and two RCP breakers must open for a reactor trip, at this power.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RCP	0300-000.00S-RCPUMP-01	V.C.3	37		9
REACTOR PROTECTION SYSTEM	0300-000.00S-RXPROT-00	V.A	33		12

aterial Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: Charging pump flow controller in MANUAL

Given the following conditions on Unit 2:

- Reactor power is 50%
- Pressurizer level is at programmed level
- 22 Charging Pump is running
- The Master Flow Controller is in MANUAL
- Charging and letdown are balanced

Which one of the following describes the effect on the plant if the Master Flow Controller is maintained in MANUAL as power is raised to 100%?

a. Pressurizer level will rise.

b. Pressurizer level will remain the same.

c. VCT level will lower.

d. An eventual reactor trip on low pressure when the pressurizer goes empty

Answer a **Exam Level** B **Cognitive Level** Comprehension
Record Number: 29 **RO Number:** 20 **SRO Number:** 24
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

004 Chemical and Volume Control System (CVCS)

A3. Ability to monitor automatic operations of the CVCS including:

A3.10 PZR level and pressure 3.9 3.9

Explanation: a. – Correct. Pressurizer level will rise due to the decrease in RCS density as the temperature increases and the equivalent volumetric increase of the water in the RCS. b. – It cannot remain the same w/o flow adjustments. c&d. - VCT level and charging flow would not be affected by manual operation without changes in other parameters such as letdown or RCS pressure

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CVCS	0300-000.00S-CVCS00-01	V.B.2.z.2	87		4

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: Letdown temperature/pressure control

Given the following conditions on Unit 2:

- Reactor power is 50%
- The operating 21 Charging Pump has tripped
- No operator action is taken

Which one of the following will occur?

a. Letdown isolation valves, CV2 & CV277, will immediately CLOSE.

b. Charging flow control valve, CV55, will fully CLOSE until 22 Charging Pump is started

c. Letdown heat exchanger outlet temperature control valve, CC71, will OPEN.

d. Letdown heat exchanger outlet temperature control valve, CC71, and Letdown pressure control valve, will both close.

Answer d **Exam Level** B **Cognitive Level** Application
Record Number: 30 **RO Number:** 21 **SRO Number:** 25
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1
004 Chemical and Volume Control System (CVCS)
K1. Knowledge of the physical connections and/or cause-effect relationships between the CVCS and the following:
K1.18 CCWS 2.9 3.2

Explanation: d. – Correct. With all charging pump breakers open, the letdown orifice isol (CV-3/4/5) will shut. When this occurs, pressure and flow will decay in the letdown line causing both valves to close and attempt to raise pressure and temperature. a. - No signal is present to close these valves. However, the orifice isol will close. b. – CV55 is not interlocked with the charging pumps. c - With letdown isolated, temperature in the letdown line will begin to lower. CC-71 will go closed, reducing CC flow through the HX in response to the temperature drop. This is a likely choice (Reg. HX outlet temp. rising) for those not recognizing that letdown isolates.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CHEMICAL AND VOLUME CONTROL SYSTEM	0300-000.00S-CVCS00-01	IV.4.B.5 & C.10.e	34, 36	4	

Material Required for Examination
Question Source: NRC Exam Bank **Question Modification Method:**
Question Source Comments:

Question: RHR HX outlet valve failure

Given the following:

- Unit 1 is shutdown with RHR providing shutdown cooling
- The 11 RHR pump and 11 RHR Heat Exchanger are in service
- RCS pressure is 320 psig
- RCS temperature is 300°F
- RCS cooldown rate is 20°F/hr
- RHR total flow is 2000 gpm

Which one of the following will occur if the air pressure regulator to 11RH18, 11 RHR HX Outlet Flow Control Valve, failed such that air was lost to the valve operator?

a. RCS cooldown rate will rise.

b. RHR HX Component Cooling outlet temperature will lower.

c. RCS pressure will slowly rise.

d. RHR flow will lower.

Answer a **Exam Level** B **Cognitive Level** Comprehension
Record Number: 31 **RO Number:** 22 **SRO Number:** 26
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3
005 Residual Heat Removal System (RHRS)
A1. Ability to predict and/or monitor changes in parameters associated with operating the RHRS controls including:
A1.01 Heatup/cooldown rates 3.5 3.6

Explanation: a. - Correct, Loss of air to the outlet control valve will cause the valve to open. This increases overall RHR flow rate. Increased flow through the HX will increase the cooldown rate of the RCS. b. - As increased flow is applied to the HX, CCW outlet temperature will increase. c. - Pressure is not expected to rise due to this event. d. - RHR flow will rise through HX.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RHR	0300-000.00S-RHR000-01	IV.B.5.a.6)	29		4e

Material Required for Examination
Question Source: NRC Exam Bank **Question Modification Method:**
Question Source Comments:

Question: VCT suction valve ops

Which one of the following would occur if SJ1, Charging Suction from RWST, failed at the 75% open position when a safety injection signal was received?

a. Gas binding in the charging pumps when the VCT empties

b. Lower than expected boron concentration in ECCS due to dilution from VCT makeup

c. No effect, both CV40&41 close, isolating the VCT

d. Backflow from the RWST to the VCT to the in-service CVC HUT, reducing the available inventory to inject into the reactor vessel.

Answer c **Exam Level** B **Cognitive Level** Comprehension
Record Number: 32 **RO Number:** 23 **SRO Number:** 27

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

006 Emergency Core Cooling System (ECCS)

K1. Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following:

K1.08 CVCS 3.6 3.9

Explanation: c. Correct - Either SJ-1 or SJ-2 may be OPEN for the normal suction from CV-40/41 to CLOSE.
a,b & d. - CV-40/41 will go closed because SJ-2 will open.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CVCS	0300-000.00S-CVCS00-01	IV.C.18	41	6	

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: ESF setpoints

Which one of the following correctly identifies the setpoints and coincidences for the low RCS pressure automatic safety injection signal and the associated automatic unblock?

a. 2/4 ≤1765; 3/4 ≥1915

b. 2/3 ≤1765; 1/3 ≥1915

c. 2/4 ≤1765; 2/4 ≥1915

d. 2/3 ≤1765; 2/3 ≥1915

Answer d **Exam Level** B **Cognitive Level** Memory
Record Number: 33 **RO Number:** 24 **SRO Number:** 28
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2
006 Emergency Core Cooling System (ECCS)
K4. Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following:
K4.05 Autostart of HPI/LPI/SIP. 4.3 4.4
Explanation: d. – Correct, both auto SI and unblock are 2/3; a., b., c. – One or more choices with the incorrect coincidence.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
ESF		0300-000.00S-ESF000-00	VII.B.1	50	21
Operator Fluency Manual					
Logic Diagram 221055					

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: Inputs to PRT

Unit 2 was at 100% power when an automatic reactor trip and safety injection occurred. All systems responded per design.

Which one of the following correctly describes the flow path for RCP seal leakoff?

a. All #2 Seals become film-riding seals and discharge to the Reactor Coolant Drain Tank, via the standpipe

b. A relief valve in the seal return line lifts and discharges to the PRT

c. A relief valve in the seal return line lifts and discharges to the Containment Trench

d. A relief valve in the seal return line lifts and discharges upstream of the Seal Water Heat Exchanger

Answer b **Exam Level** B **Cognitive Level** Memory
Record Number: 34 **RO Number:** 25 **SRO Number:** 29

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

007 Pressurizer Relief Tank/Quench Tank System (PRTS)

K1. Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following:

K1.03 RCS 3.0 3.2

Explanation: b. – Correct, per system drawing; a – A seal return path still exists through the relief valve. #2 Seals will not become film-riding; c. – By a recent design change, many "discharges to the PRT" have been re-routed to the containment trench; d. – This would be a bypass around the Phase A isolation valves.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
PRESSURIZER AND PRESSURIZER RELIEF TANK	0300-000.00S-PZRPRT-01	IV.B.8.	26-28		3,4
P&ID 205328, Sheet 3					

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: SEC operations for CCW

Given the following:

- 21 and 22 CCW Pumps are running
- 23 CCW is selected to AUTO

Which one of the following could cause 22 CCW Pump STOP push button to start flashing?

a. A SEC MODE III actuation and the pump breaker failed to close.

b. 28 VDC power swapped to the alternate source

c. 125 VDC control power for the pump breaker has failed.

d. A SEC MODE II actuation and the pump breaker failed to close.

Answer d **Exam Level** R **Cognitive Level** Comprehension

Record Number: 35 **RO Number:** 26 **SRO Number:**

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3

008 Component Cooling Water System (CCWS)

A3. Ability to monitor automatic operations of the CCWS including:

A3.08 Automatic actions associated with the CCWS that occur as a result of a safety injection signal 3.6 3.7

Explanation: d. – Correct. In MODE II there is a start signal and the breaker failed to close. a. – There is no CCW start signal in Mode III. b.&c – Loss of 28 or 125VDC will not cause a flashing light.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
COMPONENT COOLING WATER	0300-000.00S-CCW000-02	V.A.1.f.4).b)	34		8

'aterial Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: PZR master pressure controller operation

The following plant conditions exist:

- The reactor is at 100% power
- All Pressurizer heaters are OFF
- Both pressurizer spray valves are MODULATING
- Pressurizer PORVs are CLOSED

Which one of the following RCS pressures is appropriate for the stated conditions?

a. 2215 psig

b. 2223 psig

c. 2272 psig

d. 2340 psig

Answer c **Exam Level** R **Cognitive Level** Memory
Record Number: 36 **RO Number:** 27 **SRO Number:**

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

010 Pressurizer Pressure Control System (PZR PCS)

A1. Ability to predict and/or monitor changes in parameters associated with operating the PZR PCS controls including:

A1.07 RCS pressure 3.7 3.7

Explanation: c. - Correct, Spray valves begin to modulate at 2260 psig. a. - Backup htrs should be ON. b. - Proportional heaters are on when < 2250 psig and B/U heaters off >2220 psig. d. - Spray valves are full open at 2310 psig.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
PRESSURIZER PRESSURE AND LEVEL `ontrol	0300-000.00S-PZRP&L-01	IV.B.1.I-k.	22-24		4

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: S/G Feed pump trip and PZR pressure

Given the following for Unit 2:

- Reactor power is 85%
- A S/G Feed Pump trips

Which one of the following describes the expected initial response of the Pressurizer Pressure Control System during this event?

a. Pressurizer heaters de-energize at the –5% level deviation setpoint

b. The proportional heaters and the backup heaters turn full on to raise pressure to normal.

c. Pressurizer spray valves will modulate open to reduce pressure to normal.

d. The PORVs open and maintain pressure below the high reactor trip setpoint.

Answer c **Exam Level** B **Cognitive Level** Comprehension

Record Number: 37 **RO Number:** 28 **SRO Number:** 30

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

010 Pressurizer Pressure Control System (PZR PCS)

A3. Ability to monitor automatic operations of the PZR PCS including:

A3.02 PZR pressure 3.6 3.5

Explanation: c. - During a runback, Pzr level is expected to rise due to heatup of the RCS. This compresses the Pzr bubble, raising Pzr pressure. a. – Reciprocal of BUH response on a +5% deviation b. - Proportional heaters will remain off as pressure remains high. The backup heaters may energize in this instance if the Pzr level insurge exceeds the level setpoint by 5%. d. – Transient is well within the design capabilities of the pressure control system.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
AIN FEEDWATER/CONDENSATE SYSTEM ABNORMALITY	0300-000.00S-ABCN01-00				4B
PRESSURIZER PRESSURE AND LEVEL CONTROL	0300-000.00S-PZRP&L-01	III.B	15		2

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: PZR pressure control channel fails high

The following conditions exist:

- The reactor is at 100% power
- The controlling Pressurizer Level Channel fails HIGH

Which one of the following will be the result from this failure if no operator action is taken?

a. Actual Pressurizer level will start rising due to MAXIMUM charging flow and the reactor will trip on HIGH Pressurizer level.

b. Actual Pressurizer level will lower due to reduced charging flow and the reactor will trip on LOW Pressurizer pressure.

c. Actual Pressurizer level will initially lower, then rise until the reactor trips on HIGH Pressurizer level.

d. Actual Pressurizer level will initially rise until PORVs open, then lower due to loss of RCS inventory until the reactor trips on LOW Pressurizer pressure.

Answer c **Exam Level** B **Cognitive Level** Application
Record Number: 38 **RO Number:** 29 **SRO Number:** 31

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

011 Pressurizer Level Control System (PZR LCS)

A1. Ability to predict and/or monitor changes in parameters associated with operating the PZR LCS controls including:

A1.02 Charging and letdown flows 3.3 3.5

Explanation: c. - Correct, Charging flow will drop to minimum, level initially decreases, until letdown isolates. With charging flow at minimum, level will then slowly rise to the high level reactor trip. a. - Level will not increase until letdown isolation occurs. b. - Reactor will not trip on low pressure, heaters will remain available except for the short time when Pzr level is at its lowest (heater cutout). d. - Level will not initially increase nor are PORVs expected to open.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
PRESSURIZER PRESSURE AND LEVEL CONTROL	0300-000.00S-PZRP&L-01	IX.B.2.b	41		6

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: PZR heater power supplies and interlocks

Given the following plant conditions:

- Unit 1 is at 100% power
- All major controls are in AUTO
- PZR HTR 11 BU Group is powered from the 1C 460V Vital Bus during surveillance testing

Which one of the following explains the effect, if any, this power source alignment will have on PZR Heater Control?

a. 11 BU Heater Group will NOT automatically cycle and it will not de-energize on low pressurizer level.

b. 11 BU Heater Group will automatically cycle but it will not de-energize on low pressurizer level.

c. Pressurizer heater control is not affected by power source alignment. All automatic controls and interlocks continue in effect.

d. 11 BU Heater Group can only be controlled via the LOCAL control on El. 78, Switchgear Room. There are no automatic controls or interlocks in effect.

Answer a **Exam Level** S **Cognitive Level** Application
Record Number: 39 **RO Number:** **SRO Number:** 32

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

011 Pressurizer Level Control System (PZR LCS)

K6. Knowledge of the effect of a loss or malfunction of the following will have on the PZR LCS:

K6.03 Relationship between PZR level and PZR heater control circuit 2.9 3.3

Explanation: a. – Correct. When powered from the 1C 460V Vital Bus, Pzr Htr 11 BU Group will NOT automatically cycle on Pressurizer pressure and will NOT deenergize on low Pressurizer level. b. - Incorrectly states automatic heater cycling is maintained. c. Incorrectly states automatic controls and interlocks remain in effect. d. – Emergency power supply breaker can still be cycled from the control room. LOCAL is only necessary during Control Rm Evac.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
PRESSURIZER PRESSURE AND LEVEL CONTROL	0300-000.00S-PZRP&L-01	IV.B.1.j	22		4, 9
PRESSURIZER BACKUP HEATERS POWER SUPPLY TRANSFER	S1.OP-SO.PZR-0010(Q)	CAUTION 2, 5.1	4	4	

Material Required for Examination

Question Source: Previous 2 NRC Exams **Question Modification Method:**

Question Source Comments: cstar - #66

Question: PZR level master controller fail

The unit is at 100% power with 21 Charging Pump running and 2CV55, Charging Flow Control Valve, in AUTO when the controlling pressurizer level channel fails. The RO has placed the Charging Flow Master Controller in MANUAL in accordance with alarm response procedures.

Which one of the following correctly describes what will happen if the RO misunderstands an order and lowers Flow Demand to ZERO?

a. All charging flow will be through the RCP seals

b. Charging header flow lowers to zero but the miniflow valves open to maintain cooling flow through the pump

c. 2CV55 will close to the minimum stop position

d. 2CV55 will fully close then shift to MANUAL and go to the minimum stop position

Answer c **Exam Level** R **Cognitive Level** Comprehension
Record Number: 40 **RO Number:** 30 **SRO Number:**

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

011 Pressurizer Level Control System (PZR LCS)

K6. Knowledge of the effect of a loss or malfunction of the following will have on the PZR LCS:

K6.04 Operation of PZR level controllers 3.1 3.1

Explanation: c. – Correct. Master controller will drive 2CV55 closed but it will stop at the electrical stop setpoint; a. – The majority but not all flow will go to the RCP seals; b. – There is a minimum open position on 2CV55, unless removed by I&C; d. – The valve will not fully close and will only shift to MANUAL on operator action.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
PRESSURIZER PRESSURE AND LEVEL CONTROL	0300-000.00S-PZRP&L-01	IX.B.2.a.	42		8
INSTRUMENT FAILURE REVIEW	0300-000.00S-ICFAIL-00	VI.C.1	27		4
	Applicable Logic Diagrams				

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: 48VDC failure effect on SSPS

Unit 1 is operating at 100% power when a 48 VDC power supply in the "A" Train SSPS Logic cabinet fails.

'hich one of the choices correctly completes the following statement?

OHA Alarm A-34 “SSPS TRN A TRBL” actuates . . . .

a. and Train A is NOT capable of automatically tripping the reactor.

b. but Train A remains capable of automatically tripping the reactor.

c. and the shunt trip coils for Reactor Trip Breaker A and Reactor Trip Bypass Breaker A are disabled.

d. and the UV coils for Reactor Trip Breaker A and Reactor Trip Bypass Breaker A are disabled.

Answer b **Exam Level** B **Cognitive Level** Comprehension

Record Number: 41 **RO Number:** 31 **SRO Number:** 33

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

012 Reactor Protection System

A2. Ability to (a) predict the impacts of the following on the Reactor Protection System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.04 Erratic power supply operation 3.1 3.2

Explanation: b. - Correct, A GENERAL WARNING (GW) is generated for "A" train SSPS when a 48 VDC (or 15 VDC) power supply fails. This alone does NOT result in any response other than the GW alarm. Only the one train is affected by this condition. a. - A loss of any single power supply in any single Train will result in GW but does not impair the capability of the Train. c&d. - Since the 48 VDC power supplies are auctioneered, no change in function for the reactor trip breaker or its associated bypass breaker will occur.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
.EACTOR PROTECTION SYSTEM	0300-000.00S-RXPROT-00	IV.C.7.6)a)	32		11
OVERHEAD ANNUNCIATORS WINDOW A	S1.OP-AR.ZZ-0001(Q)	A-34, 1.1.A	64	26	

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: SI pump power supplies

The circuit breaker (2CVIB9) providing power to the 2C Safeguards Equipment Control cabinet has been removed for replacement.

'hich one of the following describes the response of the 21 and 22 SI pumps when a valid SI signal is initiated with a concurrent loss of off-site AC power (Blackout)?

a. Both 21 and 22 SI pumps will start.

b. Only 22 SI pump will start.

c. Neither 21 nor 22 SI pumps will start.

d. Only 21 SI pump will start.

Answer d **Exam Level** S **Cognitive Level** Memory
Record Number: 42 **RO Number:** **SRO Number:** 34

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System (ESFAS)

A2. Ability to (a) predict the impacts of the following on the ESFAS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.04 Loss of instrument bus 3.6 4.2

Explanation: 21 SIP is powered from 2A and 22 SIP from 2C. d. – Correct. 2C SEC cannot function with that breaker removed. a&b. - With 2C bkr inop, 22 pump will not receive a start signal. c. 21 pump will start.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
EMERGENCY CORE COOLING SYSTEM	0300-000.00S-ECCS00-00	IV.C.4, IV.D.2	28-29		5, 7

aterial Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: CS actuation

Given the following conditions:

- A LOCA has occurred inside containment
- All busses are supplied from off-site power and all SEC's are reset
- Containment pressure has just exceeded the Hi-Hi Containment pressure setpoint

Which one of the choices correctly completes the following statement?

The Containment Spray (CS) pumps . . .

a. will start automatically and the CS valves will align automatically.

b. must be started manually and the CS valves must be manually aligned.

c. will start automatically but the CS valves must be manually aligned.

d. must be started manually but the CS valves will align automatically.

