Facility: Arkansas Nuclear One, Unit 2								Date of Exam:								
Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1	3	3	3				3	3			3	18			6
	2	2	1	1		N/A		1	2	N/A		2	9			4
	Tier Totals	5	4	4				4	5			5	27			10
2. Plant Systems	1	2	3	2	2	2	2	3	3	3	3	3	28			5
	2	1	1	1	1	1	1	1	1	0	1	1	10			3
	Tier Totals	3	4	3	3	3	3	4	4	3	4	4	38			8
3. Generic Knowledge and Abilities Categories				1		2		3		4			10	1	2	3	4	7
				2		3		3		2						

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the “Tier Totals” in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)

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

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

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

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

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

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

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

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

G* Generic K/As

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

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

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

Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/SRO)

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1									
000008 (APE 8) Pressurizer Vapor Space Accident / 3					X		AA2.23 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Criteria for throttling high-pressure injection after a small LOCA	3.6	1
000009 (EPE 9) Small Break LOCA / 3	X						EK1.02Knowledge of the operational implications of the following concepts as they apply to the small break LOCA: Use of steam tables	3.5	2
000011 (EPE 11) Large Break LOCA / 3				X			EA1.09Ability to operate and monitor the following as they apply to a Large Break LOCA: Core flood tank initiation	4.3	3
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4					X		AA2.10 Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on loss of cooling or seal injection	3.7	4
000022 (APE 22) Loss of Reactor Coolant Makeup / 2			X				AK3.05 Knowledge of the interrelations between the Loss of Reactor Coolant Makeup and the following: Need to avoid plant transients	3.2	5
000025 (APE 25) Loss of Residual Heat Removal System / 4		X					AK2.03 Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Service water or closed cooling water pumps	2.7	6
000026 (APE 26) Loss of Component Cooling Water / 8			X				AK3.04 Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Effect on the CCW flow header of a loss of CCW	3.5	7
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3		X					AK2.03 Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners	2.6	8
000029 (EPE 29) Anticipated Transient Without Scram / 1					X		EA2.05 Ability to determine or interpret the following as they apply to a ATWS: System component valve position indications	3.4	9
000038 (EPE 38) Steam Generator Tube Rupture / 3				X			EA1.17 Ability to operate and monitor the following as they apply to a SGTR: S/G sample isolation valve indicators	3.2	10
~~000040 (APE 40; BW E05;~~ CE E05; ~~W E12~~) Steam Line Rupture—Excessive Heat Transfer / 4				X			EA1.3 Ability to operate and / or monitor the following as they apply to the (Excess Steam Demand): Desired operating results during abnormal and emergency situations	3.4	11
000054 (APE 54; ~~CE E06~~) Loss of Main Feedwater /4						X	2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	3.8	12
000055 (EPE 55) Station Blackout / 6									
000056 (APE 56) Loss of Offsite Power / 6	X						AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Principle of cooling by natural convection	3.7	13
000057 (APE 57) Loss of Vital AC Instrument Bus / 6						X	2.4.31 Knowledge of annunciator alarms, indications, or response procedures	4.2	14
000058 (APE 58) Loss of DC Power / 6	X						AK1.01 Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus.	2.8	15
000062 (APE 62) Loss of Nuclear Service Water / 4			X				AK3.02 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS	3.6	16

000065 (APE 65) Loss of Instrument Air / 8						X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	4.4	17
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6		X					AK2.07 Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine / generator control	3.6	18
(W E04) LOCA Outside Containment / 3									
(W E11) Loss of Emergency Coolant Recirculation / 4									
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4									
K/A Category Totals:	3	3	3	3	3	3	Group Point Total:		18

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

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1									
000003 (APE 3) Dropped Control Rod / 1					X		AA2.01 Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod position indication to actual rod position	3.7	19
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1					X		AA2.06 Knowledge of the interrelations between Emergency Boration and the following: When boron dilution is taking place	3.6	20
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7						X	2.1.30 Ability to locate and operate components, including local controls	4.4	21
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8		X					AK2.02 Knowledge of the interrelations between the Fuel Handling Incidents and the following: Radiation monitoring equipment (portable and installed)	3.4	22
000037 (APE 37) Steam Generator Tube Leak / 3									
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9			X				AK3.04 Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Actions contained in EOP for accidental liquid radioactive-waste release	3.8	23
000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68; BW A06) Control Room Evacuation / 8									
000069 (APE 69; W E14) Loss of Containment Integrity / 5									
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4									
000076 (APE 76) High Reactor Coolant Activity / 9				X			AA1.04 Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: Failed fuel-monitoring equipment	3.2	24
000078 (APE 78*) RCS Leak / 3									
(W E01 & E02) Rediagnosis & SI Termination / 3									
(W E13) Steam Generator Overpressure / 4									
(W E15) Containment Flooding / 5									
(W E16) High Containment Radiation /9									
(BW A01) Plant Runback / 1									
(BW A02 & A03) Loss of NNI-X/Y/7									
(BW A04) Turbine Trip / 4									
(BW A05) Emergency Diesel Actuation / 6									
(BW A07) Flooding / 8									
(BW E03) Inadequate Subcooling Margin / 4									
(BW E08; W E03) LOCA Cooldown—Depressurization / 4									

(BW E09; CE A13**; W E09 & E10) Natural Circulation/4	X						AK1.2 Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation Operations): Normal, abnormal and emergency operating procedures associated with (Natural Circulation Operations)	3.2	25
(BW E13 & E14) EOP Rules and Enclosures									
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4									
(CE A16) Excess RCS Leakage / 2	X						AK1.3 Annunciators and conditions indicating signals, and remedial action associated with the (Excess RCS Leakage)	3.2	26
(CE E09) Functional Recovery						X	2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	3.7	27
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
K/A Category Point Totals:	2	1	1	1	2	2	Group Point Total:		9

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump									X			A3.01 Ability to monitor automatic operation of the RCPS, including: Seal injection flow	3.3	28
004 (SF1; SF2 CVCS) Chemical and Volume Control									X			A3.14 Ability to monitor automatic operation of the CVCS, including: Letdown and charging flows	3.4	29
005 (SF4P RHR) Residual Heat Removal						X						K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger	2.5	30
006 (SF2; SF3 ECCS) Emergency Core Cooling				X								K4.06 Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Recirculation of minimum flow through pumps	2.7	31
007 (SF5 PRTS) Pressurizer Relief/Quench Tank					X							K5.02 Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR	3.1	32
008 (SF8 CCW) Component Cooling Water			X									K3.01 Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: Loads cooled by CCWS	3.4	33
010 (SF3 PZR PCS) Pressurizer Pressure Control		X										K2.02 Knowledge of bus power supplies to the following: Controller for PZR spray valve	2.5	34
012 (SF7 RPS) Reactor Protection					X							K5.02 Knowledge of the operational implications of the following concepts as the apply to the RPS: Power Density	3.1	35
013 (SF2 ESFAS) Engineered Safety Features Actuation								X				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; LOCA	4.6	36
022 (SF5 CCS) Containment Cooling											X	2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	4.6	37
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray		X										K2.02 Knowledge of bus power supplies to the following: MOVs	3.4	38
039 (SF4S MSS) Main and Reheat Steam				X								K4.08 Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Interlocks on MSIV and bypass valves	3.3	39
059 (SF4S MFW) Main Feedwater									X			A3.02 Ability to monitor automatic operation of the MFW, including: Programmed levels of the S/G	2.9	40
061 (SF4S AFW) Auxiliary/Emergency Feedwater		X										K2.01 Knowledge of bus power supplies to the following: AFW system MOVs	4.0	41
062 (SF6 ED AC) AC Electrical Distribution	X											K1.03 Knowledge of the physical connections and/or cause effect relationships between the ac distribution system and the following systems: DC distribution	3.5	42
063 (SF6 ED DC) DC Electrical Distribution								X				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds	2.5	43
064 (SF6 EDG) Emergency Diesel Generator						X						K6.08 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks	3.2	44
073 (SF7 PRM) Process Radiation Monitoring											X	2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.1	45

076 (SF4S SW) Service Water			X									K3.07 Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads	3.7	46
078 (SF8 IAS) Instrument Air	X											K1.05 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: MSIV air	3.4	47
103 (SF5 CNT) Containment							X					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity	3.7	48
053 (SF1; SF4P ICS*) Integrated Control														
003 (SF4P RCP) Reactor Coolant Pump										X		A4.02 Ability to manually operate and/or monitor in the control room: RCP motor parameters	2.9	49
005 (SF4P RHR) Residual Heat Removal							X					A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Closed cooling water flow rate and temperature	2.5	50
010 (SF3 PZR PCS) Pressurizer Pressure Control							X					A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR PCS controls including: RCS heatup and cooldown effect on pressure	3.1	51
012 (SF7 RPS) Reactor Protection								X				A2.07 Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of dc control power	3.2	52
026 (SF5 CSS) Containment Spray										X		A4.05 Ability to manually operate and/or monitor in the control room: Containment spray reset switches	3.5	53
039 (SF4S MSS) Main and Reheat Steam										X		A4.01 Ability to manually operate and/or monitor in the control room: Main steam supply. valves	2.9	54
076 (SF4S SW) Service Water											X	2.2.37 Ability to determine operability and/or availability of safety related equipment.	3.6	55
K/A Category Point Totals:	2	3	2	2	2	2	3	3	3	3	3	Group Point Total:		28

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

Plant Systems—Tier 2/Group 2 (RO/SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive	X											Kl.05 Knowledge of the physical connections and/or cause-effect relationships between the CRDS and the following systems: NIS and RPS	4.5	56
002 (SF2; SF4P RCS) Reactor Coolant														
011 (SF2 PZR LCS) Pressurizer Level Control														
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation										X		A4.03 Ability to manually operate and/or monitor in the control room: Trip Bypasses	3.5	65
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor							X					A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM system controls including: Core Exit Temperature	3.7	57
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control					X							K5.03 Knowledge of the operational implications of the following concepts as they apply to the HRPS: Sources of hydrogen within containment	2.9	58
029 (SF8 CPS) Containment Purge														
033 (SF8 SFPCS) Spent Fuel Pool Cooling			X									K3.03 Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Spent fuel temperature	3.0	59
034 (SF8 FHS) Fuel-Handling Equipment														
035 (SF 4P SG) Steam Generator								X				A2.06 Ability to (a) predict the impacts of the following malfunctions or operations on the GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Small break LOCA	4.5	60
041 (SF4S SDS) Steam Dump/Turbine Bypass Control						X						K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS	2.7	61
045 (SF 4S MTG) Main Turbine Generator				X								K4.01 Knowledge of MT/G system design feature(s) and/or interlock(s) which provide for the following: Programmed controller for relationship between steam pressure at T/G inlet (impulse, first stage) and plant power level	2.7	62
055 (SF4S CARS) Condenser Air Removal											X	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.3	63
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring														
075 (SF8 CW) Circulating Water		X										K2.03 Knowledge of bus power supplies to the following: Emergency/essential SWS pumps	2.6	64
079 (SF8 SAS**) Station Air														
086 Fire Protection														

050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:	1	1	1	1	1	1	1	1	0	1	1	Group Point Total:		10

Facility: Arkansas Nuclear One, Unit 2		Date of Exam:				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.28	Knowledge of the purpose and function of major system components and controls.	4.1	66		
	2.1.38	Knowledge of the station's requirements for verbal communications when implementing procedures.	3.7	67		
	Subtotal		2			
2. Equipment Control	2.2.21	Knowledge of pre- and post-maintenance operability requirements.	2.9	68		
	2.2.23	Ability to track Technical Specification limiting conditions for operations.	3.1	69		
	2.2.36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	3.1	70		
	Subtotal		3			
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	71		
	2.3.11	Ability to control radiation releases.	3.8	72		
	2.3.15	2.3.15 Ability to use radiation monitoring systems, Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	73		
	Subtotal		3			
4. Emergency Procedures/Plan	2.4.4	Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	4.5	74		
	2.4.29	Knowledge of the emergency plan.	3.1	75		
	Subtotal		2			
Tier 3 Point Total				10		7

Facility: Arkansas Nuclear One, Unit 2 Date of Exam:

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1				N/A					N/A			18	3	3	6
	2												9	2	2	4
	Tier Totals												27	5	5	10
2. Plant Systems	1												28	3	2	5
	2												10	1	2	3
	Tier Totals												38	4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2	3 4	7
														1 2	2 2	

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

G* Generic K/As

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

ES-401 PWR Examination Outline Form ES-401-2 Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/SRO)									
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1									
000008 (APE 8) Pressurizer Vapor Space Accident / 3									
000009 (EPE 9) Small Break LOCA / 3						X	2.1.45 Ability to identify and interpret diverse indications to validate the response of another indication.	4.3	76
000011 (EPE 11) Large Break LOCA / 3					X		EA2.13 Ability to determine or interpret the following as they apply to a Large Break LOCA: Difference between overcooling and LOCA indications	3.7	77
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4						X	2.2.22 Knowledge of limiting conditions for operations and safety limits.	4.7	78
000022 (APE 22) Loss of Reactor Coolant Makeup / 2						X	2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.4	79
000025 (APE 25) Loss of Residual Heat Removal System / 4									
000026 (APE 26) Loss of Component Cooling Water / 8									
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3									
000029 (EPE 29) Anticipated Transient Without Scram / 1									
000038 (EPE 38) Steam Generator Tube Rupture / 3					X		EA2.09 Ability to determine or interpret the following as they apply to a SGTR: Existence of natural circulation, using plant parameters	4.2	80
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4									
000054 (APE 54; CE E06) Loss of Main Feedwater /4									
000055 (EPE 55) Station Blackout / 6									
000056 (APE 56) Loss of Offsite Power / 6									
000057 (APE 57) Loss of Vital AC Instrument Bus / 6									
000058 (APE 58) Loss of DC Power / 6					X		AA2.02 Ability to determine and interpret the following as they apply to the Loss of DC Power: 125V dc bus voltage, low/critical low, alarm	3.6	81
000062 (APE 62) Loss of Nuclear Service Water / 4									
000065 (APE 65) Loss of Instrument Air / 8									
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6									
(W E04) LOCA Outside Containment / 3									
(W E11) Loss of Emergency Coolant Recirculation / 4									
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4									
K/A Category Totals:					3	3	Group Point Total:		6

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/SRO)		Form ES-401-2
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1					X		AA2.03 Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal : Proper actions to be taken if automatic safety functions have not taken place	4.8	82
000003 (APE 3) Dropped Control Rod / 1									
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7									
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8									
000037 (APE 37) Steam Generator Tube Leak / 3									
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9									
000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68; BW A06) Control Room Evacuation / 8									
000069 (APE 69; W E14) Loss of Containment Integrity / 5									
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4						X	2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	4.6	83
000076 (APE 76) High Reactor Coolant Activity / 9									
000078 (APE 78*) RCS Leak / 3									
(W E01 & E02) Rediagnosis & SI Termination / 3									
(W E13) Steam Generator Overpressure / 4									
(W E15) Containment Flooding / 5									
(W E16) High Containment Radiation /9									
(BW A01) Plant Runback / 1									
(BW A02 & A03) Loss of NNI-X/Y/7									
(BW A04) Turbine Trip / 4									
(BW A05) Emergency Diesel Actuation / 6									
(BW A07) Flooding / 8									
(BW E03) Inadequate Subcooling Margin / 4									
(BW E08; W E03) LOCA Cooldown—Depressurization / 4									
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4									
(BW E13 & E14) EOP Rules and Enclosures									
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4					X		AA2.2 Ability to determine and interpret the following as they apply to the (RCS Overcooling): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.4	84

(CE A16) Excess RCS Leakage / 2						X	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4.2	85
(CE E09) Functional Recovery									
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
K/A Category Point Totals:					2	2	Group Point Total:		4

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump								X				A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems	3.1	86
004 (SF1; SF2 CVCS) Chemical and Volume Control														
005 (SF4P RHR) Residual Heat Removal														
006 (SF2; SF3 ECCS) Emergency Core Cooling														
007 (SF5 PRTS) Pressurizer Relief/Quench Tank														
008 (SF8 CCW) Component Cooling Water														
010 (SF3 PZR PCS) Pressurizer Pressure Control														
012 (SF7 RPS) Reactor Protection														
013 (SF2 ESFAS) Engineered Safety Features Actuation														
022 (SF5 CCS) Containment Cooling											X	2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	4.2	87
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray														
039 (SF4S MSS) Main and Reheat Steam														
059 (SF4S MFW) Main Feedwater														
061 (SF4S AFW) Auxiliary/Emergency Feedwater														
062 (SF6 ED AC) AC Electrical Distribution														
063 (SF6 ED DC) DC Electrical Distribution														
064 (SF6 EDG) Emergency Diesel Generator								X				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Load, VARS, pressure on air compressor, speed droop, frequency, voltage, fuel oil level, temperatures.	2.9	88

073 (SF7 PRM) Process Radiation Monitoring								X				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Erratic or failed power supply	2.9	89
076 (SF4S SW) Service Water											X	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4.2	90
078 (SF8 IAS) Instrument Air														
103 (SF5 CNT) Containment														
053 (SF1; SF4P ICS*) Integrated Control														
K/A Category Point Totals:								3			2	Group Point Total:		5

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive														
002 (SF2; SF4P RCS) Reactor Coolant														
011 (SF2 PZR LCS) Pressurizer Level Control											X	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4.1	91
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation														
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor											X	2.1.20 Ability to interpret and execute procedure steps.	4.9	92
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control														
029 (SF8 CPS) Containment Purge														
033 (SF8 SFPCS) Spent Fuel Pool Cooling														
034 (SF8 FHS) Fuel-Handling Equipment														
035 (SF 4P SG) Steam Generator														
041 (SF4S SDS) Steam Dump/Turbine Bypass Control														
045 (SF 4S MTG) Main Turbine Generator														
055 (SF4S CARS) Condenser Air Removal														
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring														
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air														
086 Fire Protection								X				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.01 Manual shutdown of the FPS	3.1	93
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:								1			2	Group Point Total:		3

Facility: Arkansas Nuclear One, Unit 2 Date of Exam:

Category	K/A #	Topic	RO IR	RO #	SRO-only IR	SRO-only #
1. Conduct of Operations	2.1.28	Knowledge of the purpose and function of major system components and controls.			4.1	94
	Subtotal					
2. Equipment Control	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.			4.6	95
	2.2.25	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.			4.2	96
	Subtotal					
3. Radiation Control	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.			3.1	97
	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.			3.8	98
	Subtotal					
4. Emergency Procedures/Plan	2.4.16	Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.			4.4	99
	2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.			4.4	100
	Subtotal					
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/2	CE09 2.4.1 (Q27)	No immediate action steps in the FRP. New K/A is 2.4.6
2/1	008 K3.02 (Q33)	There is no link between CCW and CEA cooling. The new KA is 008 K3.01
2/1	010 K2.03 (Q34)	Unit 2 has no PORVs. The new K/A is 010 K2.02
2/1	078 K1.02 (Q47)	There are no cross connections between station air and instrument air in Unit 2. The new KA is 078 K1.05
2/2	001 A3.01 (Q56)	Unit 2 does not operate with control rods in automatic. New K/A is K1.05.
2/2	028 K1.01 (Q58)	Could not write a discriminating question on the interrelationship between recombiner and ventilation system. The new KA is 028 K5.03
2/2	079 A4.01 (Q65)	There are no cross connections between station air and instrument air in Unit 2. All K/As associated with station air that have importance ratings above 2.5 are associated with cross connections between station and instrument air. The new KA is 015 A4.03.
1/1	009 2.1.30 (Q76)	Cannot write an SRO level question to the KA. The new KA is 009 2.1.45.
2/2	079 A2.01 (Q93)	There are no cross connections between station air and instrument air in Unit 2. All K/As associated with station air that have importance ratings above 2.5 are associated with cross connections between station and instrument air. The new KA is 086 A2.01.

Facility: ANO-2	Date of Examination: 4/22/2019
Examination Level: RO ☒ SRO ☐	Operating Test Number: 2019-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
A1. Conduct of Operations 2.1.25 RO (3.9)	R, N	Determine flow rate using 2103.011 Draining the Reactor Coolant System Attachment K, RCS level vs RWT level. A2JPM-NRC-ADMIN-RCS
A2. Conduct of Operations 2.1.23 RO (4.3)	R, D, P	Determine time to start CNTMT evacuation and closure A2JPM-NRC-ADMIN-CNTMT2
A3. Equipment Control 2.2.23 RO (3.1)	R, M	Determine any limits for CEA positions using the COLR ANO-2-JPM-NRC-ADMIN-PDIL
A4. Radiation Control 2.3.7 RO (3.5)	R, M	Review the RWP/Survey Maps and determine RWP limits, and Dose rate work location. A2JPM-NRC-ADMIN-RWP4
Emergency Plan		

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

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

Facility: ANO-2	Date of Examination: 4/22/2019
Examination Level: RO ☐ SRO ☒	Operating Test Number: 2019-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
A5. Conduct of Operations 2.1.5 SRO (3.9)	R, M	Fatigue rule calculation. ANO-2-JPM-NRC-ADMIN-WORK2
A6. Conduct of Operations 2.1.25 SRO (4.2)	R, P	Verify RPS trip set point determination for inoperable MSSV ANO-2-JPM-NRC-MSSVINOP
A7. Equipment Control 2.2.37 SRO (4.6)	R, M	Determine operability of a safety related system. ANO-2-JPM-NRC-EFWTS2
A8. Radiation Control 2.3.6 SRO (3.8)	R, D	Review and approve Containment purge gaseous release ANO-2-JPM-NRC-ADMIN-PURGE
A9. Emergency Plan 2.4.41 SRO (4.6)	R, M	Classify an event, Time critical ANO-2-JPM-NRC-EAL16

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

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

AN()2019

Facility: ANO-2 Date of Examination: 4/22/2019
Examination Level: RO ☒ SRO ☐ Operating Test Number: 2019-1

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

System/JPM Title	Type Code*	Safety Function
S1. A2JPM-RO-VCTMU2 004 A4.07, RO-3.9 / SRO 3.7. Perform Manual Makeup to the VCT	A, M, S	1 Reactivity control
S2. A2JPM-RO-EOP08 002 A 2.01, RO-4.3 / SRO 4.4 Perform Standard Attachment 22 Isolating LOCA outside containment	A, EN, L, N, S	2 Inventory Control
S3. A2JPM-RO-PZR09 010 A4.02; RO-3.6 / SRO-3.4 Perform 2103.005 Pressurizer Operations, Attachment B, Red Train Proportional Heater Test	N, S	3 Pressure Control
S4. A2JPM-RO-RCP05 003 A2.02; RO-3.7 / SRO-3.9 Perform a normal RCP start (Alternate Path),	A, D, L, S	4 Heat Removal Primary
S5. 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
S6. A2JPM-RO-AAC01 055 EA1.06; RO-4.1 / SRO-4.5 Perform an Emergency start of the AAC Diesel from 2C-14 and energize 2A3.	D, L, S	6 Electrical
S7. A2JPM-RO-AOP06 015 A2.02; RO-3.1 / SRO-3.5 Disable B channel Excore nuclear instrumentation.	D, L, P, S	7 Instrumentation
S8. A2JPM-RO-AOP7 A13 AA1.2; RO-3.1 / SRO-3.6 Shift Gland Seal Steam to Unit 1 during Natural Circ cool down.	N, S	8 Plant Service systems

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

System/JPM Title	Type Code*	Safety Function
P1. A2JPM-RO-WGDTR 071 A2.02: RO-3.3 / SRO-3.6 Perform Waste Gas Decay Tank Release	A, D, R	9 Radioactivity Release
P2. A2JPM-RO-TLOF CE E06 EA2.2; RO-3.0 / SRO-4.2 Perform Local Actions to start ‘A’ Condensate pump during a Loss of Feedwater.	D, E, L, P	4 Heat Removal Secondary
P3. A2JPM-RO- SURV01 062 A4.04; RO-2.6 / SRO-2.7 Perform 2305.016 Remote Feature Periodic test for EDG exhaust fans	EN, N, R	6 Electrical

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

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

AN

Facility: ANO-2
Date of Examination: 4/22/2019
Examination Level: RO ☐ SRO ☒
Operating Test Number: 2019-1

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

System/JPM Title	Type Code*	Safety Function
S1. A2JPM-RO-VCTMU2 004 A4.07, RO-3.9 / SRO 3.7. Perform Manual Makeup to the VCT	A, M, S	1 Reactivity control
S2. A2JPM-RO-EOP08 002 A 2.01, RO-4.3 / SRO 4.4 Perform Standard Attachment 22 Isolating LOCA outside containment	A, EN, L, N, S	2 Inventory Control
S3.		
S4. A2JPM-RO-RCP05 003 A2.02; RO-3.7 / SRO-3.9 Perform a normal RCP start (Alternate Path),	A, D, L, S	4 Heat Removal Primary
S5.		
S6.		
S7.		
S8.		

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

P1.		
P2. A2JPM-RO-TLOF CE E06 EA2.2; RO-3.0 / SRO-4.2 Perform Local Actions to start 'D' Condensate pump during a Loss of Feedwater.	D, E, L, P	4 Heat Removal Secondary
P3. A2JPM-RO-SURV01 062 A4.04; RO-2.6 / SRO-2.7 Perform 2305.016 Remote Feature Periodic test for EDG exhaust fans	EN, N, R	6 Electrical

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

* Type Codes	Criteria for RO/SRO-I/SRO-U

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

Facility: ANO-2 Scenario No.: 1 Op-Test No.: 2019-1

Examiners: ______________________ Operators: ______________________

Initial Conditions: ~100%, MOL, RED Train Maintenance Week.

Turnover: ~100%. 260 EFPD. EOOS indicates 'Minimal Risk'. Red Train Maintenance Week. Scheduled evolution: Shift Lead Charging pumps from 2P-36C to 2P-36B using 2104.002 Chemical and Volume control section 8.1 starting with step 8.1.4.

Critical Tasks: Commence Emergency boration IAW 2202.010 Standard Attachment Exhibit 1 by the completion of SPTAs. Restore Feedwater prior to both SG levels reaching 70" wide range.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP) N (SRO)	Shift Lead Charging pumps from 2P-36C to 2P-36B. **OP-2104.002, CVCS Operations.**
2	CV4651	C (ATC) C (SRO)	'A' RCP normal spray valve drifts partially open. **OP-2203.028, Pressurizer System Malfunction AOP**
3	ESF2C40B73	C (BOP) C (SRO) TS (SRO)	Inadvertent Green Train Recirculation Actuation Signal (RAS). **OP-2203.040, Inadvertent RAS AOP.**
4	CEA43DROP	C (ATC) C (BOP) C (SRO) TS (SRO)	CEA 43 fully inserts. **OP-2203.003, CEA Malfunction AOP**
5	DI_HS_4930_1 CVC2P39ANAS CVC2P39BNAS	C (ATC) C (SRO)	2CV-4930 boration valve fails to automatically open and 2P-39A and 2P-39B boric acid makeup pumps fail to start automatically. **OP-2104.003, Chemical Addition**
6	MTGTRIPLOCKO	M (ALL)	Turbine trip causing a reactor trip. **OP-2202.001, Standard Post Trip Actions (SPTAs) EOP**
7	MFWPMPBTRP EFW2P7BFLT EFWROOMB CV0340	M (ALL)	2P-1B Main Feedwater (MFW) pump trip, 2P-7B Emergency Feedwater (EFW) motor fault, and 2P-7A steam admission valve will not open. **CT-2** **OP-2202.006, Loss of Feedwater EOP**
8	CEA02STUCK CEA07STUCK CV4873	C (ATC) C (SRO)	Control Element Assemblies (CEA's) 2 and 7 will remain withdrawn requiring emergency boration. The Volume Control Tank (VCT) outlet valve will not close. **CT-1** **OP-2202.010 Standard Attachments.**
9	AFW2P75LO	C (BOP) C (SRO)	2P-75 AFW pump trips due loss of lube oil. **OP-2202.006, Loss of Feedwater EOP**
End Point		Feedwater is restored to at least one Steam Generator	
*	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor		

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

Critical Task	Justification	
Commence Emergency Boration IAW 2202.010 Standard Attachment Exhibit 1 by the completion of SPTAs.	Meeting the SFSCs prevents core damage and minimizes radiological releases to the environment, ultimately protecting the health and safety of the public. The SFSCs assume that all but one CEA is fully inserted and that the reactor is subcritical by a certain amount (required shutdown margin or SDM).	• CE EPGB Simulator CTs: CT-01, Establish Reactivity Control (SPTA-01) • TS 3.1.1.1 Shutdown margin.
Restore Feedwater prior to both SG levels reaching 70" wide range.	Without feedwater, the SG being steamed will eventually boil dry, RCS heat removal will cease, and the reactor core will begin overheating (core melt potential). Thus, it is essential to steam and feed at least one SG to continue to remove RCS decay heat.	• CE EPGB Simulator CTs: CT-08, Establish RCS Heat Removal (LOAF-02) • EOP 2202.006 Loss of Feedwater EOP • EOP 2202.006 Loss of Feedwater EOP Tech Guide
Causing an unnecessary plant trip or ESF actuation may constitute a CT failure.	Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.	NUREG-1021 Appendix D

Scenario #1 Objectives

1) Evaluate individual ability to shift lead Charging pumps
2) Evaluate individual response to a pressurizer spray valve failing partially open.
3) Evaluate individual response to an Inadvertent Recirculation Actuation Signal (RAS).
4) Evaluate individual response to a drop Control Element Assembly (CEA)
5) Evaluate individual response to failure of interlocks for boration control.
6) Evaluate individual and crew's response to a turbine trip.
7) Evaluate individual and crew's ability to restore feedwater using Loss of Feedwater EOP.
8) Evaluate individual response to Control Element Assemblies (CEAs) failing to insert.
9) Evaluate individual response to 2P-75 Auxiliary Feedwater (AFW) pump trips.

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 should shift lead Charging pumps from 2P-36C to 2P-36B.

After the Charging pumps have been shifted, and cued by lead examiner, 2CV-4652 RCP spray valve will open to ~40%. The ATC should recognize RCS pressure lowering and 2CV-4652 intermediate indication. The SRO will enter the Pressurizer Malfunctions AOP OP-2203.028 and direct actions to attempt to close 2CV-4652 Spray valve and then isolate 2CV-4652 using the PZR spray block valves. [Industry OE: SER 4-93 RCS pressure transients caused by failed open pressurizer spray valves.].

When the crew has isolated the failed Pressurizer Spray valve, an inadvertent Green Train Recirculation Actuation Signal (RAS) will occur. The SRO should enter and commence taking action of the Inadvertent RAS AOP. The BOP will override and close the inside CNTMT sump suction isolation valve. The crew will check that Service Water is still aligned to Component Cooling Water (CCW) and Auxiliary Cooling Water (ACW). The BOP will place green train ECCS pumps in PTL. The SRO should also enter Tech Spec 3.6.2.1 for CNTMT spray and 3.5.2 for ECCS components. The SRO may have to enter Tech Spec 3.5.4 and TRM 3.1.8 for RWT level. [Industry OE: SEN 268 Invalid Safety Injection with Failure to Reset, Site OE: CR-ANO-2-2013-005 Inadvertent SIAS, CCAS, And CIAS.]

Scenario #1 NARRATIVE (continued)

When the crew has closed the CNTMT sump suction valve and entered the appropriate Tech Specs or at the lead examiner's cue, 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). When the ATC starts boration, 2CV-4930 boration valve will fail to automatically open and 2P-39A boric acid makeup (BAM) pump will fail to automatically start. The ATC will manually start the BAM pump and open 2CV-4930 boration valve. [Site and industry OE: CR-ANO-2-2007-0127 dropped CEA, and NRC Event # 49601 Palo Verde dropped CEA.]

When the crew has commenced a plant shutdown, entered the appropriate Tech Spec or cued by the lead examiner, the turbine will trip. The SRO will direct the reactor to be tripped, due to RCS pressure rising. The Reactor may trip automatically prior to the crew manually tripping the reactor. The SRO should enter and direct the actions of SPTAs. Two CEAs will remain withdrawn and the ATC will commence emergency boration to maintain Shutdown Margin. When the ATC attempt gravity feed boration the VCT outlet will fail to close requiring use of the Boric Acid Make-up pumps. When EFAS is actuated 2P-7B EFW pump flange will wet the motor and cause a motor fault. 2P-7A steam driven EFW pump steam admission valve 2CV-0340-2 will be bound and not open. 2P-1A MFW pump will trip due to being interlocked with the turbine trip. Also, 2P-1B MFW pump will trip causing a loss of feedwater. The SRO should diagnose and enter Loss of Feedwater EOP. [Site OE: CR-ANO-2-2002-2173, Reactor Trip due to turbine trip. Industry OE: SEN134 Failure of Control Rods to Fully insert.]

The SRO will complete the initial actions of the Loss of Feedwater EOP to conserve inventory, then determine that AFW is the highest prioritized source of feedwater. The BOP will start the AFW pump and it will trip based on a loss of Lube Oil. The crew will transition to the next highest prioritized source of feedwater Common Feedwater (CFW) and restore feedwater using a CFW pump. [Industry OE: SOER 86-01 Reliability of PWR Auxiliary Feedwater systems][PSA Action of failure to establish flow from auxiliary feedwater pump and PRA action to align CFW to the SGs. PSA-ANO2-06-05, PRA-A2-05-004 Rev. 3]

Facility: ANO-2 Scenario No.: 2 Op-Test No.: 2019-1

Examiners: ______________ Operators: ______________

Initial Conditions: ~100%, MOL, RED Train Maintenance Week.

Turnover: ~100%. 260 EFPD. EOOS indicates 'Minimal Risk'. Red Train Maintenance Week. Scheduled evolution: None

Critical Tasks: Manually trip the reactor within 1 minute of 'A' RCP trip. All RCPs must be secured within 10 min of RCS margin to saturation remaining below minimum NPSH for RCPs (<30 degrees MTS). And Safety injection flow must be restored prior to RVLMS level 4.

Event No.	Malf. No.	Event Type*	Event Description
1	NIBUPPER	C (BOP) C (SRO) TS (SRO)	'B' channel Excore upper chamber fails high. **OP-2203.026, NI malfunction AOP.**
2	XCVLDNHXOU	I (ATC) I (SRO)	The temperature input to the letdown HX temperature controller (2TIC-4815) fails low. **OP-2203.012L, Annunciator 2K-12 Corrective Action (ACA)**
3	SGBTUBE	C (ATC) C (BOP) C (SRO) TS (SRO)	'B' SG tube leak. **OP-2203.038, Primary to Secondary leakage AOP.**
4	IAINSTAIR K12-B08	C (BOP) C (SRO)	Loss of Instrument Air. **OP-2203.021, Loss of Instrument Air AOP**
5	RCP2P32AGRN RPSRXAUTO RPSRXMAN	C (ATC) C (SRO)	'A' RCP Trip and RPS will not auto or manually trip the reactor. **CT-2** **OP-2202.001, Standard Post Trip Actions (SPTAs) EOP**
6	RCSLOCATCA	M (All)	Loss of Coolant accident. **CT-3** **OP-2202.009, Functional Recovery EOP.**
7	HPI2P89AFAL ESFK409BAF	C (BOP) C (SRO)	2P89A HPSI pump fails to start on SIAS. 2CV-5076-2 High pressure safety injection and 2CV-5077-2 Low pressure safety injection valves fail to open. **CT-1** **OP-2202.010 Standard Attachments EOP.**
End Point			After the crew has completed the entry section of Functional Recovery and restored Safety injection flow

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

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

Critical Task	Justification	References
Perform one or more of the following to establish minimum design safety injection flow. • Start 'A' or 'C' HPSI pump. • Open Green train HPSI valve 2CV-5076-2. Safety injection flow must be restored prior to RVLMS level 4.	During a loss of inventory, SI flow keeps the core covered, cooled, and borated. The inability to maintain minimum required SI flow could result in a net loss of RCS inventory, pressure control, and sub-cooling. Once sub-cooling is lost, pressurizer level is no longer a valid indication of RCS mass inventory, and a reactor head void can form, both of which complicate the event recovery. RVLMS level 3 or higher has to be maintained to ensure Natural Circulation.	• CE EPGB Simulator CTs: CT-16, Establish required SI flow (IC-03)
Manually trip the reactor within 1 minute of 'A' RCP trip.	Following a reactor trip, safety systems are designed to keep the plant in a safe state by meeting specified critical safety function criteria (SFSC). If the heat being generated by the reactor is greater than normal decay heat levels, then the heat removal capacity of the safety systems may be inadequate resulting in core damage.	• CE EPGB Simulator CTs: CT-01, Establish reactivity control (SPTA-01) • CR-ANO-2-2010-948, Critical task criteria
All RCPs must be secured within 10 min of RCS margin to saturation remaining below minimum NPSH for RCPs (<30 degrees MTS).	The out-of-limits condition could result in shaft seal damage, and then shaft seal failure could result in increased RCS leakage out the seal to the containment atmosphere, which would worsen the event severity.	• EN-OP-123 Time Critical Action/Time sensitive action program Attachment 4. • CE EPGB Simulator CTs: CT-23, Trip any RCP exceeding operating limits. (FRG-04) • CR-ANO-2-2010-948, Critical task criteria
Causing an unnecessary plant trip or ESF actuation may constitute a CT failure.	Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.	NUREG-1021 Appendix D

Scenario #2 Objectives

1) Evaluate individual response to a failure of a Nuclear Instrument.
2) Evaluate individual response to the Letdown temperature controller.
3) Evaluate individual response to a Steam Generator Tube leak.
4) Evaluate individual response to a Loss of Instrument air.
5) Evaluate individual and crew's response to Reactor Coolant Pump trip without a reactor trip.
6) Evaluate crew's ability to mitigate a Loss of Coolant Accident.
7) Evaluate individual response to ECCS component failures.

Scenario #2 NARRATIVE

When the crew has completed their control room walk down and brief, 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 2 for Reactor Protection System. The BOP will bypass points 1, 3, and 4 on channel 'B' channel PPS. [Site OE: CR-ANO-2-2002-693, D Excore failure.]

After the 'B' channel PPS points are placed in bypass or cued by lead examiner, the Letdown heat exchanger temperature input will fail low. The ATC will report that 2K12-C1 LETDOWN HX 2E29 OUTLET TEMP HI alarm is in and the letdown heat exchanger temperature is reading low on the hand indicating controller but the computer point and control board indication are reading higher than normal. The SRO will direct the ATC to take manual control of the Letdown heat exchanger temperature control valve and manually control temperature. The SRO will also refer to the ACA for letdown radiation monitor flow low 2K12 J1 RADMONITOR FLOW LO and restore letdown radiation monitor flow. [Site OE: CR-ANO-2-2018-0812, 2TIC-4815 letdown temperature controller failed to 50 degrees.]

After the ATC has taken manual control of the letdown temperature control valve, or at the lead examiner's cue, a Steam Generator (SG) Tube Leak will occur on 'A' Steam Generator. The SRO will enter OP 2203.038, Primary to Secondary Leakage AOP. The SRO will direct the ATC to perform power reduction to take the unit offline. He will also direct the BOP to isolate steam to 'A' EFW pump from the 'A' steam generator. The SRO will enter TS 3.4.6.2 Action a, RCS leakage, 3.4.5 SG tube integrity, and TS 3.7.1.2 for EFW action a when steam is isolated to 2P-7A EFW pump. [Industry OE: SOER 83-2, Steam Generator Tube Ruptures.]

After the crew has started the down power, or cued by the lead examiner, an Instrument Air (IA) dryer will malfunction. This will cause a lowering of IA header pressure. The SRO should enter the OP-2203.021, Loss of Instrument Air AOP. The BOP will check IA crosstie with Unit 1 2CV-3015 open and will open IA crosstie with Unit 1 2CV-3004. The crew should dispatch the NLO to the IA compressors, air dryers, and to look for a leak. After the NLO report the crew should bypass the air dryer. . [Industry OE: Loss of IA, SOER 88-1 Instrument Air system failures, Braidwood Unit1 poor solder joint INPO OE# 287448. Site OE: CR-ANO-2-2014-02501 Instrument air dryer malfunction.]

After the crew has cross tied IA with Unit 1, ‘A’ Reactor Coolant Pump (RCP) will trip which should cause an automatic reactor trip. RPS will not function requiring a manual reactor trip from the Diverse Scram System (DSS). The crew will trip the reactor. After the reactor is tripped, a Large Break LOCA will occur. The SRO will enter OP 2202.001, Standard Post Trip Actions (SPTAs). The crew should recognize the signs of LOCA and ensure Safety Injection Actuation Signal (SIAS) and Containment Cooling Actuation Signal (CCAS) actuated. The SRO should diagnose and enter OP-2202.009, Functional Recovery EOP due to the Steam Generator Tube leak and the Large Break LOCA. The BOP should recognize that two Safety Injection valves failed to open and open them. The ‘A’ High Pressure Safety Injection (HPSI) pump will fail to start and the BOP should start the ‘A’ or ‘C’ HPSI pump. After the crew has entered the Functional Recovery EOP, the crew will commence mitigating actions. [Industry OE: SEN-220, SEN-216, & SEN-182, RCS leakage events.]

Facility: ANO-2 Scenario No.: 3 Op-Test No.: 2019-1

Examiners: ______ Operators: ______

Initial Conditions: ~50%, MOL, Green Train Maintenance Week.

Turnover: 49 to 51% due 500KV line maintenance (Pleasant Hills). 260 EFPD. EOOS indicates 'Minimal Risk'. Red Train Maintenance Week.

Scheduled evolution: Perform Quarterly Red Train Containment isolation valve stroke test for 2CV-2201-2 section 2.8.3.

Critical Tasks: RCS CETs must be limited to less than 80 degree F heatup. EFW must be isolated to 'A' SG to prevent MTS exceeding 200 °F. Restore CCW to RCPs within 10 min of the loss of CCW cooling or secure the RCPs within the next 10 min.

Event No.	Malf. No.	Event Type*	Event Description
1	CV22012	N (BOP) N (SRO) TS (SRO)	Complete Quarterly Red Train Containment isolation valve stroke test for 2CV-2201-2. **OP-2305.005, Valve Stroke and position verification.**
2	XCV2LT4861	I (ATC) I (SRO)	Volume Control Tank level instrument fails low resulting in Refueling Water Tank being aligned to Coolant Charging Pump suction. **OP-2203.012L Annunciator 2K12 Corrective Action.**
3	CT2VSF1B	C (BOP) C (SRO) TS (SRO)	Containment Cooler 2VSF-1B trips. **OP-2203.012F Annunciator 2K06 Corrective Action** **OP-2104.033 Containment Atmosphere control**
4	XRRPZRLSP	I (ATC) I (SRO)	Reactor Reg. output to PZR level control program fails to 41%. **OP-2203.028, Pressurizer System Malfunction AOP**
5	CNDAIRLEAKHI CND2C5	C (ATC) C (BOP) C (SRO)	Condenser Air in leakage and backup vacuum pump fails to auto start. **OP-2203.019, Loss of Condenser Vacuum AOP**
6	CNDAIRLEAKHI	M (ALL)	Condenser Air-in-leakage degrades requiring a trip. **OP-2202.001, Standard Post Trip Actions (SPTAs) EOP**
7	FW2FW5AAFT	M (ALL)	Excess Steam Demand (ESD) on 'A' SG inside containment due to Feedwater line break. **CT-1, CT-3** **OP-2202.005, Excess Steam Demand.**
8	CV10251 CV10382	C (BOP) C (SRO)	Emergency Feedwater (EFW) valves to A SG fail to close. **CT-2** **OP-2202.001, Standard Post Trip Actions (SPTAs) EOP or OP-2202.005, Excess Steam Demand.**
End Point		Post blow down RCS conditions have be stabilized.	

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

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

Critical Task	Justification	
Stabilize and control RCS temperature after the ESD blowdown terminates. RCS CETs must be limited to less than 80 degree F heatup.	Rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates do not exceed the design assumptions and satisfy the stress limits for cyclic operation. Also, 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) • TS 3.4.9.1 RCS Pressure/Temperature Limits
Maintain RCS pressure within the Pressure-Temperature limits of 200°F and 30°F Margin to Saturation (MTS) throughout implementation of SPTAs and Excess Steam Demand EOP. EFW must be isolated to ‘A’ SG to prevent MTS exceeding 200 °F	RCS pressure must be maintained in these limits to allow natural circulation of the RCS and prevent over pressurizing the RCS boundary.	• CE EPGB Simulator CTs: CT-06, Establish RCS Pressure Control (SPTA-05, ESDE-07) • EOP 2202.005 Excess Steam Demand EOP.
Restore CCW to RCPs within 10 min of the loss of CCW cooling or secure the RCPs within the next 10 min.	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.	• EN-OP-123 Time Critical Action/Time Sensitive Action Program. • CE EPGB Simulator CTs: CT-23, Trip any RCP exceeding operating limits (ESDE-03)
Causing an unnecessary plant trip or ESF actuation may constitute a CT failure.	Actions taken by the applicant(s) will be validated using the methodology for critical tasks in Appendix D to NUREG-1021.	NUREG-1021 Appendix D

Scenario #3 Objectives

1) Evaluate individual ability to perform quarterly valve stroke surveillance.
2) Evaluate individual response to a failure of Reactor Regulating system input to PZR level setpoint failing low.
3) Evaluate individual response to a trip of the Containment Cooler.
4) Evaluate individual response to failure of a VCT level transmitter.
5) Evaluate individual response to Condenser Air in-leakage.
6) Evaluate individual ability to perform a power reduction.
7) Evaluate crew's and individual ability to perform standard post trip actions.
8) Evaluate crew's ability to respond to an Excess Steam Demand.
9) Evaluate individual response to a failure of EFW.

Scenario #3 NARRATIVE

Simulator session begins with the plant at 49-51% power steady state due to 500KV line maintenance.

When the crew has completed their control room walk down and brief, the BOP will commence the stroke test of 2CV-2201-2. The valve will fall outside of the required stroke time and this will require the SRO to enter TS 3.6.3.1 Containment isolation valves.

When the crew has performed the completed the valve stroke and enter TS 3.6.3.1 or at the lead examiner's cue, one of the Volume Control Tank level transmitters, 2LT-4861, will fail low. The crew will respond to VCT low low level alarm, 2K12 G5. This will result in the VCT outlet valve to the charging pump suction closing and the Refueling Water tank (RWT) suction to the charging pumps opening. RCS temperature and pressure will lower due to boration until the ATC opens VCT outlet valve manually and closes the RWT valve manually. [Site OE: CR-ANO-2-2000-0199, VCT level transmitter spiking]

After the Crew has realigned Charging pump suction to the VCT or at the lead examiner's cue, 2VSF-1B containment cooler will trip. The BOP will determine that 2VSF-1B containment cooler has tripped and refer to OP-2203.012F/G, 2K06 and 2K07 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.]

SCENARIO #3 NARRATIVE (continued)

When the crew has placed the idle Containment cooler in service or at the lead examiner's cue, the Reactor Regulating system 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 the ATC has placed letdown in automatic or at the lead examiner's cue, a condenser air leak will start. The crew will recognize the degrading condenser vacuum and enter the Loss of Condenser Vacuum AOP. The BOP will ensure both vacuum pumps are running. The crew will direct a NLO to locally investigate both Vacuum pumps and place the Vacuum pumps in the hogging mode (raising vacuum pump air removal capacity). The crew will investigate for the source of leakage into the condenser. When it is determined that condenser pressure is continuing to slowly degrade the crew will commence a power reduction. [Site OE: CR-ANO-2-2008-1350, Loss of Condenser Vacuum due to manway leak, CR-ANO-2-2003-1916 Loss of Condenser Vacuum due to dog bone seal leak.]

After the crew has commenced a power reduction, or at the lead examiner's discretion, the condenser air leakage will degrade causing the crew to manual trip the reactor. The crew will commence SPTAs. After the Reactor trips, an Excess Steam Demand (ESD) will occur due to an 'A' Main Feedwater line break inside containment. The Crew will recognize the ESD and manually actuate Main Steam Isolation Signal (MSIS) or verify that a Main Steam Isolation signal automatically actuates. The ATC will secure all RCPs when Containment Spray Actuation Signal (CSAS) actuates. This will cause the crew to respond to a natural circulation ESD. The SRO will diagnose Excess Steam Demand (ESD) EOP 2202.005. The SRO will direct the BOP to maintain post blowdown temperature and the ATC to maintain post blowdown RCS pressure. The crew will restore Service Water to Component Cooling Water. [PRA item # 9 restore Service Water to CCW] [Industry OE for Excess Steam Demand, SOER 82-7, Reactor Vessel Pressurized Thermal Shock.][PSA action to restore SW to CCW. PSA-ANO2-06-05]

When EFAS actuates to the 'A' SG the EFW block valves will fail to close. This will make the overcooling event more severe and could lead to the crew exceeding the PT limits of 200 degrees Margin to Saturation (MTS). The crew should recognize that EFW is feeding the faulted SG and secure feeding 2P-7B EFW pump.