Facility: CPNPP Date of Exam: June 20, 2019

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	2	4	3				3	3			3	18			6
	2	1	1	1		N/A		2	2	N/A		2	9			4
	Tier Totals	3	5	4				5	5			5	27			10
2. Plant Systems	1	3	2	3	3	2	2	3	3	2	2	3	28			5
	2	1	1	1	1	1	1	0	1	1	1	1	10			3
	Tier Totals	4	3	4	4	3	3	3	4	3	3	4	38			8

3. Generic Knowledge and Abilities Categories	1	2	3	4	Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	3	3	2	2	10					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 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 Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO)								Form ES-401-2
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7) Reactor Trip, Stabilization, Recovery / 1		X					Knowledge of the interrelations between a reactor trip and the following (CFR 41.7 / 45.7) EK2.02 Breakers, relays, and disconnects	2.6	11
000008 (APE 8) Pressurizer Vapor Space Accident / 3					X		Ability to determine and interpret the following as they apply to the pressurizer vapor space accident (CFR 43.5 / 45.13) AA2.25 Expected leak rate from and open PORV or code safety	2.8	12
000009 (EPE 9) Small Break LOCA / 3						X	(CFR 41.10 / 43.1 / 45.13) G.2.4.18 Knowledge of the specific bases for EOPs	3.3	13
000011 (EPE 11) Large Break LOCA / 3		X					Knowledge of the interrelations between the and the following Large Break LOCA: (CFR 41.7 / 45.7) EK2.02 Pumps	2.6*	14
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4			X				Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump malfunctions (Loss of RC Flow) (CFR 41.5 / 41.10 / 45.6 / 45.13) AK3.07 Ensuring that S/G levels are controlled properly for natural circulation enhancement	4.1	15
000022 (APE 22) Loss of Reactor Coolant Makeup / 2				X			Ability to operate and/or monitor the following as they apply to Loss of reactor Coolant Makeup (CFR 41.7 / 45.5 / 45.6) AA1.08 VCT level	3.4	16
000025 (APE 25) Loss of Residual Heat Removal System / 4						X	(CFR 41.10 / 43.5 / 45.12) G2.1.20 Ability to interpret and execute procedure steps	4.6	17
000026 (APE 26) Loss of Component Cooling Water / 8			X				Knowledge of the reasons for the following responses as they apply to Loss of Component Cooling Water (CFR 41.5 / 41.10 / 45.6 / 45.13) AK3.03 Guidance actions contained in EOP for Loss of CCW	4.0	18
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3									
000029 (EPE 29) Anticipated Transient Without Scram / 1				X			Ability to operate and/or monitor the following as they apply to ATWS (41.7 / 45.5 / 45.6) EA1.09 Manual Rod Control	4.0	19
000038 (EPE 38) Steam Generator Tube Rupture / 3	X						Knowledge of the operational implications of the following as they apply to a SGTR (CFR 41.8 / 41.10 / 45.3) EK1.01 Use of Steam tables	3.1	20
000040 (APE 40; W E12) Steam Line Rupture—Excessive Heat Transfer Uncontrolled Depressurization of all Steam Generators / 4		X					APE 040 Knowledge of the interrelations between the Steam Line Rupture and the following (CFR 41.5 / 41.10 / 45.6 / 45.13) AK2.02 Sensors and detectors	2.6	21
000054 (APE 54;) Loss of Main Feedwater /4									
000055 (EPE 55) Station Blackout / 6	X						Knowledge of the operational implications of the following as they apply to the Station Blackout (CFR 41.8 / 41.10 / 45.3) EK1.01 Effect of battery discharge rates on capacity	3.3	22
000056 (APE 56) Loss of Offsite Power / 6					X		Ability to determine and interpret the following as they apply to Loss of offsite power (CFR 43.5 / 45.13) EA2.02 Load sequencer status lights	3.5	23

000057 (APE 57) Loss of Vital AC Instrument Bus / 6				X			Ability to operate and/or monitor the following as they apply to Loss of vital AC instrument bus (41.7 / 45.5 / 45.6) AA1.05 backup instrument indications	3.2	24
000058 (APE 58) Loss of DC Power / 6			X				Knowledge of the reasons for the following responses as they apply to Loss of DC power (CFR 41.5 / 41.10 / 45.6 / 45.13) AK3.02 Actions contained in the EOP for loss of DC power	4.0	25
000062 (APE 62) Loss of Nuclear Service Water / 4									
000065 (APE 65) Loss of Instrument Air / 8					X		Ability to determine and interpret the following as they apply to Loss of instrument air (CFR 43.5 / 45.13) AA2.05 When to commence a plant shutdown if instrument air pressure is decreasing	3.4	26
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6						X	(CFR 41.10 / 41.12 / 43.5 / 45.11) G.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	27
(W E04) LOCA Outside Containment / 3									
(W E11) Loss of Emergency Coolant Recirculation / 4		X					Knowledge of the interrelations between the loss of emergency coolant recirculation and the following: (CFR 41.7 / 45.7) EK2.2 Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	3.9	28
W E05 Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4									
K/A Category Totals:	2	4	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)									
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						Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rod: (CFR 41.8 / 41.10 / 45.3) AK1.04 Effects of power level and control position on flux	3.1	47
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2					X		Ability to determine and interpret the following as they apply to a Pressurizer Level Control Malfunctions: (CFR 43.5 / 45.13) AA2.13 The actual PZR level, given uncompensated level with an appropriate graph	2.9	48
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7		X					Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: (CFR 41.7 / 45.7) AK2.01 Power supplies, including proper switch positions	2.7	49
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7									
000036 (APE 36) 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			X				Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release (CFR 41.5, 41.10 / 45.6 / 45.13) EK3.01 Termination of a release of radioactive liquid	3.5	50
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					X		Ability to determine and interpret the following as they apply to Plant Fire onsite: (CFR 43.5 / 45.13) AA2.05 Ventilation alignment necessary to secure affected area	3.2	51
000068 (APE 68; BW A06) Control Room Evacuation / 8									
000069 (APE 69; W E14) Loss of Containment Integrity / 5						X	(CFR 41.7 / 41.10 / 43.2 / 45.13) G.2.2.39 Knowledge of less than or equal to one-hour Technical Specification action statements for systems	3.9	52

000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4				X			0074 Ability to operate and monitor the following as they apply to Inadequate Core Cooling: (CFR 41.7 / 45.5 / 45.6) EA1.16 RCS in-core thermocouple indicators	4.4	53
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				X			Ability to operate and / or monitor the following as they apply to Steam Generator Overpressure (CFR 41.7 / 45.5 / 45.6) EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and manual and automatic features.	3.1	54
(W E15) Containment Flooding / 5						X	(CFR 41.10 / 43.5 / 45.13) G.2.4.6 Knowledge of EOP mitigation strategies	3.7	55
(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									
(W E09 & E10) Natural Circulation/4									
(BW E13 & E14) EOP Rules and Enclosures									
(W E08) RCS Overcooling-Pressurized Thermal Shock / 4									
(CE A16) Excess RCS Leakage / 2									
(CE E09) Functional Recovery									
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
K/A Category Point Totals:	1	1	1	2	2	2	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 (SF4P RCP) Reactor Coolant Pump										X		Ability to manually operate and / or monitor in the control room: (CFR 41.7 / 45.5 to 45.8) A4.02 RCP motor parameters	2.9	1
004 (SF1; SF2 CVCS) Chemical and Volume Control									X			SF1 Ability to monitor automatic operation of the CVCS, including” (CFR 41.7 / 45.5) A3.04 VCT pressure control	2.8	2
							X					SF2 Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls, including: (CFR 41.5 / 45.5) A1.07 Maximum specified letdown flow	2.7	3
005 (SF4P RHR) Residual Heat Removal				X								Knowledge of the RHRS design feature(s) and/or interlock(s) which provide for the following: (CFR 41.7) K4.01 Overpressure mitigation system	3.0	4
							X					Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls, including: (CFR 41.5 / 45.5) A1.02 RHR flow rate	3.3	5
006 (SF2; SF3 ECCS) Emergency Core Cooling								X				Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations (CFR 41.5 / 45.5) A2.13 Inadvertent SIS actuation	3.9	6
007 (SF5 PRTS) Pressurizer Relief/Quench Tank	X											Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: (CFR 41.2 to 41.9 / 45.7 to 45.8) K1.01 Containment System	2.9	7

008 (SF8 CCW) Component Cooling Water	X											Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: (CFR 41.2 to 41.9 / 45.7 to 45.9) K1.04 RCS, in order to determine source(s) of RCS leakage into the CCWS	3.3	8
			X									Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: (CFR 41.2 to 41.9 / 45.7 to 45.9) K3.01 Loads cooled by CCWS	3.4	9
010 (SF3 PZR PCS) Pressurizer Pressure Control						X						Knowledge of the effect that a loss or malfunction of the following will have on the PZR PCS: (CFR 41.7 / 45.5) K6.04 PRT	2.9	10
012 (SF7 RPS) Reactor Protection		X										Knowledge of bus power supplies to the following: (CFR 41.7 / 45.6) K2.01 RPS channels, components, and interconnections	3.3	29
											X	(CFR 41.10 / 43.2 / 45.12) G.2.1.32 Ability to explain and apply system limits and precautions	3.8	30
013 (SF2 ESFAS) Engineered Safety Features Actuation				X								Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: (CFR 41.7) K4.03 Main Steam Isolation System	3.9	31
					X							Knowledge of the operational implications of the following concepts as they apply to ESFAS: (CFR 41.5 / 45.7) K5.02 Safety system logic and reliability	2.9	32
022 (SF5 CCS) Containment Cooling											X	(CFR 41.5 / 43.5 / 45.12) G.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	4.2	33
025 (SF5 ICE) Ice Condenser												Not part of CP design		
026 (SF5 CSS) Containment Spray											X	(CFR 41.10 / 43.2 / 45.12) G.2.1.32 Ability to explain and apply system limits and precautions	3.8	34
039 (SF4S MSS) Main and Reheat Steam					X							Knowledge of the operational implications of the following concepts as they apply to MRSS: (CFR 41.5 / 45.7) K5.08 Effect of steam removal on reactivity	3.6	35

059 (SF4S MFW) Main Feedwater			X									Knowledge of the effect that a loss or malfunction of the MFWS will have on the following: (CFR 41.7 / 45.6) K3.02 AFW system	3.6	36
061 (SF4S AFW) Auxiliary/Emergency Feedwater		X										Knowledge of bus power supplies to the following: (CFR 41.7) K2.01 AFW system MOVs	3.2	37
								X				Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR 41.5 / 43.5 / 45.3 / 45.13) A2.04 pump failure or improper operation	3.4	38
062 (SF6 ED AC) AC Electrical Distribution			X									Knowledge of the effect that a loss or malfunction of the AC distribution system will have on the following: (CFR 41.7 / 45.6) K3.01 Major system loads	3.5	39
				X								Knowledge of AC distribution system design feature(s) and/or interlock(s) which provide for the following: (CFR 41.7) K4.03 Interlocks between automatic bus transfer and breakers	2.8*	40
063 (SF6 ED DC) DC Electrical Distribution	X											Knowledge of the physical connections and/or cause-effect relationships between the dc distribution system and the following systems: (CFR 41.2 to 41.9 / 45.7 to 45.8) K1.02 AC electrical system	2.7	41
064 (SF6 EDG) Emergency Diesel Generator						X						Knowledge of the effect that a loss or malfunction of the following will have on the EDG: (CFR 41.7 / 45.7) K6.08 Fuel oil storage tanks	3.2	42
073 (SF7 PRM) Process Radiation Monitoring							X					Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRMS controls, including: (CFR 41.5 / 45.7) A1.01 radiation levels	3.2	43
076 (SF4S SW) Service Water								X				Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR 41.5 / 43.5 / 45.3 / 45.13) A2.02 Service water header pressure	2.7	44

078 (SF8 IAS) Instrument Air									X			Ability to monitor automatic operation of the IAS, including: (CFR 41.7 / 45.5) A3.01 Air pressure	3.1	45
103 (SF5 CNT) Containment										X		Ability to manually operate and / or monitor in the control room: (CFR 41.7 / 45.5 to 45.8) A4.01 Flow control, pressure control, and temperature control valves, including pneumatic valve controller	3.2	46
053 (SF1; SF4P ICS*) Integrated Control												Not part of CP design		
K/A Category Point Totals:	3	2	3	3	2	2	3	3	2	2	3	Group Point Total:		28

ES-401												PWR Examination Outline Plant Systems—Tier 2/Group 2 (RO)		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			X									Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: (CFR 41.7 / 45.6) K3.02 RCS	3.4	56
002 (SF2; SF4P RCS) Reactor Coolant				X								SF4P Knowledge of RCS design feature(s) and/or interlock(s) which provide for the following: (CFR 41.7) K4.03 Venting the RCS	2.9	57
011 (SF2 PZR LCS) Pressurizer Level Control								X				Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR 41.5 / 43.5 / 45.3 / 45.13) A2.02 Excessive Charging	3.2	58
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation		X										Knowledge of bus power supplies to the following: (CFR 41.7) K2.01 NIS channels, components, and interconnections	3.3	59
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In Core Temperature Monitor						X						Knowledge of the effect that a loss or malfunction of the following will have on the ITM: (CFR 41.7 / 45.7) K6.01 Sensors and detectors	2.7	60
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					X							Knowledge of the operational implications of the following concepts as they apply to SDS: (CFR 41.5 / 45.7) K5.05 basis for RCS pressure design limits	2.6	61

045 (SF 4S MTG) Main Turbine Generator									X			Ability to monitor automatic operation of the MT/G system, including: (CFR 41.7 / 45.5) A3.05 Electrohydraulic control	2.6	62
055 (SF4S CARS) Condenser Air Removal	X											Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: (CFR 41.2 to 41.9 / 45.7 to 45.8) K1.06 PRM system	2.6	63
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring										X		Ability to manually operate and / or monitor in the control room: (CFR 41.7 / 45.5 to 45.8) A4.02 Major components	2.5*	64
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air											X	(CFR 41.5 / 43.5 / 45.12) G2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	65
086 Fire Protection														
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:	1	1	1	1	1	1	0	1	1	1	1	Group Point Total:		10

Facility: CPNPP		Date of Exam: June 20, 2019				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements (41.10 / 45.13)	3.8	66		
	2.1.9	Ability to direct personnel activities inside the control room (41.10 / 45.5 / 45.12 / 45.13)	2.9*	67		
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management (41.1 / 43.6 / 45.6)	4.3	68		
	Subtotal					
2. Equipment Control	2.2.13	Knowledge of tagging and clearance procedures (41.10 / 45.13)	4.1	69		
	2.2.14	Knowledge of the process for controlling equipment configuration or status (41.10 / 43.3 / 45.13)	3.9	70		
	2.2.21	Knowledge of pre and post maintenance operability requirements (41.10 / 43.2)	2.9	71		
	Subtotal					
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions (41.12 / 43.4 / 45.10)	3.2	72		
	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities (41.12 / 43.4 / 45.10)	3.4	73		
	Subtotal					
4. Emergency Procedures/Plan	2.4.14	Knowledge of general guidelines for EOP usage (41.10 / 45.13)	3.8	74		
	2.4.3	Ability to identify post-accident instrumentation. (41.6 / 45.4)	3.7	75		
	Subtotal					
Tier 3 Point Total				10		7

Facility: CPNPP Date of Exam: June 20, 2019

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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	2	3	5
	2												10	1 1	1	3
	Tier Totals												38	4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1: 2, 2: 2	3: 1, 4: 2	7

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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.)
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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 (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									
000011 (EPE 11) Large Break LOCA / 3									
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4									
000022 (APE 22) Loss of Reactor Coolant Makeup / 2									
000025 (APE 25) Loss of Residual Heat Removal System / 4					X		Ability to determine or interpret the following as they apply to a loss of RHR system (CFR 43.5 / 45.13) AA2.04 Location and isolability of leaks	3.6	76
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					X		Ability to determine and interpret the following as they apply to the Anticipated Transient Without SCRAM or TRIP: (CFR 43.5/ 45.13) EA2.01 Reactor nuclear instrumentation	4.7	77
000038 (EPE 38) Steam Generator Tube Rupture / 3									
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture-Excessive Heat Transfer / 4						X	G2.2.38 Knowledge of conditions and limitations in the facility license (CFR 41.7/41.10/43.1/45.13)	4.5	78
000054 (APE 54; CE E06) Loss of Main Feedwater /4									
000055 (EPE 55) Station Blackout / 6					X		Ability to determine and interpret the following as they apply to the Station Blackout: (CFR 43.5/ 45.13) EA2.03 Actions necessary to restore power	4.7	79
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									
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						X	G2.2.40 Ability to apply Technical Specifications for a system (CFR: 41.10 / 43.2 / 43.5 / 45.3)	4.7	80
(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						X	G2.4.6 Knowledge of EOP mitigation strategies CFR (41.10 / 43.5 / 45.13)	4.7	81
K/A Category Totals:					3	3	Group Point Total:		6

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (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	G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits CFR (41.5 / 41.7 / 43.2)	4.2	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						X	G2.2.3 Knowledge of the design, procedural, and operational differences between units CFR (41.5 to 41.7 / 41.10 / 45.12)	3.9	83
000074 (EPE 74; W E06 & **E07**) Inadequate Core Cooling / 4					X		WE07 Ability to determine and interpret the following as they apply to the (Saturated or Inadequate Core Cooling): CFR (43.5 / 45.13) EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.0	84
000076 (APE 76) High Reactor Coolant Activity / 9									
000078 (APE 78*) RCS Leak / 3									
(W/E01 & W/E02) Rediagnosis & **SI Termination** / 3					X		W/E02 Ability to determine and interpret the following as they apply to SI Termination: CFR (43.5 / 45.13) EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	4.0	85
(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									
(CE A16) Excess RCS Leakage / 2									
(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 Form ES-401-2