Answer d **Exam Level** R **Cognitive Level** Memory

Record Number: 43 **RO Number:** 32 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System (ESFAS)

A3. Ability to monitor automatic operations of the ESFAS including:

A3.02 Operation of actuated equipment 4.1 4.2

Explanation: d. - Correct, Pumps will not start because the SEC is reset but the valves will re-align from SSPS. a. - With SEC reset the pumps will not start automatically. b. Pump can be manually started, however the valves auto align at > 15 psig regardless of SEC status c. – Reciprocal of correct answer.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
SAFEGUARDS EQUIPMENT CONTROL SYSTEM	0300-000.00S-SEC000-01	IV.C.5.g.2)b)	19		4,9

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: CS actuation interlock

Which one of the following "isolations" occurs in coincidence with a MANUAL Containment Spray actuation?

a. Steamline

b. Feedwater

c. Containment Phase A

d. Containment Ventilation

Answer d **Exam Level** R **Cognitive Level** Memory
Record Number: 44 **RO Number:** 33 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System (ESFAS)

A4. Ability to manually operate and/or monitor in the control room:

A4.03 ESFAS initiation 4.5 4.7

Explanation: d. - Correct, CNMT Ventilation Isolation will result if CS/Phase B is manually actuated. a. - Steamline Isolation occurs from the CNMT pressure channel inputs which will also initiate an automatic CS signal. b. - Feedwater Isolation is associated with SI signal. c. - CNMT Phase A Isolation occurs either manually or coincident with SI.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RPS Logic Diagram	221057 B 9545	Sh. 8		17	
CONTAINMENT AND CONTAINMENT SUPPORT SYSTEMS	0300-000.00S-CONTMT-01	II.C.11.d.1).c)	33		9b

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: ESF status

Given the following conditions for Unit 2:

- Mode 3 with RCS cooldown and depressurization underway in accordance with procedures
- RCS pressure is 1900 psig
- RCS temperature is 525°F
- When a Main Steam Safety Valve on the 23 S/G fails open resulting in the following S/G pressures : 820 psig (21); 780 psig (22); 700 psig (23); 810 psig (24)

Which one of the following correctly describes the status of the ESF actuation system for the stated conditions?

a. No ESF signal has been generated.

b. Only a Safety Injection signal has been generated.

c. Only a Main Steam Line Isolation signal has been generated.

d. A Safety Injection signal and a Main Steam Line Isolation signal have been generated.

Answer b **Exam Level** R **Cognitive Level** Comprehension

Record Number: 45 **RO Number:** 34 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

013 Engineered Safety Features Actuation System (ESFAS)

K4.03 Main Steam Isolation System 3.9 4.4

Explanation: b. - Correct, A SI signal is generated for Steam Line High Differential Pressure when the setpoint of 100 psid is reached. a. - A SI has actuated to Steamline ΔP. c. - The Steam Line Isolation is NOT generated because High Steam Flow coincident with low steam header pressure is well above the required setpoint of 600 psig. d. – There is no Steamline Isolation on a Steamline ΔP SI.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
INTRODUCTION TO ENGINEERED SAFETY FEATURES AND DESIGN CRITERIA	0300-000.00S-ESF000-01	VII.B.1.b & c	50		21
RPS - Steam Generator Trip Signals Logic	221056 B 9545	sh. 7	7		
RPS - Safeguards Actuation Signals	221057 B 9545	sh. 8	17		

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments: pamexam

Question: P-A Converter failure

Unit 2 is operating at 100% power, with all systems in automatic when annunciator alarms "ROD INSERT LMT LO" (E8) and "ROD INSERT LMT LO-LO" (E16) activate. The plant is stable (no rod motion, no power changes, etc.)

vVhich one of the following correctly explains the cause of these alarms?

a. 22 RCS loop Tavg signal has failed low

b. A RCS Thot RTD has failed high

c. The P-A Converter has failed

d. Power has been lost to the IRPI's

Answer c **Exam Level** B **Cognitive Level** Comprehension
Record Number: 46 **RO Number:** 35 **SRO Number:** 35

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

014 Rod Position Indication System (RPIS)

A1. Ability to predict and/or monitor changes in parameters associated with operating the RPIS controls including:

A1.02 Control rod position indication on control room panels 3.2 3.6

Explanation: c. – Correct. P-A Converter failure would cause the alarms and a zero input to the recorder; a. - (Auct. High) Tavg input to the P-A converter is not used; b. – This failure would not change the RIL; d. – The IRPI's do not feed the OHA's.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
ROD CONTROL AND POSITION INDICATION SYSTEMS	0300-000.00S-RODS00-00	IV.B.13.f.4)d)	46		4.d,6.k

'aterial Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments:

Question: P-6 permissive

Given the following conditions on Unit 2:

- Reactor startup is in progress
- No manual blocks have been inserted
- Intermediate Channel N35 indicates 2E-10
- Intermediate Channel N36 indicates 9E-11
- Power is lost to Source Range Channel N31

Which one of the following describes the reactor response to the conditions above?

a. A reactor trip signal is generated resulting in a reactor trip

b. A reactor trip signal is generated but no trip occurs since one channel is above P-6

c. No reactor trip signal is generated since the channel has failed low

d. No reactor trip signal is generated until N36 indicates greater than 1E-10.

Answer a **Exam Level** R **Cognitive Level** Application
Record Number: 47 **RO Number:** 36 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

K3. Knowledge of the effect that a loss or malfunction of the Nuclear Instrumentation System will have on the following:

K3.01 RPS 3.9 4.3

Explanation: a. – Correct. No manual blocks have been inserted. The associated bistable trips, setting up the 1/2 SR High Flux. Trip b. - Must be manually blocked. c. – Indication fails low but the bistable trips. d. – This is the opposite of how it works.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
xcore NI	0300-00.00S-EXCORE-00	IV.C.3.j.2)	27		10

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: Calorimetric error

During performance of S2.RE-ST.ZZ-0001(Q) "Calorimetric Calculation", the feedwater temperature points utilized were reading 10°F lower than actual feedwater temperature. Power range NI's were adjusted in accordance with the directions of the calorimetric procedure.

Which one of the following correctly describes the effect of the NIS adjustment?

a. Indicated power is less than actual power; therefore, power range instruments are set conservatively.

b. Indicated power is less than actual power; therefore, power range instruments are set non-conservatively.

c. Indicated power is greater than actual power; therefore, power range instruments are set conservatively.

d. Indicated power is greater than actual power; therefore, power range instruments are set non-conservatively.

Answer c **Exam Level** R **Cognitive Level** Comprehension
Record Number: 48 **RO Number:** 37 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

015 Nuclear Instrumentation System

K5. Knowledge of the operational implications of the following concepts as they apply to the Nuclear Instrumentation System:

K5.04 Factors affecting accuracy and reliability of calorimetric calibrations 2.6 3.1

Explanation: For the calibration, the relationship between primary and secondary power are determined by the calculations: Qsec = m (hs-hf) – mhs(SGBD) and Qprim = Qcore – QRCPs. At equilibrium Qprim = Qsec. c. – Correct. With actual FW temperature lower than that used in calculation, then hf used is lower and a higher power level than actual is calculated. Therefore, the NI's were set at a higher than actual value. Since trip setpoints do NOT change, the reactor would trip with actual power less than the trip setpoint, which is conservative. a. - Indicated power is higher than actual power. b. - Indicated power is higher than actual power and the instruments are set conservatively. d. - Since the trip would occur with actual power less than that required by the analysis, the instruments are set conservatively.

Reference Title Page Revision	L. O.	Facility Reference Number		Section
EXCORE NUCLEAR INSTRUMENTATION SYSTEM	0300-000.00S-EXCORE-00	X.A.1	78	14

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Changed conditions such that CONSERVATIVE readings exists, which changed correct answer.

Question: Tc RTD failure

Unit 2 is at 100% power, steady state operation, and all control systems are in their normal at power configuration.

'hich one of the following statements correctly describes the plant response to an RCS cold leg ιemperature transmitter failure?

a. A high failure will cause rods to insert and pressurizer level to lower.

b. A high failure will cause the OPΔT setpoint to rise and pressurizer level to rise.

c. A low failure will cause the OTΔT setpoint to lower and rods will withdraw.

d. A low failure will cause the indicated ΔT to rise and pressurizer level to rise.

Answer a **Exam Level** S **Cognitive Level** Comprehension
Record Number: 49 **RO Number:** **SRO Number:** 36

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

016 Non-Nuclear Instrumentation System (NNIS)

A2. Ability to (a) predict the impacts of the following on the NNIS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.01 Detector failure 3.0 3.1

Explanation: a. – Correct. Causes Tavg to fail high. Tavg>Tref, rods insert which causes RCS temperature to decrease & Pzr level setpoint to lower; b. – PZR level will not rise; c. – Rods will insert; d. – PZR level will not rise.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Inst failure	0300-000.00S-ICFAIL-00	III.B&C	12-18-7		1

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: POPS

Given the following conditions on Unit 2:

- Reactor cooldown and depressurization is in progress
- Pzr PORV block valves PR-6 and PR-7 are closed in accordance with TSAS 3.4.5, due to seat leakage past the PORVs PR-1 and PR-2
- RCS pressure is being maintained at 800 psig due to problems isolating the SI accumulators
- RCS temperature drops to 305°F.

Which one of the following correctly describes the outcome if the operator arms the Pressurizer Overpressure Protection System (POPS) under these conditions?

PR-6 and PR-7 would . . .

a. OPEN; PR-1 and PR-2 would remain CLOSED.

b. OPEN; PR-1 and PR-2 would OPEN.

c. remain CLOSED; PR-1 and PR-2 would remain CLOSED.

d. remain CLOSED; PR-1 and PR-2 would OPEN.

Answer b **Exam Level** R **Cognitive Level** Comprehension
Record Number: 50 **RO Number:** 38 **SRO Number:**
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2
016 Non-Nuclear Instrumentation System (NNIS)
K1. Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following:
K1.01 RCS 3.4 3.4

Explanation: b. – Correct. To arm POPS the Key Control Switch is taken to ON (top switch). This sets the associated PORV (PR1, PR2) setpoint to 375 psig, opens the associated PORV block valves (PR6, PR7), and places associated PORV in AUTO. Since RCS pressure is above the POPS setpoint, PR-1 and PR-2 would open. a. - RCS pressure is above setpoint so PR1 and PR2 would open. c&d. - The act of arming sends an open signal to the PORV block valves.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Pressurizer Pressure and Level Control	0300-000.00S-PZRP&L-01	IV.B..1.c.3)	17		4, 9
Reactor Coolant System	0300-000.00S-RCS000-02	V.A.4.a.2)	28		9

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments: Cgroup

Comment Type Comment

Question: CET readings

A LOCA has occurred on Unit 1. The crew is in EOP-LOCA-1, Loss of Reactor Coolant. The RO notes that the reading on the Subcooling Margin Monitor (SCMM) is 16 °F lower than the last time he checked it but RCS temperature and pressure have not changed significantly.

Which one of the following correctly describes a reason for that change?

a. The changing containment temperature is affecting the output signals from the in-core thermocouple reference junction box

b. An in-core thermocouple has failed high

c. Rising containment pressure is lowering RCS pressure detector output, pound for pound

d. The SCMM automatically shifted to ADVERSE

Answer d **Exam Level** B **Cognitive Level** Memory
Record Number: 51 **RO Number:** 39 **SRO Number:** 37

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

017 In-Core Temperature Monitor (ITM) System

A1. Ability to predict and/or monitor changes in parameters associated with operating the ITM System controls including:

A1.01 Core exit temperature 3.7 3.9

Explanation: d. – Correct, the SCMM shifts automatically on containment pressure or radiation levels; a. – The system compensates for changes in the reference junction temperature; b. – One T/C failed high is discriminated out by the processor; c. – Containment pressure does not affect RCS pressure output and a pound for pound change would not result in a 16 degree change.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Incore Instrument System	0300-000.00S-INCORE-00, Objective 7				

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: SW to Cont. coolers

Given the following conditions on Unit 2:

- Reactor is at 100% power when a small break LOCA occurs
- Containment pressure is 4.5 psig

Which one of the following describes the status of the containment cooling system for those conditions?

Service Water flow to containment coolers . . .

a. rises to approximately 2640 gpm and all CFCU fans start in or are shifted to fast speed.

b. rises to approximately 2640 gpm and all CFCU fans start in or are shifted to slow speed.

c. lowers to approximately 960 gpm and all CFCU fan start in or are shifted to fast speed.

d. lowers to approximately 960 gpm and all CFCU fans start in or are shifted to slow speed.

Answer b **Exam Level** B **Cognitive Level** Memory

Record Number: 52 **RO Number:** 40 **SRO Number:** 38

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

022 Containment Cooling System (CCS)

K1. Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following:

K1.01 SWS/cooling system 3.5 3.7

Explanation: b. – Correct. During accident conditions, all fans are shifted to SLOW speed and SW flow is increased to 2640 gpm. a. Fans run in LOW speed. c. - SW flow increases to 2640 gpm & fans run in LOW speed. d. - SW flow to CFCUs increases.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CONTAINMENT AND CONTAINMENT `UPPORT SYSTEMS	0300-000.00S-CONTMT-01	III.H.2.a.3) a)	71		9

Material Required for Examination

Question Source: Facility Exam Bank

Question Modification Method:

Question Source Comments:

Question: SEC effect on CFCUs

Given the following conditions on Unit 2:

- A loss of offsite power has occurred
- 2A 4KV Bus de-energized on DIFF

Which one of the following correctly describes the status of the CFCUs?

a. 3 CFCU's will be operating in SLOW speed

b. 4 CFCU's will be operating in SLOW speed

c. CFCU's running before the event will restart in the speed selected

d. No CFCU's will be operating

Answer d **Exam Level** B **Cognitive Level** Application
Record Number: 53 **RO Number:** 41 **SRO Number:** 39

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

022 Containment Cooling System (CCS)

K2. Knowledge of electrical power supplies to the following:

K2.01 Containment cooling fans 3.0 3.1

Explanation: d.- Correct. In the event of "blackout" during normal operation, Breakers No. 1 and No. 2 are tripped by SEC and the CFCU must be manually restarted after the SEC is reset; a. – Choice if candidates believes two CFCU's are powered from 2A Bus; b. – Choice if candidate believes all available CFCU's start; c. – SEC trips both CFCU breakers.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CONTAINMENT AND CONTAINMENT SUPPORT SYSTEMS	0300-000.00S-CONTMT-01	III.H.1.f.3)	71		4, 5

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: Manual actuation of CS

A large break LOCA has occurred. The crew has transitioned to FRCE-1, Response to Excessive Containment Pressure, on a PURPLE Path. Prior to starting containment spray (CS) pumps, the procedure ˈ ˋes the question "Is EOP-LOCA-5 in effect?" If the answer is YES then the crew is directed to "Operate CS Pumps as directed by EOP-LOCA-5."

Which one of the following correctly describes the difference between operation of the CS Pumps in LOCA-5 as compared to FRCE-1?

a. LOCA-5 stops both CS Pumps to allow evaluation of CFCU capability to control containment pressure. CS Pumps are re-started one-at-a-time, if needed. FRCE-1 starts both CS Pumps and runs all CFCU's in LOW speed

b. LOCA-5 stops both CS Pumps if all five CFCU's are available to run in HIGH speed. FRCE-1 starts both CS Pumps and runs all CFCU's in LOW speed

c. LOCA-5 runs only one CS Pump as long as containment pressure is <47 psig. FRCE-1 starts both CS Pumps, regardless of containment pressure

d. LOCA-5 runs CS Pumps based on the combined status of RWST level, containment pressure and the number of operating CFCU's. FRCE-1 always starts both CS Pumps

Answer d **Exam Level** B **Cognitive Level** Memory
Record Number: 54 **RO Number:** 42 **SRO Number:** 40
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1
026 Containment Spray System (CSS)
A2. Ability to (a) predict the impacts of the following on the CSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
.2.03 Failure of ESF 4.1 4.4

Explanation: d – Correct. CS Pumps are run according to LOCA-5, Table C; a – CS Pumps are not stopped unless permitted by Table C or LO-LO RWST level; b – CFCU's are not run in HIGH speed in either procedure; c – If insufficient CFCU's are available then 2 CS Pumps are running, even though containment pressure may be <47 psig.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
EOP-LOCA-5, Table C					
EOP-FRCE-1					

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: CS pump start failure impact

Given the following conditions for Unit 2:

- A large-break LOCA has occurred
- The Injection Phase of SI is in progress
- 22 Containment Spray Pump is unavailable

Which one of the following correctly describes the response to the above conditions?

a. Safeguards Pumps will operate for a longer time with suction from the RWST before swapover to the containment sump.

b. The higher pressure in containment will result in overpressurizing the RHR suction piping when swapover to the containment sump occurs.

c. A portion of the 22 RHR pump discharge flow must be diverted to provide flow through the affected spray header.

d. Water level in the containment sump will NOT be sufficient to supply all ECCS pumps when the alignment for cold leg recirculation is complete.

Answer a **Exam Level** R **Cognitive Level** Comprehension
Record Number: 55 **RO Number:** 43 **SRO Number:**

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 1

026 Containment Spray System (CSS)

K1. Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following:

K1.01 ECCS 4.2 4.2

Explanation: a. Correct. During the injection phase, the CS pumps take suction from the RWST and deliver borated water (mixed with sodium hydroxide) to the Containment atmosphere. Without this pump taking suction from the RWST, the inventory of the RWST will last longer. b. - With one CS pump operating in conjunction with CFCUs, the CNMT pressure will remain below 47 psig, which is well below the design pressure for RHR suction piping. c. – RHR is used to provide spray in the recirculation mode. d. – The inventory is the same. It just takes longer to get to the recirculation initiation setpoint..

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CONTAINMENT SPRAY SYSTEM	0300-000.00S-CSPRAY-01	IV.B.2.a.2)	18		3.b.iii
EMERGENCY CORE COOLING SYSTEM	0300-000.00S-ECCS00-00	IV.A.4.a	20		4.a

Material Required for Examination

Question Source: Facility Bank **Question Modification Method:**

Question Source Comments:

Question: IRU on SI

The 21 Containment Iodine Removal Unit (IRU) was started in preparation for a planned containment entry.

/hich one of the following correctly describes the status of the IRUs if a Safety Injection signal is actuated?

21 IRU . . .

a. continues to run and 22 IRU is locked out.

b. is tripped and locked out, and 22 IRU is locked out.

c. continues to run and 22 IRU starts on SEC Mode Operation.

d. trips then restarts on SEC Mode Operation, and 22 IRU starts on SEC Mode Operation.

Answer b **Exam Level** B **Cognitive Level** Memory
Record Number: 56 **RO Number:** 44 **SRO Number:** 41

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 2

027 Containment Iodine Removal System (CIRS)

A4. Ability to manually operate and/or monitor in the control room:

A4.01 CIRS controls 3.3 3.3

Explanation: b. – Correct. Fans are non-safety related and power is locked out by SEC Mode Ops. a,b&c - SEC locks out the non-ESF breakers.

Reference Title	Facility Reference Number	Section	Page	Revision/L. O.
CONTAINMENT AND CONTAINMENT SUPPORT SYSTEMS	0300-000.00S-CONTMT-0,	IV.D.2.a.1).a)	78-79	8

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments: pamexam

Question: H2 Recombiner setting

Given the following conditions for Unit 2:

- A large break LOCA has occurred
- Prior to the LOCA, containment temperature was 90°F
- Following the LOCA, containment pressure is 5 psig
- Containment temperature is currently 120°F
- The EOPs require that a Hydrogen Recombiner be placed in service

Which one of the following values will be set on the Hydrogen Recombiner potentiometer for the above conditions?

a. 50.2 Kw

b. 52.8 Kw

c. 54.5 Kw

d. 55.9 Kw

Answer c **Exam Level** B **Cognitive Level** Application
Record Number: 57 **RO Number:** 45 **SRO Number:** 42

Tier: Plant Systems **RO Group:** 3 **SRO Group:** 2

028 Hydrogen Recombiner and Purge Control System (HRPS)

A2. Ability to (a) predict the impacts of the following on the HRPS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.01 Hydrogen recombiner power setting, determined by using plant data book 3.4 3.6

Explanation: c. – Correct. At 5 psig & 90°F, the Cp (correction factor) is approximately 1.24. Using Att 1, the calculated value is (1.24 x 44.00) = 54.5 Kw. a. - Approximates 3 psig on the 120°F curve. b. - Approximates 5 psig on the 120°F curve. d. - Approximates 5 psig on the 60°F curve.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Hydrogen Recombiner Operation	S2.OP-SO.CAN-0001	Att1&2	1	3	
CONTAINMENT AND CONTAINMENT SUPPORT SYSTEMS	0300-000.00S-CONTMT-01	IX.I.1.b	122		4, 12

Material Required for Examination S2.OP-SO.CAN-0001,Hydrogen Recombiner Operation Attachments 1 & 2

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments:

Question: Termination of containment purge

Unit 1 is in Mode 5 with a containment purge in progress. All RMS channels are operable.

Which one of the following describes a condition requiring immediate termination of the containment ırge?

a. An Auxiliary BLDG AIR D/P LOW console alarm.

b. Failure of the Plant Vent Flow Monitor with all Auxiliary Building Exhaust Fans operating.

c. Failure of RMS Channel 1R16, Plant Vent Effluent

d. Failure of RMS Channel 1R12A, Containment Noble Gas

Answer a **Exam Level** R **Cognitive Level** Memory

Record Number: 58 **RO Number:** 46 **SRO Number:**

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

029 Containment Purge System (CPS)

A3. Ability to monitor automatic operations of the Containment Purge System including:

A3.01 CPS isolation 3.8 4.0

Explanation: a. - Correct, A caution on every page of the Containment Purge procedure requires termination of the lineup if Auxiliary Bldg. D/P alarms. b.,c., &d. - None of the distracters cause automatic isolation or require immediate actions but are if/then statements in the procedure.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RADIOACTIVE WASTE GAS SYSTEM	0300-000.00S-WASGAS-01	IX.A.5	54		12
CONTAINMENT PURGE TO PLANT VENT	S1.OP-SO.WG-0006(Q)	CAUTION 5.1.14.A	9	14	

.iaterial Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments: CGroup

Question: Fuel Handling crane movement

During a Unit 2 refueling, Fuel Handling Crane area radiation monitor (2R32A) reaches the HIGH alarm setpoint when the tool and attached spent fuel assembly are being raised. The crane hoist has not yet been fully raised.

Which one of the following describes the status of the fuel assembly attached to the crane?

a. Crane controls can only lower the fuel assembly.

b. Crane controls are disabled until jumpers are installed to defeat the interlock.

c. The fuel assembly can be lowered only after pressing the BYP INT pushbutton on the crane controls.

d. Movement of the fuel assembly is terminated IAW procedures until an HP Technician completes a general area survey.

Answer a **Exam Level** B **Cognitive Level** Memory
Record Number: 59 **RO Number:** 47 **SRO Number:** 43
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 2
034 Fuel Handling Equipment System (FHES)
K4. Knowledge of Fuel Handling Equipment System design feature(s) and/or interlock(s) which provide for the following:

K4.02 Fuel movement 2.5 3.3

Explanation: a. - Correct, Fuel Handling Cranes (1R32A; 2R32A) area radiation monitors lock out all crane motion except downward movement of suspended load. b. - jumpers are not required for downward movement. c. - BYP INT not part of selectable operation for FH crane, but is on Manipulator crane. d. Procedural directions are to place the fuel assembly in safe location.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RADIATION MONITORING SYSTEM	0300-000.00S-RMS000-01	IV.B.3.q	36		6
Refueling System	0300-000.00S-REFUEL-00	V.B.2.c	46		6
RADIATION MONITORING SYSTEMS OPERATION	S2.OP-SO.RM-0001(Q)	Att. 1	4	12	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments: cgroup

Question: Affect of steam dump steam press. setpoint change

While in Hot Standby awaiting reactor startup, an operator error causes the steam dump auto steam pressure setpoint to be reduced from 1005 psig to 940 psig.

'hich one if the following is the resulting RCS Tavg maintained by the steam dumps?

a. 536°F

b. 539°F

c. 543°F

d. 547°F

Answer c **Exam Level** R **Cognitive Level** Comprehension
Record Number: 60 **RO Number:** 48 **SRO Number:**

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

039 Main and Reheat Steam System (MRSS)

A2. Ability to (a) predict the impacts of the following on the MRSS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.04 Malfunctioning steam dump 3.4 3.7

Explanation: c. Correct - Temperature interlock auto closure of dumps at 543°F; a. - Possible error if interpolation of the Steam Tables is not done; b. - Temperature for saturation at 940 psig. d. - No-load Tave. for 1005 psig.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Steam Dump	0300-000.00S-STDUMP-02	V.A.6.c	22		9e

Material Required for Examination Steam Tables

'uestion Source: New **Question Modification Method:**

Question Source Comments:

Question: Steam Dump Failure

Given the following conditions for Unit 2:

- Reactor power is 100%
- 2A 115 VAC Vital Bus power is lost

Which one of the following correctly describes the reason the operator is directed to shift the Steam Dump controller from TAVG to MS PRESS CONT?

a. The steam dumps are armed. The steam dump valves will open due to the signal from the load rejection controller if Tavg exceeds Tref by 5°F.

b. A steam dump demand signal is generated from the plant trip controller. If an arming signal is generated, the steam dump valves will open to the demanded position.

c. A steam dump demand signal is generated from the load rejection controller. If an arming signal is generated, the steam dump valves will open to the demanded position.

d. The steam dumps CANNOT be armed from the turbine first stage pressure signal. If ONE reactor trip breaker fails to open on a trip, the steam dumps would be inoperable in TAVG Mode.

Answer c **Exam Level** B **Cognitive Level** Comprehension
Record Number: 61 **RO Number:** 49 **SRO Number:** 44
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3
041 Steam Dump System (SDS) and Turbine Bypass Control
K3. Knowledge of the effect that a loss or malfunction of the SDS will have on the following:
K3.02 RCS 3.8 3.9

Explanation: c. Correct. First Stage Impulse Pressure Channel PT-505 is lost, generating a demand signal on the Load Rejection circuit. a. - No arming signal present since PT-506 provides arming signal for steam dumps and remains operable. b. - With trip breakers closed, demand signal is from Load Rejection Controller. When trip breakers open, demand is from Plant Trip Controller. d. – PT-506 provides the arming signal.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
STEAM DUMP SYSTEM	0300-000.00S-STDUMP-02	V.A.5.a), V.A.7.c	19, 226		9, 10
LOSS OF 2A, 2B, 2C AND 2D 115V VITAL INSTRUMENT BUS	0300-000.00S-AB1151-01	III.C.5	9		3
Steam Dump Logic	221059 B 9545			10	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments: Pamexam

Question: Stm dmp interlock on low vacuum

Given the following conditions for Unit 2:

- Turbine Trip has occurred from 20% Reactor Power
- Maintenance activities have resulted in a break in the condenser rupture disk
- Condenser Air Removal Pumps are unable to handle the in-leakage volume and condenser vacuum is at 19 inches of Hg
- Tavg is at 547°F
- All Circulating Water System Pumps are in service.

Which one of the following correctly describes the status of the Condenser Steam Dump System?

a. The Load Rejection Controller will be modulating the steam dump valves

b. Low condenser vacuum is blocking steam dump valve operation

c. The Plant Trip Controller will be modulating the steam dump valves

d. Tavg is blocking steam dump valve operation

Answer b **Exam Level** B **Cognitive Level** Application
Record Number: 62 **RO Number:** 50 **SRO Number:** 45

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

055 Condenser Air Removal System (CARS)

K3. Knowledge of the effect that a loss or malfunction of the CARS will have on the following:

K3.05 SDS 2.3 2.6

Explanation: b. – Correct. Steam Dump operation is blocked when either the East or West condenser vacuum switches indicate vacuum is less than 20 inches of Hg. a. - The Load Rejection Controller would modulate if BLOCK was not present. c. – At this power, a turbine trip will not cause a reactor trip. d. - Tave block does not occur until Tave reaches 543°F.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
STEAM DUMP SYSTEM	0300-000.00S-STDUMP-02	V.A.7.e	23		10

Material Required for Examination

Question Source: Previous 2 NRC Exams **Question Modification Method:**

Question Source Comments: CStar - #75

Question: Condensate sys break

Given the following conditions for Unit 2:

- Reactor is at 100% power
- A break occurs just downstream of 22 Condensate Pump

Which one of the following correctly describes the subsequent plant response?

a. The affected Steam Generator Feed Pump (SGFP) trips if suction pressure lowers to either: 215 psig after a 10-second delay, or 190 psig instantaneous.

b. The Condensate Polisher Bypass valves CN-108s automatically open to restore SGFP suction pressure. If pressure lowers to 190 psig then any affected pump trips.

c. At 275 psig on the suction of either SGFP a 10-second timer starts. If automatic opening of the CN-108 valves does not restore suction pressure then any affected pump trips.

d. At 275 psig the "Condensate Suction Pressure LO" console alarm actuates. If suction pressure remains <275 psig for >10 seconds then any affected pump trips.

Answer a **Exam Level** B **Cognitive Level** Memory
Record Number: 63 **RO Number:** 51 **SRO Number:** 46

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

056 Condensate System

K1. Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following:

Explanation: a. – Correct. The affected Feedwater Pump will trip if its suction pressure lowers to either: 215 psig, after a 10 second delay, or 190 psig instantaneous. b. - The operators have to manually open this valve (CN108s). c. - Console alarm CONDENSATE SUCTION PRESSURE LO setpoint: is 275 psig at the suction of either Feedwater Pump. There is no interlock associated with the alarm. d. – 275 psig is the alarm setpoint. It does not start the timer.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CONDENSATE AND FEEDWATER SYSTEM	0300-000.00S-CN&FDW-01	V.C.5.I	64		8

Material Required for Examination

Question Source: Previous 2 NRC Exams **Question Modification Method:**

Question Source Comments: Cstar - #55

Question: SG High Level permissive

Given the following:

- Reactor power is 18% with a turbine startup in progress
- S/G 1 NR levels are 68%
- S/G 2 NR levels are 69%
- S/G 3 NR levels are 66%
- S/G 4 NR levels are 65%

Which one of the following correctly lists automatic actions that are all a DIRECT result of the stated conditions?

a. Reactor trip, Main Turbine trip, Feedwater Interlock

b. Main Turbine trip, Feedwater Interlock, SG Feed Pump trip

c. Reactor trip, Feedwater Isolation, SG Feed Pump trip

d. Main Turbine trip, Feedwater Isolation, SG Feed Pump trip

Answer d **Exam Level** S **Cognitive Level** Memory
Record Number: 64 **RO Number:** **SRO Number:** 47

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

059 Main Feedwater (MFW) System

A2. Ability to (a) predict the impacts of the following on the MFW System and (b) based on those predictions, use

A2.03 Overfeeding event 2.7 3.1

Explanation: d. - Correct, P-14 generates a turbine trip, FWI, and SGFP trip. a. – The reactor will not trip. b. - FW Interlock ocurs with Low -Tavg interlock and a reactor trip. c. – The reactor will not trip.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CONDENSATE AND FEEDWATER SYSTEM	0300-000.00S-CN&FDW-01	VIII.A.5	75		9

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:**

Question Source Comments:

Question: Coordinated SGFP speed/FRV control

Given the following conditions for Unit 2:

- Reactor is at 100% power
- All BF19 and BF40 valves, and both feed pumps are in AUTO
- The operator places the feed pump MASTER controller in MANUAL and lowers the demand setting.

Which one of the following describes the result of this action?

Feed pump speed will lower resulting in . . .

a. all BF19's closing down to maintain ΔP and the resultant continuous lowering of S/G levels

b. all BF19's opening further to maintain programmed S/G levels.

c. a reduction in feed flow and a possible reactor trip on steam flow-feed flow mismatch in coincidence with a low steam generator level

d. an ADFWCS alarm on LOW ΔP and shifting of all BF19 and BF40 controllers to MANUAL

Answer b **Exam Level** R **Cognitive Level** Application
Record Number: 65 **RO Number:** 52 **SRO Number:**
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

059 Main Feedwater (MFW) System
A3. Ability to monitor automatic operations of the MFW System including:
A3.07 ICS 3.4 3.5

Explanation: b. - Correct, A SGFP speed lowers, feed flow and S/G levels will drop. The BF19s will open to maintain level. a. – SGFP speed is automatically controlled on BF19 DP. c. – This trip has been eliminated. d. – ADFWCS shifts controllers to MANUAL on various controller failures

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
ADVANCED DIGITAL FEEDWATER CONTROL SYSTEM	0300-000.00S-ADFWCS-00	II.B	15		8

Material Required for Examination
Question Source: Facility Exam Bank **Question Modification Method:** Changed a, c, d
Question Source Comments:

Question: FRV interlock reset

Given the following conditions for Unit 2:

- The Unit tripped from 100% 4 hours ago
- Unit is in MODE 3 at normal operating pressure and temperature
- Level in 22 S/G rose to 72% due to misoperation of 22AF21, AFW Discharge Flow Control Valve
- Level has been restored to 35%

Which one of the following states the minimum actions necessary to perform stroke testing of the BF40s, FWRV Bypass valves?

a. Cycle the reactor trip breakers

b. Reset the Feedwater Interlock Signal

c. Cycle the reactor trip breakers and then reset the Feedwater Interlock Signal

d. Reset the Feedwater Isolation signal

Answer c **Exam Level** R **Cognitive Level** Comprehension
Record Number: 66 **RO Number:** 53 **SRO Number:**
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1
059 Main Feedwater (MFW) System
K4. Knowledge of MFW System design feature(s) and/or interlock(s) which provide for the following:
K4.19 Automatic feedwater isolation of MFW 3.2 3.4

Explanation: With RCS temperature less than 554°F and P-4 present (reactor trip breakers open), the FWRV Bypass valves and the FWRVs are closed on Feedwater Interlock. Additionally when S/G level went above 67%, a FWI signal was generated. The generation of this signal cleared when S/G level was returned to normal. However, with the trip breakers open, either an SI signal or P-14 signal will seal in the closure signal to the FWRV Bypass and FWRVs. To clear this both the SI signal and the P-14 (if they exist) must be reset/blocked and/or clear AND the trip breakers must be cycled to clear the seal-in. c. - Correct, No SI is present. The P-14 has cleared but the FWI is sealed in by P-4. Also, the Feedwater Interlock is active. To move the FWRV Bypass valves, the reactor trip breakers must be cycled and then the Feedwater Interlock must be reset. a. - This action will not clear the FWI. It is all that is required following a "normal" reactor trip. b. – The trip breakers must be cycled. d. The Feedwater Isolation signal cleared when SG levels returned to normal.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
ADVANCED DIGITAL FEEDWATER CONTROL SYSTEM	0300-000.00S-ADFWCS-00	V.I.	37-38		10, 13a
REACTOR PROTECTION SYSTEM	0300-000.00S-RXPROT-00 0	VII.B.7	50		13
Feedwater Control & Isolation Logic	221062 B 9545		1	6	

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: AFW pump x-connect and runout protection

Given the following conditions for Unit 2:

- Unit is in Mode 3 with Tavg at 547°F
- The 23 AFW pump is NOT available
- The 21 AFW pump has been just stopped due to unusual motor noises
- The 22 AFW pump is running with normal parameters
- An operator has been dispatched to open 21AF923 and 22AF923 to allow cross-tie of the AFW headers from the motor-driven AFW pumps.
- The CRS wants to maintain all SG levels within normal operating band

Which one of the choices correctly completes the following statement?

Once the AF923 valves are open, the PO...

a. only needs to throttle the AF21 valve to each SG.

b. must depress the PRESS OVERRIDE DEFEAT for both AFW Pumps, then throttle the AF21 valve to each SG

c. must depress the PRESS OVERRIDE DEFEAT for 22 AFW Pump, and then throttle the AF21 valve to each SG.

d. must depress the PRESS OVERRIDE DEFEAT for 21 AFW Pump, and then throttle the AF21 valve to each SG

Answer d **Exam Level** R **Cognitive Level** Comprehension

Record Number: 67 **RO Number:** 54 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

61 Auxiliary / Emergency Feedwater (AFW) System

A1. Ability to predict and/or monitor changes in parameters associated with operating the AFW System controls including:

A1.01 S/G level 3.9 4.2

Explanation: d. - Correct, Runout protection for the MD AFW pumps is provided by respective AF21 valve. With 21 AFW pump disch header pressure <1150 psig, the associated AF21 valves remain closed. Feeding the S/Gs when pump discharge is <1350 psig, the operator will depress PRESS OVERRIDE DEFEAT pushbutton to remove runout protection for the idle pump and allows operator control of AF21 valves. Since the headers are tied and 21 AFW Pump is not running, the operator must override 23 and 24AF21. a. - Must actuate PRESS OVRD DEFEAT to open 23& 24AF21. b. - PRESS OVRD DEFEAT only needed for valves associated with idle pump (21, 23 & 24AF21). c. - PRESS OVRD DEFEAT not required for valves associated with running pump (22, 21 & 22AF21).

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
AUXILIARY FEEDWATER SYSTEM	0300-000.00S-AFW000-02	IV.B.3.g.2).d); IV.B.3.g.3).a)	27		4.h; 8.c

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments: Pamexam

Question: TDAFWP LOCAL control

Unit 1 is at 100% power. 13 AFW Pump is presently selected to LOCAL while a technician tests the control circuit.

Which one of the choices correctly completes the sentence regarding the status of 13 AFW Pump if a reactor trip and loss of off-site power occurs?

13 AFW Pump will . . . . .

a. start automatically on group bus undervoltage

b. start but only if 2/3 levels on 2/4 SG's drop below the setpoint

c. will NOT be started automatically and cannot be started from the control room until the local selector is returned to the REMOTE position

d. will NOT be started automatically but can be started manually from the control room by depressing the pump LOCAL-MANUAL pushbutton then depressing the START pushbutton

Answer c **Exam Level** S **Cognitive Level** Comprehension
Record Number: 68 **RO Number:** **SRO Number:** 48
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1
061 Auxiliary / Emergency Feedwater (AFW) System
A2. Ability to (a) predict the impacts of the following on the AFW System and (b) based on those predictions, use

A2.04 pump failure or improper operation 3.4 3.8
Explanation: c. – Correct. With the SDAFW Pp selected to LOCAL all automatic and remote starts are blocked; a.&b – All automatic starts are blocked; d. – the LOCAL-MANUAL PB is for alarm/indication only.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
AUXILIARY FEEDWATER SYSTEM	0300-000.00S-AFW000-02	IV.B.6.b.	31		8

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: MDAFW Pp local controls

The unit is at 100% power. 22 AFW Pump is stopped but selected to LOCAL while a technician performs a circuit test.

Which one of the following correctly describes the status of 22 AFW Pump if 2B 4KV Vital Bus de-energizes when the transfer relay fails during a swap to the alternate SPT?

a. 22 AFW Pump will not start when 2B SEC loads the EDG

b. 22 AFW Pump will not start since 2B SEC will not actuate

c. 22 AFW Pump will start when 2B SEC loads the EDG and the associated AF21 valves will stroke open

d. 22 AFW Pump will start when 2B SEC loads the EDG but the associated AF21 valves will remain closed on pressure override due to lower SG pressure at HFP

Answer c **Exam Level** R **Cognitive Level** Comprehension
Record Number: 69 **RO Number:** 55 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1
061 Auxiliary / Emergency Feedwater (AFW) System
A3. Ability to monitor automatic operations of the AFW System including:
A3.01 AFW startup and flows 4.2 4.2

Explanation: c. – Correct. 2B 4KV Bus will load onto the EDG in Mode II*. MDAFW Pumps will start on SEC actuation, even when selected to LOCAL; a. – SEC starts are still functional; b. – SEC actuates in Mode II*; d. – Pressure is sensed upstream of the associated AF21's.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
AUXILIARY FEEDWATER SYSTEM	0300-000.00S-AFW000-02	IV.B.4.c.6) a)(1)	31		6,9

Material Required for Examination
Question Source: NRC Exam Bank **Question Modification Method:**
Question Source Comments:

Question: Electrical failure impact on CW/Stm Dump

Unit 2 is at 40% power. Operators had been raising power but a SW leak has developed on the MTLO Cooler. The following conditions exist:

- #23B Circulating Water (CW) Pump is OOS while electricians test the 4KV breaker, outside of the breaker cubicle
- The generator is synchronized to the grid
- Operators are preparing to trip the turbine due to the SW leak on the MTLO Cooler

Which one of the following correctly identifies the number of steam dump valves capable of operating if the DIFF Relay on 2CW Bus Section 23 actuates following the turbine trip?

a. 0

b. 4

c. 8

d. 12

Answer c **Exam Level** S **Cognitive Level** Comprehension
Record Number: 70 **RO Number:** **SRO Number:** 49

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

062 A.C. Electrical Distribution System

A2. Ability to (a) predict the impacts of the following on the A.C. Electrical Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.01 Types of loads that, if de-energized, would degrade or hinder plant operation 3.4 3.9

Explanation: The bus failure leaves 21B and 22B Circulators in service. c. – Correct, four valves continue to operate on each condenser with an operating CW Pump; a., b., d. – Numbers of valves that could be selected if the candidate believes the bus swaps on DIFF actuation, does not know the pump power supplies, or thinks only the two valves on each condenser box with an operable pump will open.

Reference Title	Facility Reference Number/Section/Page/Revision/L.O.
Lesson Plan: Steam Dump	0300-000.00S-STMDMP, Obj. 10
Lesson Plan: 4160 ELECTRICAL SYSTEM	0300-000.00S-4KVAC0-01, Obj. 4.b, 6.d
Logic Diagram 221059	

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: AC to DC transfer

Which one of the following describes the mechanism used by the safety-related portion of the 115 VAC system to transfer power from its normal AC source to the 125 VDC source?

a. An auctioneering circuit transfers power from the rectifier output to the 125 VDC supply.

b. A static switch transfers power from AC regulator to 125 VDC supply.

c. When low voltage is sensed from the AC regulator, the 125 VDC input breaker is automatically shut.

d. A static switch transfers power from the rectifier output to 125 VDC if low 115 VAC bus voltage is sensed.

Answer a **Exam Level** B **Cognitive Level** Memory
Record Number: 71 **RO Number:** 56 **SRO Number:** 50

Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2

062 A.C. Electrical Distribution System

K4. Knowledge of A.C. Electrical Distribution System design feature(s) and/or interlock(s) which provide for the following:

K4.10 Uninterruptable ac power sources 3.1 3.5

Explanation: a. - Correct, An auctioneered circuit is used with 115 VAC power rectified and 125 VDC bus power as inputs. The AC rectified voltage is normally higher than the incoming DC line voltage at the auctioneering circuit.
b&d. - A static switch is not used. c. - Breakers are not utilized to transfer power sources. Both the AC input and DC input breakers are normally closed.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
115VAC ELECTRICAL SYSTEMS	0300-000.00S-115VAC-01	V.A.2.a	18		4

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments:

Question: EDG voltage control

The 2A Diesel Generator (EDG) is running and paralleled to the grid during a surveillance test.

Which one of the following correctly states the result of operating the EDG Voltage Control Switch in the `anner described?

Positioning the VOLTAGE CONTROL switch...

a. to LOWER raises generator amperage but has no effect on either real or reactive load.

b. to LOWER has no effect because voltage control is automatic when the EDG is synchronized.

c. to RAISE causes the generator to pick up a larger share of the real load.

d. to RAISE causes the generator to pick up a larger share of the reactive load.

Answer d **Exam Level** R **Cognitive Level** Memory
Record Number: 72 **RO Number:** 57 **SRO Number:**
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2
064 Emergency Diesel Generator (ED/G) System
A4. Ability to manually operate and/or monitor in the control room:
A4.02 Adjustment of exciter voltage (using voltage control switch) 3.3 3.4

Explanation: d. – Correct. Operating the VOLTAGE CONTROL will affect reactive load. KVARs will rise when taken to RAISE. a. - Reactive load will change. b. – Voltage control is not automatic with the EDG synchronized. c. - Operation of VOLTAGE CONTROL does not affect generator real load.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
EMERGENCY DIESEL GENERATORS	0300-000.00S-EDGOOO-01	IV.B.11.c.6) & 9).b)	73-75		4,7

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: Isol of liquid rad waste

Which one of the following correctly describes TWO conditions that will independently cause automatic closure of 2WL51, Liquid Waste Discharge Valve?

a. High discharge flow rate or high radiation sensed in the release header.

b. High discharge flow rate or loss of power to RMS Channel R-18

c. Loss of power to the flow recorder or loss of control air

d. High radiation sensed in the release header or loss of 125 VDC control power to the valve.

Answer d **Exam Level** B **Cognitive Level** Memory
Record Number: 73 **RO Number:** 58 **SRO Number:** 51

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

068 Liquid Radwaste System (LRS)

A4. Ability to manually operate and/or monitor in the control room:

A4.04 Automatic isolation 3.8 3.7

Explanation: d. - Correct, 2WL51 closes automatically on High rad in the discharge stream, loss of 125 VDC control power or 28 VDC control power, and loss of air to the valve . a. – There is no interlock on high flow rate since flow rate is specifically determined for each release. b. – See explanation for a. and WL51 closes on high radiation sensed by R-18, not power. c. – There is no interlock between the flow recorder and WL51.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RADIOACTIVE LIQUID WASTE SYSTEM	0300-000.00S-WASLIQ-01	V.B.3.b.5)	56		6e

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Significantly Modified (jkl)

Question Source Comments: Braidwood June 1999 NRC exam.

Question: PRT drain path

Given the following conditions for Unit 2:

- A Pressurizer Safety Valve has been leaking
- Procedural actions being taken to prevent overpressurizing the PRT are generating liquid rad waste
- 2WL13 is open

Which one of the following correctly describes the flow path of water from the PRT after a NCO opens 2PR14, PRT Drain Valve?

a. The PRT gravity drains to the in-service CVC HUT.

b. The PRT gravity drains to the RCDT. The RCDT pumps automatically cycle on RCDT level, pumping to the in-service CVC HUT.

c. The RCDT Pump in AUTO cycles to control PRT level whenever 2PR14 is open.

d. RCDT pumps start on interlock with 2WL12, directing flow to the in-service CVC HUT.

Answer d **Exam Level** R **Cognitive Level** Memory

Record Number: 74 **RO Number:** 59 **SRO Number:**

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

068 Liquid Radwaste System (LRS)

K1. Knowledge of the physical connections and/or cause-effect relationships between the Liquid Radwaste System and

K1.07 Sources of liquid wastes for LRS 2.7 2.9

Explanation: d. – Correct. Opening PR14 causes both RCDT pumps to start if WL12 and WL13 are open. PR14 aligns the PRT to the suction of the RCDT pumps, which are normally aligned to the CVCS Holdup Tanks; a. – The PRT does not drain directly to the CVCHUT; b. - PRT drains to the suction of the pumps downstream of a check valve, not directly to the RCDT; c. – The pumps cycle on valve position interlock, not PRT level.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RADIOACTIVE LIQUID WASTE SYSTEM	0300-000.00S-WASLIQ-01	IV.C.13.d.2).e) IV.D.1	36; 43		3.b.l; 6.c
No. 2 Unit Waste Disposal Liquid	205339	C-2	Sh. 3	28	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments: Pamexam

Question: Waste Gas system pressure

Which one of the following correctly states the basis for the inlet pressure limitation on WG41, Waste Gas Release Valve?

a. Maintain pressure less than the maximum design pressure for the packing in WG41

b. At higher pressures leaks may develop in the release line components, leading to an unmonitored release

c. Maintain D/P across the valve less than design to ensure the valve can close automatically

d. The calculated release flow rate could be exceeded at higher pressures

Answer d **Exam Level** B **Cognitive Level** Memory
Record Number: 75 **RO Number:** 60 **SRO Number:** 52

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

071 Waste Gas Disposal System (WGDS)

A3. Ability to monitor automatic operations of the Waste Gas Disposal System including:

A3.02 Pressure-regulating system for waste gas vent header 2.8 2.8

Explanation: d. - Correct, Pressure regulating valve 2WG38 is set to maintain a constant pressure upstream of 2WG41 so that a constant flow is maintained while discharging to the Plant Vent. a. – Pressure is maintained much lower than the design capabilities of the valve. b. - Pressure is not limited to prevent leaks. c. – D/P across the valve is very low, even if the procedural requirement is not maintained.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RADIOACTIVE WASTE GAS SYSTEM	0300-000.00S-WASGAS-01	IV.B.3.f	25		4k
DISCHARGE OF 21 GAS DECAY TANK TO PLANT VENT	S2.OP-SO.WG-0008(Q)	3.7	3	18	

Material Required for Examination

uestion Source: Facility Exam Bank **Question Modification Method:**

Question Source Comments:

Question: 1R1B-1 failure

Which one of the following describes the system response when Control Room Air Intake Radiation Monitor 1R1B-1 fails high?

a. No system response since a 2/4 coincidence is necessary for actuation.

b. Only Unit 1 EACS equipment and Unit 1 EACS Outside Air Dampers are actuated.

c. Both Unit 1 and Unit 2 EACS equipment are actuated, but only the Unit 2 EACS Outside Air Dampers are actuated.

d. The CAAC system normal supply duct dampers (1CAA40 & 1CAA43) will close but no other system configuration changes occur.

Answer c **Exam Level** S **Cognitive Level** Memory
Record Number: 76 **RO Number:** **SRO Number:** 53

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

072 Area Radiation Monitoring (ARM) System

A2. Ability to (a) predict the impacts of the following on the ARM system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:

A2.02 Detector failure 2.8 2.9

Explanation: c. - Correct, the failure actuates CRIX Train B only, which actuates one EACS fan per Unit and aligns the Unit 2 Outside Air intake. a. - Coincidence is not required. c&d. - EACS fans on both Units are started and outside air is provided from Unit 2 EACS intake.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RADIATION MONITORING SYSTEM	0300-000.00S-RMS000-01	IV.B.1.a.2).d)	19, 85		6a
CONTROL AREA VENTILATION SYSTEM	0300-000.00S-CAVENT-0 0	V.B. 1 & 3.a	34, 35-36		4, 8
CONTROL AREA VENTILATION OPERATION	S1.OP-SO.CAV-0001(Q)	3.8, 3.9	4	23	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Original Q based on previous design for CR vent when Unit 1 and Unit 2 automatic actuations were separate. Changed 2 selections to reflect single unit no longer goes to Pressurized Mode.

Question: ARM Adverse Containment

While performing the Emergency Operating Procedures, a step is encountered which states, "Control PZR level between 25% (33% adverse) and 77% (74% adverse) by adjusting charging and letdown flows." Containment pressure has risen to 5.5 psig and dropped back down to 3.3 psig, containment ·diation levels have risen to 3E5 R/hr and have dropped back down to 6.7E4 R/hr.

Which one of the following states the indicated PZR levels that must be maintained?

a. Minimum of 25%; Maximum of 77%

b. Minimum of 33%; Maximum of 74%

c. As specified by the Shift Technical Advisor

d. As specified by the Operations Support Center

Answer b **Exam Level** B **Cognitive Level** Memory
Record Number: 77 **RO Number:** 61 **SRO Number:** 54
Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1
072 Area Radiation Monitoring (ARM) System
2.1 Conduct Of Operations
2.1.32 Ability to explain and apply all system limits and precautions. 3.4 3.8

Explanation :b. Correct, Adverse containment values are only required to be used if pressure remains above 4 psig. If pressure later drops to less than 4 psig, then normal values may be used. If radiation levels ever exceed 1E5 R/hr, then adverse values must continue be used, unless the TSC grants permission to use normal values. a. - Normal values are not to be used since radiation levels exceeded 1E5 R/hr. c&d. - If radiation levels exceeded 1E5 R/hr, then only the TSC can give permission to use other values after evaluation.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
'SE AND CONTROL OF PROCEDURES	0300-000.00S-PROCED-02	III.E.12	26		4
USE OF PROCEDURES	SC.OP-AP.ZZ-0102(Q)	5.3.10.B	17	7	

Material Required for Examination
Question Source: Facility Exam Bank **Question Modification Method:**
Question Source Comments: CGroup

Question: CFCU protection on LOOP

Which one of the following correctly describes the purpose of the accumulators installed in the Service Water System?

a. Maintain SW flow to the CFCU motor coolers while the SW Pumps are started and SW pressure recovers in SEC Mode 3

b. Maintain the CFCU's full of water to prevent damage from water hammer when the CFCU's and Service Water Pumps are automatically started in SEC Mode 3

c. Provide an in-surge/out-surge volume to ensure pressure in all CFCU's remain within design limits while it is isolated and waiting for a SEC Mode 3 start signal

d. Maintain SW pressure in the CFCU's to prevent the all SW57's, CFCU Inlet Pressure Control Valves, from opening too rapidly when a CFCU is started and causing runout on SW Pumps starting in SEC Mode 3

Answer b **Exam Level** B **Cognitive Level** Memory
Record Number: 78 **RO Number:** 62 **SRO Number:** 55
Tier: Plant Systems **RO Group:** 3 **SRO Group:** 3
076 Service Water System (SWS)
2.1.28: Knowledge of the purpose and function of major system components and controls. 3.2/3.3
Explanation: b. – Correct. Accumulators maintain the CFCU's full during the SEC Mode 3 CFCU/SW starting delays; a. – No need to maintain SW flow to the motor coolers when the CFCU is not running; c. – Another piece of the same design change ensures the CFCU is not over-pressurized. The accumulators are not intended as an in-surge volume; d. – Another piece of the same design change involved a SW flow-related travel stop on the SW57's but the reason is incorrect.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
SERVICE WATER - NUCLEAR HEADER	0300-000.00S-SW0NUC-01	IV.B.2.a 1) & 5)	16-18	4	

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: ECAC auto actions

Due to concurrent problems with off-site power and the Station Air Compressors, control air header pressure is lowering on both units.

'hich one of the following correctly describes the associated automatic actions for the conditions above?

a. At 85 psig, #1 ECAC automatically starts to supply "A" Header and #2 ECAC automatically starts to supply "B" Header

b. At 85 psig, #1 ECAC automatically starts to supply "B" Header and #2 ECAC automatically starts to supply "A" Header

c. At 80 psig, #1 ECAC automatically starts to supply "A" Header and #2 ECAC automatically starts to supply "B" Header

d. At 80 psig, #1 ECAC automatically starts to supply "B" Header and #2 ECAC automatically starts to supply "A" Header

Answer b **Exam Level** B **Cognitive Level** Memory
Record Number: 79 **RO Number:** 63 **SRO Number:** 56
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2
079 Station Air System (SAS)
K4. Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following:
K4.01 Cross-connect with IAS 2.9 3.2

Explanation: b. - Correct, Emergency Control Air Compressors (ECAC) start automatically when header pressure reaches 85 psig. #1 ECAC supplies the "B" Header, #2 ECAC supplies the "A" header. a. - #1 ECAC supplies the "B" Header, #2 ECAC supplies the "A" header. c&d. - Procedure directs the operators to trip the reactor if control air header pressure reaches 80 psig. The ECAC's start at 85 psig.

'eference Title	Facility Reference Number	Section	Page	Revision	L. O.
Control Air System	0300-000.00S-CONAIR-00	V.B.1.f.1)	37		4, 9

Material Required for Examination
Question Source: Facility Exam Bank **Question Modification Method:**
Question Source Comments: CGroup

Question: EDG CO_2

Which one of the following correctly describes a difference in response for an AUTO as compared to a MANUAL actuation of the DG Area CO_2 system?

a. AUTO CO_2 actuation is blocked on a SEC start

b. MANUAL CO_2 actuation trips the associated, running EDG

c. AUTO CO_2 actuation is blocked when the associated EDG is in LOCKOUT

d. On a MANUAL actuation, there is no CO_2 discharge delay

Answer d **Exam Level** B **Cognitive Level** Comprehension
Record Number: 80 **RO Number:** 64 **SRO Number:** 57
Tier: Plant Systems **RO Group:** 2 **SRO Group:** 2
086 Fire Protection System (FPS)
K4. Knowledge of Fire Protection System design feature(s) and/or interlock(s) which provide for the following:
K4.06 CO2 3.0 3.3

Explanation: d. – Correct. When a CO_2 system actuates, the system performs a planned sequence of events as follows: 1) First, a system timer is energized to start timing a predischarge period; 2) Once the predischarge period is timed out the CO_2 discharge starts. The CO_2 System may be discharged manually by using the operating lever installed on the electro-manual pilot operating cabinet. This method of operation bypasses the timer functions. a. - SEC will not block DG CO_2 actuation. b. - Neither auto nor manual CO_2 actuation will trip the DG, but it does trip DG Ventilation fans. c. - There are no interlocks between EDG switch positions and CO_2 actuation.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
FIRE PROTECTION SYSTEM	0300-000.00S-FIRPRO-00	IV.B.2.p.15).g)	55, 58		4.c.v, 7.b & c
DIESEL GENERATOR AREA VENTILATION OPERATION	S2.OP-SO.DGV-0001(Q)	3.7	3	5	

Material Required for Examination
Question Source: Facility Exam Bank **Question Modification Method:**
Question Source Comments: Pamexam

Question: Uncontrolled control rod withdrawal

Unit 2 is at 50% power with all major controls in AUTO.

/ uming no operator action, which one of the following failures will cause a continuous rod withdrawal?

a. NIS Power Range Channel N-41 fails low

b. RCS Loop 23 Thot RTD fails low

c. RCS Loop 23 Tcold RTD fails high

d. Turbine First Stage Pressure Transmitter PT-505 fails high

Answer d **Exam Level** S **Cognitive Level** Application
Record Number: 81 **RO Number:** **SRO Number:** 58
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1
001 Continuous Rod Withdrawal
AA2. Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal:
AA2.05 Uncontrolled rod withdrawal, from available indications 4.4 4.6

Explanation: d. – Correct. AUTO Rod Control believes turbine power is much higher than nuclear power; a. – No motion since PRNIS signal is from auctioneered high; b. – No motion since Tave signal is from auctioneered high; c. – Inward motion since Tave is failed high.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CONTINUOUS ROD MOTION TECHNICAL BASES DOCUMENT	S2.OP-AB.ROD-0003(Q)	2.2	3	7	
CONTINUOUS ROD MOTION	0300-000.00S-ABROD3-00	II.A.2.d	8		1, 4.A

Material Required for Examination
Question Source: NRC Exam Bank **Question Modification Method:** Significantly Modified
Question Source Comments: Braidwood 6/7/1999 NRC Exam Q58. Modified conditions and distractor 'b'.

Question: Rod recovery alarms

Given the following conditions on Unit 2:

- Reactor power - 25%
- Control rod 2D2 in Control Bank D has fully dropped.
- Recovery of the dropped rod is in progress per S2.OP-AB.ROD-0002(Q) "DROPPED ROD"
- All Disconnect Switches in Control Bank D are in DISCONNECT except for 2D2

Which one of the following describes an alarm that will actuate and the affect that alarm actuation will have on recovering the dropped control rod?

a. A Non-Urgent Failure will actuate; rod recovery can proceed without additional operator action

b. A Non-Urgent Failure will be received; rod recovery can proceed after depressing the ALARM RESET pushbutton on the console

c. An Urgent Failure will actuate; rod recovery can proceed without additional operator action

d. An Urgent Failure will actuate; rod recovery can proceed after depressing the ALARM RESET pushbutton on the console

Answer c **Exam Level** B **Cognitive Level** Comprehension
Record Number: 82 **RO Number:** 65 **SRO Number:** 59
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1
003 Dropped Control Rod
AK2. Knowledge of the interrelations between the Dropped Control Rod and the following:
AK2.05 Control rod drive power supplies and logic circuits 2.5 2.8

Explanation: c. Correct – The alarm is from the opposite group not moving. With the RBSS selected to CBD, rod motion continues. a.&b. – A Non-Urgent Failure alarm does not actuate. d. – Rod motion continues because of the position of the RBSS.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
DROPPED ROD	S2.OP-AB.ROD-0002(Q)	3.33 & NOTE	5	5	
DROPPED ROD	0300-000.00S-ABROD2-00	II.C.1.d&e	8		4.A

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: Misaligned rod indication

Unit 1 is at 100% power when a SGFP trip results in an automatic rapid power reduction.

Which one of the following correctly identifies an alarm actuation that would be indicative of a single immovable control rod?

a. Auxiliary Annunciator "DELTA I/EXCEEDS TARGET BAND"

b. OHA E-38, UPPER SECT DEV ABV 50% PWR

c. OHA E-40, ROD BANK URGENT FAILURE

d. TAVE/TREF DEV Console Alarm

Answer b **Exam Level** B **Cognitive Level** Comprehension
Record Number: 83 **RO Number:** 66 **SRO Number:** 60
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1
005 Inoperable/Stuck Control Rod
AA2. Ability to determine and interpret the following as they apply to the Inoperable/Stuck Control Rod:
AA2.01 Stuck or inoperable rod from in-core and ex-core NIS, in-core or loop temperature measurements 3.3 4.1
Explanation: b. – Correct. A single immovable rod affects quadrant power; a. – This alarm is indicative of bank motion/misalignment; c. – This should only be indicative of a rod bank problem; d. – The moving rods will still correct Tave/Tref deviations.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
IMMOVABLE/MISALIGNED CONTROL RODS TECHNICAL BASES DOCUMENT	S2.OP-AB.ROD-0001(Q)	3.5	5	6	
ROD POSITION INDICATION FAILURE TECHNICAL BASES DOCUMENT	S2.OP-AB.ROD-0004(Q)	2.2	2	5	
IMMOVABLE/MISALIGNED CONTROL ROD	0300-000.00S-ABROD1-01	II.D.1	11		3.A.4

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: RTB failure to open

Given the following conditions for Unit 2:

- A reactor trip has occurred from 100% power
- Train A reactor trip breaker failed to open
- I&C has NOT installed the P-4 jumpers

Which one of the following correctly describes a consequence of Train A reactor trip breaker remaining closed?

a. Train A SI signal reset and automatic SI block capability is lost.

b. Condenser steam dumps will NOT function in Average Temperature mode.

c. Main Steamline Isolation will NOT function on Steam Flow High coincident with Low Tave.

d. Control room operators will have to manually close the Feedwater Regulating Valves (BF19's) and Feedwater Regulating Bypass Valves (BF40's).

Answer a **Exam Level** S **Cognitive Level** Comprehension
Record Number: 84 **RO Number:** **SRO Number:** 61
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
007 Reactor Trip
EA2. Ability to determine and interpret the following as they apply to a reactor trip:
EA2.03 Reactor trip breaker position 4.2 4.4

Explanation: a. – Correct. The P-4 signal must be available from each train to reset and block the SI signal (for that train). b. - RTB A provides the signal that arms Steam Dumps on the trip; however, steam dumps will still function from the load rejection signal (turbine impulsepressure). c. - Main Steamline Isolation does not receive an input from P-4. d. - The Feedwater Isolation will still occur with either reactor trip breaker open (and low Tave) due to the arrangement of the solenoid valves.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RPS - Safeguards Actuation Signals	221057	2-D	8	17	
EOP-TRIP-2, REACTOR TRIP RESPONSE	0300-000.00S-TRP002-02	3.3.6.1	14-15		2
REACTOR PROTECTION SYSTEM	0300-000.00S-RXPROT-00	VII.B.6.b-d	50		10

Material Required for Examination
Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified
Question Source Comments: Pam Exam. Changed wording to change position of correct answer.

Question: SR response following Rx trip

Which one of the following is the expected Source Range indication following a reactor trip? (No operator action taken)

The Source range channels read approximately...

a. 4000 cps at 10 minutes, post trip.

b. 6000 cps at 20 minutes, post trip

c. 0 cps at 30 minutes, post trip.

d. 1000 cps at 40 minutes, post trip.

Answer b **Exam Level** R **Cognitive Level** Memory
Record Number: 85 **RO Number:** 67 **SRO Number:**

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

007 Reactor Trip

EK1. Knowledge of the operational implications of the following concepts as they apply to the reactor trip:

EK1.05 Decay power as a function of time 3.3 3.8

Explanation: The SR instruments will automatically actuate when both IR Channels are less than 7E-11 Amps. From the normal "at power" IR level, this takes about 13-18 minutes with -80 second period (-1/3 dpm). b. - Correct, This is the approximate count rate on SR when it re-energizes 15 minutes following a trip. a. - Expected count rate is reasonable. However, the time for automatic SR reinstatement is premature. c. - The SR would be energized by this time and is expected to read about 100-500 cps. d. - The count rate is too high for the time following trip.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
REACTOR TRIP RESPONSE	2-EOP-TRIP-2	22	4	23	
EXCORE NUCLEAR INSTRUMENTATION SYSTEM	0300–000.00S–EXCORE–00	IV.B, IV.D.2.h	22, 29-31		2, 7.c

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments:

Question: PORV leak indication

Given the following conditions for Unit 2:

- Reactor power is 87%
- Pzr pressure is 2235 psig
- Pzr PORV 2PR1 is leaking
- PRT pressure is 5 psig
- PORV discharge temperature has stabilized near 230°F

Considering each one individually, which one of the following directly causes PORV discharge temperature to rise?

a. PRT pressure is allowed to rise to 10 psig

b. PORV leak rate rises by 2 gpm

c. Pzr vapor space temperature rises by 1°F

d. The PRT rupture disk fails

Answer a **Exam Level** B **Cognitive Level** Application
Record Number: 86 **RO Number:** 68 **SRO Number:** 62
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
AK3. Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident:
AK3.02 Why PORV or code safety exit temperature is below RCS or PZR temperature 3.6 4.1

Explanation: The leaking PORV can be considered a isenthalpic process. The temperature downstream is based on the downstream pressure at the determined enthalpy for the Pzr pressure. a. - Correct, If PRT pressure rises, the saturation temperature of the PRT increases. Since both conditions result in saturated conditions temperature rises from 230°F to 235°F. b. - Increase in the leak rate does not significantly affect the throttling (isenthalpic) process. c. - Increase in Pzr temperature results in an increase in pressure. This results in a lower enthalpy for the steam. However, since pressure has not changed in the PRT, tailpipe temperature does NOT change. d. – Lowering PRT pressure lowers tailpipe temperature.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Steam Tables					
PRESSURIZER PRESSURE MALFUNCTION	0300-000-00S-ABPZR1-01	III.C.1	9		1, 3

Material Required for Examination Steam Tables
Question Source: NRC Exam Bank **Question Modification Method:** Direct From Source
Question Source Comments: Prairie Island 1996 NRC exam.

Question: RCS pressure & temperature changes

Given the following conditions for Unit 2:

- A Small Break LOCA has occurred
- All ECCS pumps are operating as designed
- Twenty minutes after the initial transient, the following conditions exist:
  - No RCPs running
  - Core Exit TCs read 580°F
  - Pzr level indicates 0%
  - RCS pressure is at 1310 psig
- The operators begin drawing more steam from all S/Gs and increase AFW flow to maintain level

Which one of the following describes how and why ECCS flow changes as a result of these operator actions?

a. As the plant cools down, RCS pressure lowers and ECCS flow rises

b. ECCS flow will not change until the pressurizer begins to refill, then ECCS flow will lower

c. For this set of conditions, cooldown has no effect on ECCS flow. RCS pressure cannot change unless one or more ECCS pumps are stopped

d. ECCS flow will not change because the SI and Charging Pumps all are operating at their maximum flow rate

Answer a **Exam Level** B **Cognitive Level** Comprehension

Record Number: 87 **RO Number:** 69 **SRO Number:** 63

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

009 Small Break LOCA

A2. Ability to determine and interpret the following as they apply to a small break LOCA:

EA2.01 Actions to be taken, based on RCS temperature and pressure, saturated and superheated 4.2 4.8

Explanation: No subcooling a.- Correct. Pzr pressure control is unavailable with 0% level. As the cooldown continues RCS temperature and pressure will fall. This will allow more flow from the ECCS pumps. b. – ECCS flow will change before the pressurizer refills. c. – In a saturated system, a temperature reduction affects pressure. This choice is more likely in a solid system. d. – Neither set of pumps is at their maximum flow rate.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
EOP-LOCA-2, POST LOCA COOLDOWN AND DEPRESSURIZATION	0300-000.00S-LOCA02-01	I.C.4; III.C.10.b	7, 23		1, 5

Material Required for Examination Steam Tables

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: SOQL0002.

Question: Hot Leg recirc basis

EOP-LOCA-1, Loss of Reactor Coolant, Step 28 reads "WAIT UNTIL 14 HOURS HAVE ELAPSED SINCE SI ACTUATION". The following arrow box reads "EOP-LOCA-4, TRANSFER TO HOT LEG RECIRCULATION".

Which one of the following correctly describes the basis for transitioning to EOP-LOCA-4 after 14 hours?

a. Eliminate steam voids that may be hindering heat removal in the upper core

b. Wash fission product particulates back into solution for processing in the CVCS demineralizers

c. Preclude the potential for boron precipitation to hinder core cooling

d. Cover the core above the hot leg elevation to establish natural circulation flow to the SG's

Answer c **Exam Level** B **Cognitive Level** Memory
Record Number: 88 **RO Number:** 70 **SRO Number:** 64

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1
011 Large Break LOCA
EA1. Ability to operate and/or monitor the following as they apply to a Large Break LOCA:
EA1.11 Long-term cooling of core 4.2 4.2

Explanation: c. – Correct, per EOP-LOCA-4 Basis Document; a. – Steaming is a means of heat removal rather than a hindrance; b. – At this point, containment rather than processing waste is the concern; d. – SG's are not re-established as the heat sink.

Reference Title Facility Reference Number	Section	Page	Revision L. O.
LOSS OF REACTOR COOLANT	2-EOP-LOCA-4, Basis Document		
EOP-LOCA-01, LOSS OF REACTOR COOLANT AND LOSS OF COOLANT ACCIDENT ANALYSIS	0300-000.00S-LOCA04, Objective 4		

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: Contmt. Sump indications for LBLOCA

Which one of the following correctly lists indications that are evaluated as possible sources of excessive inventory in the containment sump, per EOP-FRCE-2, Response to High Containment Sump Level?

a. Fire Protection water flow, Demineralized Water Storage Tank, Primary Water Storage Tank

b. Fire Protection water flow, Auxiliary Feedwater Storage Tank, Component Cooling Water Surge Tank

c. CFCU SW flow, CVCS Volume Control Tank, Component Cooling Water Surge Tank

d. CFCU SW flow, Demineralized Water Storage Tank, Boric Acid Storage Tank

Answer a **Exam Level** R **Cognitive Level** Memory
Record Number: 89 **RO Number:** 71 **SRO Number:**

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

011 Large Break LOCA

2.1 Conduct Of Operations

2.1.31 Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup. 4.2 3.9

Explanation: a. – Correct, per FRCE-2; b. – AFWST is incorrect; c. – VCT is incorrect; d. – BAST is incorrect

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
EOP-FRCE-2, Response to Excessive Containment Sump Level					
Lesson Plan 300-000.00S-FRCE00, Objective 5					

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: LOCA Contiuous Action

A LOCA has occurred on Unit 2. After stopping all ECCS pumps except 21 Charging Pump, the control room crew properly transitioned from EOP-TRIP-3, SI Termination to EOP-LOCA-2, Post-LOCA Cooldown and Depressurization. The following conditions are observed:

- 21 Charging Pump running, drawing from the RWST
- Group Busses were lost during the automatic transfer
- RCS Subcooling is 20 °F
- RCS Pressure 1700 psig, slowly rising
- Pressurizer (PZR) Level 21%, slowly rising
- Containment Pressure 3 psig
- VCT Level is 19%

Which one of the following is the correct crew action if the leak rate begins to rise?

a. If PZR level reaches ≤19% and RCS subcooling = 0 °F, then start ECCS Pumps as necessary to restore PZR level or subcooling

b. If PZR level reaches ≤19% or RCS subcooling = 0 °F, then start ECCS Pumps as necessary to restore PZR level and subcooling

c. If PZR level reaches ≤11% or RCS subcooling = 0 °F, then start ECCS Pumps as necessary to restore PZR level and subcooling

d. If PZR level reaches ≤11% and RCS subcooling = 0 °F, then start ECCS Pumps as necessary to restore PZR level or subcooling

Answer c **Exam Level** S **Cognitive Level** Application
Record Number: 90 **RO Number:** **SRO Number:** 65
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1
011 Large Break LOCA
2.4 Emergency Procedures / Plan
2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. 4.0 4.0

Explanation: c. – Correct. LOCA-2 CAS; a., b., d. – Combinations of either adverse values for non-adverse conditions or misplaced and/or statements.

Reference Title	**Facility Reference Number, Section**	**Page**	**Revision**	**L. O.**
EOP-LOCA-2, Post-LOCA Cooldown and Depressurization, CAS				
Lesson Plan 300-000.00S-LOCA02, Obj. 4, 5				

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question Indication with #2 seal in control

The unit is at 100% power.

W...ch one of the following set of indications would occur if 21CV104, Seal Leakoff Isolation Valve, fails closed while operating at 100% power?

a. #1 Seal D/P indicates low and PRT level is rising

b. #1 Seal D/P indicates low and Seal Leakoff Flow is zero

c. #1 Seal D/P indicates high and RCDT level is rising

d. #1 Seal D/P indicates high and Seal Leakoff Flow is zero

Answer b **Exam Level** B **Cognitive Level** Comprehension
Record Number: 91 **RO Number:** 72 **SRO Number:** 66
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1
015 Reactor Coolant Pump (RCP) Malfunctions
AK2. Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions and the following:
AK2.10 RCP indicators and controls 2.8 2.8

Explanation: b. Correct. Closure of the seal leakoff results in the following as the #2 seal becomes the full pressure drop boundary: #1seal D/P goes to ZERO, seal leakoff flow goes to ZERO since line is isolated, #2 seal leakoff increases, Standpipe high level alarm may be expected as #2 seal leakoff flow increases (backs up into standpipe). a. - PRT level increase is expected only if a seal leakoff containment isolation valve (2CV116 or 2CV284) closes with CV104 valve open. c.&d. - D/P would NOT indicate high because the pressure drop is across #2 seal, not #1 seal.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
REACTOR COOLANT PUMP ABNORMALITY TECHNICAL BASIS DOCUMENT	S2.OP-AB.RCP-0001(Q)	2.4.D	4	12	
CONTROL CONSOLE 2CC1	S2.OP-AR.ZZ-0011(Q)	Alarm 2-5, 3.1;	129	26	
REACTOR COOLANT PUMP ABNORMALITY	0300-000.00S-ABRCP1-01	VII.E.7.e-g, F.3	25-26, 28		4.B

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Prairie Island 5/1999 NRC exam.

Question: Makeup system failure

Given the following conditions on Unit 2:

- Reactor power is 65%
- Auto makeup initiated to the VCT
- Shortly after AUTO Makeup started, boric acid filter clogging caused the BORIC ACID FLOW DEVIATION console alarm actuate

Assuming no operator action, which one of the following correctly describes what will occur?

a. Control rods will insert in AUTO to control Tave

b. Reactor power will rise slightly and level off.

c. The running Boric Acid Transfer Pump will trip

d. VCT level will drop until charging suction swaps to the RWST.

Answer d **Exam Level** B **Cognitive Level** Comprehension

Record Number: 92 **RO Number:** 73 **SRO Number:** 67

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

022 Loss of Reactor Coolant Makeup

AA1. Ability to operate and/or monitor the following as they apply to the Loss of Reactor Coolant Pump Makeup:

AA1.08 VCT level 3.4 3.3

Explanation: d. Correct. With the deviation alarm in for over 60 sec., a signal is sent to close 2CV185, Charging Suction Make-Up Stop Valve. This terminates makeup. a. - Assumes dilution only continues, but this is stopped when 2CV185 closes. b. - Power constant due to turbine load. Also dilution is stopped when 2CV185 closes. c. – No interlock to trip the BAT Pump.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
ONTROL CONSOLE 2CC2	S2.OP-AR.ZZ-0012(Q)	3-16	19	12	
CHEMICAL AND VOLUME CONTROL SYSTEM	0300-000.00S-CVCS00-01	V.B.2,v.2) & gg.2)	83, 103		4.d, 8

Material Required for Examination

Question Source: Other Facility **Question Modification Method:** Editorially Modified

Question Source Comments: Prairie Island exam bank

Question: Boric Acid Pump ops

Given the following conditions for Unit 2:

- Power is 90%
- A plant transient has occurred that results in Control Bank D rods inserting beyond their insertion limits
- The RO is in the process of initiating a rapid boration,
- One Boric Acid Transfer Pump (BATP) is running in FAST speed
- 2CV175, Rapid Borate Stop Valve is open
- Charging flow is 75 gpm on 2FI-128B

Which one of the following actions will significantly raise the RCS boration rate?

a. Starting the second BATP in FAST speed.

b. Closing 21/22CV160 Boric Acid Tank Recirc valves.

c. Throttle further open 2CV71, Seal Pressure Control valve.

d. Close 2VCV175 and align the Charging Pumps suction to the RWST

Answer b **Exam Level** B **Cognitive Level** Comprehension

Record Number: 93 **RO Number:** 74 **SRO Number:** 68

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

024 Emergency Boration

AA1. Ability to operate and/or monitor the following as they apply to the Emergency Boration:

AA1.20 Manual boration valve and indicators 3.2 3.3

Explanation: b. - Correct. Closing the recirculation valves forces more flow through the discharge line to the charging pump suction. This action is directed in the procedure. a. - Due to system limitations, boron addition rate is essentially same whether one or both Boric Acid Transfer Pumps are operating. c. This will increase charging flow to the RCS and reduce flow to the RCP seals. Adjusting CV71 just changes the direction of flow and has no effect on boration rate. d. - The flowpath from the RWST 1s an alternate rapid boration flowpath. However, the boration rate is lower because the RWST is at a lower boron concentration.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RAPID BORATION	S2.OP-SO.CVC-0008(Q)	3.2, 5.1.5/6	2-3	2	
CHEMICAL AND VOLUME CONTROL SYSTEM	0300-000.00S-CVCS00-01	8.b & c	134-135		4.d, 12

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: Loss of RHR Alternate cooling

Given the following conditions:

- Plant in Mode 5
- Highest CET temperature is 190°F
- RCS pressure is 325 psig
- 21 RHR loop is in service, 22 RHR loop is out of service for repairs
- RCS is intact with 20% Pzr level indicated
- 21 RHR Pump experiences a seal failure and is isolated from the RCS

Which one of the following is the preferred method of core cooling if a RHR cannot be restored and RCS temperatures are rising?

a. Natural or forced RCS flow while steaming intact S/Gs with a level of equal to or greater than 70% NR.

b. Fill from RWST via one SI Pump and the Hot Leg Injection Isolation valves, and spill through the Pzr PORVs.

c. Fill via RWST gravity flow through RHR and reflux cooling to any S/G with level equal to or greater than 70% NR.

d. Fill from RWST via one charging pump and the BIT Isolation valves, and spill via both PORVs and Reactor Head Vent Solenoid Valves.

Answer d **Exam Level** B **Cognitive Level** Comprehension
Record Number: 94 **RO Number:** 75 **SRO Number:** 69
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
025 Loss of Residual Heat Removal System (RHRS)
K3. Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System:
AK3.01 Shift to alternate flowpath 3.1 3.4

Explanation: d. Correct. With Core Exit TCs < 200°F, the preferred alternate path is Cold Leg Injection Feed & Bleed. a. - Natural or forced flow cooling via S/G is an allowed method but not preferred (fails to minimize temperature rise). b. - Hot leg injection feed and bleed is a preferred path if CETs are greater than or equal to 200°F. c. - Reflux cooling is used only if RCS is depressurized & RCPs NOT available. (least limiting of temperature rise).

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF RHR	S2.OP-AB.RHR-0001(Q)	3.31, Attach 8	9, 1-3	9	
LOSS OF RHR	0300-000.00S-ABRHR1-01	III.D.7.a	16		3, 4.c

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: CCW leak

Given the following conditions for Unit 2:

- A LOCA has occurred
- Actions of 2-EOP-LOCA-3 "TRANSFER TO COLD LEG RECIRCULATION" have been completed
- Two CCW Pumps are running

Which one of the following correctly identifies a consequence resulting from a tube leak in the Seal Water Heat Exchanger?

a. Both CCW Pumps will eventually trip on loss of NPSH.

b. A high level alarm will actuate for the CCW Surge tank.

c. The CCW Pump supplying the safety related header will eventually trip due to loss of NPSH.

d. The CCW Pump supplying the non-safety related header will eventually trip due to loss of NPSH.

Answer d **Exam Level** B **Cognitive Level** Comprehension
Record Number: 95 **RO Number:** 76 **SRO Number:** 70

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

026 Loss of Component Cooling Water (CCW)

AA2. Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water:

AA2.02 The cause of possible CCW loss 2.9 3.6

Explanation: d. Correct. At the completion of EOP-LOCA-3, the CCW systems are aligned as independent loops with 22 CC loop supplying the non-safety loads. A leak in the Seal Water HX will allow CCW in-leakage to CVCS. The CCW Surge Tank internal baffle indicated level will prevent loss of suction (NPSH) to the 21 CCW Pump but the side from which 22 CC Pump takes suction from will fall and NPSH is lost. d. - Correct, 22 CCW Pump is supplying non-safety header a. - Only 22 CCW Pump affected to point where NPSH lost. b. – Leakage is in the opposite direction. c. - 21 CCW Pump will maintain adequate NPSH due to baffle in CCW Surge Tank.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
No. 2 Unit Component Cooling P&ID	205331		1 & 2	48, 35	
COMPONENT COOLING WATER	0300-000.00S-CCW000-02	IV.B.1.b.3) & 4)	19		3.c, 4.a
EOP-LOCA-03, TRANSFER TO COLD LEG RECIRCULATION	0300-000.00S-LOCA3-U2-00	IX.C.39	38-39		2

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Changed LTDN HX to Seal Water HX

Question Source Comments: Pam NRC 1998 re-exam.

Question: POPS input failure

Given the following conditions on Unit 2:

- RCS Tave - 150°F
- RCS pressure - 280 psig

Which one of the following describes the response of the PORVs if PT-405 wide range loop pressure transmitter fails high?

a. Both PORVs open because the coincidence is 1/2 with POPS armed.

b. Only the PORV fed by that channel opens

c. Neither PORV opens because the enabling signal from the other channel is NOT met

d. Neither PORV opens because both open when 2/2 WR pressure channels are >setpoint

Answer b **Exam Level** B **Cognitive Level** Memory
Record Number: 96 **RO Number:** 77 **SRO Number:** 71

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 2

027 Pressurizer Pressure Control (PZR PCS) Malfunction

AK2. Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following:

AK2.03 Controllers and positioners 2.6 2.8

Explanation: b. - Correct, RCS loop 1 hot leg wide-range, PT-405, Channel I provides the pressure input for POPS operation of 2PR1. a. - The circuits have separate temperature & input channels with control 1/1 for each PORV. c. – There is no feed from the opposite channel. d. -Each pressure transmitter feeds a separate POPS channel.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
PRESSURIZER PRESSURE AND LEVEL 'ONTROL	0300-000.00S-PZRP&L-01	IV.B.3, V.E.1	27, 34		4.I, 8

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: PZR level alarms

Given the following conditions on Unit 2:

- Turbine load is 19% and Rod Control is in AUTO
- Charging flow controller has failed high

Which one of the following identifies the approximate value for actual pressurizer level when OHA E-20, PZR HTR ON LVL HI, actuates?

a. 28%

b. 33%

c. 55%

d. 70%

Answer b **Exam Level** B **Cognitive Level** Application
Record Number: 97 **RO Number:** 78 **SRO Number:** 72
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3
028 Pressurizer (PZR) Level Control Malfunction
AA2. Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions:
AA2.01 PZR level indicators and alarms 3.4 3.6

Explanation: Controller will turn on heaters when level is 5% above the level setpoint. Level changes from 22% (22.3) at 0% power to 49.7% at 100% power. Level setpoint at 19% power is 27-28% (27.5). b. Correct. OHA E-20 comes in & heaters actuate at 28+5 = 33%. a. – Approx. level setpoint at 19% power. c. – 5% above the 100% setpoint. d. - Bezel alarm for Hi Pzr level actuates at 70%.

Reference Title	Facility Reference Number	Section	Page	Revision	L.O.
OVERHEAD ANNUNCIATORS WINDOW E	S2.OP-AR.ZZ-0005(Q)	E-20	32	12	
PRESSURIZER PRESSURE AND LEVEL ONTROL	0300-000.00S-PZRP&L-01	IV.B.2.c.1)	26		6.g, 9

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Wolf Creek 1993 NRC. Changed from temperature to power value. Changed power level.

Question: FRSM actions for boration

Which one of the following is a correct statement regarding operation of the charging pump(s) during the implementation of FRSM-1, Response to Nuclear Power Generation.

a. A manual safety injection is initiated to start both centrifugal charging pumps and thereby ensure the maximum possible boron injection rate

b. If RCS pressure exceeds 2335 psig then a second centrifugal charging pump is started to ensure the maximum possible boron injection rate

c. Both charging pumps are placed in-service to provide the maximum possible charging flow and boron injection rate

d. Only one charging pump is run. Running only one pump prevents excessive charging from contributing to a RCS pressure rise that actually lowers the boration rate

Answer c **Exam Level** S **Cognitive Level** Memory
Record Number: 98 **RO Number:** **SRO Number:** 73

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

029 Anticipated Transient Without Scram (ATWS)

AA2. Ability to determine and interpret the following as they apply to a ATWS:

AA2.04 CVCS centrifugal charging pump operating indication 3.2 3.3

Explanation: c. Correct, per FRSM-1 and FRSM-1 Basis Document; a. – Not initiating SI to avoid a SGFP trip is discussed in the Basis Document; b. – If RCS pressure exceeds 2335 then PZR PORV operation is initiated; d. – Two charging pumps are always started.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RESPONSE TO NUCLEAR POWER GENERATION	2-EOP-FRSM-1	4	1	23	
EOP-FRSM-1 and 2 RESPONSE TO NUCLEAR POWER GENERATION	0300-000.00S-FRSM00-02	3.2.4	21-22		4.A, 5.A

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: ATWS conditions

Which one of the following correctly describes the reason why it is worse for a full-power ATWS event to occur at the Beginning-of-Life (BOL) as compared to the End-of-Life (EOL)?

a. The additional burnable poisons provide less heat conduction; therefore, the fuel pin outer clad temperatures are higher.

b. The effective delayed neutron fraction is higher; therefore, the rate of power reduction is slower.

c. The Moderator Temperature Coefficient (MTC) is less negative; therefore, the reactor power reduction due to heat addition is less.

d. The higher boron concentration in the RCS causes the emergency boration to be less effective; therefore, it takes longer to achieve adequate Shutdown Margin (SDM).

Answer c **Exam Level** B **Cognitive Level** Memory
Record Number: 99 **RO Number:** 79 **SRO Number:** 74

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

029 Anticipated Transient Without Scram (ATWS)

EK1. Knowledge of the operational implications of the following concepts as they apply to the ATWS:

EK1.05 Definition of negative temperature coefficient as applied to large PWR coolant systems 2.8 3.2

Explanation: c. Correct, An ATWS event is more severe early in core life (least negative MTC) due to RCS pressure response; pressure could exceed safety setpoints. a. - Reduced heat conduction will raise fuel temperature which is better for Doppler coefficient. b. - Delayed neutron fraction is lower at BOL. d. - Boron addition rates do not mitigate the peak transient concerns.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
EOP-FRSM-1 and 2 Response to Nuclear Power Generation	0300-000.00S-FRSM00-02	1.3.4; 1.4.5	8-9		2.A, 5.A

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: PAM NRC 6/1998 re-exam

Question: SR fuse failure

With Unit 2 in MODE 3, an I&C technician is troubleshooting a power supply problem at NIS Source Range N-32 drawer. The following indications are received at the main control boards:

- OHA E-5 SR DET VOLT TRBL actuates
- OHA E-13 SR HI FLUX AT S/D actuates
- OHA F-25 SR FLUX HI actuates
- Source range counts - 50 cps (N31), 0 cps(N32)

Which one of the following occurred?

The I&C technician ...

a. removed power to TWO Power Range channels.

b. activated the RPS input for the SOURCE RANGE BLOCK.

c. pulled the instrument power fuses for N32 with the Level Trip switch in NORMAL.

d. pulled the control power fuses for N32 after placing the Level Trip switch in BYPASS.

Answer c **Exam Level** S **Cognitive Level** Comprehension
Record Number: 100 **RO Number:** **SRO Number:** 75
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
032 Loss of Source Range Nuclear Instrumentation
AA2. Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation:
AA2.05 Nature of abnormality, from rapid survey of control room data 2.9 3.2

Explanation: c. Correct. Loss of instr power with the Level Trip Sw in NORMAL will generate a trip of the drawer bistables (Flux trip, Hi Flux at S/D, Loss of Det volts) and result in loss of indication (0 cps or downscale). a&b. - Either action has the same affect in that voltage is removed from the affected SR detector(s). a. - Would affect both SR NIS channels indication but trip & S/D bistables would NOT actuate. b. - Could affect only one channel but trip & S/D bistables would NOT actuate. d. - The level trip switch will NOT prevent a trip signal or actuation of the indicated bistables but SR indication would NOT be affected.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
NUCLEAR INSTRUMENTATION SYSTEM MALFUNCTION	S2.OP-AB.NIS-0001(Q)	3.19	5	3	
NUCLEAR INSTRUMENTATION SYSTEM MALFUNCTION	0300-000.00S-ABNIS1-00	19	13		1
EXCORE NUCLEAR INSTRUMENTATION SYSTEM	0300–000.00S–EXCORE–00	V.A.1.a, b, c	53		8

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments:

Question: IR overcompensated

Which one of the following describes the effect of having compensating voltage set too high on N35 during a unit startup?

a. N35 indicates lower than N36; P-6 is not affected since the coincidence is 1/2.

b. N35 indicates higher than N36; the Source Range High Flux Trip will occur prior to reaching P-6.

c. N35 indicates lower than N36 but the SUR will be the same.

d. N35 indicates higher than N36 and P-6 will energize prior to achieving proper Source Range/Intermediate Range overlap on the correctly reading channel

Answer a **Exam Level** B **Cognitive Level** Memory
Record Number: 101 **RO Number:** 80 **SRO Number:** 76

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

033 Loss of Intermediate Range Nuclear Instrumentation

AA2. Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation:

AA2.02 Indications of unreliable intermediate-range channel operation 3.3 3.6

Explanation: a. Correct, Overcompensation occurs resulting in a lower reading for the affected channel. Also the SUR for that channel will be higher since the curve slope increases faster. However P-6 is NOT affected because only one channel is required to BLOCK the SR channels. b. This may be true if BOTH IR channels are overcompensated. c. - N36 reads higher and the relative difference is not constant since the rate of change (SUR) for the affected channel is higher. d. – N-35 will be reading lower.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
EXCORE NUCLEAR INSTRUMENTATION SYSTEM	0300–000.00S–EXCORE–00	IV.D.2.h.5).b)	31		5.b

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Concept Used

Question Source Comments: SOQL0189. Changed from shutdown to startup conditions.

Question: Worst case FH accident

Which one of the following identifies the most limiting fault occurrence, involving potential release of radioactivity to the public, for a fuel handling accident in which a fuel assembly is dropped?

The fuel assembly involved is a...

a. spent fuel assembly dropped into the core.

b. spent fuel assembly dropped into the spent fuel pit.

c. new fuel assembly dropped into the core at the end of a refueling operation.

d. new fuel assembly dropped into the spent fuel pit at the beginning of a refueling operation.

Answer b **Exam Level** S **Cognitive Level** Memory
Record Number: 102 **RO Number:** **SRO Number:** 77

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

036 Fuel Handling Incidents

AA2. Ability to determine and interpret the following as they apply to the Fuel Handling Incidents:

AA2.03 Magnitude of potential radioactive release 3.1 4.2

Explanation: Fuel Handling Accident (Condition IV event): Dropping of a spent fuel assembly onto the spent fuel pit floor resulting in the rupture of the cladding of all fuel rods in the assembly. b. - Correct, This is the most limiting event due to the lack of containment when the assembly is dropped. a. - The accident in containment has the containment barrier to reduce release outside the plant. c&d. - The new fuel does not have the radioactive fission product loading of spent fuel.

Reference TitleFacility Reference Number	Section	Page	Revision	L. O.
0300-000.00S-ABFUEL1-00	V.A	16		4.A

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: SOQL1132.

Question: SG Tube leak vs. rupture

Which one of the following conditions differentiates between a S/G tube leak, which is addressed in S2.OP-AB.SG-0001(Q) "STEAM GENERATOR TUBE LEAK", and a steam generator tube rupture (SGTR), which is addressed in the Emergency Procedures (EOPs)?

a. Affected S/G level is controlled at the programmed level, in automatic

b. RCS pressure is stable or rising with all PZR heaters energized and no load change in progress

c. PZR level can be maintained stable or rising

d. Affected S/G Blowdown Radiation Monitor (R-19) remains below the alarm setpoint

Answer c **Exam Level** S **Cognitive Level** Memory
Record Number: 103 **RO Number:** **SRO Number:** 78

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

037 Steam Generator (S/G) Tube Leak

2.4 Emergency Procedures / Plan

2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. 4.0 4.3

Explanation: c. - Correct. The capability to maintain Pzr level and VCT level are the parameters reviewed to determine if the unit should be tripped & SI actuated; a. - S/G level can be controlled in AUTO even though a relatively large difference may exist between SF and FF; b. While the two are related, the trip criteria is based on the capability to control inventory, not pressure; d. – The criteria is PZR/VCT level, not RMS alarms.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
STEAM GENERATOR TUBE LEAK	S2.OP-AB.SG-0001(Q)	Attach 1, CAS	1	13	
STEAM GENERATOR TUBE LEAK	0300-000.00S-ABSG01-01	III.C.2-4	11		4

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: SGTR basis for RCS temperature control

Given the following conditions for Unit 2:

- A S/G tube rupture has been identified on 23 S/G
- SI has been actuated
- The crew has completed the initial cooldown actions of 2-EOP-SGTR-1

Which one of the following conditions could occur if the RCS temperature established is higher than the target temperature stipulated by the EOP?

a. Pzr level will go solid (100%) during the subsequent RCS depressurization

b. Pressure of the ruptured S/G rises with resultant lifting of a S/G Safety Valve

c. Pressure of the non-ruptured S/Gs rises with resultant opening of the MS10's

d. RCS subcooling may be lost before RCS and ruptured S/G pressures are equalized

Answer d **Exam Level** B **Cognitive Level** Memory

Record Number: 104 **RO Number:** 81 **SRO Number:** 79

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

038 Steam Generator Tube Rupture (SGTR)

EK3. Knowledge of the reasons for the following responses as they apply to the SGTR:

EK3.06 Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures 4.2 4.5

Explanation: d. - Correct, Target temperature for the cooldown is selected based on ruptured S/G pressure, which is the target pressure for the RCS depressurization. The cooldown of the RCS establishes sufficient subcooling for ruptured S/G pressure. a. - This will not occur as a direct result of the cooldown; however, any delay in equalizing pressures delays SI termination, increasing the likelihood of reaching a solid Pzr condition. b&c. - Once the cooldown is completed, even if the temperature isn't as low as required, the operator is directed to maintain that temperature, preventing these conditions from occurring. If leakage to the ruptured S/G is NOT controlled (by pressure equalization), then the S/G pressure is expected to increase to that of the RCS once the S/G is filled. This resultant pressure should be below the MS10 setpoints (1045 psig).

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
STEAM GENERATOR TUBE RUPTURE	2-EOP-SGTR-1	15 Table D	3	23	
EOP-SGTR-1, STEAM GENERATOR TUBE RUPTURE	0300-000.00S-SGTR01-02	III.A.1, 3.c	20-21		3, 7

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Byron 9/98 NRC exam

Question LOCA/STEAM leak indications

Which one of the following parameters can be used shortly after event initiation to differentiate between a secondary steam leak and a small primary loss-of-coolant accident, both inside containment?

a. Pzr level.

b. RCS pressure.

c. T-cold temperatures.

d. ECCS injection flow rates.

Answer c **Exam Level** R **Cognitive Level** Comprehension
Record Number: 105 **RO Number:** 82 **SRO Number:**

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

040 Steam Line Rupture

AK1. Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture:

AK1.06 High-energy steam line break considerations 3.7 3.8

Explanation: c. Correct. T-cold is the discriminator, being lower on steam break due to cooldown. a. - Pressurizer level would be low on each break. b. - RCS pressure changes on either break. d. - ECCS flow would be high as RCS pressure changes for both accidents.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
EOP-LOSC-1, Loss Of Secondary Coolant	0300-000.00S-LOSC01-03	II.C.2	12		1

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Direct From Source

Question Source Comments: Braidwood 6/1999 NRC exam.

Question: Loss of vacuum action basis

Abnormal procedure S2.OP-AB.COND-0001, Loss of Condenser Vacuum, requires load reductions in accordance with Attachment 4, Condenser Back Pressure Limits, to stabilize condenser vacuum at or greater than the OPERATING LIMIT. However, it is possible to stabilize vacuum at that value but still be required to initiate a turbine trip based on the 5 MINUTE OPERATING LIMIT.

Which one of the following correctly describes the basis for the 5 MINUTE OPERATING LIMIT?

a. During low load-low vacuum conditions, extraction steam temperatures rise causing excessive thermal stresses in and possible failure of the feedwater heaters.

b. During low load-low vacuum conditions, the turbine-condenser "boot" overheats rapidly and may fail.

c. Low steam flow-low vacuum conditions can cause non-synchronous blade vibration (flutter) in the final stage of turbine blades and irreversible damage will occur.

d. At low loads, one or more turbine governor valves may be closed. If the closed governor valves are in alternate quadrants then "double-shocking" of the first stage turbine blades occurs and the turbine may fail.

Answer c **Exam Level** B **Cognitive Level** Memory
Record Number: 106 **RO Number:** 83 **SRO Number:** 80
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1
051: Loss of Condenser Vacuum
2.1 Conduct Of Operations
2.1.32 Ability to explain and apply all system limits and precautions. 3.4 3.8
Explanation: c. – Correct, per AB.COND-1 Basis Document; a. – Extraction steam temperatures will rise but the turbine limits are the most restrictive element of the procedure; b. – Exhaust temperatures rise but this is controlled by sprays; d. This is a valid problem for a turbine but governor valve failure is necessary for this to occur.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF CONDENSER VACUUM	S2.OP-AB.COND-0001(Q), Basis Document				
LOSS OF CONDENSER VACUUM	0300-000.00S-ABCOND-01, Obj.3				

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: FW line break evaluation

Unit 2 is at 100% power. A break has occurred in the main feedwater line just upstream of 23BF22, Feedwater Stop Valve.

Assuming no operator action, which one of the choices correctly completes the following statement?

The reactor will trip on low SG level and 23 SG will . . . .

a. completely blowdown. 23AF21 will remain closed on pressure override.

b. be maintained at steam header pressure. AFW flow will maintain a level.

c. be maintained at steam header pressure. AFW flow will be out the break.

d. depressurize, causing an AUTO SI when steam pressure drops to 100 psi less than the other SG's.

Answer b **Exam Level** B **Cognitive Level** Comprehension
Record Number: 107 **RO Number:** 84 **SRO Number:** 81
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
054 Loss of Main Feedwater (MFW)
AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW):

AK1.01 MFW line break depressurizes the S/G (similar to a steam line break) 4.1 4.3
Explanation: b. – Correct. 23BF22 is a MOV stop check valve and will terminate reverse flow. The AFW connection is downstream; a. – BF22 will close and terminate reverse flow from the SG and isolating AFW from the break; c. – It will not boil dry since AFW flow is maintained; d. – It will not de-pressurize because the remaining MS167's are open.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Lesson Plan: FEED and CONDENSATE	0300-000.00S-CN&FDW, Obj. 3, 4.s				
P&ID 205302, Sheet 3					

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: LOPA recovery

Given the following conditions for Unit 2:

- A loss of all AC power has occurred
- The SECs have been deenergized in accordance with 2-EOP-LOPA-1, "LOSS OF ALL AC POWER"
- RCP Seal Cooling isolation has been completed
- Buses 2B and 2C vital buses were just re-energized

Which one of the following sets of parameters/conditions is used to select the appropriate recovery procedure?

a. SI actuation status

b. RCS subcooling and Pzr level

c. The vital buses power sources

d. Pzr pressure and S/G pressures

Answer b **Exam Level** S **Cognitive Level** Memory
Record Number: 108 **RO Number:** **SRO Number:** 82
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1
055 Loss of Offsite and Onsite Power (Station Blackout)
2.4 Emergency Procedures / Plan
2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. 4.0 4.3

Explanation: Two criteria are used to determine whether the recovery actions include the need for SI or not. b. - Correct, The parameters for recovery without SI are RCS subcooling >0°F and Pzr level >11% (19% ADVERSE). These define the status of the RCS when AC power is restored; a. - SI is in title of recovery procedures but as "required" or "not required", not whether (or not) it has actuated; c. - The vital power source is only a concern for loading of DG; d. - Pzr pressure & S/G pressure provide for actuation of SI but are NOT directly addressed by the evaluation for recovery procedure selection.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF ALL AC POWER	2-EOP-LOPA-1	48	4	23	
EOP-LOPA-1, 2, 3; LOSS OF ALL AC POWER AND RECOVERY	0300-000.00S-LOPA00-02	4.3.48	53		9.B, 10

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Prairie Island 5/1999 NRC exam. Modified conditions & equipment for Salem.

Question: LOPA action basis

In accordance with 2-EOP-LOPA-1 "LOSS OF ALL AC POWER", which one of the following is the basis for maintaining S/G Narrow Range levels above 9% when the RCS is being cooled to 310°F Cold Leg temperature?

a. Ensures the capability to cooldown once AC power is restored

b. Ensure proper thermal stratification layer in the S/Gs in the event of a S/G tube rupture

c. Narrow Range level is the only indication of S/G inventory available after a loss of all AC power

d. Ensure sufficient heat transfer capability exists to remove heat from the RCS via natural circulation

Answer d **Exam Level** R **Cognitive Level** Memory
Record Number: 109 **RO Number:** 85 **SRO Number:**

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1
055 Loss of Offsite and Onsite Power (Station Blackout)
2.4 Emergency Procedures / Plan
2.4.7 Knowledge of event based EOP mitigation strategies. 3.1 3.8

Explanation: d. – Correct. Maintaining the U-tubes covered in at least one S/G will ensure that sufficient heat transfer capability exists to remove heat from the RCS via either natural circulation or reflux boiling after the RCS saturates. a. - Cooldown capability following power restoration is not a concern since equipment will be available to establish S/G feed flow. b. – A SGTR is no more or less likely during a LOPA. c. - Wide Range level indication is also available.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF ALL AC POWER	2-EOP-LOPA-1	37	3	23	
EOP-LOPA-1, 2, 3; LOSS OF ALL AC POWER AND RECOVERY	0300-000.00S-LOPA00-02	4.3.37.3	48		7, 8

Material Required for Examination
Question Source: NRC Exam Bank **Question Modification Method:** Editorially Modified
Question Source Comments: Braidwood 6/1999 NRC exam. Changed values for Salem specific.

Question: SEC ops

Which one of the following describes SEC operation if a Safety Injection actuation (Mode I) occurs while Blackout (Mode II) loading is in progress?

a. Mode II loading is completed. Operators must reset the SEC and start any Mode III loads not started in Mode II

b. Mode II loading stops, all loads are shed, and Mode III loading begins

c. Mode II loading stops, the SEC resets to Mode III and any ESF loads not already running are sequentially started.

d. Mode II loading is completed. The SEC will then shed any non-Mode III loads and start any Mode III loads not already running

Answer b **Exam Level** B **Cognitive Level** Memory

Record Number: 110 **RO Number:** 86 **SRO Number:** 83

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 3

056 Loss of Offsite Power

AA1. Ability to operate and/or monitor the following as they apply to the Loss of Offsite Power:

AA1.21 Reset of the ESF load sequencers 3.3 3.3

Explanation: When mode change is called for: 1) SEC stops where it is, 2) Sequencer currently in use is reset (by the SEC), 3) all loads previously started are shed, and 4) new loading sequence is initiated. Automatic mode changes are restricted and only certain changes can occur: 1) Mode I to Mode III, 2) Mode I to Mode IV, and 3) Mode II to Mode III. No other modes changes can occur until SEC is reset. b. – Correct. The mode change is allowed from MODE II (blackout) to MODE III (SI w/Blackout). The process is as described above. a. - This would be true if the Mode change was NOT allowed. c.&d. - These both incorrectly describe operation of the SEC in controlling sequencers.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
AFEGUARDS EQUIPMENT CONTROL SYSTEM	0300-000.00S-SEC000-01	IV.D.1	21		8

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Braidwood 6/1999 NRC exam.

Question: PZR heater control

Procedure S2.OP-AB.115-0002, "LOSS OF 2B 115V VITAL INSTRUMENT BUS", directs the installation of a jumper to energize pressurizer level comparator 2LC460D-C from an alternate source.

'hich one of the following describes the operation of the Pzr Backup Heaters during the interim?

Until the jumper is installed, the heaters...

a. cannot be energized

b. can be operated using the LOCAL control

c. can be operated using the normal console pushbuttons

d. can only be operated by transferring to the emergency power supplies

Answer b **Exam Level** B **Cognitive Level** Comprehension

Record Number: 111 **RO Number:** 87 **SRO Number:** 84

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

057 Loss of Vital AC Electrical Instrument Bus

AA1. Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus:

AA1.06 Manual control of components for which automatic control is lost 3.5 3.5

Explanation: b. - The affected level comparator provides for Letdown Isolation and Pressurizer Heater Interlock. The heaters must be energized in local control as long as the comparator remains de-energized. a. - The heaters can be operated in local control. c. - The interlock prevents manual Control Room operation (manual and auto). d. – Only is incorrect and the emergency power supply has a limited heater capacity.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF 2B 115V VITAL INSTRUMENT BUS	S2.OP-AB.115-0002(Q)	3.11 NOTE	3	8	
LOSS OF 2A, 2B,2C and 2D 115V VITAL INSTRUMENT BUS	0300-000.00S-AB1151-01	III.D.11	12		3, 4.b

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Modified choices

Question Source Comments: SOQL1488

Question: Loss of 125 VDC

Which one of the following describes how a loss of 125 VDC affects the Reactor Trip Breakers (RTBs)?

a. The breaker is not capable of opening on a signal to the shunt trip coil

b. The loss of voltage causes a shunt trip actuation and the breaker opens

c. The breaker is not capable of opening on a signal to the UV trip coil

d. The loss of voltage de-energizes the UV coil and the breaker opens

Answer a **Exam Level** B **Cognitive Level** Memory

Record Number: 112 **RO Number:** 88 **SRO Number:** 85

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

058 Loss of DC Power

AA2. Ability to determine and interpret the following as they apply to the Loss of DC Power:

AA2.03 DC loads lost; impact on ability to operate and monitor plant systems 3.5 3.9

Explanation: a. - Correct, The shunt trips are provided with external 125 VDC supplies and are energized to actuate. b. - The shunt trip energizes to actuate. c&d. - The undervoltage trip power to the reactor trip breakers is supplied from the respective SSPS cabinet 48 VDC and is not affected by the 125VDC.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
125VDC GROUND DETECTION	S2.OP-SO.125-0004(Q)	Attach 2,	5	7	
REACTOR PROTECTION SYSTEM	0300-000.00S-RXPROT-00	IV.C.4.c&e, C.7.a	28-29, 31		10, 11

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: CGroup NRC exam.

Question: WL permit conditions

Unit 2 is in Mode 5 and Unit 1 is at 100% power. A radioactive liquid release is in progress from 21 CVCS Monitor Tank, via 22 CCHX to Unit 1 Circulating Water. 12A Circulating Water Pump is OOS, all others are c rating.

Which one of the following correctly describes the required action if 11B Circulator trips?

a. The release can continue, the minimum dilution flow is still available

b. The release shall be terminated, immediately

c. The release may continue but only if RMS Channel R-18 is in service

d. The Unit 2 CRS shall direct the responsible NEO to reduce the release flow rate by 50%

Answer a **Exam Level** S **Cognitive Level** Comprehension
Record Number: 113 **RO Number:** **SRO Number:** 86

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1
059 Accidental Liquid Radwaste Release
AA2. Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release:
AA2.02 The permit for liquid radioactive-waste release 2.9 3.9

Explanation: a. – Correct. There is still one circulator running in the release path; b. – Both pumps in the dilution medium have not been lost; c. – R-18 has nothing to do with the dilution flow rate or path; d. – While the dilution flow has been reduced by 50%, the minimum dilution flow is still available.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
S2.OP-SO.WL-0001(Q), RELEASE OF RADIOACTIVE LIQUID WASTE					
S1.OP-AB.CW-0001(Q), Circulating Water Malfunction					
Lesson Plan: RADIOACTIVE LIQUID WASTE SYSTEM, 0300-000.00S-WASLIQ-01, Obj. 3.b, 12					

Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: SWS leak

Given the following conditions for Unit 2:

- Power is at 100%
- OHAs for 21 SW HDR PRESS LO (B-13) and 22 SW HDR PRESS LO (B-14) actuated
- Pressure on both Service Water (SW) header pressure indicators is lowering
- Actions of S2.OP-AB.SW-0001(Q), "Loss of Service Water Header Pressure" are being performed
- After closing 21&22SW17, SW Bay tie valves, and 21&22SW23 Nuclear Header tie valves, both SW header pressure meters still indicate a slow pressure reduction

Which one of the following components would the NEOs be directed to check for leaks and proper operation?

a. 21 CFCU Piping.

b. 2SW308, SW Bay 2 Pressure Control Valve.

c. Leakage into the 22SW valve and piping compartment.

d. Emergency Diesel Generator SW supply header piping.

Answer d **Exam Level** B **Cognitive Level** Comprehension
Record Number: 114 **RO Number:** 89 **SRO Number:** 87

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

062 Loss of Nuclear Service Water

AA2. Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water:

AA2.01 Location of a leak in the SWS 2.9 3.5

Explanation: d. – Correct. Once 21 & 22 headers have been separated, a continued decrease in pressure would be in common piping. This is either common piping to the DGs or 23 CFCU. a. - Only 23 CFCU has common SW piping. b. - Only one header affected. c. - Only one header affected and would be accompanied by other alarms

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF SERVICE WATER HEADER PRESSURE	S2.OP-AB.SW-0001(Q)	3.21	9	6	
LOSS OF SERVICE WATER HEADER PRESSURE	0300-000.00S-ABSW01-00	III.C	11		1, 4
Service Water Nuclear Area	205342	G-6	sh. 3	64	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: SOQL1871. Modified layout of premise

Question: Actions for waste release in progress

During a complete loss of control air, S2.OP-AB.CA-0001(Q) "LOSS OF CONTROL AIR" the operator is directed to stop any liquid or gaseous radioactive releases in progress by closing 2WL51, Liquid Release Stop, and 2WG41, Gas Decay Tanks Vent Isolation.

. /hich one of the following correctly describes the reason for closing these valves?

a. Ensure a positive closing signal while some air pressure is available

b. Without air pressure, neither valve is capable of closing on interlock from their respective RMS channel

c. This action terminates the open signal. Otherwise, these valves will re-open when air pressure is restored

d. Ensures a release is not continued while degrading air pressure may be causing a change in the dilution medium flow rate

Answer d **Exam Level** B **Cognitive Level** Memory
Record Number: 115 **RO Number:** 90 **SRO Number:** 88

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 3 **SRO Group:** 2

065 Loss of Instrument Air

AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air:

AK3.08 Actions contained in EOP for loss of instrument air 3.7 3.9

Explanation: d. – Correct. This step insures that, on a gradual depressurization, a release is not in progress when the dilution medium flowrate may be changing. a.&b. – These valves fail closed. c. – When these valves fail closed a closed limit switch terminates the open signal.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF CONTROL AIR	S2.OP-AB.CA-0001(Q)	3.23	4	5	
' OSS OF CONTROL AIR	0300-000.00S-ABCA01-01	III.D.23	12		2

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified (jkl)

Question Source Comments: SOQL1045

Question: CNMT fire response

Given the following conditions for Unit 2:

- Power is at 100%
- OHA, FIRE PROT FIRE (A-7) alarms
- 2RP5 is checked and indicates the following:
  - Zone 59, Air and Water Deluge, Containment El. 100 Panel 335 is flashing
  - Zone 74, Smoke and Fire Detector, Containment El. 100 Panel 335 is lit

Which one of the following describes the status of the fire protection system?

a. Fire protection water is being delivered via deluge valves

b. A single manual valve in the Mechanical Penetration Area must be opened to initiate fire protection water flow via the open deluge valves

c. The containment isolation valve must be opened from the control room to initiate fire protection water flow via the open deluge valves

d. The Panel 335-related deluge valves located in the Mechanical Penetration Area must be manually opened to initiate fire protection water flow

Answer c **Exam Level** B **Cognitive Level** Memory
Record Number: 116 **RO Number:** 91 **SRO Number:** 89

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

067 Plant Fire on Site

AA1. Ability to operate and/or monitor the following as they apply to the Plant Fire on Site:

AA1.06 Fire alarm 3.5 3.7

Explanation: c. - Correct, The operator is procedurally directed "if at any time" RP-5 fire indication for both Zones 59 and 74 on are received, open 2FP147, Fire Protection Containment Isolation. a. - 2FP147 must be opened before flow into CNMT is established. b.&d. - The valve required to be opened is remotely operated from the RP-5 Panel.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
FIRE PROTECTION SYSTEM	0300-000.00S-FIRPRO-00	V.B.5 & 7	67-68		8.c
OVERHEAD ANNUNCIATORS WINDOW A	S2.OP-AR.ZZ-0001(Q)	A-7, 3.2	17	24	

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: B Group NRC exam (Q76)

Question: CCW indications

Given the following conditions for Unit 2:

- A control room evacuation has occurred due to habitability concerns
- All immediate actions have been completed for S2.OP-AB.CR-0001(Q) "CONTROL ROOM EVACUATION"
- CCW system was aligned with 21 and 23 Pumps running and 22 Pump in AUTO
- After leaving the control room but prior manning the Hot Shutdown Panel (HSD), CCW header pressure dropped to 65 psig and then recovered.
- No actions were taken to alter CCW Pump status when control was established at the Hot Shutdown Panel

Which one of the following describes the CCW Pump indication status the operator would observe when re-establishing control in the control room?

a. The START backlight for only 21 and 23 Pumps will be lit.

b. The START backlight for 21and 23 Pumps will be lit. The START backlight for 22 Pump will be flashing and the audible group alarm will be sounding.

c. No CCW Pump indications will be observable until the respective HSD Panel switches are returned to REMOTE, then the START backlight for only 21 and 23 Pumps will be lit.

d. No CCW Pump indications will be observable until the respective HSD Panel switches are returned to REMOTE, then the START backlight for 21and 23 Pumps will light, the START backlight for 22 Pump will flash and the audible group alarm will sound.

Answer b **Exam Level** B **Cognitive Level** Comprehension
Record Number: 117 **RO Number:** 92 **SRO Number:** 90

er: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

068 Control Room Evacuation

AA1. Ability to operate and/or monitor the following as they apply to the Control Room Evacuation:

AA1.21 Transfer of controls from control room to shutdown panel or local control 3.9 4.1

Explanation: The AUTO start feature on low header pressure for CCW is blocked once the Control Switch on the HSD Panel is taken to LOCAL. Control room indications, and group alarm functions remain operable. b. – Correct. 22 CCW pump auto started which activated the backlight flash & group alarm. a. – This would be the indication if the pressure dropped after taking LOCAL control. c. – Incorrectly assumes that taking LOCAL control removes CR indication. d. – Incorrectly assumes that taking LOCAL control removes CR indication and that the AUTO start signal is retained.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
CONTROL ROOM EVACUATION	S2.OP-AB.CR-0001(Q)	Attach 6, 3.0; Attach 16, 6.0	1, 2-3	8	
CONTROL ROOM EVACUATION	0300-000.00S-ABCR01-00	III.D.16	17		1, 3.c
COMPONENT COOLING WATER	0300-000.00S-CCW000-02	V.A.1.d, f	33-34		8, 9

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: PTS RCP starting criteria

Given the following conditions for Unit 2:

- A LOCA has occurred
- A Core Cooling RED Path exists and 2-EOP-FRCC-1, "RESPONSE TO INADEQUATE CORE COOLING" is being implemented.
- Steam Generator depressurization was ineffective in restoring core cooling.
- All RCPs are stopped
- TEN CETs indicate temperatures above 1200°F

In order to provide core cooling, which one of the following conditions must be established prior to starting an RCP?

a. RVLIS Full Range level is greater than 39%.

b. Seal injection flow for the selected RCP is greater than 6 gpm.

c. Level in the S/G in the loop for the selected RCP is greater than 15% NR.

d. The RCP Oil Lift Pump on selected pump is running for greater than 2 minutes.

Answer c **Exam Level** B **Cognitive Level** Memory
Record Number: 118 **RO Number:** 93 **SRO Number:** 91

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

074 Inadequate Core Cooling

EK3. Knowledge of the reasons for the following responses as they apply to the Inadequate Core Cooling:

EK3.07 Starting up emergency feedwater and RCPs 4.0 4.4

Explanation: For the stated conditions the RCP is started without requiring "normal" RCP prerequisites. c. - Correct, RCPs should only be started in loops with S/G levels >15% ADVERSE to ensure heat removal by secondary. a. - RVLIS level is an evaluation criteria for exiting this procedure at Step 8, Core Cooling Check. During a LOCA, RVLIS or Pzr level requirements must be met to start RCP. b. - Normal RCP starting requirements are for >6 gpm seal injection flow. d. - The LO pump must be running to start the RCP. Normal requirements are for the Lift Oil pump to run for at least 2 minutes prior to RCP start. . The run time is NOT specified in FRCC and a procedure CAUTION statement indicates the RCP must be started when directed.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RESPONSE TO INADEQUATE CORE COOLING	2-EOP-FRCC-1	25	3	21	
EOP-FRCC-1, 2, and 3 ACCIDENT MITIGATION STRATEGY	0300-000.00S-FRCC00-01	II.C.6, III.B.25	22, 50		3.a,7

Material Required for Examination

Question Source: Other Facility **Question Modification Method:** Significantly Modified

Question Source Comments: Prairie Island exam bank

Question: RCS temperature reduction basis

Given the following conditions for Unit 2:

- R31, Letdown Line-Failed Fuel Process Rad Monitor, indication is rising
- S2.OP-AB.RC-0002(Q), "HIGH ACTIVITY IN REACTOR COOLANT" was entered
- High RCS activity is confirmed
- As directed by the CONTINUOUS ACTION SUMMARY, the CRS directs the Unit to be shutdown and RCS temperature reduced to 500°F.

Which one of the below identifies the bases for reducing Tave below this value?

a. Lowers the expected peak containment pressure in the event of a LOCA

b. Ensures S/G pressures remain below the lift setpoint for the MS10s in the event of a SGTR

c. This lowers CVCS letdown temperature to increase the effectiveness of the demineralizers in removing activated corrosion products

d. Reduces migration of radioactive nuclides through existing cracks or breaks in the clad by lowering clad tensile stresses

Answer b **Exam Level** B **Cognitive Level** Memory
Record Number: 119 **RO Number:** 94 **SRO Number:** 92

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

076 High Reactor Coolant Activity

AK3. Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity:

AK3.06 Actions contained in EOP for high reactor coolant activity 3.2 3.8

Explanation: b. - Correct, Reducing Tave to less than 500°F prevents an activity release should a steam generator tube rupture occur, since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves (MS10s). a. - Action will reduce heat input to CNMT reducing the peak pressure, but the CNMT is a pressure boundary in the event of a break. c. - CVCS letdown rates are increased in the procedure but activated corrosion product removal is via filtering rather than ion exchange and is unaffected by temperature. d. - Tensile stress on cladding may be reduced this does not necessarily reduce the migration of radioactive material through existing cladding flaws.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
Salem Unit 2 Technical Specification Bases		3/4.4.9	B 3/4 4-6	112	
HIGH ACTIVITY IN REACTOR COOLANT SYSTEM	0300-000.00S-ABRC02-00	C.2	9		3

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Combination of SOQL1885 & SOQL0953. Changed layout of premise.

Question: SEC response during SI termination

The following conditions exist on Unit 2:

- An electronic failure and technician error caused an inadvertent SI
- The crew has transitioned to TRIP-3, SI Termination
- SI and Phase A are reset
- 2B and 2C SEC are reset
- 2A SEC failed to reset

Which one of the following correctly describes the expected response of the 4KV vital buses if a loss of off-site power occurs before 2A SEC can be de-energized?

a. Blackout loading occurs on all buses

b. Accident+Blackout loading occurs on all buses

c. Blackout loading occurs on 2B and 2C buses. Accident+Blackout loading occurs on 2A Bus

d. Blackout loading occurs on 2B and 2C buses. 2A bus is de-energized.

Answer c **Exam Level** B **Cognitive Level** Comprehension
Record Number: 120 **RO Number:** 95 **SRO Number:** 93
Tier: Emergency and Abnormal Plant Evolutions **RO Group:**2 **SRO Group:**1
E02, SI Termination
EK3. Knowledge of the reasons for the following responses as they apply to the SI Termination:
EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency 3.9/3.9

Explanation: The SECs normally respond to loss of offsite power in MODE II. When 2A SEC failed to RESET, the operator would block and attempt to reset the SEC. If SI is NOT present when a block switch is taken to "Block", both trains of SI will be blocked to the respective SEC Cabinet as long as the switches are held in the "BLOCK" position. As soon as the switches are released (returned to center) "Block" clears and circuitry return to normal. c. – Correct. 2B and 2C SEC see a Blackout. 2A SEC sees a SI followed by a Blackout; a. – 2A is still actuated in the SI Mode; b. – 2B and 2C SEC's are reset, setup for the next actuation; d. – This would be the case if 2A SEC was de-energized prior to the loss of power.

Reference Title **Facility Reference Number/Section/Page/L.O.**
SAFETY INJECTION TERMINATION, 2-EOP-TRIP-3/Steps 1-3/1/24/NA
SAFEGUARDS EQUIPMENT CONTROL, 0300-000.00S-SEC000-01/IV.D.4, F.4/22,24,25/8,9,13
EOP-TRIP-3, SAFETY INJECTION TERMINATION, 0300-000.00S-TRP003-01/3.3.3/13/3,10.A.5
Material Required for Examination
Question Source: New (jkl) **Question Modification Method:**
Question Source Comments:

Question: Evaluation of conditions for stopping SI Pumps

Given the following conditions for Unit 2:

- A LOCA has occurred
- The crew is performing actions of 2-EOP-LOCA-2 "POST LOCA COOLDOWN AND DEPRESSURIZATION"
- After stopping ONE Charging pump the following parameters exist:
  - RCS pressure is 1025 psig stable
  - Pzr level is 28%
  - RCS temperature (CETs) are reading 480°F
  - Containment pressure is 4.4 psig

Which one of the following describes the action to be taken for these conditions?

a. SI should be manually re-initiated.

b. Re-start the Charging pump based on subcooling less than 38°F.

c. Stopping of ONE SI Pump should be evaluated using NORMAL values for subcooling and Pzr level.

d. Stopping of ONE SI Pump should be evaluated using ADVERSE values for subcooling and Pzr level.

Answer d **Exam Level** B **Cognitive Level** Application
Record Number: 121 **RO Number:** 96 **SRO Number:** 94
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
E03 LOCA Cooldown and Depressurization
2.4 Emergency Procedures / Plan
2.4.14 Knowledge of general guidelines for EOP flowchart use. 3.0 3.9

Explanation: Adverse CNMT conditions are 1) Containment pressure greater than 4 psig, or 2) Containment radiation greater than 1E5 R/hr. d. – Correct. With the Charging pump stopped, the next step is to evaluate stopping one of the running SI pumps. ADVERSE & NORMAL sets of numbers are provided. Since CNMT PRESS is ADVERSE, use ADVERSE values for both parameters. a. - SI actuation criteria, which directs operation of ECCS pumps as necessary (NOT SI re-actuation), is covered in the CAS based on subcooling of 0°F or Pzr level of 11% (19%). b. - 38°F is a NORMAL value used in the evaluation of stopping the first charging pump. Once that is satisfied, pump is NOT restarted unless the CAS values are exceeded. c. – Use of NORMAL values is incorrect and may result in stopping SI pump when conditions do not warrant.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
POST LOCA COOLDOWN AND DEPRESSURIZATION	2-EOP-LOCA-2	22	3	22	
EOP-TRIP-1, REACTOR TRIP OR SAFETY INJECTION AND INTRODUCTION TO THE USE OF EOPs	0300-000.00S-TRP001-01	2.14	27		1.G
EOP-LOCA-2, POST LOCA COOLDOWN AND DEPRESSURIZATION	0300-000.00S-LOCA02-01	III.C.22	32		7

Material Required for Examination EOP-CFST-1, Table A and B
Question Source: Previous 2 NRC Exams **Question Modification Method:** Significantly Modified
Question Source Comments: DGroup NRC 2/1999. Modified conditions such that correct answer changed.

Question: Eval of LOCA location

Given the following conditions for Unit 2:

- Unit is in MODE 4 cooling down on RHR
- RCS Temperature - 340°F
- RCS pressure - 300 psig lowering
- PZR level - 22% lowering
- CNMT pressure - 0.2 psig
- 2R16, Plant Vent Effluent Monitor is in ALERT
- R41B, Plant Vent Iodine Monitor radiation levels are trending higher
- S/G levels stable at - 42% (21); 40% (22); 43% (23); 40% (24)
- S/G pressures stable at - 100 psig (21), 95 psig (22), 100 psig (23), 98 psig (24)

Which one of the following events is taking place?

a. POPS actuated and one PORV is stuck open.

b. A LOCA has occurred in the area of the Regenerative Heat Exchanger

c. A LOCA has occurred on the suction of the RHR pump.

d. Letdown line pressure control valve 2CV18 has failed open.

Answer c **Exam Level** B **Cognitive Level** Comprehension
Record Number: 122 **RO Number:** 97 **SRO Number:** 95

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 1

E04 LOCA Outside Containment

EK1. Knowledge of the operational implications of the following concepts as they apply to the LOCA Outside Containment:

EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the LOCA Outside Containment. 3.5 3.9

Explanation: c. – Correct. During any LOCA, RCS pressure & inventory will fall. Rising indication on the Aux Bldg radiation monitors is indicative of the LOCA outside containment (RHR pump suction). a. - When POPs operation results in a PORV opening, RCS pressure will drop, but PZR level should rise due to voiding in the reactor vessel head. Initial conditions do NOT support POPs auto operation. b. – This leak is inside containment. d. - Failure of CV18 open results in increased diversion of RHR flow (RCS inventory) to letdown but the flow is contained and would not lead to rising Aux. Bldg. RMS indications.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF REACTOR COOLANT	2-EOP-LOCA-1	16	2	24	
EOP-LOCA-6 LOCA OUTSIDE CONTAINMENT	0300-000.00S-LOCA06-01	3.1	7		3
EOP-LOCA-01, LOSS OF REACTOR COOLANT AND LOSS OF COOLANT ACCIDENT ANALYSIS	0300-000.00S-LOCA01-01				9

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: Braidwood 1997 NRC exam

Question: Core cooling evaluation

Which one of the following is the reason that 2-EOP-FRHS-1 "RESPONSE TO LOSS OF SECONDARY HEAT SINK" directs transition to the procedure and step in effect if RCS pressure is less than all intact or ruptured S/G pressures?

a. Feeding S/Gs under these conditions may halt natural circulation core cooling

b. Core decay heat is being removed by means other than the secondary heat sink

c. Under these conditions, initiating feed flow can cause a substantial reverse delta-P that may result in a S/G tube rupture

d. RCS subcooling must be restored prior to the initiation of feed and bleed

Answer b **Exam Level** B **Cognitive Level** Memory
Record Number: 123 **RO Number:** 98 **SRO Number:** 96

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2

E05 Loss of Secondary Heat Sink

EK2. Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following:

EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. 3.9 4.2

Explanation: b. - Correct, The S/Gs are no longer functioning as a heat sink. Core heat is being removed by break flow. a. -Control of feed is a concern for natural circulation flow. However, in this case, natural circ will not be affected by feeding. c. - A SGTR is not the concern in this situation. d. – There is no means of restoring subcooling until a heat sink is recovered.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RESPONSE TO LOSS OF SECONDARY HEAT SINK	2-EOP-FRHS-1	3	1	24	
OP-FRHS-1, 2, 3, 4, and 5 HEAT SINK FUNCTIONAL RESTORATION	0300-000.00S-FRHS00-03 3	5.2.3	21-22		10

Material Required for Examination

Question Source: NRC Exam Bank **Question Modification Method:** Editorially Modified

Question Source Comments: Braidwood 6/1999 NRC exam.

Question: Identification of heat removal process

The following conditions exist on Unit 2:

- A major steam leak occurred and has now been isolated
- The crew has entered 2-EOP-FRTS-1 "RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIONS"
- SI has been reset
- All RCPs are stopped
- Conditions required to support a RCP start are met

Which one of the choices correctly completes the following statement regarding operation of a RCP?

Under the current conditions a RCP will...

a. NOT be started to avoid causing additional thermal stresses in stagnant loops.

b. NOT be started to avoid the pressure surge that will aggravate the PTS condition.

c. be started to establish normal spray for a controlled depressurization.

d. be started to provide mixing of the ECCS injection flow, decreasing the likelihood of PTS.

Answer d **Exam Level** S **Cognitive Level** Comprehension
Record Number: 124 **RO Number:** **SRO Number:** 97

Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

E08 Pressurized Thermal Shock

EA2. Ability to determine and interpret the following as they apply to the Pressurized Thermal Shock:

EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations. 3.4 4.2

Explanation: d. – Correct, per FRTS Basis Document. In order to mix the relatively cold ECCS flow with the warm reactor coolant and thereby decrease the likelihood of a PTS condition, an RCP re-start is attempted. a.&b. - Starting a RCP was shown not to result in any further flaw propagation and loss of vessel integrity. c. – The procedure just calls for starting a RCP; not start 21 or 23 RCP. There are alternative paths for depressurization if a RCP start conditions are not met.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITIONS	2-EOP-FRTS-1	14.4	3	24	
EOP-FRTS-1 AND 2, RESPONSE TO PRESSURIZED THERMAL SHOCK CONDITIONS	0300-000.00S-FRTS00-01	3.2.14.3	29		2, 9

Material Required for Examination

Question Source: New **Question Modification Method:**

Question Source Comments:

Question: Natural circulation cooldown comparison

Which one of the following correctly describes a major philosophical difference between EOP-TRIP-5 (TRIP-5), Natural Circulation Rapid Cooldown Without RVLIS, and EOP-TRIP-6 (TRIP-6), Natural Circulation Rapid Cooldown With RVLIS?

a. In TRIP-5 steps are taken to prevent bubble formation in the reactor vessel head. In TRIP-6 steps are taken if a bubble forms in the reactor vessel head.

b. In TRIP-5 the cooldown rate is limited to 50°F/hr. In TRIP-6 the cooldown rate is 100°F/hr.

c. In TRIP-5 the cooldown and depressurization is performed in discrete steps. In TRIP-6 the cooldown is continuous and steps are taken if RVLIS indicates excessive reactor vessel head bubble formation.

d. TRIP-6 permits use of a PZR PORV for depressurization while TRIP-5 does NOT. Reactor vessel head bubble formation cannot be accurately inferred from indicated pressurizer level while a PORV is open.

Answer c **Exam Level** B **Cognitive Level** Comprehension
Record Number: 125 **RO Number:** 99 **SRO Number:** 98
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1
E10 Natural Circulation with Steam Void in Vessel with/without RVLIS
EK1. Knowledge of the operational implications of the following concepts as they apply to the Natural Circulation with Steam Void in Vessel with/without RVLIS:
EK1.2 Normal, abnormal and emergency operating procedures associated with Natural Circulation with Steam 3.4 3.6
Explanation: c. – Correct. In TRIP-5, the cooldown is terminated at plateaus and depressurization and corrective action for bubble formation is accomplished. In TRIP-6, the cooldown is continuous with corrective actions based on RVLIS indication; a – Reactor vessel head bubble formation is expected in both procedures; b. – InTRIP-6, the cooldown rate is 50°F/hr for only the first leg; d. – Aux. Spray is preferred but use of a PORV is permitted in both procedures.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
NATURAL CIRCULATION RAPID COOLDOWN WITHOUT RVLIS	2-EOP-TRIP-5	7, 10, 19, 26	1,2,3	21	
NATURAL CIRCULATION RAPID COOLDOWN WITH RVLIS	2-EOP-TRIP-6	7	1	21	
EOP-TRIP 4, 5, 6; NATURAL CIRCULATION COOLDOWN	0300-000.00S-TRP004-01	5.3.7, 5.3.8, 7.3.7	44, 46, 73		5, 6

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments:

Question: Procedure transition when elec. power is restored

Given the following conditions for Unit 2:

- A LOCA has occurred
- While performing actions of 2-EOP-LOCA 1"LOSS OF REACTOR COOLANT", 22 RHR pump motor seizes and power is lost to the 21 RHR pump
- The crew enters 2-EOP-LOCA-5, "LOSS OF EMERGENCY RECIRCULATION"
- A cooldown as been initiated as directed in 2-EOP-LOCA-5
- During the cooldown, the crew restores power to 21 RHR pump.

Based on current plant conditions, which one of the following represents the correct mitigation strategy?

a. Return to 2-EOP-LOCA-1 and continue recovery actions with the step previously in effect

b. Start the RHR pump to provide makeup flow to the RCS and continue recovery via 2-EOP-LOCA-5

c. Continue with the cooldown and start the RHR pump when directed in 2-EOP-LOCA-5

d. Immediately start the RHR pump and transition to 2-EOP-LOCA-3, Transfer to Cold Leg Recirculation

Answer a **Exam Level** S **Cognitive Level** Memory
Record Number: 126 **RO Number:** **SRO Number:** 99
Tier: Emergency and Abnormal Plant Evolutions **RO Group:** 2 **SRO Group:** 2
E11 Loss of Emergency Coolant Recirculation
2.4 Emergency Procedures / Plan
2.4.8 Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-based EOPs. 3.0 3.7

Explanation: a. – Correct. Continuous action step states that if any train of emergency recirculation capability is restored then the crew should return to the procedure and step in effect. This is consistent with the organization of the EOPs. b. - The RHR pump should be started but that direction is provided in LOCA-1. c. - Continuation of the cooldown in LOCA-5 is not required. The purpose of the procedure is mitigation and recovery of recirculation capability. d. – A transfer to LOCA-3 is not initiated until RWST level is evaluated in LOCA-1.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
LOSS OF EMERGENCY RECIRCULATION	2-EOP-LOCA-5	5.1	1	22	
EOP-LOCA-5, LOSS OF EMERGENCY RECIRCULATION	0300-000.00S-LOCA05-01	2.1.1, 3.3.5.3	7, 15		1, 7.B

Material Required for Examination

Question Source: Facility Exam Bank **Question Modification Method:** Significantly Modified

Question Source Comments: SOQL0117

Question: AFW flow to multiple faulted S/Gs

Given the following conditions for Unit 2:

- A steamline break occurred on the 22 S/G 25 minutes ago
- All MSIVs failed to close
- RCS pressure is 1050 psig
- RCS temperature (SPDS) average 370°F
- RCS Tcolds: 310°F (21), 280°F (22), 320°F (23), 320°F (24)
- CNMT pressure has stabilized at 8 psig
- S/G WR levels - 50% (21); 8% (22); 48% (23); 55% (24)

The operating crew is performing the Safeguards Reset Actions in accordance with 2-EOP-LOSC-2, "MULTIPLE STEAM GENERATOR DEPRESSURIZATION" when the STA reports a PURPLE Path for Thermal Shock Status Tree.

Which one of the following correctly describes the AFW flow strategy for both procedures?

a. Maintain flow at 1.0E04 lb/hr to each S/G, to limit cooldown and prevent S/G tube dryout.

b. Maintain total flow >22E04 lb/hr but only feed 21, 23, 24 S/G's, to maintain an adequate heat sink but limit cooldown

c. Maintain flow to 21, 23, 24 SG at 1.0E04 lb/hr each, to limit cooldown and prevent SG tube dryout

d. Maintain total flow at 22E04 lb/hr, feeding all S/Gs to maintain an adequate heat sink but limit cooldown

Answer a **Exam Level** B **Cognitive Level** Comprehension
Record Number: 127 **RO Number:** 100 **SRO Number:** 100

:r: Emergency and Abnormal Plant Evolutions **RO Group:** 1 **SRO Group:** 1

E12 Uncontrolled Depressurization of all Steam Generators

EK3. Knowledge of the reasons for the following responses as they apply to the Uncontrolled Depressurization of all Steam Generators:

EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations. 3.5 3.7

Explanation: a. Correct, If all S/G are faulted or if any faulted S/G is necessary for RCS temperature control, feed flow is controlled at a minimum measurable value to minimize the effects of the RCS cooldown and prevent steam generator tube dryout. b. – Any SG <9% NR is required to be fed at 1E04; c. - If only one faulted S/G, the normal action is to isolate the faulted S/G and maintain 22E04 to the other S/Gs; d. – 22E04 is the minimum AFW flow value in most other procedures.

Reference Title	Facility Reference Number	Section	Page	Revision	L. O.
MULTIPLE STEAM GENERATOR DEPRESSURIZATION	2-EOP-LOSC-2	4	1	24	
EOP-FRTS-1 AND 2, RESPONSE TO PRESSURIZED THERMAL SHOCK CONDITIONS	0300-000.00S-FRTS00-01	3.2.5	19		3.a
EOP-LOSC-2, MULTIPLE STEAM GENERATOR DEPRESSURIZATION	0300-000.00S-LOCS02-02	4	16		4

Material Required for Examination

Question Source: New (jkl) **Question Modification Method:**

Question Source Comments: