ES-401	PWR Examination Outline	FORM ES-401-2

Facility Name:Arkansas Nuclear One Unit 2 Date of Exam:2/21/2014

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

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

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the “Tier Totals” in each K/A category shall not be less than two).

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.

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

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

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

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

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

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

ES-401							PWR Examination Outline		Form ES-401-2
Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO)									
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	IR	#
000007 Reactor Trip / 1									1
CE/E02 Reactor Trip Recovery / 1	03						Annunciators and conditions indicating signals, and remedial actions associated with the (Reactor Trip Recovery).	3.0	
000008 Pressurizer Vapor Space Accident / 3					30		Inadequate core cooling	4.3	1
000009 Small Break LOCA / 3									0
000011 Large Break LOCA / 3						01.20	Ability to interpret and execute procedure steps.	4.6	1
000015 RCP Malfunctions / 4 000017 RCP Malfunctions (Loss of RC Flow) / 4	01						Natural circulation in a nuclear reactor power plant	4.4	1
000022 Loss of Rx Coolant Makeup / 2	03						Relationship between charging flow and PZR level	3.0	1
000025 Loss of RHR System / 4				03			LPI pumps	3.4	1
000026 Loss of Component Cooling Water / 8			02				The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS	3.6	1
000027 Pressurizer Pressure Control System Malfunction / 3		03					Controllers and positioners	2.6	1
000029 ATWS / 1				12			M/G set power supply and reactor trip breakers	4.1	1
000038 Steam Gen. Tube Rupture / 3				44			Level operating limits for S/Gs	3.4	1
000040 Steam Line Rupture / 4									1
CE/E05 Excessive Steam Demand / 4					02		Adherence to appropriate procedures and operation within the limitations in the Facility's license and amendments.	3.4	
000054 Loss of Main Feedwater / 4									1
CE/E06 Loss of Feedwater / 4		01					Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.3	
000055 Station Blackout / 6			02				Actions contained in EOP for loss of offsite and onsite power	4.3	1
000056 Loss of Off-site Power / 6					47		Proper operation of the ED/G load sequencer	3.8	1
000057 Loss of Vital AC Inst. Bus / 6					04		ESF system panel alarm annunciators and channel status indicators	3.7	1
000058 Loss of DC Power / 6			02				Actions contained in EOP for loss of dc power	4.0	1
000062 Loss of Nuclear Svc Water / 4									0
000065 Loss of Instrument Air / 8			03				Knowing effects on plant operation of isolating certain equipment from instrument air	2.9	1
000077 Generator Voltage and Electric Grid Disturbances / 6	03						Under-excitation	3.3	1
K/A Category Totals:	4	2	4	3	4	1	Group Point Total:		18

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

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000001 Continuous Rod Withdrawal / 1									0
000003 Dropped Control Rod / 1				05			Reactor power - turbine power	4.1	1
000005 Inoperable/Stuck Control Rod / 1									0
000024 Emergency Boration / 1		04					Pumps	2.6	1
000028 Pressurizer Level Malfunction / 2									0
000032 Loss of Source Range NI / 7			01				Startup termination on source-range loss	3.2	1
000033 Loss of Intermediate Range NI / 7									0
000036 Fuel Handling Accident / 8	02						SDM	3.4	1
000037 Steam Generator Tube Leak / 3					11		When to isolate one or more S/Gs	3.8	1
000051 Loss of Condenser Vacuum / 4									0
000059 Accidental Liquid RadWaste Rel. / 9									0
000060 Accidental Gaseous Radwaste Rel. / 9									0
000061 ARM System Alarms / 7									0
000067 Plant Fire On-site / 9 8						04. 31	Knowledge of annunciator alarms, indications, or response procedures.	4.2	1
000068 Control Room Evac. / 8			18				Actions contained in EOP for control room evacuation emergency task	4.2	1
000069 Loss of CTMT Integrity / 5									0
000074 Inad. Core Cooling / 4				27			ECCS valve control switches and indicators	4.2	1
000076 High Reactor Coolant Activity / 9									0
CE/A13 Natural Circ. / 4									0
CE/A11 RCS Overcooling / 4									0
CE/A16 Excess RCS Leakage / 2									0
CE/E09 Functional Recovery	02						Normal, abnormal and emergency operating procedures associated with (Functional Recovery).	3.2	1
									0
									0
									0
									0
									0
									0
									0
									0
									0
K/A Category Totals:	2	1	2	2	1	1	Group Point Total:		9

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

Plant Systems - Tier 2/Group 1 (RO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
003 Reactor Coolant Pump			02									S/G	3.5	1
004 Chemical and Volume Control					30							Relationship between temperature and pressure in CVCS components during solid plant operation	3.8	1
005 Residual Heat Removal		01					01					RHR pumps; Heatup/cooldown rates	3; 3.5	2
006 Emergency Core Cooling				09				13				Valve positioning on safety injection signal; Inadvertent SIS actuation	3.9; 3.9	2
007 Pressurizer Relief/Quench Tank							02					Maintaining quench tank pressure	2.7	1
008 Component Cooling Water	05										02.44	Sources of makeup water; Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect	3; 4.2	2
010 Pressurizer Pressure Control						01						Pressure detection systems	2.7	1
012 Reactor Protection										03		Channel blocks and bypasses	3.6	1
013 Engineered Safety Features Actuation			01									Fuel	4.4	1
022 Containment Cooling				04								Cooling of control rod drive motors	2.8	1
025 Ice Condenser														0
026 Containment Spray	02			08								Cooling water; Automatic swapover to containment sump suction for recirculation phase after LOCA (RWST low-low level alarm)	4.1; 4.1	2
039 Main and Reheat Steam								05		04		Increasing steam demand, its relationship to increases in reactor power; Emergency feedwater pump turbines	3.3; 3.8	2
059 Main Feedwater			03					07				S/Gs; Tripping of MFW pump turbine	3.5; 3	2
061 Auxiliary/Emergency Feedwater					02							Decay heat sources and magnitude	3.2	1
062 AC Electrical Distribution		01						12				Major system loads; Restoration of power to a system with a fault on it	3.3; 3.2	2
063 DC Electrical Distribution				02								Breaker interlocks, permissives, bypasses and cross-ties	2.9	1
064 Emergency Diesel Generator						08						Fuel oil storage tanks	3.2	1
073 Process Radiation Monitoring	01											Those systems served by PRMs	3.6	1
076 Service Water									02			Emergency heat loads	3.7	1
078 Instrument Air											01.30	Ability to locate and operate components, including local controls.	4.4	1
103 Containment							01					Containment pressure, temperature, and humidity	3.7	1
K/A Category Totals:	3	2	3	4	2	2	3	4	1	2	2	Group Point Total:		28

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

Plant Systems - Tier 2/Group 2 (RO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
001 Control Rod Drive														0
002 Reactor Coolant											04.11	Knowledge of abnormal condition procedures.	4.0	1
011 Pressurizer Level Control		02										PZR heaters	3.1	1
014 Rod Position Indication														0
015 Nuclear Instrumentation					19							Heat balance	2.9	1
016 Non-nuclear Instrumentation			02									PZR LCS	3.4	1
017 In-core Temperature Monitor														0
027 Containment Iodine Removal														0
028 Hydrogen Recombiner and Purge Control								03				The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment damage in containment	3.4	1
029 Containment Purge				03								Automatic purge isolation	3.2	1
033 Spent Fuel Pool Cooling														0
034 Fuel Handling Equipment										01		Radiation levels	3.3	1
035 Steam Generator														0
041 Steam Dump/Turbine Bypass Control									01			RCS T-ave. meter (cooldown rate)	3.2	1
045 Main Turbine Generator	06											RCS, during steam valve test	2.6	1
055 Condenser Air Removal														0
056 Condensate														0
068 Liquid Radwaste														0
071 Waste Gas Disposal														0
072 Area Radiation Monitoring														0
075 Circulating Water														0
079 Station Air														0
086 Fire Protection										05		Deluge valves	3.0	1
K/A Category Totals:	1	1	1	1	1	0	0	1	1	2	1	Group Point Total:		10

ES-401							PWR Examination Outline		Form ES-401-2
							Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (SRO)		
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000007 Reactor Trip / 1									0
CE/E02 Reactor Trip Recovery / 1									
000008 Pressurizer Vapor Space Accident / 3									0
000009 Small Break LOCA / 3					06		Whether PZR water inventory loss is imminent	4.3	1
000011 Large Break LOCA / 3									0
000015 RCP Malfunctions / 4 000017 RCP Malfunctions (Loss of RC Flow) / 4									0
000022 Loss of Rx Coolant Makeup / 2									0
000025 Loss of RHR System / 4									0
000026 Loss of Component Cooling Water / 8									0
000027 Pressurizer Pressure Control System Malfunction / 3									0
000029 ATWS / 1									0
000038 Steam Gen. Tube Rupture / 3									0
000040 Steam Line Rupture / 4						04.02	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	4.6	1
CE/E05 Excessive Steam Demand / 4									
000054 Loss of Main Feedwater / 4									0
CE/E06 Loss of Feedwater / 4									
000055 Station Blackout / 6					01		Existing valve positioning on a loss of instrument air system	3.7	1
000056 Loss of Off-site Power / 6									0
000057 Loss of Vital AC Inst. Bus / 6									0
000058 Loss of DC Power / 6					03		DC loads lost; impact on to operate and monitor plant systems	3.9	1
000062 Loss of Nuclear Svc Water / 4						01.32	Ability to explain and apply system limits and precautions.	4.0	1
000065 Loss of Instrument Air / 8									0
000077 Generator Voltage and Electric Grid Disturbances / 6						02.36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	4.2	1
K/A Category Totals:	0	0	0	0	3	3	Group Point Total:		6

ES-401							PWR Examination Outline		Form ES-401-2
Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (SRO)									
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000001 Continuous Rod Withdrawal / 1									0
000003 Dropped Control Rod / 1									0
000005 Inoperable/Stuck Control Rod / 1						02.22	Knowledge of limiting conditions for operations and safety limits.	4.7	1
000024 Emergency Boration / 1					01		Whether boron flow and/or MOVs are malfunctioning, from plant conditions	4.1	1
000028 Pressurizer Level Malfunction / 2									0
000032 Loss of Source Range NI / 7									0
000033 Loss of Intermediate Range NI / 7									0
000036 Fuel Handling Accident / 8									0
000037 Steam Generator Tube Leak / 3									0
000051 Loss of Condenser Vacuum / 4									0
000059 Accidental Liquid RadWaste Rel. / 9									0
000060 Accidental Gaseous Radwaste Rel. / 9									0
000061 ARM System Alarms / 7									0
000067 Plant Fire On-site / 9 8									0
000068 Control Room Evac. / 8									0
000069 Loss of CTMT Integrity / 5					02		Verification of automatic and manual means of restoring integrity	4.4	1
000074 Inad. Core Cooling / 4									0
000076 High Reactor Coolant Activity / 9									0
CE/A13 Natural Circ. / 4									0
CE/A11 RCS Overcooling / 4					01		Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	3.3	1
CE/A16 Excess RCS Leakage / 2									0
CE/E09 Functional Recovery									0
K/A Category Totals:	0	0	0	0	3	1	Group Point Total:		4

ES-401	PWR Examination Outline Plant Systems - Tier 2/Group 1 (SRO)												Form ES-401-2	
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
003 Reactor Coolant Pump														0
004 Chemical and Volume Control														0
005 Residual Heat Removal														0
006 Emergency Core Cooling														0
007 Pressurizer Relief/Quench Tank														0
008 Component Cooling Water														0
010 Pressurizer Pressure Control														0
012 Reactor Protection								03				Incorrect channel bypassing	3.7	1
013 Engineered Safety Features Actuation											02.22	Knowledge of limiting conditions for operations and safety limits.	4.7	1
022 Containment Cooling														0
025 Ice Condenser														0
026 Containment Spray											01.20	Ability to interpret and execute procedure steps.	4.6	1
039 Main and Reheat Steam														0
059 Main Feedwater														0
061 Auxiliary/Emergency Feedwater								04				pump failure or improper operation	3.8	1
062 AC Electrical Distribution														0
063 DC Electrical Distribution														0
064 Emergency Diesel Generator								01				Failure modes of water, oil, and air valves	3.3	1
073 Process Radiation Monitoring														0
076 Service Water														0
078 Instrument Air														0
103 Containment														0
K/A Category Totals:	0	0	0	0	0	0	0	3	0	0	2	Group Point Total:		5

ES-401												PWR Examination Outline Plant Systems - Tier 2/Group 2 (SRO)		Form ES-401-2
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
001 Control Rod Drive								18				Incorrect rod stepping sequence	3.8	1
002 Reactor Coolant														0
011 Pressurizer Level Control														0
014 Rod Position Indication														0
015 Nuclear Instrumentation														0
016 Non-nuclear Instrumentation											01.07	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.7	1
017 In-core Temperature Monitor														0
027 Containment Iodine Removal														0
028 Hydrogen Recombiner and Purge Control														0
029 Containment Purge														0
033 Spent Fuel Pool Cooling														0
034 Fuel Handling Equipment				01								Fuel protection from binding and dropping	3.4	1
035 Steam Generator														0
041 Steam Dump/Turbine Bypass Control														0
045 Main Turbine Generator														0
055 Condenser Air Removal														0
056 Condensate														0
068 Liquid Radwaste														0
071 Waste Gas Disposal														0
072 Area Radiation Monitoring														0
075 Circulating Water														0
079 Station Air														0
086 Fire Protection														0
K/A Category Totals:	0	0	0	1	0	0	0	1	0	0	1	Group Point Total:		3

ES-401	Generic Knowledge and Abilities Outline (Tier 3)	Form ES-401-3

Facility Name:Arkansas Nuclear One Unit 2 Date of Exam:2/21/2014

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1. 07	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	2.9	1		
	2.1. 21	Ability to verify the controlled procedure copy.	3.9	1		
	2.1. 44	Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	3.9	1		
	2.1.					
	2.1. 01	Knowledge of conduct of operations requirements.			4.2	1
	2.1. 39	Knowledge of conservative decision making practices.			4.3	1
	Subtotal			3		2
2. Equipment Control	2.2. 06	Knowledge of the process for making changes to procedures.	3.0	1		
	2.2. 07	Knowledge of the process for conducting special or infrequent tests.	2.9	1		
	2.2. 43	Knowledge of the process used to track inoperable alarms.	3.0	1		
	2.2. 14	Knowledge of the process for controlling equipment configuration or status.			4.3	1
	2.2. 17	Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.			3.8	1
	2.2.					
	Subtotal			3		2
3. Radiation Control	2.3. 11	Ability to control radiation releases.	3.8	1		
	2.3. 13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	3.4	1		
	2.3.					
	2.3. 14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.			3.8	1
	2.3.					
	2.3.					
	Subtotal			2		1
4. Emergency Procedures / Plan	2.4. 05	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.	3.7	1		
	2.4. 32	Knowledge of operator response to loss of all annunciators.	3.6	1		
	2.4.					
	2.4. 14	Knowledge of general guidelines for EOP usage.			4.5	1
	2.4. 28	Knowledge of procedures relating to a security event.			4.1	1
	2.4.					
	Subtotal			2		2
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO T1/G1 QID # 5	0022 G2.4.1	There are no immediate actions (as defined by plant procedures) associated with Loss of Reactor Coolant Makeup (Loss of Charging AOP) Selected 0022 K1.03 as the replacement K/A
RO T1/G1 QID #10	0038 G2.2.42	Overlap with Operating exam (there is a SGTR event with Tech Spec Calls) Selected 0038 A1.44 as the replacement K/A
RO T1/G1 QID #16	058 A1.01	Question does not apply. Unit does not have the ability to cross tie Vital DC buses. Selected 058 AK3.02 as the replacement K/A
RO T1/G2 QID #21	033 AK 3.01	Startup Channels serve the purpose of Intermediate range instrumentation during a reactor startup. Selected 032 AK3.01 as the replacement K/A
RO T2/G1 QID #39	0013 A3.02	System over sample concerns between Tier 1 and Tier 2. Selected 026 K4.08 as the replacement K/A
RO T2/G1 QID #46	061 A3.03	System over sample concerns. Selected 005 A1.01 as the replacement K/A
RO T2/G1 QID #51	064 A4.06	System over sample concerns. Selected 039 A2.05 as the replacement K/A
RO T2/G2 QID #65	086 A1.01	Unit 2 does not control the stations fire pumps or have indications in the control room for the fire water system. The fire water system is operated by Unit 1. Selected 086 A 4.05 as the replacement K/A
RO T3 QID #66	G2.1.5	Rejected original G2.1.5 due to being an SRO duty. Selected G2.1.7 as the replacement K/A
RO T3 QID #67	G2.1.19	Does not lend itself to a generic question (directs monitoring plant components or systems). Selected G2.1.21 as the replacement K/A
RO T3 QID #71	G2.2.42	Does not lend itself to a generic question (specific system parameters that are entry level conditions for Tech Specs). Selected G2.2.43 as the replacement K/A
SRO T1/G2 QID #83	024 G2.4.41	There is not an E-Plan associated with Emergency Boration. Selected 024 AA2.01 as the replacement K/A
SRO T3 QID #99	G2.4.23	Difficultly of matching K/A. Selected G2.4.14 as the replacement K/A

Rev 2

Facility: Arkansas Nuclear One Unit 2 Date of Examination: 02/10/2014
Examination Level: RO X SRO ☐ Operating Test Number: 2014-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
A1. Conduct of Operations 2.1.20 RO(4.6)	D/R	Spent Fuel Pool Makeup Calculation ANO-2-JPM-NRC-ADMIN-SFPMU2
A2. Conduct of Operations 2.1.23 RO (4.3)	D/P/R	Calculate Time to Boil using Computer Program ANO-2-JPM-NRC-ADMIN-TTBCRO
A3. Equipment Control 2.2.12 RO (3.7)	N/R	Evaluate Containment Atmospheric Conditions ANO-2-JPM-NRC-ADMIN-CNTMT
A4. Radiation Control 2.3.7 RO (3.5)	D/R	Review Emergency RWP and Perform Evolution ANO-2-JPM-NRC-ADMIN-RWP2
Emergency Procedures/Plan		

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

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

Facility: Arkansas Nuclear One Unit 2 Date of Examination: 02/10/2014

Examination Level: RO ☐ SRO X Operating Test Number: 2014-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
A5. Conduct of Operations 2.1.20 SRO (4.6)	D/R	Review and Approve Spent Fuel Pool Makeup Calculation ANO-2-JPM-NRC-ADMIN-SFPMU
A6. Conduct of Operations 2.1.40 SRO (3.9)	N/R	Determine Shutdown Operations Protection Plan Condition ANO-2-JPM-NRC-ADMIN-SOPP1
A7. Equipment Control 2.2.14 SRO (4.3)	D/P/R	Supervisory Review of Maintenance Activities for Configuration Control ANO-2-JPM-NRC-ADMIN-MAINT
A8. Radiation Control 2.3.7 SRO (3.6)	M/R	Review Emergency RWP ANO-2-JPM-NRC-ADMIN-RWP3
A9. Emergency Procedures/Plan 2.4.38 SRO (4.4)	N/R	EOF Evacuation Determination ANO-2-JPM-NRC-ADMIN-EOFEVAC

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

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

Facility: Arkansas Nuclear One Unit 2 Date of Examination: 02/10/2014

Exam Level: RO X SRO-I ☐ SRO-U ☐ Operating Test No.: 2014-1

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U, including 1 ESF)

	System / JPM Title	Type Code*	Safety Function
S1.	ANO-2-JPM-NRC-CNTCL 022 A4.03 RO-3.2/SRO-3.2 Verify Containment Coolers in Emergency Mode	A/D/EN/L/S	5 Containment
S2.	ANO-2-JPM-NRC-ELEC06 062 A4.01 RO-3.3/SRO-3.1 Transfer Auxiliaries from SU#2 to SU#3 for 2A-1	A/M/S	6 Electrical
S3.	ANO-2-JPM-NRC-CVCS2 004 A4.07 RO-3.9/SRO3.7 Perform Emergency Boration	A/D/L/S	1 Reactivity control
S4.	ANO-2-JPM-NRC-EFW01 061 A1.01 RO-3.9/SRO4.2 Shutdown EFW Train 'A' with EFAS Signal Present	D/EN/L/S	4 Heat Removal Secondary
S5.	ANO-2-JPM-NRC-FWCS1 035 A4.01 RO-3.7/SRO-3.6 Place Feedwater Control system in Automatic	D/S	4 Heat Removal Primary
S6.	ANO-2-JPM-NRC-CVCS12 004 A4.06 RO-3.6/SRO-3.1 Verification of Minimum Letdown Flow	N/S	2 Inventory Control
S7.	ANO-2-JPM-NRC-EOP6 012 A2.06 RO-4.4/SRO-4.7 Manually Trip the Reactor	A/D/S	7 Instrumentation
S8.	ANO-2-JPM-NRC-PZR01 010 A4.01 RO-3.7/SRO-3.5 Equalize RCS and Pressurizer Boron	D/S	3 Pressure Control

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

	System / JPM Title	Type Code*	Safety Function
P1.	ANO-2-JPM-NRC-PRHTR 068 AA1.07 RO-4.1/SRO-4.2 Perform Local Operations of the Proportional Heaters	D/E/L	3 Pressure Control
P2.	ANO-2-JPM-NRC-EDDCS 064 A4.01 RO-4.0/SRO-4.3 Startup Diesel Generator Without DC Control Power (2K-4A)	D/E/L	6 Electrical
P3.	ANO-2-JPM-NRC-WGDTR 071 A2.02 RO-3.3/SRO-3.6 Perform Waste Gas Decay Tank Release	A/N/R	9 Rad Control

@ All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

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

Facility: Arkansas Nuclear One Unit 2 Date of Examination: 02/10/2014

Exam Level: RO ☐ SRO-I X SRO-U ☐ Operating Test No.: 2014-1

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U, including 1 ESF)

System / JPM Title	Type Code*	Safety Function
S1. ANO-2-JPM-NRC-CNTCL 022 A4.03 RO-3.2/SRO-3.2 Verify Containment Coolers in Emergency Mode	A/D/EN/L/S	5 Containment
S2. ANO-2-JPM-NRC-ELEC06 062 A4.01 RO-3.3/SRO-3.1 Transfer Auxiliaries from SU#2 to SU#3 for 2A-1	A/M/S	6 Electrical
S3. ANO-2-JPM-NRC-CVCS2 004 A4.07 RO-3.9/SRO3.7 Perform Emergency Boration	A/D/L/S	1 Reactivity control
S4. ANO-2-JPM-NRC-EFW01 061 A1.01 RO-3.9/SRO4.2 Shutdown EFW Train 'A' with EFAS Signal Present	D/EN/L/S	4 Heat Removal Secondary
S5. ANO-2-JPM-NRC-FWCS1 035 A4.01 RO-3.7/SRO-3.6 Place Feedwater Control system in Automatic	D/S	4 Heat Removal Primary
S6. ANO-2-JPM-NRC-CVCS12 004 A4.06 RO-3.6/SRO-3.1 Verification of Minimum Letdown Flow	N/S	2 Inventory Control
S7. ANO-2-JPM-NRC-EOP6 012 A2.06 RO-4.4/SRO-4.7 Manually Trip the Reactor	A/D/S	7 Instrumentation
S8.		

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

System / JPM Title	Type Code*	Safety Function
P1. ANO-2-JPM-NRC-PRHTR 068 AA1.07 RO-4.1/SRO-4.2 Perform Local Operations of the Proportional Heaters	D/E/L	3 Pressure Control
P2. ANO-2-JPM-NRC-EDDCS 064 A4.01 RO-4.0/SRO-4.3 Startup Diesel Generator Without DC Control Power (2K-4A)	D/E/L	6 Electrical
P3. ANO-2-JPM-NRC-WGDTR 071 A2.02 RO-3.3/SRO-3.6 Perform Waste Gas Decay Tank Release	A/N/R	9 Rad Control

@ All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

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

Facility: Arkansas Nuclear One Unit 2	Date of Examination: 02/10/2014
Exam Level: RO ☐ SRO-I ☐ SRO-U X	Operating Test No.: 2014-1

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U, including 1 ESF)

System / JPM Title	Type Code*	Safety Function
S1. ANO-2-JPM-NRC-CNTCL 022 A4.03 RO-3.2/SRO-3.2 Verify Containment Coolers in Emergency Mode	A/D/EN/L/S	5 Containment
S2.		
S3.		
S4. ANO-2-JPM-NRC-EFW01 061 A1.01 RO-3.9/SRO4.2 Shutdown EFW Train 'A' with EFAS Signal Present	D/EN/L/S	4 Heat Removal Secondary
S5.		
S6.		
S7.		
S8.		

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

System / JPM Title	Type Code*	Safety Function
P1. ANO-2-JPM-NRC-PRHTR 068 AA1.07 RO-4.1/SRO-4.2 Perform Local Operations of the Proportional Heaters	D/E/L	3 Pressure Control
P2. ANO-2-JPM-NRC-EDDCS 064 A4.01 RO-4.0/SRO-4.3 Startup Diesel Generator Without DC Control Power (2K-4A)	D/E/L	6 Electrical
P3. ANO-2-JPM-NRC-WGDTR 071 A2.02 RO-3.3/SRO-3.6 Perform Waste Gas Decay Tank Release	A/N/R	9 Rad Control

@ All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

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

Facility: ANO-2	Scenario No.: 1 (New)	Op-Test No.: 2014-1
Examiners:	Operators:	

Initial Conditions:
100%, 260 EFPD. RED Train Maintenance Week.

Turnover:
EOOS indicates 'Minimal Risk'.
Evolution scheduled: Shift Control Element Drive Mechanism (CEDM) fans from 2VSF-35D to 2VSF-35C IAW 2104.033 starting with 10.6.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP) N (SRO)	Shift Control Elements Drive Mechanism (CEDM) fans. **OP-2104.033, Containment Atmosphere Control.**
2	XCVLDNHXOU K12D01	I (ATC) I (SRO)	The temperature input to the letdown HX temperature controller (2TIC-4815) fails Hi. **OP-2203.012L, Annuciator 2K-12 Corrective Action (ACA)**
3	CT2VSF1D	C (BOP) C (SRO) TS (SRO)	2VSF-1D Containment cooler trips. TS for SRO. **OP-2203.012D/E, 2K-04 and 2K05 ACAs**
4	CEA43DROP	R (ATC) C (BOP) C (SRO) TS (SRO)	CEA 43 fully inserts. TS for SRO. **OP-2203.003, CEA Malfunction AOP**
5	RCP2P32ALOS	C (ATC) C (SRO)	'A' RCP oil leak. **OP-2203.025, RCP Emergencies AOP**
6	MSSGBLK	M (ALL)	Excess Steam Demand inside containment on 'B' SG. **OP-2202.001, Standard Post Trip Actions (SPTAs) EOP and OP-2202.009, Functional Recovery EOP.**
7	CV4652	C (ATC) C (SRO)	'B' RCP normal spray valve fails open. **OP-2202.010, Standard Attachments EOP or OP-2203.0028, Pressurizer System Malfunction AOP**
8	EFW2P7BFLT EFW2P7ACOU	M (ALL)	2P-7B EFW pump motor fault on start, 2P-7A EFW pump coupling failure. **OP-2202.009, Functional Recovery EOP.**
9	CV0760 DO_CV_0760_1 DO_CV_0760_2 CV0761 DO_CV_0760_1 DO_CV_0760_2	C (BOP) C (SRO)	The selected AFW flow path discharge valve (2CV-0760 or 2CV-0761) breaker trip. **OP-2202.010, Standard Attachments EOP.**
End point			Feedwater is restored to 'A' SG.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Target Quantitative Attributes (Section D.5.d)	Actual Attributes
Total Malfunctions (5-8)	8
Malfunctions after EOP entry (1-2)	2
Abnormal Events (2-4)	2
Major Transients (1-2)	2
EOPs entered requiring substantive actions (1-2)	1
EOP contingencies requiring substantive actions (0-1)	1
Critical Tasks (2-3)	3

Critical Task	Justification	References
‘A’ RCP must be secured within 10 min of the reactor trip.	Exceeding operating limits has the potential to degrade the RCS pressure boundary. RCPs should be maintained in an available condition for last-resort use if needed.	• 1015.050 Time Critical Operation action program, Attachment C • CE EPGB Simulator CTs: CT-23, Trip any RCP exceeding operating limits (ESDE-03, FRG-04)
Stabilize and control RCS temperature after the ESD blowdown terminates. Maintain RCS pressure within the Pressure-Temperature limits of 200°F and 30°F Margin to Saturation throughout implementation of SPTAs and Functional Recovery EOP.	If RCS heatup is allowed after SG blowdown, the RCS could over pressurize and result in lifting PZR and SG safeties. These pressure stresses added to thermal stresses of rapid cooldown could present PTS concerns.	• CE EPGB Simulator CTs: CT-07, Establish RCS temperature Control (SPTA-07, ESDE-05, HR-05)
Restore Feedwater prior to both SG levels reaching 70” wide range.	Inventory in the unaffected SG is required to remove decay heat from the reactor core (core melt potential).	• CE EPGB Simulator CTs: CT-08, Establish RCS Heat Removal (ESDE-08, HR-01) • EOP 2202.009 Functional Recovery • EOP 2202.006 Loss of Feedwater EOP Tech Guide

Scenario #1 Objectives

1) Evaluate individual ability to transfer CEDM fans.
2) Evaluate individual response to a failure of a temperature input to the letdown heat exchanger and ability to manually control temperature.
3) Evaluate individual response to a trip of a Containment Cooling fan.
4) Evaluate individual response to a CEA Malfunction.
5) Evaluate individual response to a Reactor Coolant pump oil leak (RCP emergencies).
6) Evaluate crew ability to mitigate an Excess Steam Demand.
7) Evaluate crew ability to mitigate a Loss of Feedwater.
8) Evaluate individual ability to combat events using the Functional Recovery procedure.
9) Evaluate individual ability to respond to a failure of an AFW pump discharge valve.
10) Evaluate individual ability to respond to RCP spray valve failing open.

SCENARIO #1 NARRATIVE

Simulator session begins with the plant at 100% power steady state.

When the crew has completed their control room walk down and brief, the BOP will shift Control Elements Drive Mechanism (CEDM) fans from 2VSF-35D to 2VSF-35C.

When the CEDM fans have been shifted or cued by lead examiner, the temperature input (2TE-4815) to the letdown heat exchanger temperature controller will fail high. The ATC will report that the letdown heat exchanger temperature is reading high on the hand indicating controller but the computer point and control board indication are reading lower than normal due to excessive cooling flow. The SRO will direct the ATC to take manual control of the Letdown heat exchanger temperature control valve and manually control temperature for the duration of the scenario.

After the letdown temperature controller has been placed in manual and cued by the lead examiner, 2VSF-1D containment cooler will trip. The BOP will determine that 2VSF-1D containment cooler has tripped and refer to OP-2203.012D/E, 2K04 and 2K05 Annunciator Corrective Actions. The BOP will start the idle containment cooler to maintain containment temperature and pressure in the acceptable region of operation. The SRO will enter Tech Spec 3.6.2.3 Action a. [Site OE: CR-ANO-2-2006-2444, 2VSF-1A motor failure and breaker trip.]

After the BOP has started the idle containment cooling fan and cued by lead examiner, CEA 43 will drop into the core due to faulty timing card. The SRO will enter OP-2203.003, CEA malfunction AOP. The SRO should check that less than 2 CEAs are inserted and then commence a down power within 15 minutes. The BOP should complete attachment C DNBR/LPD log. The SRO will enter Tech Specs for CEA position (3.1.3.1 Action d) and Aztilt (3.2.3). [Site and industry OE: CR-ANO-2-2007-0127 dropped CEA, and NRC Event # 49601 Palo Verde dropped CEA.]

After the crew has completed the required reactivity manipulation, entered the appropriate tech specs, and cued by the lead examiner, 'A' RCP oil leak will start that causes oil level to lower and bearing temperatures to rise. The CRS will enter OP-2203.025, RCP Emergencies AOP. The crew will monitor the 'A' RCP oil level trend and bearing temperatures. After bearing temperatures begin to rise (trip criteria >18^0F/min.) the ATC should trip the reactor and secure the 'A' RCP. The crew may elect to secure a RCP in the 'B' S/G loop to balance flows. Securing a RCP not satisfying operating limits is a time critical operator action per OP-1015.050 Time Critical Operator Action Program. [Site OE: RCP oil leaks CR-ANO-2-2013-1602, CR-ANO-2-2013-587, CR-ANO-2-2013-58.]

The crew will implement OP-2202.001, Standard Post Trip Actions (SPTA) EOP. After the reactor trips a Main Steam line break ('B' SG) inside containment will cause an Excess Steam Demand. Main Steam Isolation (MSIS) and Containment Spray (CSAS) will actuate tripping Main Feedwater pumps, Condensate pumps, AFW pump, closing the MSIVs and feedwater block valves. The 2P-7B EFW pump motor will fail to start and 2P-7A EFW pump coupling will break causing a loss of feedwater event. The ATC will secure all the Reactor Coolant pumps due to the Containment Spray actuation. When the 'B' RCP spray valve (2CV-4652) handswitch is placed in manual, the valve will fail open. The ATC must recognize this and isolate the spray valve using the associated block valve. If the spray valve is not isolated, the ATC's ability to control RCS pressure will be limited. [Industry OE for Excess Steam Demand, SOER 82-7, Reator Vessel Pressurized Thermal Shock.]

SCENARIO #1 NARRATIVE (continued)

After completing SPTAs, The SRO will diagnose an Excess Steam Demand and Loss of Feedwater event and enter OP-2202.009, Functional Recovery EOP. The crew will maintain post blowdown temperature and pressure of the RCS to prevent pressurized thermal shock. The BOP will steam ‘A’ S/G using the upstream Atmospheric Dump valve when ‘B’ S/G blows dry. The ATC should use Auxiliary Spray to maintain RCS pressure. The Crew will restore Feedwater from the AFW pump (2P-75) after removing the MSIS and CSAS trip. The selected feed path valve from AFW will trip its breaker when the valve is opened requiring use of the alternate flow path. [Loss of feedwater events industry OE: SOER 86-01 Reliability of PWR Auxiliary feedwater systems, and PRA operator action # 3 Establish flow to SGs from AFW to the SGs given a los of both EFW and MFW flow to the SGs.]

Facility: ANO-2	Scenario No.: 2 (New)	Op-Test No.: 2014-1

Examiners:	Operators:

Initial Conditions:
~40 %. MOL. 'C' channel Excore has failed and PPS points 1 through 4 are in bypass. RED Train Maintenance Week. 'B' Component Cooling Water CCW pump in service.

Turnover:
EOOS indicates 'Minimal Risk'. Hold power 39- 41 % until S/G Chloride less than 10 ppb. SG blowdown ~120 gpm per SG for cleanup. Reactor Engineering is developing reactivity plan for power escalation. 'C' channel Excore has failed and PPS points 1 through 4 are in bypass and all required actions are complete (TS 3.3.1.1 action 2 entered).
Evolution scheduled: Perform Red Train Proportional Heater test starting with step 2.1.

Event No.	Malf. No.	Event Type*	Event Description
1		N (ATC) N (SRO)	Perform Red Train Proportional Heater test. **OP-2103.005 Pressurizer Operations.**
2	NIBUPPER	C (BOP) C (SRO) TS (SRO)	'B' channel Excore upper chamber fails high. TS for SRO. **OP-2203.026, NI malfunction AOP.**
3	XRCCHBPCNT	I (ATC) I (BOP) I (SRO)	'B' Pressurizer pressure control channel fails high. **OP-2203.028, Pressurizer System Malfunction AOP**
4	CCW2P33BPWR CCW2P33CPWR	C (BOP) C (SRO)	2P-33B CCW pump trips and 2P-33C CCW pump fails to start. **OP-2203.025, RCP Emergencies AOP**
5	RCP2P32CSLK	R (ATC) N (BOP) N (SRO) TS (SRO)	'C' Reactor Coolant Pump (RCP) develops an intersystem LOCA from the RCS to CCW of 15 gpm. TS for SRO. **OP-2203.016, Excess RCS leakage AOP**
6	RCP2P32CSLK ESFK202AAF ESFK202BAF	M (All)	'C' RCP intersystem LOCA degrades to 250 gpm. CCW to RCPs fail to auto close on CIAS. **OP-2202.001, Standard Post Trip Actions (SPTA), and OP-2202.003, Loss of Coolant Accident EOP**
7	RCSHTRON	C (ATC) C (SRO)	Pressurizer Backup Heaters fail to de-energize on low pressurizer level.
8	CV0231	C (BOP) C (SRO)	Gland seal regulator 2PCV-0231 fails closed. **2203.012B, Annuciator 2K-02 Corrective Action (ACA)**
End point			CCW to RCP has been isolated, a RCS cooldown has been started and condenser vacuum maintain by operation of 2CV-0233 gland seal regulator.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Target Quantitative Attributes (Section D.5.d)	Actual Attributes
Total Malfunctions (5-8)	7
Malfunctions after EOP entry (1-2)	2
Abnormal Events (2-4)	4
Major Transients (1-2)	1
EOPs entered requiring substantive actions (1-2)	1
EOP contingencies requiring substantive actions (0-1)	0
Critical Tasks (2-3)	3

Critical Task	Justification	
Isolate RCS from leaving Containment by closing CCW to containment valves within 10 min. of the Reactor Trip.	Isolating CCW to Containment prevents release of radioactivity by bypassing Containment. Failure to establish a containment boundary could result in violating exposure limits.	• CE EPGB Simulator CTs: CT-09, Establish Containment Isolation (LOCA-07) • 10CFR20 • 10CFR100
Commence an RCS cooldown within 30 minutes of entry into OP-2202.003, LOCA EOP.	Cooling down and depressurizing the RCS removes decay heat and lowers the DP at the break, slowing the leak rate and reducing makeup volume required. SDC entry conditions are also required for long-term cooling.	• CE EPGB Simulator CTs: CT-20, Cool down and depressurize RCS (LOCA-09) • CR-ANO-2-2010-948, Critical task times
Establish RCS pressure control to maintain RCS subcooling. Maintain pressure and temperature within the PT limits of <200^0 F and >30^0F MTS throughout implementation of OP-2202.003, LOCA EOP.	Once RCS subcooling is lost, PZR level is no longer a valid indication of RCS inventory. A reactor head void can form, and if left uncontrolled, could result in core uncovery and fuel damage.	• CE EPGB Simulator CTs: CT-06, Establish RCS Pressure Control (LOCA-12)

Scenario #2 Objectives

1) Evaluate individual ability to perform Proportional heater surveillance.
2) Evaluate individual response to a failure of a Nuclear Instrument.
3) Evaluate individual response to a Pressurizer System Malfunction (Pressure channel failure).
4) Evaluate individual response to a failure of a Component Cooling water pump.
5) Evaluate individual response to an intersystem Loss of Coolant Accident. (LOCA)
6) Evaluate crew ability to mitigate an intersystem LOCA.
7) Evaluate individual response to failure of a gland seal regulator.
8) Evaluate individual response to a failure of the pressurizer backup heaters to de-energize on low level.

SCENARIO #2 NARRATIVE

Simulator session begins with the plant at 40% power steady.

When the crew has completed their control room walk down and brief, they will perform the Red Train Proportional Heater surveillance.

SCENARIO #2 NARRATIVE (continued)

When the Red Train Proportional Heater has been placed to auto or when cued by the lead examiner, Channel B Excore upper chamber will fail high. The SRO will enter the OP-2203.026, NI Malfunction AOP and the crew should determine that B channel linear power is failed but log power is still functional by monitoring output for the three chambers. The SRO will also enter Tech Spec 3.3.1.1 Action 3 for Reactor Protection System. The BOP will trip points 1, 3, and 4 on channel ‘B’ by using the linear calibrate switch. The points must be tripped because Channel C is in bypass. [Site OE: CR-ANO-2-2002-693, D Excore failure.]

When the SDBCS permissives have been aligned and cued by the lead examiner, the ‘B’ pressurizer pressure control channel will fail high causing the spray valves to open and RCS pressure to lower. The CRS should enter the OP-2203.028, Pressurizer System Malfunction AOP. The crew will place the other pressurizer pressure controller in service, verify that both spray valves close, and the pressurizer heaters restore RCS pressure. The BOP will place a maximum of one Steam Dump and Bypass Control System (SDBCS) valve permissive in manual and all other permissives to off. [Site OE: CR-ANO-2-2011-1605, Pressurizer pressure failing high.]

After the BOP has tripped points 1, 3, and 4, and cued by lead examiner, 2P-33B CCW pump will trip and 2P-33C CCW pump will fail to start automatically or manually. The SRO will enter OP-2203.025, RCP Emergencies AOP. The BOP should call NLOs to investigate the CCW pump trip. The SRO should direct the BOP to start 2P-33C CCW pump but it will fail to start. The SRO will then direct opening all CCW cross-tie valves and start 2P-33A CCW pump. [Site OE: CR-ANO-2-2007-313, Trip of 2P-33B CCW pump with 2P-33C out of service for maintenance.]

After the crew has restored CCW flow to the RCPs, and cued by the lead examiner, a 15 gpm RCS to CCW leak will start. The crew should notice that CCW Surge Tank level is rising. The crew’s recognition of the leak may be delayed because the ‘B’ Surge Tank level would normally rise from the different pump configuration. Also the CCW letdown radiation monitor will alarm indicating RCS to CCW leakage. The SRO will enter OP-2203.016, Excess RCS Leakage AOP, and direct the board operator actions. The crew should perform leak rates, isolate letdown to verify the leak is not in letdown and determine the need for a plant shutdown using normal boration. The SRO should enter Attachment A of Excess RCS Leakage, align the CCW surge tanks to the gas collection header and direct the NLO to control surge tank level. The crew will perform a power reduction such that the plant will be taken off line. The SRO should enter Tech Spec 3.4.6.2 Action a for RCS leakage. The ATC will borate the RCS and reduce turbine load to maintain Tave-Tref within 2°F. The BOP will make preparations to remove secondary plant equipment from service as power is reduced. [Industry OE: NRC information notice 92-36 Intersystem LOCA outside containment. Industry OE: SEN-220, SEN-216, & SEN-182, RCS leakage events.]

After the required reactivity manipulations are complete and cued by the lead examiner, the RCS to CCW will degrade to 250 gpm. The SRO will direct the reactor to be tripped, actuate SIAS & CCAS, secure RCPs, and isolate CCW to the RCPs. The CCW to RCPs valves will fail to auto close on a valid CIAS. The SRO should enter and direct the actions of SPTAs.

SCENARIO #2 NARRATIVE (continued)

The crew will implement OP-2202.001, Standard Post Trip Actions (SPTA) EOP. The ATC should recognize that the pressurizer backup heaters failed to de-energize on low pressurizer level. Also, the crew should place the SDBCS master controller in Auto Local and lower the set point to maintain margin to saturation.

The SRO will diagnose and enter OP-2202.003, Loss of Coolant Accident EOP. After the crew has entered the LOCA EOP and cued by the lead examiner, 2PCV-0231 gland seal pressure control valve will fail closed. The BOP will manually control 2CV-0233 gland seal bypass valve to maintain gland seal header pressure and condenser vacuum. The crew will commence a cooldown to allow depressurization and refilling the pressurizer. The BOP will restore Service Water to Component Cooling Water and Auxiliary Cooling water. [Site OE: for 2PCV-0231 gland seal pressure control valve CR-ANO-2-2009-719, CR-ANO-2-2009-311, and CR-ANO-2-2006-1406.]

Facility: ANO-2	Scenario No.: 3 (New)	Op-Test No.: 2014-1
Examiners:	Operators:	

Initial Conditions:
98% MOL; RED Train Maintenance Week.

Turnover:
Mabelvale transmission line out of service and Unit 2 output is limited to 1035 MW gross, 995 MW net.
EOOS indicates 'Minimal Risk'.
Evolution scheduled: Shift running vacuum pumps.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP) N (SRO)	Shift running vacuum pumps. **OP-2106.010 Condenser Vacuum System.**
2	XRRPZRLSP	I (ATC) I (SRO)	Reactor Reg. output to PZR level control program fails to 41%. **OP-2203.028, Pressurizer System Malfunction AOP**
3	DO_HS_8259_G CV82591 XRI2RITS8231A DO_RITS8231_10	C (BOP) C (SRO) TS (SRO)	2RITS-8271-2 Containment Atmosphere Monitor (CAMS) coupling fails and 2RITS-8231-1 CAMS particulate detector fails. TS for SRO. **OP-2203.012J, Annunciator 2K-10 Corrective Action (ACA), and OP-2203.012K, 2K-11 ACA**
4	MFWPMPBTRP	R (ATC) C (BOP) C (SRO) TS (SRO)	'B' Main Feed Water pump trips. TS (Tcold out of range high) for SRO. **OP-2203.027, Loss of Main Feedwater pump AOP**
5	SGBTUBE	M (ALL) TS (SRO)	'B' Steam Generator Tube Rupture ramps up to 300 gpm over 20 min. TS for SRO **OP-2203.038, Primary to Secondary leakage AOP, OP-2201.001 Standard Post Trip Actions EOP, and 2202.004 Steam Generator Tube Rupture EOP**
6	ESFSIAS2 CV48211	C (ATC) C (BOP) C (SRO)	Green Train SIAS fails to actuate and letdown isolation 2CV-4821-1 fails open. **OP-2202.010 Standard Attachments EOP**
7	CV0302 CV0303 CV0306	C (ATC) C (SRO)	Steam dump turbine bypass valve fails closed. **OP-2105.008, SDBCS operations**
End Point			'B' Steam Generator is isolated.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Target Quantitative Attributes (Section D.5.d)	Actual Attributes
Total Malfunctions (5-8)	6
Malfunctions after EOP entry (1-2)	2
Abnormal Events (2-4)	3
Major Transients (1-2)	1
EOPs entered requiring substantive actions (1-2)	1
EOP contingencies requiring substantive actions (0-1)	0
Critical Tasks (2-3)	2

Critical Task	Justification	
Perform one or more of the following to maintain/restore Margin to Saturation (MTS) > 30 degrees F. • Start the Green train HPSI pump and open HPSI injection valve(s) • Isolate letdown • Adjust RCS Cooldown rate MTS must be restored >30 degrees F within 10 min.	Once RCS subcooling is lost, PZR level is no longer a valid indication of RCS inventory. A reactor head void can form, and if left uncontrolled, could result in core uncovery and fuel damage. RCP operating limits require MTS to be >30^0F.	• CE EPGB Simulator CTs: CT-06, Establish RCS Pressure Control (SGTR-10) • 1015.050 Time Critical Operation Actions, Attachment C
Conduct an RCS cooldown to Thot <535 ^{0}F and isolate ‘B’ SG (2202.010 Attachment 10 completed) within 1 hour after the Reactor trip. Assumption is that the operator will diagnose within 30 minutes and then isolate within next 30 minutes after entry into 2202.004, SGTR EOP	Reduce Thot below 535^0F is necessary to prevent a MSSV from lifting (535^0F), thus preventing an offsite release and exceeding 10CFR100 exposure limits at the site boundary.	• CE EPGB Simulator CTs: CT-20, Cooldown and depressurize RCS (SGTR-05) CT-14, Isolate most affected SG (SGTR-09). • SAR Section 15.1.18 • 1015.050 Time Critical Operation Actions, Attachment C • EOP 2202.004, SGTR Tech Guide

Scenario #3 Objectives

1) Evaluate individual ability to perform a vacuum pump swap.
2) Evaluate individual response to a failure of a Containment Air monitor sample pump.
3) Evaluate individual response to a failure of a Containment Air monitor radiation monitor.
4) Evaluate individual response to a Pressurizer system malfunction involving pressurizer level failing high.
5) Evaluate individual response to a failure of loss of main feedwater pump.
6) Evaluate crew’s ability to mitigate a Steam Generator Tube Rupture.
7) Evaluate individual response to Green Train SIAS failure to actuate.
8) Evaluate individual response to a failure of letdown to automatically isolate.
9) Evaluate individual response to a steam dump turbine bypass valve failing closed.

SCENARIO #3 NARRATIVE

Simulator session begins with the plant at ~98% power.

When the crew has completed their control room walk down and brief, they will shift running vacuum pumps.

When the vacuum pumps have been shifted or when cued by the lead examiner, the Reactor Reg pressurizer level program output will fail to minimum (41%). The SRO will enter the OP-2203.028, PZR System Malfunctions AOP. The ATC will take manual control of letdown to control pressurizer level. The ATC must take control of PZR heaters to control RCS pressure (All heaters will be energized) The ATC should place the PZR level controller to Auto and Local then adjust the setpoint to programmed setpoint. Then Letdown should be placed back in automatic. This failure will also prevent manual start of back up charging pumps if needed to control PZR level.

When letdown has been restored to automatic or cued by lead examiner, the in-service Containment Air Monitor System (CAMS) unit 2RITS-8271-2 coupling will fail. The ATC should report the 2K-11 H10 CNTMT Air Monitor trouble alarm and refer to the ACA. The BOP should investigate and determine that 2RITS-8271-2 has low flow. When contacted, the NLO will report the coupling has failed. The BOP should use OP-2104.033 Ventilation System Operations to place the standby CAMS unit in service. When the standby CAMS unit is placed in service, the particulate detector will fail requiring entry into Tech Spec 3.4.6.1 Action a. [Site OE: CR-ANO-2-2013-1880, CAMS particulate detector failure, CR-ANO-2-2011-2691, for CAMS unit low air flow and CR-ANO-2-2006-1191, for sample motor failure.]

When all actions due to the CAMS failure have been completed, or cued by the lead examiner, ‘B’ MFWP will trip. The SRO will enter and implement OP-2203.027, Loss of Main Feedwater Pump AOP. This will result in steam flow exceeding feed flow and SG levels lowering. The crew will manually and rapidly reduce turbine load, insert group 6 and group P CEAs, borate using emergency boration to the RCS until feed flow is greater than steam flow. The SRO will be required to enter Tech Spec 3.2.6 for Tc out of range high. Then start a normal boration power reduction to less than 80%. [Industry OE: INPO event # OE31445 Loss of a Main feedwater pump. Site OE: CR-ANO-2-2009-3744, ‘B’ Main Feedwater pump trip.]

After the crew has restored feedwater flow greater than steam flow or cued by lead examiner, a Steam Generator Tube Leak will occur on ‘B’ Steam Generator. The SRO will enter OP 2203.038, Primary to Secondary Leakage AOP. The SRO should enter TS 3.4.6.2 Action a, RCS leakage, and TS 3.7.1.2 for EFW when steam is isolated to 2P-7A EFW pump. When the leak rate exceeds 44 gpm, the crew should determine that the leak rate is greater than 44 gpm. They will trip the reactor, actuate SIAS, and CCAS. [Industry OE: SOER 83-2, Steam Generator Tube Ruptures. Steam Generator Tube Rupture response is a time critical operator action per OP-1015.050 Time Critical Operator action program.]

SCENARIO #3 NARRATIVE (continued)

The Crew will implement OP-2202.001, Standard Post Trip Actions (SPTA) EOP. When SIAS is actuated the Green train components will fail to reposition. The crew should recognize the failure of Green train SIAS to actuate. The BOP should have a NLO check the breaker and pump motors for the Green train High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection pumps (LPSI) pumps. After the NLO report, the BOP should manually start 2P-89B HPSI pump and open all injection valves. Also, 2CV-4821-1 Red train letdown isolation valve will fail to close leaving letdown aligned. The ATC should recognize that letdown is aligned and close a Green train isolation to help maintain RCS inventory. The crew will align Service Water to CCW to maintain forced circulation. The crew may lower Steam Dump Master Controller setpoint during SPTAs to aid in maintaining margin to saturation. [Industry OE: SOER 83-9, Valve inoperability cause by motor operator failures for 2CV-4821-1.]

The SRO will diagnose and enter OP-2202.004, Steam Generator Tube Rupture (SGTR) EOP. The ATC should commence cool down of the RCS to allow isolation of ‘B’ steam generator. The BOP will override Service Water to Auxiliary Cooling Water to maintain condenser vacuum. During the cooldown, 2CV-0303, 2CV-0302, or 2CV-0306 Steam dump valve (depending on which is being used) will fail closed impacting the cooldown rate. The ATC will notice that the cooldown has stopped and adjust the cooldown rate to ensure the steam generator is isolated within the 30 minute required time. Once T_{hot} is less than 535 degrees F, the BOP should isolate ‘B’ steam generator. [Site OE: CR-ANO -2-2010-558, CR-ANO -2-2009-3780, CR-ANO -2-2008-1190, Steam dump valve failure to open.]