Plant Systems—Tier 2/Group 1 (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														
004 (SF1; SF2 CVCS) Chemical and Volume Control								X				Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: (CFR: 41.5 / 43.5 / 45.3 / 45.5) A2.09 High primary and/or secondary activity	3.9	86
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											X	G2.4.8 Knowledge of how AOPs are used in conjunction with EOPs CFR (41.10 / 43.5 / 45.13)	4.5	87
010 (SF3 PZR PCS) Pressurizer Pressure Control														
012 (SF7 RPS) Reactor Protection								X				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: (CFR: 41.5 / 43.5 / 45.3 / 45.5) A2.01 Faulty bistable operation	3.6	88
013 (SF2 ESFAS) Engineered Safety Features Actuation														
022 (SF5 CCS) Containment Cooling											X	G2.2.37 Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)	4.6	89
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											X	G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits CFR (41.5 / 41.7 / 43.2)	4.2	90
063 (SF6 ED DC) DC Electrical Distribution														
064 (SF6 EDG) Emergency Diesel Generator														
073 (SF7 PRM) Process Radiation Monitoring														
076 (SF4S SW) Service Water														
078 (SF8 IAS) Instrument Air														
053 (SF1; SF4P ICS*) Integrated Control														
K/A Category Point Totals:								2			3	Group Point Total:		5

ES-401												PWR Examination Outline Plant Systems—Tier 2/Group 2 (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											X	G2.2.12 Knowledge of surveillance procedures CFR (41.10 / 45.13)	4.1	91
011 (SF2 PZR LCS) Pressurizer Level Control								X				Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: CFR (41.5 / 43.5 / 45.3 / 45.13) A2.11 Failure of PZR level instrument - low	3.6	92
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation														
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In‑Core Temperature Monitor														
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control														
029 (SF8 CPS) Containment Purge														
033 (SF8 SFPCS) Spent Fuel Pool Cooling														
034 (SF8 FHS) Fuel‑Handling Equipment							X					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: CFR (43.7) A1.02 Water level in the refueling canal	3.7	93
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														
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:							1	1			1	Group Point Total:		3

Facility: CPNPP		Date of Exam: June 20, 2019				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.34	Knowledge of primary and secondary plant chemistry limits (CFR: 41.10 / 43.5 / 45.12)			3.5	94
	2.1.36	Knowledge of procedures and limitations involved in core alterations (CFR: 41.10 / 43.6 / 45.7)			4.1	95
	Subtotal					
2. Equipment Control	2.2.5	Knowledge of the process for making design or operating changes to the facility (CFR: 41.10 / 43.3 / 45.13)			3.2	96
	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 (CFR: 41.10 / 43.5 / 45.13))			3.8	97
	Subtotal					
3. Radiation Control	2.3.11	Ability to control radiation releases (CFR: 41.11 / 43.4 / 45.10)			4.3	98
	Subtotal					
4. Emergency Procedures/Plan	2.4.30	Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator (CFR: 41.10 / 43.5 / 45.11)			4.1	99
	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc (CFR: 41.7 / 43.5 / 45.12)			4.6	100
	Subtotal					
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO EXAM		
2/1	004A1.12	Question 3 A question on the de-borating demineralizer at CPNPP would be inappropriate as it is not used for its intended design function. It is only used certain cycles at EOL during coastdown. Replaced original K/A with new K/A 004A1.07
1/1	011EK1.01	Question 14 Could not write a question about Natural Circulation / Reflux boiling during a LB LOCA because it is not a concern. Replaced original K/A with new K/A 011EK2.02.
2/1	013K4.18	Question 31 Could not write a question about using jumpers for CS pumps during CS train testing because CP does not use jumpers (it places switches in Pull Out). Replaced original K/A with new K/A 013K4.03.
2/1	061A2.08	Question 38 Could not write an operationally valid question on expected flow rates based on AFW valve position. Replaced original K/A with new K/A 061A2.04.
1/2	028AA2.14	Question 48 Could not write a question about PRZR level during reflux boiling due to conditions in the Pressurizer. Replaced original K/A with new K/A 028AA2.13.
1/2	059EK3.04	Question 50 Could not write a question about the reason for actions contained in the EOP (ABN/ALM) as they apply to accidental liquid Radwaste release because the EOPs are the only procedures at CPNPP containing the reasons for actions (bases document) and the EOPs do not cover actions for this event. The ABNs/ALMs do not contain reasons for the actions that could be tested on. Replaced original K/A with new K/A 059EK3.01
1/2	W E15G2.1.28	Question 55 Could not write a question about purpose/function of major system components/controls for containment flooding because the EOP only contains isolation of various systems. Replaced original K/A with new K/A W E15G2.4.6.

2/2	072A4.01	Question 64 Could not write a question about ARM setpoint checks and adjustments because they are made by I&C, not operations. Replaced original K/A with new K/A 072A4.02.
2/2	079G2.2.3	Question 65 Could not write a question about unit differences between Unit 1 & 2 Station (Service) Air systems because there are not differences. Replaced original K/A with new K/A 079G2.2.44.
3	2.4.45	Question 75 Could not write a generic question about alarm prioritization without being alarm/unit specific that is greater than LOD 1. Replaced original K/A with new K/A 2.4.3.
3	2.1.8	Question 67 Could not write a question at the RO level for directing activities outside the control room as this is the responsibility of the SRO. Replaced original K/A with new K/A 2.1.9

Tier / Group	Randomly Selected K/A	Reason for Rejection
SRO Exam		
2/2	011A2.02	Question 92 Cannot write a question to this KA without overlapping with RO Question 58 because it is the same K/A. Replaced original K/A with new K/A 011A2.11.

Facility: **CPNPP Units 1 and 2**	Date of Examination: **June 2019**
Examination Level: RO ☒ SRO ☐	Operating Test Number: **NRC**

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations (RA1)	M, S	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (4.3) JPM: Calculate BOL Boration for Long Term Use (RO1307D)
Conduct of Operations (RA2)	D, R	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. (3.9) JPM: Respond to Voids in Reactor Vessel. (RO7030)
Equipment Control (RA3)	M, R	2.2.12 Knowledge of surveillance procedures. (3.7) JPM: Perform a Manual Quadrant Power Tilt Ratio Calculation (RO1803D).
Radiation Control (RA4)	D, R	2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (3.2) JPM: Determine Containment Stay Time Requirements. (RWT029)
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 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)

RA1 The applicant will use the simulator to obtain plant reference data and calculate the number of gallons of Boric Acid required to obtain a new measured RCS boron concentration of 1510 ppm from current value. The applicant will then use this final Boron Concentration to calculate new BOL Boration volumes and pot settings for long term use per SOP-104A, REACTOR MAKEUP AND CHEMICAL CONTROL SYSTEM, Attachment 2, BOL Boration for Long-Term Use. Critical steps include calculating number of gallons of Boric Acid required to achieve new boron concentration, determining Reactor Coolant System corrected boron, gallons of Reactor Makeup Water to offset boron, and potentiometer settings for the Chemical and Volume Control System. This is a modified from bank JPM. (K/A 2.1.23 – IR 4.3)

RA2 The applicant will determine the maximum allowable venting time for venting the reactor vessel using FRI-0.3A, Response to Voids in Reactor Vessel, Attachment 5. Critical steps include various stages of the calculation, including the final determination of allowable venting time. This is a direct from bank JPM. (K/A 2.1.25 – IR 3.9)

RA3 The applicant will perform a manual Quadrant Power Tilt Ratio calculation per OPT-302, CALCULTING POWER TILT RATIO, and determine whether Acceptance Criteria are met. The critical steps include recording data, accurately performing calculations and applying Acceptance Criteria. This is a modified from bank JPM. (K/A 2.2.12 – IR 3.7)

RA4 The applicant will calculate maximum allowable stay time within Containment per STA-620, CONTAINMENT ENTRY, STA-655, EXPOSURE MONITORING PROGRAM, STA-657, ALARA JOB PLANNING/DEBRIEFING, STA-660, CONTROL OF HIGH RADIATION AREAS, and STI-211.07, HEAT STRESS MANAGEMENT. Critical steps include calculating allowable gamma dose, neutron dose, Heat Stress Stay Time, and determining the limiting stay time. This is a direct from bank JPM. (K/A 2.3.12 – IR 3.2)

Facility: **CPNPP Units 1 and 2** Date of Examination: **June 2019**

Examination Level: RO ☐ SRO ☒ Operating Test Number: **NRC**

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations (SA1)	N, R	2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc. (3.8) JPM: Review Medical Status and Determine Ability to Continue Watch Standing. (SO1004)
Conduct of Operations (SA2)	M, R	2.1.25 Ability to interpret reference materials such as graphs, curves, tables, etc. (4.2) JPM: Restore Refueling Water Storage Tank Level and Evaluate Technical Specifications. (SO1211)
Equipment Control (SA3)	D, R	2.2.14 Knowledge of the process for controlling equipment configuration or status. (4.3) JPM: Determine Fire Compensatory Measures for an Emergent Condition. (SO1048)
Radiation Control (SA4)	D, R	2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (3.7) JPM: Select Volunteer for Emergency Exposure. (SO1142A)
Emergency Procedures/Plan (SA5)	M, R	2.4.44 Knowledge of emergency plan protective action recommendations. (4.4) JPM: Determine Protective Action Requirements. (SO1140)

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

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

SA1 The applicant is presented with a record of three operators with the number of watches stood and the current status of their physicals. In accordance STA-121, License Operator Physicals and License Application Process, and ODA-315, Licensed Operator Maintenance Tracking, the applicant will evaluate circumstances and state current watch standing restrictions as well as what, if anything, is required of each operator to assume licensed duties at a future date. Critical steps include determining 2 of the 3 operators can currently assume license duties as well as determining 2 of the 3 operators must perform subsequent actions to assume license duties at the future date identified in the JPM. This is a new JPM. (K/A 2.1.4 – IR 3.8)

SA2 The applicant will restore Refueling Water Storage Tank (RWST) per SOP-104A, REACTOR MAKEUP AND CHEMICAL CONTROL SYSTEM, Section 5.2.7, Makeup to RWST. Critical steps include calculating the required volume of borated water necessary to raise RWST level, Boric Acid Flowrate, total gallons of Boric Acid, and potentiometer settings for the Flow Control Valves and then evaluating Technical Specifications when it is determined that RWST temperature is out of specification. This is a modified from bank JPM. (K/A 2.1.25 – IR 4.2)

SA3 The applicant will evaluate a Fire Protection Impairment per STA-738, FIRE PROTECTION SYSTEM/EQUIPMENT IMPAIRMENTS. The critical steps are to determine Fire Watch Implementation and other Compensatory Measures. This is a direct from bank JPM. (K/A 2.2.14 – IR 4.3)

SA4 The applicant is given accident conditions involving the need for a volunteer to attempt a lifesaving activity. Using the guidance in EPP-305, EMERGENCY EXPOSURE GUIDELINES AND PERSONNEL DOSIMITRY, the applicant will evaluate a series of potential volunteers and select the preferred volunteer from this list. The critical steps are evaluation and elimination of volunteers who do not meet the criteria required for the activity, and then final selection of the preferred volunteer. This is a direct from bank JPM. (K/A 2.3.4 – IR 3.7)

SA5 The applicant will determine the required Protective Action recommendation based on a General Emergency declared at the site. The critical steps include determining the correct Protective Action, highlighting the correct decision flowpath on the PAR attachment, and completing the correct Minimum Affected Area. This a modified from bank JPM. (K/A 2.4.44 – IR 4.4)

Facility: **CPNPP 1 & 2**	Date of Examination: **June 2019**
Exam Level: RO SRO(I) **SRO (U)**	Operating Test Number: **NRC**

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
S-1	**001 – Control Rod Drive System (RO1024C)** **Initiate restoration of Control Rods following a manual Turbine Runback**	**A,N,S**	**1**
S-2	**E11 – Loss of Emergency Coolant Recirculation (RO1506N)** **Transfer ECCS from Injection Phase to Cold Leg Recirculation with Sump Blockage**	**A,D,EN,L,P,S**	**2**
S-3	010 – Pressurizer Pressure Control System (RO1205B) PORV Block Valve Operability Test	A,M,S	3
S-4	076 – Loss of Nuclear Service Water (RO3705A) Respond to a SSW leak (Low Discharge Press) on discharge of SSW Pump 1-01	A,EN,M,S	4S
S-5	062 – AC Electrical Distribution (RO4201B) Shift Normal Bus 1A4 from 1ST to 1UT	M,S	6
S-6	015 – Nuclear Instrumentation System (RO1820) Respond to a Power Range Channel Malfunction	D,S	7
S-7	008 – Component Cooling Water System (RO3605) Fill the CCW Surge Tank (RO Only)	A,N,S	8
S-8	060 – Accidental Gaseous Radwaste Release (RO4006) Perform a Containment Pressure Reduction	A,D,EN,S	9

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

P-1	**040 – Steam Line Rupture-Excessive Heat Transfer (AO6424B)** **Manually Isolate AFW to a Faulted Steam Generator**	**D,E,L,R**	**4P**
P-2	**067 – Plant Fire on Site (AO5408)** **Actions for Fire in Safeguard Building Fire Area 1SB/2SB**	**N,E,L,R**	**8**
P-3	**057 – Loss of Vital AC Electrical Instrument Bus (AO4204D)** **Energize Instrument Bus using the spare (swing) inverter**	**N,E,L**	**6**

*	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 **(6)** / 4-6 **(5)** / 2-3 **(2)**
(C)ontrol room	**(0)** / **(0)** / **(0)**
(D)irect from bank	≤ 9 **(4)** / ≤ 8 **(4)** / ≤ 4 **(2)**
(E)mergency or abnormal in-plant	≥ 1 **(3)** / ≥ 1 **(3)** / ≥ 1 **(3)**
(EN)gineered safety feature	≥ 1 **(3)** / ≥ 1 **(3)** / ≥ 1 **(1)** (control room system)
(L)ow-Power / Shutdown	≥ 1 **(4)** / ≥ 1 **(4)** / ≥ 1 **(4)**
(N)ew or (M)odified from bank including 1(A)	≥ 2 **(7)** / ≥ 2 **(6)** / ≥ 1 **(3)**
(P)revious 2 exams	≤ 3 **(1)** / ≤ 3 **(1)** / ≤ 2 **(1)** (randomly selected)
(R)CA	≥ 1 **(2)** / ≥ 1 **(2)** / ≥ 1 **(2)**
(S)imulator	**(0)** / **(0)** / **(0)**

NRC JPM Examination
Summary Description

S-1 The reactor is critical at 100% power when a Heater Drain Pump trip requires a manual turbine runback. The operator will manually initiate a Main Turbine Runback to 700 MWe. The control rods will automatically insert during the runback to restore Tave to within 1.5°F of Tref. Control Rods will automatically insert below the Lo-Lo Rod Insertion Limit and the operator will be required to initiate a boration to restore Shutdown Margin. Critical Steps will include manually initiating a Main Turbine Runback as required and initiating boration to restore SDM. This is a new JPM. This JPM is under the Reactivity Control Safety Function. (K/A 001 A4.05 – IR 3.7 / 3.7)

S-2 A large break LOCA has occurred and the criteria has been met to line up Cold Leg Recirculation using EOS-1.3A, TRANSFER TO COLD LEG RECIRCULATION. The Train ‘A’ RHR Pump is tagged out. The operator will realign for single train ECCS recirculation with the ‘B’ RHR pump supplying both trains CCP’s and SIP’s suction source. The ‘B’ RHR pump will start to cavitate due to sump blockage on the train ‘B’ sump (Alternate Path). The operator must secure at least one of the four pumps (‘A’ or ‘B’ CCP, ‘A’ or ‘B’ SIP) to remove the cavitation. The operator must not secure the ‘B’ RHR pump as this is the only pump cooling the core. Critical Steps will include aligning ECCS for single train operation in the Cold Leg Recirculation mode and stopping any one of the CCPs or SIPs taking suction from the Train ‘B’ RHR Pump when cavitation is evident. This JPM was used on the 2018 NRC exam (designated previous). This JPM is under the Reactor Coolant System Inventory Control Safety Function.
(K/A E11 EA1.1 – IR 3.9 / 4.0)

S-3 The operator will be provided with OPT-109A-R11-P-0, PORV BLOCK VALVE TEST and will be required to perform the Operability Test. When PORV Block Valve 1/1-8000B is re-opened as part of the test then PORV 1/1-PCV-456 partially opens requiring the operator to take action to isolate the open PORV, however PORV Block Valve 1/1-8000B will fail to close the second time (Alternate Path). This will require the operator to trip the Reactor prior to an automatic Reactor Trip occurring at 1880 psig. Critical Steps will include closing and opening each PORV Block Valve per the OPT. then attempting to close PORV-456 and block valve 1-8000B after PORV-456 fails open. There is an additional critical step to trip the Reactor following the PORV failing open. This is a modified from bank JPM. This JPM is under the Reactor Pressure Control Safety Function. (K/A 010 A4.03 – IR 4.0 / 3.8)

S-4 With the unit at 100% power, SSW Pump 1-01 develops a leak on the pump discharge. The low SSW header pressure annunciator comes into alarm. ABN-501, STATION SERVICE WATER SYSTEM MALFUNCTION, Section 3.0, SSW SYSTEM HEADER PRESSURE LOW is entered. The affected SSW Pump discharge pressure < 32 psig requires the operator to stop the pump and GO TO Section 2.0, SSW PUMP TRIP. The standby CCW pump will fail to auto-start and a manual start must be attempted, but will not be successful (Alternate Path). This will require the operator to trip the Reactor then trip all RCPs, upon completion of the Immediate Operator Actions. Critical Steps will include stopping the affected SSW Pump, tripping the Reactor, securing all RCPs, and placing the affected Diesel Generator in Pull-Out. This is a modified from bank JPM. This JPM is under the Heat Removal from Reactor Core - Secondary Safety Function. (K/A 076 A2.02 – 2.7 / 3.1)

S-5 The operator will shift Normal Bus 1A4 from the Unit Auxiliary Transformer 1UT to the Station Service Transformer 1ST per SOP-603A, 6900 V SWITCHGEAR, Section 5.3.2, TRANSFERRING A 6.9 KV NORMAL BUS FROM UNIT 1 AUXILIARY TRANSFORMER 1UT TO STATION SERVICE TRANSFORMER 1ST. Critical Steps will include turning on the Synchroscope for Bus 1A4 and closing the incoming breaker from 1ST. This is a modified from bank JPM. This JPM is under the Electrical Safety Function. (K/A 062 A4.07 – IR 3.1* / 3.1*)

S-6 Following a Power Range Instrument failure. The operator is required to perform the actions of ABN-703, POWER RANGE INSTRUMENT MALFUNCTION. Critical Steps will include several repositions on the NI Detector cabinets to defeat the failed instrument, defeating the N-16 Channel on CB-05, and the T_{AVE} channel on CB-07. This is a direct from bank JPM. This JPM is under the Instrumentation Safety Function. (K/A 015 A2.01 – IR 3.5 / 3.9)

S-7 The operator is directed to fill the CCW Surge Tank with RMUW to 72% in accordance with ABN-502, COMPONENT COOLING WATER SYSTEM MALFUNCTIONS, Section 3.0, LEAKAGE OUT OF THE CCW SYSTEM. The applicant will attempt to fill the CCW Surge Tank using RMUW. The RMUW supply valve will fail to open (Alternate Path) and the applicant must use Demin Water to fill the Surge Tank to 72%. The critical steps will include opening the tank makeup supply valves, opening the Demin Water fill valve and then closing the Demin Water fill valve at 72% +/- 2.0%. This is a new JPM. This JPM is under the Plant Service Systems Safety Function.
(K/A 008 A2.02 – IR 3.2 / 3.5)

S-8 The operator will be directed to place the Containment Pressure Relief System in operation per SOP-801A, CONTAINMENT VENTILATION SYSTEM. A containment high radiation annunciator will alarm alerting the operator of a high radiation condition. The operator will be required to manually isolate containment ventilation utilizing ALM-0032A, ALARM PROCEDURE DRMS as the system will fail to automatically isolate (Alternate Path). Critical Steps will include aligning the Containment Pressure Relief System and isolating the Containment release path upon receipt of a high radiation alarm. This is a direct from bank JPM. This JPM is under the Radioactivity Release Safety Function.
(K/A 060 AA2.05 – IR 3.7 / 4.2)

P-1 The operator will be directed to locally isolate AFW flow to SG u-02 from the appropriate MDAFWP and the TDAFWP. Critical Steps will include closing at least one AFW flow isolation valve to SG u-02 from u-01 MDAFWP and the TDAFWP. This is a direct from bank JPM. This JPM is under the Heat Removal from Reactor Core-Primary Safety Function. (K/A 040 AA1.03 – IR 4.3 /4.3)

P-2 In accordance with ABN-804A/B, Response to Fire in the Safeguards Building, the operator will be directed to align Unit 1/2 plant equipment per Attachment 3, Actions to Be Taken for a Fire in Fire Area uSB (NEO #2). The critical steps include opening the RHR to CL 2-03/2-04 Injection Isolation Valve motor breaker (U2 only), opening the RCP Seal Water ORC Return Isolation Valve motor breaker, opening the motor breakers to each TDAFWP Discharge to each SG Isolation Valve, opening the Sample Valve Control Panel Train B Supply Breaker, opening the uEB3 to 480 VAC MCC uEB3-1 Feeder Breaker, opening the RCP Seal Water Return Isolation Valve motor breakers 1 and 2, and closing each SG TDAFWP Discharge Isolation Valve locally. This is a new JPM. This JPM is under the Plant Service Systems Safety Function. (K/A 067 AK3.04 – IR 3.3 / 4.1)

P-3 The operator will energize the spare (swing) inverter, IVuEC1/3, and supply distribution panel uEC1 in accordance with SOP-607A/B, 118VAC DISTRIBUTION SYSTEM AND INVERTERS, Section 5.1.3, ENERGIZING IVuEC1/3 TO SUPPLY DISTRIBUTION PANEL uEC1. Critical Steps will include shutting down IVuEC1, ensuring the battery is in service, and powering distribution panel uEC1 from IVuEC1/3. This is a new JPM. This JPM is under the Electrical Safety Function.
(K/A 057 AA1.01 – IR 3.7* / 3.7)

Facility:	CPNPP 1 & 2	Scenario No.:	1	Op Test No.:	June 2019 NRC
Examiners:	______			Operators:	______
	______				______
	______				______

Initial Conditions: 55% power

Turnover: Ramp to 100% power after Chemistry Hold (See Turnover Sheet)

Critical Tasks:

CT-1 - Identify and Isolate the Ruptured Steam Generator Prior to Commencing a Cooldown per EOP-3.0A, Steam Generator Tube Rupture.

CT-2 - Initiate Cooldown of Reactor Coolant System Prior to Lifting Any Main Steam Safety Valve on the Ruptured Steam Generator.

Event No.	Malf. No.	Event Type*	Event Description
1	RP05B	I (RO, SRO) TS (SRO)	RCS Loop 1 Tcold Narrow Range 1-TI-411A Fails low
2	ED04F	C (RO, BOP, SRO) TS (SRO)	Supply Breaker 1EB1-1 (Safeguards 480V bus 1EB1) Trips Open
3	RX04B	I (BOP, SRO) TS(SRO)	Steam Generator 1-02 Level Channel (LT-552) fails high
4	CC06	R (RO, SRO)	Seal Water HX tube leak
5	FW16	C (BOP, RO, SRO)	Lowering Condenser Vacuum Requiring Reactor Trip
6	SG01A	M (RO, BOP, SRO)	Steam Generator 1-01 Tube Rupture
7	MS08A	C (RO, SRO)	1-HV-2333A, Steam Generator 1-01 MSIV Fails to Close
8	OVRDE	C (BOP)	1-HS-6082 and 1-HS-6084, Chill Water Return Isolation Valves Fail to Automatically Close on Phase A

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

Actual	Target Quantitative Attributes
8	Total malfunctions (5-8)
2	Malfunctions after EOP entry (1-2)
5	Abnormal events (2-4)
1	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
2	Critical tasks (2-3)

SCENARIO 1 SUMMARY

*** Event 1 - RCS Loop 1 Tcold Narrow Range 1-TI-411A Fails low**
RCS loops 1 Tcold Narrow Range instrument 1-TI-411A fails low. Crew responds per ABN-704, Tc / N16 Instrumentation Malfunction, Section 2.0. Crew selects out inoperable channel and restores pressurizer level to automatic. Control Rods will remain in manual due to failed input channel. Technical Specification, LCO 3.3.1.E, Reactor Trip System Instrumentation Function 6, and 7.

*** Event 2 - Breaker 1EB1-1 Trips Open**
Breaker 1EB1-1 trips open. Crew responds per ABN-602, Response to a 6900/480V System Malfunction Section 6.0, Safeguards 480V Bus Fault. Crew may attempt one breaker re-closure, then re-energize bus from bus tie breaker. Once bus is re-energized, crew will re-start Containment Fan Cooler #1 and resets group C Pressurizer heaters. Technical Specification 3.8.9 Condition A.

*** Event 3 - Steam Generator 1-02 Level Channel (LT-552) fails high**
Steam Generator (1-02) Level Channel (LT-552) fails high. Crew responds per ABN-710, Steam Generator Level Instrumentation Malfunction Section 2.0. Places Steam Generator (SG) Level Control in Manual, stabilizes level, aligns Alternate Channel, and transfers Level Control back to AUTO. Technical Specifications 3.3.1 Condition E Function 14, 3.3.2 Condition D Function 6c, 3.3.2 Condition I Function 5b.

*** Event 4 - Seal Water HX tube leak**
Seal Water Return Heat Exchanger tube leak from CCW into CVCS . Crew responds per ABN-105, Chemical and Volume Control System Malfunction section 8.0, Dilution Anomaly. Crew, determines Seal Return Heat Exchanger is leaking, then bypass and isolates heat exchanger.

*** Event 5 - Lowering Condenser Vacuum Requiring Reactor Trip**
Condenser Vacuum lowers due to air inleakage. Crew responds per ABN-304, Main Condenser and Circulating Water System Malfunction Section 3.0, Main or Auxiliary Condenser Vacuum Decreasing. They will start standby Condenser Exhaust Vacuum (CEV) pump, reduce turbine load in an attempt to maintain vacuum, condenser vacuum will continue to deteriorate until a Reactor Trip is required. Crew enters EOP-0.0A, Reactor Trip or Safety Injection and transitions to EOS-0.1A, Reactor Trip Response.

*** Event 6 - Steam Generator 1-01 Tube Rupture**
At EOS-0.1A, Reactor Trip Response Step 1, Event 6 is initiated. Crew re-enters EOP-0.0A, Reactor Trip or Safety Injection. Steam Generator Tube Rupture is diagnosed then transitions to EOP-3.0A, Steam Generator Tube Rupture.

Event 7 - 1-HV-2333A, Steam Generator 1-01 MSIV Fails to Close
SG 1-01 MSIV fails to close, crew closes all remaining MSIVs via MSLI activation, disables Steam Dumps, and closes the Main Steam to Auxiliary Steam Supply Valve. The RCS cooldown is performed with intact SG ARVs.

Event 8 - 1-HS-6082 and 1-HS-6084, Chill Water Return Isolation Valves Fail to Automatically Close on Phase A
Chill Water Return Isolation Valves Fail to Automatically Close on Phase A. BOP manually actuates Phase A from CB-02 and CB-07, then manually isolates either 1-HS-6082, CH WTR RET ISOL VLV ORC or 1-HS-6083, CH WTR RET ISOL VLV IRC.

***** - *On Lead Examiner’s Cue*

Terminating Criteria
The scenario will be terminated when the operators have performed CT-2, Initiate Cooldown of Reactor Coolant System Prior to Lifting Any Main Steam Safety Valve on the Ruptured Steam Generator, or at the Examiner's discretion.

Risk Significance:

• Failure of risk important system prior to trip:	Breaker 1EB1-1 Tripping Open
• Risk significant core damage sequence:	SGTR
• Risk significant operator actions:	Isolation of Ruptured Steam Generator complicated by MSIV failure to close Failure of Containment Isolation Phase A

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT-1 - Identify and Isolate the Ruptured Steam Generator Prior to Commencing a Cooldown per EOP-3.0A, Steam Generator Tube Rupture.	Take actions to isolate ruptured SG from non-ruptured SG to minimize the release of radioactive steam to the environment due to a failure of Ruptured SG MSIV closure and isolate normal steam release paths from ruptured SG.	Procedurally driven from EOP-3.0A to isolate the ruptured SG to prevent further release of radioactivity from the SG. In addition, isolation is necessary to establish a pressure differential between the ruptured and non-ruptured steam generators in order to cool the RCS and stop primary-to-secondary leakage.	The crew will attempt to close SG 1-01 MSIV, the MSIV will fail to close and all other MSIVs must be closed. ARV for SG 1-01 will be adjusted to 1160 psig, steam Supply to the TDAFWP from MSL #1 must be placed in Pull-Out and 1-HS 3228, MS TO AUX STM SPLY VLV must be closed. AFW flow to SG 1-01 must be stopped when Narrow Range SG level is > 43%.	Indication of MSIV closure at CB-08 from valve position indicating lights. TDAFWP steam supply from MSL #1 Control Board Handswitch in Pull-Out at CB-09. Indication of 1-HS 3228 closure at CB-01. Indication of all AFW flow stopped to SG 1-01 with valve position indicating lights at CB-09.
CT-2 - Initiate Cooldown of Reactor Coolant System Prior to Lifting Any Main Steam Safety Valve on the Ruptured Steam Generator.	Take actions that would stop RCS leakage into the Rupture SG to prevent over filling the ruptured SG and result in lifting the SG Safety valves.	Procedurally driven from EOP-3.0A to commence cooldown to reduce the overall temperature of the RCS.	The operator will increase dumping steam from the SGs via the ARVs to reduce RCS temperature. Steam Dumps will be unavailable as all intact SG MSIVs will be closed.	Lowering SG pressures and lowering RCS temperatures beginning with the cold leg temperatures.

NOTE: (Per NUREG-1021, Appendix D) If an operator or the Crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.

Facility: CPNPP 1 & 2	Scenario No.: 2		Op Test No.: June 2019 NRC
Examiners:		Operators:	
Initial Conditions: 2-3% Power, BOL			
Turnover: Raise Reactor Power to 6% - 8%, RWST is recirculating with Containment Spray Pump 1-02 (See Turnover Sheet)			

Critical Tasks:

CT-1 – Manually initiate Train A Safety Injection due to Failure to Automatically Actuate prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.

CT-2 – Manually start Safety Injection Pump 1-01 due to an automatic start failure on Safety Injection, prior to RVLIS 79 ” above Core Plate Light going DARK.

CT-3 – Trip RCPs within 5 minutes upon a Loss of Subcooling per EOP-0.0A, Reactor Trip or Safety Injection or EOP-1.0A, Loss of Reactor or Secondary Coolant

Event No.	Malf. No.	Event Type*	Event Description
1	-	R (RO, SRO)	Raise Reactor Power to 6% - 8%.
2	SW01B	C (RO, BOP, SRO) TS (SRO)	SSW Pump 1-02 Trip
3	RX12	I (BOP, SRO) TS (SRO)	Main Steam Header Transmitter PT-507 Fails High
4	RX08A RX05A RC12	I (RO, SRO) TS (SRO)	Pressurizer Common Instrument Line Failure
5	RC13	C (RO, SRO) TS (SRO)	40 gpm Pressurizer Leak
6	RC12 RC13 RP07A	M (RO, BOP, SRO)	Spurious Safety Injection Train B, Automatic Safety Injection Train A Failure with a SBLOCA
7	SI04D	C (BOP)	Safety Injection Pump 1-01 Auto Start Failure on Safety Injection Signal

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

Actual	**Target Quantitative Attributes**
6	Total malfunctions (5-8)
1	Malfunctions after EOP entry (1-2)
4	Abnormal events (2-4)
1	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
0	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

SCENARIO 2 SUMMARY

Event 1 - Raise Reactor Power to 6%-8%
Crew raises Reactor Power to 6%-8% per IPO-003A, Power Operations, Section 5.1, Warmup and Synchronization of the Turbine Generator.

*** Event 2 - Service Water Pump 1-02 Trip**
Crew responds per ABN-501, Station Service Water System Malfunction Section 2.0, Station Service Water Pump Trip and ABN-502, Component Cooling Water Systems Malfunctions Section 2.0, CCW Pump Trip. Emergency Diesel 1-02 and Train B equipment are placed in Pull-Out. Technical Specifications 3.7.8 Condition B, 3.8.1 Condition B.

*** Event 3 - Main Steam Pressure Transmitter PT-507 Fails High**
Crew responds per ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st Stage Pressure, and Feed Header Pressure Instrument Malfunction, Section 3.0 Steam Header Pressure Malfunction. Crew takes manual control of Steam Dump Pressure Controller 1-PK-507 controlling RCS Temperature. Technical Specification 3.4.1 Condition A

***Event 4 - Pressurizer Common Instrument Line Failure**
Crew takes manual control of Pressurizer Level and Pressure and responds per ABN-706, Pressurizer Level Instrument Malfunction, Section 2.0 Pressurizer Level Instrument Malfunction and ABN-705, Pressurizer Pressure Malfunction, Section 2.0 Pressurizer Pressure Instrument Malfunction. Both failed channels are bypassed then restored to automatic control. Technical Specifications 3.3.1 Condition E Function 6 and 8b, 3.3.2 Condition D Function 1d, and Condition L Function 8b.

**** Event 5 - 40 gpm Pressurizer Leak**
When Crew defeats failed pressurizer pressure and level channels, the Pressurizer steam space leak will be initiated. Crew responds per ABN-103, Excessive Reactor Coolant Leakage Section 2.0, Excessive Reactor Coolant Leakage. Crew reduces letdown to 45 gpm restoring Pressurizer Level. Technical Specification 3.4.13 Condition A

*** Event 6 - Spurious Safety Injection Train B, Automatic Safety Injection Train A Failure with a SBLOCA**
Crew responds per EOP-0.0A, Reactor Trip or Safety Injection, manually initiates Safety Injection then transitions to EOP-1.0A, Loss of Reactor or Secondary Coolant.

Event 7 - Safety Injection Pump 1-01 Auto Start Failure on Safety Injection Signal
Crew starts SIP 1-01 during performance of EOP-0.0A, Attachment 2, Safety Injection Actuation Alignment.

***** *- On Lead Examiner's Cue*
****** *- Starts automatically or on Lead Examiners Cue*

Termination Criteria
Scenario will be terminated when the operators transition to EOS 1.2, Post LOCA Cooldown and Depressurization or at the Lead Examiner's discretion.

Risk Significance:

- Failure of risk important system prior to trip: Loss of SSWP 1-02
  Pressurizer Common Instrument Line Failure
- Risk significant core damage sequence: Loss of a Safety Train
  SBLOCA
- Risk significant operator actions: Manually Initiate Safety Injection
  Manually start the 1-01 SIP
  Trip all RCP's on loss of subcooling

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT-1 - Manually initiate Train A Safety Injection due to Failure to Automatically Actuate prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.	Recognize a failure of Train A Safety Injection automatic actuation with Train B SI OOS due to a loss of cooling.	Procedural direction at EOP- 0.0A Step 4 to determine if a Safety Injection is required and annunciators indicating that an SI should have occurred yet did not occur.	The operator will manually actuate Safety Injection using either the handswitch on CB- 07 or CB-02.	PCIP Window 1.8 annunciates indicating both trains of SI have actuated. Numerous equipment changes of state.
CT-2 - Manually start Safety Injection Pump 1-01 due to an automatic start failure on Safety Injection, prior to RVLIS 79 ” above Core Plate Light going DARK.	Recognize a failure or an incorrect automatic actuation of SIP 1-01 to start, to provide adequate injection capability/core cooling for a SBLOCA with Train B SI OOS.	Procedurally driven from EOP-0.0A, Attachment 2 to provide makeup inventory to the RCS during accident conditions.	The operator will start SI Pump 1-01 using the handswitch on CB-02.	Indication pump start including light indication, flow and discharge pressure on CB-02.
CT-3 - Trip RCPs within 5 minutes upon a Loss of Subcooling per EOP-0.0A, Reactor Trip or Safety Injection or EOP-1.0A, Loss of Reactor or Secondary Coolant.	Take one or more actions that would prevent a challenge ability to cool the core during a SBLOCA.	Procedurally driven from EOP-0.0A and EOP-1.0A Foldout pages. Availability of Subcooling indication both on meters and computer.	The operator will secure ALL RCPs using the handswitches on CB-05.	Indication of pump stop including light indication, flow and motor current.
<u>**NOTE:**</u> (Per NUREG-1021, Appendix D) If an operator or the Crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.				

Facility: CPNPP 1 & 2 Scenario No.: 3 Op Test No.: June 2019 NRC

Examiners: ____________ Operators: ____________

Initial Conditions: 100% power, RHR Pump 1-02 OOS, TS 3.5.2 B

Turnover: Maintain Power and Availability. SIP 1-01 is running in preparation to raise SI Accumulator 1-03 level. (See Turnover Sheet)

Critical Tasks:

CT-1 – Ensure Control Rods are inserting and rod speed does not fall below 48 Steps / Minute for greater than an accumulated time of 36 seconds during Reactor Trip Failure, until Reactor Power is below 5% Power Range indication, per FRS-0.1A, Response to Nuclear Power Generation / ATWT.

CT-2 – Due to the Reactor failing to manually trip, initiate Emergency Boration from the RWST via 1/1-LCV-112D OR 1/1-LCV-112E, since BOTH Boric Acid Pumps failed to start, prior to exiting FRS-0.1A, Response to Nuclear Power Generation / ATWT.

Event No.	Malf. No.	Event Type*	Event Description
1	Normal	N (BOP, SRO)	Raise SI Accumulator 1-03 Level
2	RC23A	C (RO)	Reactor Vessel Head Flange Inner Seal Leak
3	RX09A	I (RO, SRO) TS (SRO)	1st Stage Pressure Transmitter PT-505 Fails High
4	TC08C	C (BOP, SRO) TS (SRO)	High Pressure Turbine Stop Valve # 3 (UV-2430A) Fails Closed
5	RX05A	I (RO, SRO) TS (SRO)	Pressurizer Level Channel 459 Fails Low
6	RC03C RP01 RP13C OVRDE	M (RO, BOP, SRO)	RCP 1-03 High Vibrations, Reactor Fails to Trip
7	OVRDE CV19B	C (BOP, SRO)	Both Boric Acid Transfer Pumps Fail to Start
8	RC08A2	M (RO, BOP, SRO)	Loop 1 Hot Leg LBLOCA

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

Actual	Target Quantitative Attributes
7	Total malfunctions (5-8)
2	Malfunctions after EOP entry (1-2)
4	Abnormal events (2-4)
2	Major transients (1-2)
2	EOPs entered/requiring substantive actions (1-2)
1	EOP contingencies requiring substantive actions (0-2)
2	Critical tasks (2-3)

SCENARIO 3 SUMMARY

Event 1 - Raise SI Accumulator 1-03 Level
Crew raises SI Accumulator 1-03 level per SOP-202A, Safety Injection Accumulators Section 5.4.1, Raising Accumulator Level to 58%.

*** Event 2 - Reactor Vessel Head Flange Inner Seal Leak**
Crew responds per 1-ALB-5C, Window 1.1, RV FLANGE LKOFF TEMP HI closing 1/1-8032, RV SEAL LKOFF VLV.

*** Event 3 - First Stage Pressure Transmitter PT-505 Fails High**
Crew responds per ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st Stage Pressure, and Feed Header Pressure Instrument Malfunction, Section 4.0 Turbine Impulse Pressure Instrument Malfunction. Crew places Rod Control in Manual, bypasses the failed Turbine 1st Stage Pressure channel, restores Tave-Tref deviation then returns Control Rods to automatic. Technical Specification 3.3.1 Condition T Function 18f.

*** Event 4 - High Pressure Turbine Stop Valve # 3 (UV-2430A) Fails Closed**
Crew responds per ABN-401, Main Turbine Malfunction Section 9.0, Inadvertent Closure of an HP or LP Stop or Control Valve. Crew reduces turbine load 50MWe until all operable HP Control Valves are ≤ 98%. Control Rods will insert below the LO LO Control Rod Insertion Limit for the current power level. Technical Specification 3.1.6 Condition A.

*** Event 5 - Pressurizer Level Channel 459 Fails Low**
Crew responds per ABN-706, Pressurizer Level Instrumentation Malfunction Section 2.0, Pressurizer Level Instrument Malfunction. Crew reduces charging to RCP Seals only, bypasses the failed channel, restores letdown then returns pressurizer level control to automatic. Technical Specification 3.3.1 Condition M Function 9.

*** Event 6 - RCP 1-03 High Vibrations, Reactor Fails to Trip**
Crew responds per ABN-101, Reactor Coolant Trip/Malfunctions Section 6.0, Excessive Reactor Coolant Pump Vibration. Crew attempts Reactor Trip then responds per EOP-0.0A, Reactor Trip or Safety Injection and transitions to FRS-0.1A, Response to Nuclear Power Generation / ATWT. Reactor is tripped locally, RCP 1-03 is secured then the crew transitions back to EOP-0.0A, Reactor Trip or Safety Injection.

Event 7 - Both Boric Acid Transfer Pumps Fail to Start
Crew responds per FRS-0.1A, Response to Nuclear Power Generation / ATWT Emergency Boration Att 1F, Initiation Emergency Boration. Crew lines up Emergency Boration from the RWST.

*** Events 8 - Loop 1 Hot Leg LBLOCA**
Crew responds per EOP-0.0A, Reactor Trip or Safety Injection, stops all RCP's. Crew transitions to EOP-1.0A, Loss of Primary or Secondary Coolant then transitions to FRP 0.1A Response to Imminent Pressurized Thermal Shock Condition and FRZ 0.1A Response to High Containment Pressure transitions back to EOP-1.0A, Loss of Primary or Secondary Coolant resets SI Signal, SI Sequencers and secures the EDG's.

***** *- On Lead Examiner's Cue*

Terminating Criteria
Scenario will be terminated when the Diesel Generators are shutdown in EOP 1.0 Loss of Reactor or Secondary Coolant or at Lead Examiners discretion.

Risk Significance:

- Failure of risk important system prior to trip: Rx Vessel Head Flange Leak; Pressurizer Level Channel Failure; MT 1st Stage Pressure Transmitter Failure
- Risk significant core damage sequence: ATWT; LBLOCA
- Risk significant operator actions: Insertion of control rods during ATWT; Boration from RWST during ATWT

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT-1 - Ensure Control Rods are inserting and rod speed does not fall below 48 Steps / Minute for greater than an accumulated time of 36 seconds during Reactor Trip Failure, until Reactor Power is below 5% Power Range indication, per FRS-0.1A, Response to Nuclear Power Generation / ATWT.	Take actions to insert Control Rods at the fastest rate speed available to prevent challenges to plant's heat removal capability and fuel integrity.	Procedurally driven from FRS-0.1A to reduce the heat input to the RCS to only that of decay heat and reactor coolant pump heat.	If the reactor is not tripped, the intent of the RNO actions is to insert the rods at the fastest rate available. If RCS temperature has increased above the current reference temperature, then the rods should automatically be driven in by the Rod Control System. When the Rod Control System signal decreases to less than the speed for manual rod insertion, the operator is required to manually insert rods to maximize negative reactivity addition.	Neutron flux lowering and indication of rods driving into the core by DRPI and step counters.
CT-2 - Due to the Reactor failing to manually trip, initiate Emergency Boration from the RWST via 1/1-LCV-112D OR 1/1-LCV-112E, since BOTH Boric Acid Pumps failed to start, prior to exiting FRS-0.1A, Response to Nuclear Power Generation / ATWT.	Take actions to add negative reactivity with normal emergency boration flow path failure, to prevent challenges to the plant's heat removal capability and fuel integrity.	Procedurally driven from FRS-0.1A to add negative reactivity to the core	The operator will ensure a charging pump is running, open the charging pump suctions to the RWST, close the charging pump suctions to the VCT, and adjust the charging flow control valve to establish maximum charging flow.	Indication of charging flow on 1-FI-121A, CHRG FLO with either 1/1-LCV-112D or 1/1-LCV-112E open.
NOTE: (Per NUREG-1021, Appendix D) If an operator or the Crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.				

Facility:	CPNPP 1 & 2	Scenario No.: 4	Op Test No.: June 2019 NRC
Examiners:		Operators:	

Initial Conditions: 100% power. XST1 and MDAFWP1-02 out of service. TS LCOs 3.8.1 A and 3.7.5 B.

Turnover: Maintain Power and availability, Reduce Letdown to 75 gpm, Start PDP, Shutdown CCP 1-01 per SOP-103A starting at Step 5.3.1.F (See Turnover Sheet)

Critical Tasks:

CT-1 - Trip all RCPs in accordance with FRH-0.1A, Response To Loss Of Secondary Heat Sink, prior to actuating SI.

CT-2 - Initiate RCS Bleed and Feed in accordance with FRH-0.1A, Response To Loss Of Secondary Heat Sink, prior to All SG WR levels lowering to 14%

Event No.	Malf. No.	Event Type*	Event Description
1	Normal	N (RO)	Start PDP and Shutdown CCP 1-01
2	TP01A OVRDE	C (BOP, SRO)	TPCW leak, TPCW Makeup Valve Fails to Automatically Open
3	RP04A	TS (SRO)	RCS Loop 1 Flow Transmitter Fails Low
4	ED03A	C (BOP, SRO) R(RO) TS(SRO)	Loss of 345 KV Transformer XST2, CCP 1-02 Trips on Overcurrent
5	FW05A	I (BOP, SRO)	MFP 1A GE Speed Controller Fails Low
6	OVRDE FW08G	M (RO, SRO)	1-HV-2136, FWIV 3 Fails Closed – Manual Reactor Trip, MDAFWP 1-01 trips on overcurrent
7	FW09A	M (RO, BOP, SRO)	TDAFWP trips on overspeed, Loss of All AFW Flow (Loss of Heat Sink)
8	RX16B	C (BOP, SRO)	PORV 456 Fails to Open Manually or Automatically

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

Actual	Target Quantitative Attributes
7	Total malfunctions (5-8)
3	Malfunctions after EOP entry (1-2)
4	Abnormal events (2-4)
2	Major transients (1-2)
1	EOPs entered/requiring substantive actions (1-2)
1	EOP contingencies requiring substantive actions (0-2)
2	Critical tasks (2-3)

SCENARIO 4 SUMMARY

Event 1 – Start PDP and Shutdown CCP 1-01
RO reduces letdown to 75 gpm commencing with SOP 103A, Chemical and Volume Control System Section 5.2.4, Lowering/Securing Letdown Flow, starts Positive Displacement Pump (PDP) and transfers flow from CCP 1-01 then secures CCP 1-01.

*** Event 2 – TPCW leak, TPCW Makeup Valve Fails to Automatically Open**
Crew responds per 1ALB-9A Window 2.10, TPCW HEAD TK LVL LO and ABN-306, TPCW System Malfunction, Section 2.0, Excessive TPCW System Leakage. BOP operator manually initiates TPCW Makeup and dispatches personnel to locate and isolate leak.

*** Event 3 – RCS Loop 1 Flow Transmitter Fails Low**
Crew responds per ABN-713, RCS Loop Flow Instrument Malfunction. Technical Specification 3.3.1 Condition M Function 10.

*** Event 4 – Loss of 345 KV Transformer XST2, CCP 1-02 Trips on Overcurrent**
Transformer failure causes loss of offsite power sources to both safety buses. EDGs start and power 1EA1 and 1EA2. Crew responds per ABN-601, 138/345 KV System Malfunction Section 2.0, Loss of Startup/Station Service Transformers and section 3.0, Plant Recovery from a Blackout Sequencer Signal (If Reactor power is above 100% power the crew will perform a load reduction per the beginning of shift brief). CCP 1-02 trips on Overcurrent and PDP load sheds on Blackout Sequencer Signal. CCP 1-01 starts and provides charging flow. Technical Specifications 3.5.2 Condition A and 3.8.1 Condition C.

*** Event 5 – MFP 1A GE Speed Controller Fails Low**
Crew responds per ABN-302, Feedwater, Condensate, Heater Drain System Malfunction Section 9.0, Feedwater Pump Control System Malfunction. BOP operator places both MFP GE speed controllers in manual, adjusts feed pumps to control differential pressure on program.

*** Event 6 – 1-HV-2136, FWIV 3 Fails Closed – Manual Reactor Trip, MDAFWP 1-01 Trips**
Crew manually trips Reactor and responds per EOP-0.0A, Reactor Trip or Safety Injection. MDAFWP 1-01 trips on overcurrent.

*** Event 7 – TDAFWP trips on overspeed, Loss of All AFW Flow (Loss of Heat Sink)**
Crew transitions to EOS-0.1A, Reactor Trip Response. *TDAFWP trips on overspeed per Lead Examiner's Cue at Step 2 of EOS-0.1A, causing a loss of Heat Sink. Crew transitions to FRH-0.1A, Response to Loss of Secondary Heat Sink.

Event 8 – PORV 456 Fails to Open Manually or Automatically
Crew responds per FRH-0.1A, Response to Loss of Secondary Heat Sink opening all reactor vessel head and pressurizer vents.

***** - *On Lead Examiner's Cue*

Terminating Criteria
Scenario terminated when bleed and feed is initiated in accordance with FRH-0.1A; or at the discretion of the lead examiner.

Risk Significance:

• Failure of risk important system prior to trip:	CCP 1-02 Trip Automatic MFP Control Loss of 345KV Transformer XST2
• Risk significant core damage sequence:	Loss of Heat Sink
• Risk significant operator actions:	Manually trip Reactor Initiation of Bleed and Feed

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
CT-1 - Trip all RCPs in accordance with FRH-0.1A, Response To Loss Of Secondary Heat Sink, prior to actuating SI.	Without a source of water to provide a heat sink on the secondary side of SGs, RCPs are tripped to extend effectiveness of remaining water inventory in SGs.	Procedural direction at FRH-0.1A Step 2 RNO a. to immediately stop all RCPs.	Operator will manually stop RCPs using handswitches on CB-05.	Control board light and flow indications, along with loss of flow annunciators that RCPs have stopped.
CT-2 - Initiate RCS Bleed and Feed in accordance with FRH-0.1A, Response To Loss Of Secondary Heat Sink, prior to All SG WR levels lowering to 14%	Actuating SI ensures feed path of cool water to RCS and isolates containment to confine any RCS releases from bleed flow. Bleed flow through a PORV/Vent valve ensures enough cool water feeds from ECCS flow path to remove sufficient decay heat.	AFW flow will not be indicated on any AFW flow meter. Also, no AFW pumps will be running. A RED path exists on CSFST for heat sink. A heat sink is needed as indicated by RCS temperature and pressure.	Actuates SI, ensures at least one CCP and SI pump running with flow indicated providing a feed path to RCS. PRZR PORV as well as PRZR and Reactor Vessel vent valves open providing a bleed path for RCS.	Flow indicated on both a CCP and an SI pump. PRZR PORV open with block valve open. PRZR and Reactor Vessel vents open. RCS pressure lowering and CETs indicating core cooling.
NOTE: (Per NUREG-1021, Appendix D) If an operator or the Crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.				

UNIT: 1

PART I TO BE PREPARED BY THE OFF-GOING UNIT SUPERVISOR.
1.0 SHIFT ACTIVITIES: