

Facility: CPNPP													Date of Exam: July 18, 2016				
Tier	Group	RO K/A Category Points											SRO-Only Points				
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	Total	A2	G*	Total	
1. Emergency & Abnormal Plant Evolutions	1	2	3	4	N/A			3	3	N/A			3	18			
	2	2	1	2	N/A			1	2	N/A			1	9			
	Tier Totals	4	4	6	N/A			4	5	N/A			4	27			
2. Plant Systems	1	3	3	3	2	2	2	2	3	3	3	2	28				
	2	1	0	1	1	1	1	1	1	1	1	1	10				
	Tier Totals	4	3	4	3	3	3	3	4	4	4	3	38				
3. Generic Knowledge and Abilities Categories					1		2		3		4		10				
					3		2		2		3						

Notes:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 Radiation Control K/A is allowed if the K/A is replaced by a K/A from another Tier 3 Category).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted 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' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in 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

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	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
000007 Reactor Trip - Stabilization - Recovery / 1				X			EA1.10 Ability to operate and monitor the following as they apply to a reactor trip: S/G pressure	3.7	39 (1)
000008 Pressurizer Vapor Space Accident / 3									
000009 Small Break LOCA / 3						X	2.4.6 Knowledge of EOP mitigation strategies	3.7	40 (2)
000011 Large Break LOCA / 3		X					EK2.02 Knowledge of the interrelations between the Large Break LOCA and the following: Pumps	2.6*	41 (3)
000015/17 RCP Malfunctions / 4									
000022 Loss of Rx Coolant Makeup / 2			X				AK3.03 Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: Performance of lineup to establish excess letdown after determining need	3.1	42 (4)
000025 Loss of RHR System / 4									
000026 Loss of Component Cooling Water / 8					X		AA2.03 Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The valve lineups necessary to restart the CCWS while bypassing the portions of the system causing the abnormal condition	2.6	43 (5)
000027 Pressurizer Pressure Control System Malfunction / 3	X						AK1.02 Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases.	2.8	44 (6)
000029 ATWS / 1		X					EK2.06 Knowledge of interrelations between the following and an ATWS: Breakers, relays and disconnects.	2.9*	45 (7)
000038 Steam Gen. Tube Rupture / 3						X	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	46 (8)
000040 (W/E12) Steam Line Rupture - Excessive Heat Transfer / 4			X				AK3.04 Knowledge of the reasons for the following responses as they apply to the Steam Line Rupture: Actions contained in EOPs for steam line rupture	4.5	54 (16)
000054 Loss of Main Feedwater / 4			X				AK3.01 Knowledge of the reason for the following responses as they apply to the Loss of Main Feedwater (MFW): Reactor and/or turbine trip, manual or automatic	4.1	47 (9)
000055 Station Blackout / 6					X		EA2.04 Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only dc battery power available.	3.7	48 (10)
000056 Loss of Off-site Power / 6	X						AK1.03 Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Definition of subcooling: use of the steam tables to determine it.	3.1*	49 (11)

ES-401

PWR Examination Outline
Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1 (RO / SRO)

Form ES-401-2

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
000057 Loss of Vital AC Inst. Bus / 6				X			AA1.03 Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Feedwater pump speed to control pressure and level in S/G.	3.6*	50 (12)
000058 Loss of DC Power / 6						X	2.1.20 Ability to interpret and execute procedure steps.	4.6	51 (13)
000062 Loss of Nuclear Svc Water / 4			X				AK3.04 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Effect on the nuclear service water discharge flow header on a loss of CCW	3.5	52 (14)
000065 Loss of Instrument Air / 8				X			AA1.04 Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: Emergency air compressor.	3.5	53 (15)
W/E04 LOCA Outside Containment / 3									
W/E11 Loss of Emergency Coolant Recirc. / 4									
W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4		X					EK2.2 Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	3.9	55 (17)
000077 Generator Voltage and Electric Grid Disturbances / 6					X		AA2.04 Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: VARs outside the capability curve.	3.6	56 (18)
K/A Category Totals:	2	3	4	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	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
000001 Continuous Rod Withdrawal / 1	X						AK1.21 Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: Integral rod worth.	2.9	57 (19)
000003 Dropped Control Rod / 1									
000005 Inoperable/Stuck Control Rod / 1									
000024 Emergency Boration / 1									
000028 Pressurizer Level Malfunction / 2			X				EK3.05 Knowledge of the reasons for the following responses as they apply to the Pressurizer level Control malfunctions: Actions contained in EOP for PZR level malfunctions.	3.7	58 (20)
000032 Loss of Source Range NI / 7						X	2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	3.8	59 (21)
000033 Loss of Intermediate Range NI / 7				X			AA1.01 Ability to operate and / or monitor the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Power-available indicators in cabinets or equipment drawers	2.9	60 (22)
000036 Fuel Handling Accident / 8									
000037 Steam Generator Tube Leak / 3					X		AA2.11 Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: When to isolate one or more S/Gs	3.8	61 (23)
000051 Loss of Condenser Vacuum / 4			X				AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capacity upon loss of condenser vacuum	2.8*	62 (24)
000059 Accidental Liquid Radwaste Rel. / 9									
000060 Accidental Gaseous Radwaste Rel. / 9									
000061 ARM System Alarms / 7									
000067 Plant Fire On-site / 8									
000068 Control Room Evac. / 8									
000069 Loss of CTMT Integrity / 5									
000074 (W/E06&E07) Inad. Core Cooling / 4									
000076 High Reactor Coolant Activity / 9					X		AA2.02 Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Corrective actions required for high fission product activity in RCS	2.8	64 (26)
W/E01 & E02 Rediagnosis & SI Termination / 3									
W/E13 Steam Generator Over-pressure / 4									
W/E15 Containment Flooding / 5		X					EK2.1 Knowledge of the interrelations between the (Containment Flooding) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	2.8	65 (27)

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		Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2 (RO / SRO)							
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
W/E16 High Containment Radiation / 9									
W/E09 & 10 Natural Circulation with Steam Void in Vessel with/without RVLIS/4	X						EK1.2 Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Normal, abnormal and emergency operating procedures associated with (Natural Circulation with Steam Void in Vessel with/without RVLIS).	3.4	63 (25)
K/A Category Point Totals:	2	1	2	1	2	1	Group Point Total:		9

ES-401	PWR Examination Outline Plant Systems - Tier 2 / Group 1 (RO / SRO)												Form ES-401-2	
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
003 Reactor Coolant Pump							X					A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP motor stator winding temperatures	2.6	1 (28)
004 Chemical and Volume Control											X	2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm.	4.1	2 (29)
005 Residual Heat Removal	X											K1.06 Knowledge of the physical connections and/or cause effect relationships between the RHRS and the following systems: ECCS	3.5	3 (30)
006 Emergency Core Cooling					X							K5.11 Knowledge of the operational implications of the following concepts as they apply to ECCS: Basic heat transfer equation	2.5	4 (31)
007 Pressurizer Relief/Quench Tank								X				A2.06 Ability to (a) predict the impacts of the following malfunctions or operations on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Bubble formation in PZR	2.6	5 (32)
007 Pressurizer Relief/Quench Tank										X		A4.04 Ability to manually operate and/or monitor in the control room: PZR vent valve	2.6*	6 (33)
008 Component Cooling Water				X								K4.09 Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: The "standby" feature for the CCW pumps	2.7	7 (34)
010 Pressurizer Pressure Control	X											K1.05 Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: PRTS	3.4	8 (36)
010 Pressurizer Pressure Control		X										K2.03 Knowledge of bus power supplies to the following: Indicator for PORV position	2.8*	9 (35)
012 Reactor Protection							X					A1.01 Ability to predict and/or monitor Changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: Trip setpoint adjustment	2.9	10 (38)
012 Reactor Protection									X			A3.07 Ability to monitor automatic operation of the RPS, including: Trip breakers	4.0	11 (37)
013 Engineered Safety Features Actuation	X											K1.18 Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: Premature reset of ESF actuation	3.7	12 (39)

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System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
022 Containment Cooling				X								K4.04 Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Cooling of control rod drive motors	2.8	14 (40)
026 Containment Spray		X										K2.02 Knowledge of bus power supplies to the following: MOVs	2.7*	15 (42)
039 Main and Reheat Steam					X							K5.08 Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity	3.6	16 (43)
059 Main Feedwater									X			A3.02 Ability to monitor automatic operation of the MFW, including: Programmed levels of the S/G	2.9	17 (44)
061 Auxiliary/Emergency Feedwater						X						K6.02 Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps	2.6	24 (52)
061 Auxiliary/Emergency Feedwater								X				A2.06 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: Back leakage of MFW	2.7	18 (45)
062 AC Electrical Distribution										X		A4.03 Ability to manually operate and/or monitor in the control room: Synchroscope, including an understanding of running and incoming voltages	2.8	19 (46)
063 DC Electrical Distribution											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.	4.2	20 (47)
064 Emergency Diesel Generator			X									K3.01 Knowledge of the effect that a loss or malfunction of the ED/G Systems controlled by automatic loader	3.8*	21 (48)
064 Emergency Diesel Generator						X						K6.07 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers	2.7	22 (49)
073 Process Radiation Monitoring								X				A2.02 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: Detector failure	2.7	23 (50)
073 Process Radiation Monitoring			X									K3.01 Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent release	3.6	13 (41)

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System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
076 Service Water										X		A4.02 Ability to manually operate and/or monitor in the control room: SWS valves	2.6	25 (51)
078 Instrument Air		X										K2.01 Knowledge of bus power supplies to the following: Instrument air compressor	2.7	26 (53)
078 Instrument Air									X			A3.01 Ability to monitor automatic operation of the IAS, including: Air pressure	3.1	27 (54)
103 Containment			X									K3.01 Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under shutdown conditions	3.3*	28 (55)
K/A Category Point Totals:	3	3	3	2	2	2	2	3	3	3	2	Group Point Total:		28

ES-401	PWR Examination Outline Plant Systems - Tier 2 / Group 2 (RO / SRO)											Form ES-401-2		
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
001 Control Rod Drive										X		A3.07 Ability to monitor automatic operation of the CRDS, including: Boration/dilution.	4.1	38 (65)
002 Reactor Coolant					X							K5.08 Knowledge of the operational implications of the following concepts as they apply to the RCS: Why PRZ level should be kept within the programmed level.	3.4	31 (58)
011 Pressurizer Level Control											X	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.3	29 (56)
014 Rod Position Indication				X								K4.06 Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following: Individual and group misalignment	3.4	37 (64)
015 Nuclear Instrumentation							X					A1.02 Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the NIS controls including: SUR	3.5	30 (57)
016 Non-Nuclear Instrumentation														
017 In-Core Temperature Monitor														
027 Containment Iodine Removal														
028 Hydrogen Recombiner and Purge Control										X		A4.01 Ability to manually operate and/or monitor in the control room: HRPS controls	4.0*	32 (59)
029 Containment Purge														
033 Spent Fuel Pool Cooling														
034 Fuel Handling Equipment														
035 Steam Generator						X						K6.01 Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: MSIVs	3.2	33 (60)
041 Steam Dump/Turbine Bypass Control			X									K3.02 Knowledge of the effect that a loss or malfunction of the SDS will have on the following: RCS	3.8	34 (61)
045 Main Turbine Generator	X											K1.20 Knowledge of the physical connections and/or cause effect relationships between the MT/G system and the following systems: Protection system	3.4	35 (62)
055 Condenser Air Removal														
056 Condensate														
068 Liquid Radwaste														
071 Waste Gas Disposal														
072 Area Radiation Monitoring														

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System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
075 Circulating Water								X				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the Circulating Water System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of circulating water pumps	2.5	36 (63)
079 Station Air														
086 Fire Protection														
K/A Category Point Totals:	1	0	1	1	1	1	1	1	1	1	1	Group Point Total:		10

Facility: CPNPP			Date of Exam: July 18, 2016			
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.15	Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.	2.7	66		
	2.1.38	Knowledge of the station's requirements for verbal communications when implementing procedures.	3.7	67		
	2.1.41	Knowledge of the refueling process.	2.8	68		
		Subtotal			3	
2. Equipment Control	2.2.13	Knowledge of tagging and clearance procedures.	4.1	69		
	2.2.38	Knowledge of conditions and limitations in the facility license.	3.6	70		
		Subtotal			2	
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	71		
	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	72		
		Subtotal			2	
4. Emergency Procedures / Plan	2.4.39	Knowledge of RO responsibilities in emergency plan implementation.	3.9	73		
	2.4.46	Ability to verify that the alarms are consistent with the plant conditions.	4.2	74		
	2.4.50	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.2	75		
		Subtotal			3	
Tier 3 Point Total				10		

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1. Emergency & Abnormal Plant Evolutions	1															3	3	6	
	2					N/A					N/A					2	2	4	
	Tier Totals															5	5	10	
2. Plant Systems	1															3	2	5	
	2														0	2	1	3	
	Tier Totals														5	3	8		
3. Generic Knowledge and Abilities Categories				1	2	3	4								1	2	3	4	7
															2	1	2	2	

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000007 Reactor Trip - Stabilization - Recovery / 1					X		EA2.05 Ability to determine or interpret the following as they apply to a reactor trip: Reactor trip first-out indication	3.9	76
000008 Pressurizer Vapor Space Accident / 3						X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and interpretation.	4.7	77
000009 Small Break LOCA / 3									
000011 Large Break LOCA / 3									
000015/17 RCP Malfunctions / 4									
000022 Loss of Rx Coolant Makeup / 2									
000025 Loss of RHR System / 4									
000026 Loss of Component Cooling Water / 8									
000027 Pressurizer Pressure Control System Malfunction / 3									
000029 ATWS / 1						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	78
000038 Steam Gen. Tube Rupture / 3									
000040 Steam Line Rupture - Excessive Heat Transfer / 4					X		AA2.05 Ability to determine and interpret the following as they apply to the Steam Line Rupture: When ESFAS systems may be secured	4.5	79
000054 Loss of Main Feedwater / 4									
000055 Station Blackout / 6									
000056 Loss of Off-site Power / 6									
000057 Loss of Vital AC Inst. Bus / 6						X	2.4.11 Knowledge of abnormal condition procedures.	4.2	80
000058 Loss of DC Power / 6									
000062 Loss of Nuclear Svc Water / 4									
000065 Loss of Instrument Air / 8									
W/E04 LOCA Outside Containment / 3									
W/E11 Loss of Emergency Coolant Recirc. / 4									
W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4					X		EA2.1 Ability to determine and interpret the following as they apply to the (Excessive Heat Transfer): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	4.4	81
000077 Generator Voltage and Electric Grid Disturbances / 6									
K/A Category Totals:					3	3	Group Point Total:		6

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000001 Continuous Rod Withdrawal / 1									
000003 Dropped Control Rod / 1						X	AA2.04 Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod motion stops due to dropped rod	3.6	82
000005 Inoperable/Stuck Control Rod / 1						X	2.1.32 Ability to explain and apply system limits and precautions.	4.0	83
000024 Emergency Boration / 1									
000028 Pressurizer Level Malfunction / 2									
000032 Loss of Source Range NI / 7									
000033 Loss of Intermediate Range NI / 7									
000036 Fuel Handling Accident / 8									
000037 Steam Generator Tube Leak / 3									
000051 Loss of Condenser Vacuum / 4									
000059 Accidental Liquid Radwaste Rel. / 9									
000060 Accidental Gaseous Radwaste Rel. / 9									
000061 ARM System Alarms / 7									
000067 Plant Fire On-site / 8						X	2.2.38 Knowledge of conditions and limitations in the facility license.	4.5	84
000068 Control Room Evac. / 8									
000069 Loss of CTMT Integrity / 5									
000074 (W/E06&E07) Inad. Core Cooling / 4									
000076 High Reactor Coolant Activity / 9									
W/E01 & E02 Rediagnosis & SI Termination / 3									
W/E13 Steam Generator Over-pressure / 4									
W/E15 Containment Flooding / 5									
W/E16 High Containment Radiation / 9									
W/E 08 RCS Overcooling - PTS / 4						X	EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.2	85
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	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
003 Reactor Coolant Pump														
004 Chemical and Volume Control														
005 Residual Heat Removal														
006 Emergency Core Cooling								X				A2.13 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: Inadvertent SIS actuation	4.2	86
007 Pressurizer Relief/Quench Tank														
008 Component Cooling Water											X	2.2.3 (multi-unit license) Knowledge of the design, procedural, and operational differences between units.	3.9	87
010 Pressurizer Pressure Control														
012 Reactor Protection														
013 Engineered Safety Features Actuation								X				A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; 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	4.2	88
022 Containment Cooling														
026 Containment Spray														
039 Main and Reheat Steam								X				A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioning steam dump	3.7	89
059 Main Feedwater											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	90
061 Auxiliary/Emergency Feedwater														
062 AC Electrical Distribution														
063 DC Electrical Distribution														
064 Emergency Diesel Generator														

ES-401

PWR Examination Outline
 Plant Systems - Tier 2 / Group 1 (RO / **SRO**)

Form ES-401-2

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
073 Process Radiation Monitoring														
076 Service Water														
078 Instrument Air														
103 Containment														
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	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
001 Control Rod Drive														
002 Reactor Coolant														
011 Pressurizer Level Control														
014 Rod Position Indication														
015 Nuclear Instrumentation														
016 Non-Nuclear Instrumentation														
017 In-Core Temperature Monitor											X	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits	4.2	91
027 Containment Iodine Removal														
028 Hydrogen Recombiner and Purge Control														
029 Containment Purge														
033 Spent Fuel Pool Cooling														
034 Fuel Handling Equipment								X				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Dropped fuel element	4.4	93
035 Steam Generator								X				A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the S/GS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulted or ruptured S/Gs	4.6	92
041 Steam Dump/Turbine Bypass Control														
045 Main Turbine Generator														
055 Condenser Air Removal														
056 Condensate														
068 Liquid Radwaste														
071 Waste Gas Disposal														
072 Area Radiation Monitoring														
075 Circulating Water														
079 Station Air														
086 Fire Protection														
K/A Category Point Totals:								2			1	Group Point Total:		3

Facility: CPNPP			Date of Exam: July 18, 2016			
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.2	Knowledge of operator responsibilities during all modes of plant operation.			4.4	94
	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	95
		Subtotal				2
2. Equipment Control	2.2.42	Ability to recognize system parameters that are entry-level conditions for Technical Specifications.			4.6	96
		Subtotal				1
3. Radiation Control	2.3.11	Ability to control radiation releases.			4.3	97
	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.			3.7	98
		Subtotal				2
4. Emergency Procedures / Plan	2.4.29	Knowledge of the emergency plan			4.4	99
	2.4.40	Knowledge of SRO responsibilities in emergency plan implementation.			4.5	100
		Subtotal				2
Tier 3 Point Total						7

Tier/Group	Randomly Selected K/A	Reason for Rejection
1/1	009 G.2.4.6	Question 40 Small Break LOCA: Unable to write an operationally valid question to RO tasks performed outside the control room as the recovery procedures do not have any actions which are performed by the ROs outside of the control room. For the few instances that field actions are taken these would not be performed by the ROs. Randomly/Systematically Replaced K/A 009 G.2.4.34 with K/A 009 G.2.4.6.
1/1	022 AK3.03	Question 42 Loss of Rx Coolant Makeup: Unable to write an operationally valid question that could not be challenged as CPNPP procedure guidance does not exist for reasons to avoid plant transients, thus no plausible distractors could be excluded. Randomly/Systematically Replaced K/A 022 AK3.05 with K/A 022 AK3.03.
1/1	027 AK1.02	Question 44 Pressurizer Pressure Control Malfunctions: Could not write a discriminating question based on the content of the original K/A. Randomly/Systematically Replaced K/A 027 AK1.01 with K/A 027 AK1.02
1/1	038 G.2.4.4	Question 46 Steam Generator Tube Rupture: Could not write a question to the original K/A as no immediate action steps exist for a SGTR. Randomly/Systematically Replaced K/A 038 G.2.4.1 with K/A 038 G.2.4.4.
1/1	054 AK3.01	Question 47 Loss of Main Feedwater: Could not write an operationally valid RO level question on PORVs cycling as this requires operational errors in performing the actions of FRH-0.1A. Additionally, this is a Loss of Heat Sink which is beyond the Loss of Main Feedwater. Randomly/Systematically Replaced K/A 054 AK3.05 with AK3.01.
1/1	057 AA1.03	Question 50 Loss of Vital AC Inst. Bus: Unable to an develop operationally valid question as no guidance is provided for use of backup instrument indications. Randomly/Systematically Replaced K/A 057 AA1.05 with 057 AA1.03.
1/1	G.2.1.20	Question 51 Loss of DC Power: Unable to write operationally valid question for original K/A. Randomly/Systematically Replaced K/A 058 G.2.2.12 with 058 G.2.1.20.
1/1	065 AA1.04	Question 53 Loss of Instrument Air: CPNPP design does not include manual loaders. Randomly/Systematically Replaced K/A 065 AA1.01 with K/A 065 AA1.04.

1/1	040 AK3.04	Question 54 LOCA Outside Containment: This ECA is three steps in length and the details of this procedure are beyond RO knowledge. Randomly/Systematically Replaced K/A W/E 04 EK3.02 with 040 AK3.04
1/2	033 AA1.01	Question 60 Loss of Intermediate Range NI: Unable to write an operationally valid question with plausible distractors based on CPNPP design for the original K/A. Randomly/Systematically Replaced K/A 033 AA1.02 with K/A 033 AA1.01.
1/2	W/E10 EK1.2	Question 63 Loss of Containment Integrity: Unable to write an operationally valid question on the effect of pressure on leak rate as licensed operators are not required to perform tasks during normal or emergency operations of calculating containment leakage rates. Randomly/Systematically Replaced K/A 069 AK1.01 with W/E10 EK1.2.
2/1	004 G.2.4.45	Question 2 Chemical and Volume Control: Original K/A was not RO level knowledge. Randomly/Systematically Replaced K/A 004 G.2.4.41 with K/A 004 G.2.4.45.
2/1	073 K3.01	Question 13 Containment Cooling: Original K/A concerning equipment damage to containment equipment by temperature, humidity and pressure was beyond RO knowledge level as these concerns are addressed via the Technical Specifications and Technical Requirement Manual below the line of RO knowledge. Randomly/Systematically Replaced K/A 022 K3.01 with K/A 073 K3.01.
2/1	026 K2.02	Question 15 Containment Spray: Original K/A was on power supply to the pumps. As the pumps are powered directly from the respective train related 6.9 kV Safeguards Bus an operationally discriminating question could not be written. Randomly/Systematically Replaced K/A 026 K2.01 with K/A 026 K2.02.
2/1	061 K6.02	Question 24 Service Water: Original K/A was intended to be a K6 per the random selection process and K4.03 was listed as the topic. This placed the Tier Totals out of balance. Randomly/Systematically Replaced K/A 076 K4.03 with K/A 061 K6.02.
2/1	078 K2.01	Question 26 Instrument Air: Original K/A involved securing Service Air upon loss of cooling water. The instrument and service air systems at CPNPP are not connected and thus this K/A is not applicable to the plant. Randomly/Systematically Replaced K/A 078 K4.03 with K/A 078 K2.01.

2/2	002 K5.08	Question 31 In-core Temperature Monitor: Could not write an operationally valid question on the operational implications of the ITM system with respect to saturation and subcooling of water without overlapping other questions which existed on the exam. Randomly/Systematically Replaced K/A 017 K5.02 with K/A 002 K5.08.
2/2	075 A2.02	Question 36 Waste Gas Disposal: Replaced K/A as CPNPP design has any stuck open relief valve either discharging to another Gas Decay Tank or discharging via an unisolable path to the plant vent thus no procedural actions exist for the failure. Randomly/Systematically Replaced K/A 071 A2.09 with K/A 075 A2.02.
2/2	014 K4.06	Question 37 Area Radiation Monitoring: Replace K/A as CPNPP Area Radiation Monitoring system does not provide isolations or interlocks to operations. Randomly/Systematically Replaced K/A 072 K4.02 with K/A 014 K4.06.
2/2	001 A3.07	Question 38 Circulating Water: Replaced K/A as CPNPP design does not have emergency essential SSW pumps in the circulating water system. Randomly/Systematically Replaced K/A 075 K2.03 with K/A 001 A3.07.
3	G. 2.2.13	Question 69 ROs do not manage maintenance activities and therefore this generic was not applicable to the RO position. Randomly/Systematically Replaced Generic G.2.2.18 with G.2.2.13.
3	G.2.3.12	Question 72 Question was unsat based on overlap with Operating Test and LOD was rated by NRC as 1 based on looking up on an RWP. Randomly/Systematically Replaced Generic G.2.3.7 with G.2.3.12.
1/1	008 G.2.1.7	Question 77 Pressurizer Vapor Space Accident: Original K/A was an oversampling of G.2.1.32 with Question 83. Randomly/Systematically Replaced K/A 008 G.2.1.32 with K/A 008 G.2.1.7.
1/1	040 AA2.05	Question 79 Steam Line Rupture – Excessive Heat Transfer: Original K/A asked for the ability to determine and interpret the difference between a Steam Line Rupture and LOCA. Was unable to write a question which met the NUREG-1021 criteria for SRO only as the determination aligned itself with identification and entry into Major EOPs as delineated in NUREG-1021. Randomly/Systematically Replaced K/A 040 AA2.03 with K/A 040 AA2.05.

1/1	057 G.2.4.11	Question 80 Loss of Main Feedwater: Original K/A did not have a corresponding 10CFR 55.43(b) tie. Randomly/Systematically Replaced K/A 054 G.2.4.31 with K/A 054 G.2.4.11. Unable to write acceptable SRO level question to selected K/A. Randomly/Systematically Replaced K/A 054 G.2.4.11 with K/A 057 G.2.4.11.
1/2	W/E 08 A2.01	Question 85 Steam Generator Over-pressure: Original K/A was based on a Westinghouse Yellow path FRG. This procedure has few steps and does not offer discriminating value for an SRO only question. Randomly/Systematically Replaced K/A W/E 13 A2.02 with W/E 08 A2.01.
2/2	017 G.2.2.25	Question 91 Containment Iodine Removal: As the Preaccess filtration system is the only approved Iodine Removal System at CPNPP, was unable to write an operationally valid SRO level question to a non-safety system with limited administrative requirements. Randomly/Systematically Replaced K/A 027 G.2.4.50 with 017 G.2.2.25.
2/2	035 A2.01	Question 92 Spent Fuel Pool Cooling: Unable to write an SRO discriminatory question to the K/A. Randomly/Systematically Replaced K/A 033 A2.02 with K/A 035 A2.01.
3	G.2.1.4	Question 95 Unable to write a question at the SRO level for locating and operating components that is not at the RO level or is not tied to a specific system or event which is contrary to the Tier 3 requirements. Randomly/Systematically Replaced K/A G.2.1.30 with K/A G.2.1.4.
3	G.2.4.29	Question 99 Unable to write an SRO level question to the original K/A. Randomly/Systematically Replaced K/A G.2.4.17 with K/A G.2.4.18. Unable to write an acceptable Tier 3 question for the replaced K/A. Randomly/Systematically Replaced K/A G.2.4.18 with K/A G.2.4.29.

Facility: CPNPP Units 1 and 2		Date of Examination: July 2016
Examination Level: RO <input checked="" type="checkbox"/> SRO <input type="checkbox"/>		Operating Test Number: NRC
Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations (RA1)	D,R	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)	M,R	2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant temperature, secondary plant, fuel depletion etc. (4.1) JPM: Determine Reactivity Effects When Starting Positive Displacement Charging Pump. (RO1310E)
Equipment Control (RA3)	D,R	2.2.1 Ability to perform pre-startup procedures including operating those controls associated with plant equipment that could affect reactivity. (4.5) JPM: Perform a 1/M Plot and Predict Critical Conditions. (RO1003A)
Radiation Control (RA4)	D,R	2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions. (3.5) JPM: Determine Entry Conditions for Radiation Area Clearance. (RWT056B)
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 calculate BOL Boration for long term use per SOP-104A, Reactor Makeup and Chemical Control System, Attachment 2, BOL Boration for Long-Term Use. Critical steps include 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 direct from bank JPM. (K/A 2.1.23 - IR 4.3)
- RA2 The applicant is presented with information pertaining to the boron concentration in the suction piping to the Positive Displacement Charging Pump and the Reactor Coolant System. The applicant will use SOP-103A, Chemical and Volume Control System to determine the reactivity effects of the planned evolution. The critical steps will be to calculate the change in RCS boron concentration and RCS temperature. The JPM is modified from a previous version by changing the RCS boron concentration and the concentration in the PDP suction line. The prior version was a resulting boration. The modified version is a resulting dilution. This is a modified JPM. (K/A 2.1.43 - IR 4.1)
- RA3 The applicant will perform a 1/M plot for a Reactor Startup per IPO-002A, Plant Startup From Hot Standby, Attachment 2, Inverse Count Rate Ratio Calculation. The critical steps include calculating and plotting 1/M, predicting critical conditions, and identifying action for criticality above the power dependent insertion limit. This is a direct from bank JPM. (K/A 2.2.1 - IR 4.5)
- RA4 The applicant will determine the radiological requirements for implementing a Clearance in a Radiological Controlled Area per STA-656, Radiation Work Control, RPI-602, Radiological Surveillance and Posting, and RPI-606, Radiation Work and General Access Permits. Critical tasks include identifying Dose Monitoring Requirements, Protective Clothing Requirements, highest contamination level, and highest dose rate. This is a direct from bank JPM. (K/A 2.2.37 - IR 3.5)

Task Summary

Facility: CPNPP Units 1 and 2		Date of Examination: July 2016	
Examination Level: RO <input type="checkbox"/> SRO <input checked="" type="checkbox"/>		Operating Test Number: NRC	
Administrative Topic (See Note)	Type Code*	Describe activity to be performed	
Conduct of Operations (SA1)	N,R	2.1.26	Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperatures, high pressure, caustic, chlorine, oxygen and hydrogen). (3.6)
		JPM:	Determine Electrical Safe Work Practices Requirements. (SO1028)
Conduct of Operations (SA2)	N,R	2.1.37	Knowledge of procedures, guidelines or limitations associated with reactivity management. (4.6)
		JPM:	Determine Reactivity Management Severity and Notifications. (SO1017B)
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	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions. (3.6)
		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. (4.1)
		JPM:	Determine Entry Conditions for Radiation Area Clearance and Reporting Requirements. (SO1112B)
Emergency Procedures/Plan (SA5)	D,R	2.4.41	Knowledge of emergency action level thresholds and classifications. (4.6)
		JPM:	Classify an Emergency Plan Event. (SO1136I)

Task Summary

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)

- SA1 The applicant is presented with a task to determine as the Unit Supervisor, the Personnel Protective Equipment and Safety Boundaries for emergent work of racking the Rx trip breaker from disconnect to remove in accordance with STA-124, Electrical Safe Work Practices. The critical steps will be to identify the Hazard/Risk Category, Clothing requirements and Boundaries. In addition, the applicant will be required to determine if their position has approval authority for the task. This is a new JPM. (K/A 2.1.26 - IR 3.6)
- SA2 The applicant is presented with a plant transient event and response. As Unit Supervisor, the applicant is required to take necessary actions for a reactivity management event in accordance with STA-102, Reactivity Management Program. The critical steps will be to make a determination of the Severity Level and determine the written and verbal notifications. This is a new JPM. (K/A 2.1.37 - IR 4.6)
- SA3 The applicant will evaluate a Fire Protection Impairment per STA-738, Fire Protection Systems/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 will determine the radiological requirements for implementing a Clearance in a Radiological Controlled Area per STA-656, Radiation Work Control, RPI-602, Radiological Surveillance and Posting, RPI-606, Radiation Work and General Access Permits and STA-501, Nonroutine Reporting. Critical steps include identifying Dose Monitoring Requirements, Protective Clothing Requirements, highest contamination level, highest dose rate and determination of proper oral and written notifications due to an overexposure event. This is a direct from bank JPM. (K/A 2.3.7 - IR 3.6 & K/A 2.4.30 - IR 4.1)
- SA5 The applicant will determine the appropriate Emergency Plan Classification in accordance with EPP-201, Assessment of Emergency Action Levels, Emergency Classification, and Plan Activation. The critical step will be the determination of the correct classification. This is a direct from bank JPM. (K/A 2.4.41 - IR 4.6)

Facility: CPNPP Units 1 and 2		Date of Examination: July 2016	
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	003 – Dropped Control Rod (RO1024A) Respond to Control Rod Misalignment	M,S	1
S-2	010 – Pressurizer Pressure Control System (RO1205) PORV Block Valve Operability Test	A,D,S	3
S-3	002 – Reactor Coolant System (RO1412C) Respond to a Shutdown Loss of Coolant	L,M,S	4P
S-4	045 – Main Turbine Generator System (RO3113) Perform Pre-Startup Turbine Trip Checks	A,L,N,S	4S
S-5	026 – Containment Spray System (RO2002C) Transfer Containment Spray to Recirculation with Cavitation	A,D,EN,L,S	5
S-6	064 – Emergency Diesel Generator System (RO4215B) Restore Safeguards Bus 1EA1 to Offsite Power	A,D,P,S	6
S-7	015 – Nuclear Instrumentation System (RO1820) Respond to a Power Range Channel Malfunction	D,S	7
S-8	067 – Plant Fire On-site (RO4405) Respond to Fire in the Safeguards Building	D,S	8
In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)			
P-1	004 – Chemical and Volume Control (AO5202A) Perform Local Actions to Restart the Positive Displacement Pump	A,D,E,R	2
P-2	055 – Loss of All AC Power (RO4217H) Perform Attachment 2A DC Load Shedding	N,E,L	6
P-3	068 – Control Room Evacuation (AO5115B) Emergency Borate from the Remote Shutdown Panel	D,E,L,R	8

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

NRC JPM Examination
Summary Description

- S-1 Control Rod H-8 which is part of Control Bank D is misaligned from its bank. Control Rod H-8 is at 204 steps as indicated on DRPI and Control Bank D indicates 216 steps. The applicant is provided ABN-712, Rod Control System Malfunction and is required to realign Control Rod H-8 using the DRPI Method. The critical steps include selecting the proper bank, withdrawing the entire bank to a known position, deselecting the non-misaligned rods from moving, aligning Control Rod H-8, resetting the Rod Control Urgent Failure alarm, returning the entire bank to its pre-malfunction position and restoring the Control Rod system for continued operation. This is a modified from bank JPM as a recent procedural change added the directions which are to be used for clearing the Rod Control Urgent Failure alarm if present and which this JPM now exercises. This JPM is under the Control Rod Drive System – Reactivity Control Safety Function. (K/A 003.AA1.02 - IR 3.6 / 3.4)
- S-2 The applicant will be provided with OPT-109A, PORV Block Valve Test and will be required to perform the Operability Test. This is an Alternate Path JPM because when PORV Block Valve 1/1-8000B is reopened as part of the test, the PORV partially opens requiring the applicant to take action to isolate the open PORV. The critical steps include closing each PORV Block Valve, performing the stroke test of each PORV and restoring the original configuration. An additional critical step of isolating the stuck open PORV follows the malfunction. PORV Block Valves are provided to isolate a PORV if excessive leakage develops and are discussed in FSAR 15.4.13.2. This is a direct from bank JPM under the Pressurizer Pressure Control System – Reactor Pressure Control Safety Function. (K/A 010.A4.03 - IR 4.0 / 3.8)

- S-3 The applicant will respond to a lowering Pressurizer level with the Residual Heat Removal System in service per ABN-108, Shutdown Loss of Coolant, Section 2.0, Shutdown Loss of Coolant. This is a modified JPM under the Residual Heat Removal System – Primary System Heat Removal from Reactor Core Safety Function. The modification consists of a different plant configuration as the Initial Conditions which do not require performance of an Alternate Path. (K/A 025.AA1.02- IR 3.8 / 3.9)
- S-4 The applicant will use OPT-410A, Pre-Startup Turbine Trip Checks to perform the task. This is an Alternate Path JPM as the Turbine speed will increase above the allowable procedural guidance while the HP Stop Valves are opening. This speed increase requires that the turbine be tripped in accordance with OPT-410A. The critical steps will include resetting the turbine trip, latching the turbine, opening the HP Stop Valves and tripping the turbine when speed increases. This is a new JPM under the Main Turbine Generator System – Heat Removal from Reactor Core Secondary Systems Safety Function. (K/A 045.A4.01 - IR 3.1 / 2.9)
- S-5 Following a LBLOCA, the applicant will transfer the Containment Spray System from the Injection mode to Recirculation in accordance with EOS-1.3A, Transfer to Cold Leg Recirculation. This is an Alternate Path JPM as the applicant will not be able to open the containment sump valves to the Train B Containment Spray Pumps. This will require the applicant to secure Train B. Critical steps will include transferring Train A suction to the containment sump and securing both Train B pumps when suction cannot be realigned. Transferring Containment Spray to Recirculation Mode is considered a Time Significant Action. STI-214.01, Control of Timed Operator Actions, TSA-2.8 requires Containment Spray transferred to Recirculation Mode within 70 seconds of RWST level reaching 6%. This Time Significant Action is performed to avoid the requirement to secure Containment Spray Pumps due to losing suction supply when RWST level reaches 0%. This is a direct from bank JPM under the Containment Spray System – Containment Integrity Safety Function. (K/A 026.A4.01 - IR 4.5 / 4.3)
- S-6 The applicant will restore Offsite Power to 6.9 kV Safeguards Bus 1EA1 from Transformer XST1 in accordance with SOP-609A, Diesel Generator System, Section 5.7, Transferring From DG Supplying Alone to Normal or Alternate Supply. The alternate path occurs when a lowering frequency requires separating the Emergency Diesel Generator from the grid. This is a bank JPM, previously used on the 2014 NRC operating test, under the Emergency Diesel Generator System – Electrical Safety Function. (K/A 064.A4.07 - IR 3.4 / 3.4)
- S-7 Following a Power Range Instrument failure. The applicant is required to perform the actions of ABN-703, Power Range Instrument Malfunction. Critical steps 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 under the Nuclear Instrumentation System – Instrumentation Safety Function. (K/A 015.A2.01 - IR 3.5 / 3.9)

- S-8 A fire has been identified in the Safeguards Building. The applicant is directed to respond to the fire in accordance with ABN-804A, Response to a Fire in the Safeguards Building. Critical steps include performing an emergency start of Diesel Generator 1-02, performing CVCS realignments and starting CCP 1-02. Comanche Peak has commitments within ABN-804A, Response to a Fire in the Safeguards Building, to maintain CCP suction due to possible Gas Intrusion as noted in SOER 97-01, Loss of HP Injection & Charging from Gas Intrusion. This is a direct from bank JPM under the Plant Fire On-site – Plant Service Systems Safety Function. (K/A 067.AA2.16 - IR 3.3 / 4.0)
- P-1 Following a loss of instrument air, the applicant is required to reset control air to the Positive Displacement Charging Pump in accordance with ABN-301, Instrument Air System Malfunction and restore the PDP to operation in accordance with SOP-103A, Chemical and Volume Control System. This JPM is Alternate Path as the Stuffing Box Coolant Tank level is out of specification during the pump restart and requires filling. Critical steps include resetting the air to the hydraulic speed changer, repositioning the fill valve to the coolant tank and opening the pump discharge valve. This is a direct from bank JPM under the Chemical and Volume Control System – Reactivity Control System Inventory Control Safety Function. (K/A 004.A4.08 - IR 3.8 / 3.4)
- P-2 During a complete loss of All AC Power, the applicant is required to perform ECA-0.0A, Loss of All AC Power Attachment 2A which is Initial DC Load Shed. Critical steps include performing several operations on Distribution Panels to properly align equipment from Unit 2 where possible and shed loads where required. This is a new JPM as DC Load Shedding has been redeveloped following BDBEE considerations. (K/A 055.EA1.04 3.5/3.9)
- P-3 During a Control Room evacuation due to a security threat, the applicant is required to take action to place the plant in control of the operators from outside the control room. Actions will be performed using ABN-905B, Loss of Control Room Habitability. The critical steps include transferring control of equipment from the Control Room to the Hot Shutdown Panel, starting a Boric Acid Transfer Pump and opening the emergency borate valve. This is a direct from bank JPM under the Control Room Evacuation System – Plant Service Systems Safety Function. (K/A 068.AA1.11 - IR 3.9 / 4.1)

Facility:	CPNPP 1 & 2	Scenario No.:	2	Op Test No.:	July 2016 NRC
Examiners:	_____	Operators:	_____		
	_____		_____		
	_____		_____		
Initial Conditions: 53% power MOL - RCS Boron is 1054 ppm					
Turnover: 600 MWe due to a B MFP trip. 1-02 MDAFWP in Pull Out (DANGER tagged) with breaker de-energized for scheduled maintenance. Restored B MFP following the trip but 1-PV-2286 was damaged and is DANGER tagged out for repairs. Hold power per load dispatch.					
Critical Tasks: CT-1 Initiate Emergency Boration prior to exiting EOS-0.1A, Reactor Trip Response within a maximum of 15 minutes following the loss of DRPI.					
CT-2 Establish an RCS bleed and feed path prior to exiting FRH-0.1A, Response to a Loss of Secondary Heat Sink.					
Event No.	Malf. No.	Event Type*	Event Description		
1	RX05A	I (RO, SRO) TS (SRO)	PRZR level instrument LT-459 fails low		
2	RX18	I (BOP, SRO)	Feed Header Pressure Transmitter (PT-508) Fails High		
3	RX09A	I (RO, BOP, SRO) TS (SRO)	Main Turbine 1 st Stage Pressure (PT-505A) Fails Low		
4	CH03	C (BOP, SRO)	Neutron Detector Well Fan 9 trips on motor overload		
5	FW06A	C (BOP, SRO)	Main Feed Pump A Recirc valve fails open		
6	ED02	TS (SRO)	Loss of XST1 Transformer		
7	ED01 EG06A	M (RO,BOP,SRO)	Loss of offsite power Failure of the DG 1-01 to start (air start failure)		
8			Emergency Boration due to loss of DRPI		
9	FW09A	M (RO,BOP,SRO)	Loss of all AFW TDAFWP Overspeed Trip		
* (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)
6	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 Event Description
NRC Scenario 2

SCENARIO SUMMARY NRC 2

Turnover:

The plant is at 600 MW following a B MFP trip. Reactor power is being held stable per instruction of the Load Dispatcher. MDAFWP 1-02 is Danger tagged for planned maintenance. When the B MFP tripped it caused damage to 1-PV-2286, Low Pressure Feedwater Heater Bypass Valve. The MFP has been restored to operation but 1-PV-2286 is Danger tagged to complete repairs.

Event 1 (Key 1)

The first event will be a PRZR level channel (LT-459) failing low. Entry into ABN-706, PRZR Level Instrumentation Malfunction, section 2.0, will be required. Letdown will isolate, charging will be placed in manual to control PRZR level. Actions will include selecting an operable channel, restoring letdown, and then restoring PRZR level to program and placing controls back in automatic. The SRO will determine the loss of this channel is a TS entry for LCO 3.3.1, Reactor Trip System Instrumentation; function 9, Condition M.

Event 2 (Key 2)

The next event is Main Feedwater (MFW) Header Pressure Transmitter (PT-508) will fail high. Entry into ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st Stage Pressure, and Feed Header Pressure Instrument Malfunction Section 5.0, is required. Section 5.0 is designated for Feed Header Pressure Malfunction. Actions include placing the MFW Pump Turbine Master Speed Controller in MANUAL. This controller will remain in MANUAL for the duration of the scenario and require monitoring/adjustment. If Pressurizer Pressure falls below 2220 psig, the DNB TS 3.4.1 should be entered.

Event 3 (Key 3)

Once the plant is stabilized, the next event is a Turbine 1st Stage Pressure Transmitter (PT-505) will fail low. Crew actions are per ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st Stage Pressure, and Feed Header Pressure Instrument Malfunction, Section 4.0 is required. Section 4.0 is designated for Turbine 1st Stage Pressure Malfunction. Actions include placing Rod Control in Manual and bypassing the failed Turbine 1st Stage Pressure channel. The SRO will refer to Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation; Function 18f, Condition T.

Event 4 (Key 4)

The next event will be a trip of the running Neutron Detector Well Fan #9. This will alarm 2.1 CNTMT FN MASTER TRIP. The ALM will direct the crew to determine which fan has tripped and start the other fan as required using SOP-801A, Containment Ventilation System. The crew will place the tripped fan handswitch in Pull Out or Stop as applicable.

Event 5 (Key 5)

The next event is MFP 'A' Recirculation Valve, 1-FCV-2289, opening. ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, section 11.0 will be entered. Since earlier in this scenario the main feedwater header pressure transmitter failed MFP speed control is in manual. Manual speed control is required to restore S/G levels and stabilize the plant. The RO must ensure rods are in auto for this event. The crew will dispatch an operator to isolate the failed open recirculation valve. Once the failed valve is isolated, the BOP will adjust MFP speed again for the current plant configuration.

Event 6 (Key 6)

The next event is a loss of XST1 which is the alternate offsite power source for Unit 1. The ALM will have the crew enter ABN-601, Response to a 138/345 KV System Malfunction, as well as a TS entry for 3.8.1, Electrical Power Systems, AC Sources – Operating, Condition A.

Scenario Event Description
NRC Scenario 2

Event 7 (Key 7)

The major event is a loss of all offsite power causing a reactor trip. The crew will enter EOP-0.0A, Reactor Trip or Safety Injection. Coincident with the loss of offsite power DG 1-01 will fail to auto start and cannot be manually started due to an air start failure. This will cause a complete loss of all safeguards train 'A' power. The crew will transition to EOS-0.1A, Reactor Trip Response, to continue with recovery efforts.

Event 8 – CT-1 (Auto)

Due to the Loss of Offsite Power, DRPI is lost and per EOP-0.0A and EOS-0.1A, Reactor Trip Response, Attachments 1.A, an Emergency Boration will be required. The crew will then perform CT-1; Initiate Emergency Boration prior to exiting EOS-0.1A, Reactor Trip Response. This will be completed by entering ABN-107, Emergency Boration.

Event 9 (Auto Triggered when 1/1-8104 is placed in open per ABN-107)

After the crew has commenced the emergency boration the TDAFWP will trip on Overspeed. This combined with the loss of all Safeguards Train 'A' power as well as the inoperability of MDAFWP 1-02 will place the crew in a loss of heat sink event. The crew will enter FRH-0.1A, Response to Loss of Secondary Heat Sink, and actuate SI. The crew will then perform CT-2, Establish an RCS bleed and feed path prior to exiting FRH-0.1A, Response to a Loss of Secondary Heat Sink.

Termination Criteria

This scenario is terminated after the crew establishes a bleed and feed path per FRH-0.1A. One CCP and one SI pump running with both PRZR PORVs open.

Scenario Event Description
NRC Scenario 2

Risk Significance Determination

Risk Significance	Event	Guidance
Failure of risk important systems prior to Reactor Trip	Loss of Transformer XST1	FSAR 8.2.1.2.1 – Two independent power sources are available on an immediate basis following a DBA to ensure operation of the vital safety functions. The second offsite power source will no longer be available on loss of XST1.
Risk significant core damage sequence	<p>FSAR 15.2.6.3 Loss of Non-emergency AC power to the station auxiliaries</p> <p>Loss of Secondary Heat Sink</p>	<p>EOS-0.2A, Natural Circulation Cooldown – For Units 1 and 2, the analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.</p> <p>CPNPP Accident Sequence Quantification, R&R-PN-022 – Loss of secondary heat removal, not related to ventilation failures, accounts for about 9% of CDF.</p>
Risk significant operator actions	Initiation of Boration to Add Negative Reactivity to the Core (TSA 2.14)	STI 214.01; ABN-107, Emergency Boration; WCAP-1687 1-P, Section 6.3.5; TRM Bases 13.1.31 – Within 15 minutes, when local alignment is required to establish boration flow. Boration is initiated within the prescribed time. When local manual control credited, admin controls are utilized to ensure personnel are aware/designated to perform alignment to establish boration flow.

Scenario Event Description
NRC Scenario 2

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
<p>Initiate Emergency Boration prior to exiting EOS-0.1A, Reactor Trip Response within a maximum of 15 minutes following the loss of DRPI.</p>	<p>Shutdown Margin must be maintained. Since there are NO DRPI lights lit the bases states to borate at least 3600 gallons of 7000 ppm borated water to ensure shutdown margin is maintained. This gallon value corresponds to 2 of the most reactive rods stuck out.</p>	<p>After the loss of offsite power and the failure of the DG 1-01, DRPI will be dark and no CCP will be running. Per attachment 1.A of EOS-0.1A, ABN-107 will be performed.</p>	<p>Started CCP 1-02, started Boric Acid Transfer Pump 1-02, and opened 1/1-8104, EMER BORATE VLV.</p>	<p>Boration flow will be indicated on 1-FI-183A, EMER BORATE FLO.</p>
<p>Establish an RCS bleed and feed path prior to exiting FRH-0.1A, Response to a Loss of Secondary Heat Sink.</p>	<p>Actuating SI will ensure a feed path of cool water to the RCS (core) and isolate the containment to confine any RCS releases from the bleed flow. The bleed flow through both PORVs will ensure that enough cool water will feed from the ECCS flow path to remove sufficient decay heat.</p>	<p>AFW flow will not be indicated on any AFW flow meter. Also no AFW pumps will be running. A RED path showing on CSFST for heat sink. The need for a heat sink as indicated by RCS temperature and pressure.</p>	<p>Actuated SI, ensured at least one CCP and SI pump is running with flow indicated providing a feed path for the RCS. Both PRZR PORVs open providing a bleed path for the RCS.</p>	<p>Flow indicated on both a CCP and an SI pump. PRZR PORVs open with block valves open. RCS pressure and temperature lowering.</p>

Facility:	CPNPP 1 & 2	Scenario No.:	2	Op Test No.:	July 2016 NRC
Examiners:	_____	Operators:	_____		
	_____		_____		
	_____		_____		
Initial Conditions: 53% power MOL - RCS Boron is 1054 ppm					
Turnover: 600 MWe due to a B MFP trip. 1-02 MDAFWP in Pull Out (DANGER tagged) with breaker de-energized for scheduled maintenance. Restored B MFP following the trip but 1-PV-2286 was damaged and is DANGER tagged out for repairs. Hold power per load dispatch.					
Critical Tasks: CT-1 Initiate Emergency Boration prior to exiting EOS-0.1A, Reactor Trip Response within a maximum of 15 minutes following the loss of DRPI.					
CT-1 Manually control the Main Feedwater Master Speed Controller to prevent receiving an automatic reactor trip due to low steam generator levels, or a trip of Main Feed Pumps due to low suction pressure, and subsequent manual reactor trip.					
CT-2 Establish an RCS bleed and feed path prior to exiting FRH-0.1A, Response to a Loss of Secondary Heat Sink.					
Event No.	Malf. No.	Event Type*	Event Description		
1	RX05A	I (RO, SRO) TS (SRO)	PRZR level instrument LT-459 fails low		
2	RX18	I (BOP, SRO)	Feed Header Pressure Transmitter (PT-508) Fails High		
3	RX09A	I (RO, BOP, SRO) TS (SRO)	Main Turbine 1 st Stage Pressure (PT-505A) Fails Low		
4	CH03	C (BOP, SRO)	Neutron Detector Well Fan 9 trips on motor overload		
5	FW06A	C (BOP, SRO)	Main Feed Pump A Recirc valve fails open		
6	ED02	TS (SRO)	Loss of XST1 Transformer		
7	ED01 EG06A	M (RO,BOP,SRO)	Loss of offsite power Failure of the DG 1-01 to start (air start failure)		
8			Emergency Boration due to loss of DRPI		
9	FW09A	M (RO,BOP,SRO)	Loss of all AFW TDAFWP Overspeed Trip		
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Technical Specifications					

Actual	Target Quantitative Attributes
8 7	Total malfunctions (5-8)
2	Malfunctions after EOP entry (1-2)
6 5	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 SUMMARY NRC 2

Turnover:

The plant is at 600 MW following a B MFP trip. Reactor power is being held stable per instruction of the Load Dispatcher. MDAFWP 1-02 is Danger tagged for planned maintenance. When the B MFP tripped it caused damage to 1-PV-2286, Low Pressure Feedwater Heater Bypass Valve. The MFP has been restored to operation but 1-PV-2286 is Danger tagged to complete repairs.

Event 1 (Key 1)

The first event will be a PRZR level channel (LT-459) failing low. Entry into ABN-706, PRZR Level Instrumentation Malfunction, section 2.0, will be required. Letdown will isolate, charging will be placed in manual to control PRZR level. Actions will include selecting an operable channel, restoring letdown, and then restoring PRZR level to program and placing controls back in automatic. The SRO will determine the loss of this channel is a TS entry for LCO 3.3.1, Reactor Trip System Instrumentation; function 9, Condition M.

Event 2 (Key 2)

The next event is Main Feedwater (MFW) Header Pressure Transmitter (PT-508) will fail high. Entry into ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st Stage Pressure, and Feed Header Pressure Instrument Malfunction Section 5.0, is required. Section 5.0 is designated for Feed Header Pressure Malfunction. Actions include placing the MFW Pump Turbine Master Speed Controller in MANUAL. This controller will remain in MANUAL for the duration of the scenario and require monitoring/adjustment. If Pressurizer Pressure falls below 2220 psig, the DNB TS 3.4.1 should be entered.

Event 3 (Key 3)

Once the plant is stabilized, the next event is a Turbine 1st Stage Pressure Transmitter (PT-505) will fail low. Crew actions are per ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st Stage Pressure, and Feed Header Pressure Instrument Malfunction, Section 4.0 is required. Section 4.0 is designated for Turbine 1st Stage Pressure Malfunction. Actions include placing Rod Control in Manual and bypassing the failed Turbine 1st Stage Pressure channel. The SRO will refer to Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation; Function 18f, Condition T.

Event 4 (Key 4)

The next event will be a trip of the running Neutron Detector Well Fan #9. This will alarm 2.1 CNTMT FN MASTER TRIP. The ALM will direct the crew to determine which fan has tripped and start the other fan as required using SOP-801A, Containment Ventilation System. The crew will place the tripped fan handswitch in Pull Out or Stop as applicable.

Event 5 (Key 5)

The next event is MFP 'A' Recirculation Valve, 1-FCV-2289, opening. ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, section 11.0 will be entered. Since earlier in this scenario the main feedwater header pressure transmitter failed MFP speed control is in manual. Manual speed control is required to restore S/G levels and stabilize the plant. The RO must ensure rods are in auto for this event. The crew will dispatch an operator to isolate the failed open recirculation valve. Once the failed valve is isolated, the BOP will adjust MFP speed again for the current plant configuration.

Event 6 (Key 6)

~~The next event is a loss of XST1 which is the alternate offsite power source for Unit 1. The ALM will have the crew enter ABN 601, Response to a 138/345 KV System Malfunction, as well as a TS entry for 3.8.1, Electrical Power Systems, AC Sources—Operating, Condition A.~~

Scenario Event Description
NRC Scenario 2

Event 7 (Key 7)

The major event is a loss of all offsite power causing a reactor trip. The crew will enter EOP-0.0A, Reactor Trip or Safety Injection. Coincident with the loss of offsite power DG 1-01 will fail to auto start and cannot be manually started due to an air start failure. This will cause a complete loss of all safeguards train 'A' power. The crew will transition to EOS-0.1A, Reactor Trip Response, to continue with recovery efforts.

Event 8 — CT-1 (Auto)

~~Due to the Loss of Offsite Power, DRPI is lost and per EOP-0.0A and EOS-0.1A, Reactor Trip Response, Attachments 1.A, an Emergency Boration will be required. The crew will then perform CT-1; Initiate Emergency Boration prior to exiting EOS-0.1A, Reactor Trip Response. This will be completed by entering ABN-107, Emergency Boration.~~

Event 9 (Auto Triggered when 1/1-8104 is placed in open per ABN-107)

After the crew has commenced the emergency boration the TDAFWP will trip on Overspeed. This combined with the loss of all Safeguards Train 'A' power as well as the inoperability of MDAFWP 1-02 will place the crew in a loss of heat sink event. The crew will enter FRH-0.1A, Response to Loss of Secondary Heat Sink, and actuate SI. The crew will then perform CT-2, Establish an RCS bleed and feed path prior to exiting FRH-0.1A, Response to a Loss of Secondary Heat Sink.

Termination Criteria

This scenario is terminated after the crew establishes a bleed and feed path per FRH-0.1A. One CCP and one SI pump running with both PRZR PORVs open.

Scenario Event Description
NRC Scenario 2

Risk Significance Determination

Risk Significance	Event	Guidance
Failure of risk important systems prior to Reactor Trip	Loss of Transformer XST1	FSAR 8.2.1.2.1 – Two independent power sources are available on an immediate basis following a DBA to ensure operation of the vital safety functions. The second offsite power source will no longer be available on loss of XST1.
Risk significant core damage sequence	<p>FSAR 15.2.6.3 Loss of Non-emergency AC power to the station auxiliaries</p> <p>Loss of Secondary Heat Sink</p>	<p>EOS-0.2A, Natural Circulation Cooldown – For Units 1 and 2, the analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.</p> <p>CPNPP Accident Sequence Quantification, R&R-PN-022 – Loss of secondary heat removal, not related to ventilation failures, accounts for about 9% of CDF.</p>
Risk significant operator actions	Initiation of Boration to Add Negative Reactivity to the Core (TSA 2.14)	STI 214.01; ABN-107, Emergency Boration; WCAP-1687 1-P, Section 6.3.5; TRM Bases 13.1.31 – Within 15 minutes, when local alignment is required to establish boration flow. Boration is initiated within the prescribed time. When local manual control credited, admin controls are utilized to ensure personnel are aware/designated to perform alignment to establish boration flow.

Scenario Event Description
NRC Scenario 2

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
<p>Initiate Emergency Boration prior to exiting EOS-0.1A, Reactor Trip Response within a maximum of 15 minutes following the loss of DRPI.</p>	<p>Shutdown Margin must be maintained. Since there are NO DRPI lights lit the bases states to borate at least 3600 gallons of 7000 ppm borated water to ensure shutdown margin is maintained. This gallon value corresponds to 2 of the most reactive rods stuck out.</p>	<p>After the loss of offsite power and the failure of the DG 1-01, DRPI will be dark and no CCP will be running. Per attachment 1.A of EOS-0.1A, ABN-107 will be performed.</p>	<p>Started CCP 1-02, started Boric Acid Transfer Pump 1-02, and opened 1/1-8104, EMER BORATE VLV.</p>	<p>Boration flow will be indicated on 1-FI-183A, EMER BORATE FLO.</p>
<p>Manually control the Main Feedwater Master Speed Controller to prevent receiving an automatic reactor trip due to low steam generator levels, or a trip of Main Feed Pumps due to low suction pressure, and subsequent manual reactor trip.</p>	<p>Result of improper operator action or inaction, i.e., such as an unintentional RPS or ESF actuation.</p>	<p>After the Main Feed Pump A recirc valve fails open, S/G levels will begin decreasing. Manual control of the feed pump speed will maintain S/G levels on program.</p>	<p>S/G levels maintained on program without tripping the reactor or tripping the Main Feed Pumps on low suction Pressure, followed by a manual reactor trip (ABN-302, immediate operator action).</p>	<p>Neither reactor nor Main Feed Pumps do not trip.</p>
<p>Establish an RCS bleed and feed path prior to exiting FRH-0.1A, Response to a Loss of Secondary Heat Sink.</p>	<p>Actuating SI will ensure a feed path of cool water to the RCS (core) and isolate the containment to confine any RCS releases from the bleed flow.</p>	<p>AFW flow will not be indicated on any AFW flow meter. Also no AFW pumps will be running. A RED path showing on CSFST for heat sink. The need for a heat sink as indicated by RCS</p>	<p>Actuated SI, ensured at least one CCP and SI pump is running with flow indicated providing a feed path for the RCS. Both PRZR PORVs</p>	<p>Flow indicated on both a CCP and an SI pump. PRZR PORVs open with block valves open. RCS pressure and temperature lowering.</p>

Scenario Event Description
NRC Scenario 2

	The bleed flow through both PORVs will ensure that enough cool water will feed from the ECCS flow path to remove sufficient decay heat.	temperature and pressure.	open providing a bleed path for the RCS.	
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Facility:	CPNPP 1 & 2	Scenario No.:	3	Op Test No.:	July 2016 NRC
Examiners:	_____	Operators:	_____		
	_____		_____		
	_____		_____		
Initial Conditions: 100% power MOL - RCS Boron is 924 ppm					
Turnover: Maintain steady-state power conditions.					
Critical Tasks: CT-1 Manually Trip Reactor due to Failure to Automatically Trip prior to exiting EOP-0.0A, Reactor Trip or Safety Injection.					
CT-2 Identify and Isolate the Ruptured Steam Generator Prior to Commencing an Operator Induced Cooldown per EOP-3.0A, Steam Generator Tube Rupture.					
Event No.	Malf. No.	Event Type*	Event Description		
1	RP05D	I (RO, SRO) TS (SRO)	Cold Leg Loop 4 NR Temperature Transmitter Failure (TE-441B) Fails High		
2	RP03A	I (BOP, SRO) TS (SRO)	Steam Generator (1-01) Steam Line Pressure Instrument (PT-514) Fails Low.		
3	Override	C (RO, SRO)	Letdown HX Outlet Flow Controller Failure (TK-130) Fails Low		
4	FW22	C (BOP, SRO) TS (SRO)	Station Service Water Pump 1-01 Trip		
5	TC08C	C (BOP, SRO)	High Pressure Turbine Stop Valve #3 (UV-2430A) Fails Closed		
6	SG02C	M (RO, BOP, SRO)	Steam Generator 1-03 Tube Rupture		
7	RP15E	C (BOP, SRO)	Automatic Reactor Trip Failure, Manual Trip from 1B3 and 1B4		
8	MS08C	C (RO, SRO)	Steam Generator 1-03 MSIV Fails to Close		
* (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 Event Description
NRC Scenario 3

SCENARIO SUMMARY NRC 3

Event 1 (Key 1)

The crew will assume the watch at 100% power with no scheduled activities per IPO-003A, Power Operations. The first event is a failure high of a Reactor Coolant System Loop 4 Narrow Range Temperature (TE=441B) element. Crew actions are per ABN-704, T₀/N-16 Instrumentation Malfunction, Section 2.0. Section 2.0 is designated for T₀/N-16 Instrumentation Malfunction. Actions include placing the Control Rods in MANUAL and defeating the failed channel. Control Rods will be restored in Manual to their pre-failure position and remain in Manual until restored to Operable per ABN-704. The SRO will refer to Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation (Functions 6 & 7); Condition E, One channel inoperable.

Event 2 (Key 2)

The next event is a failure low of Main Steam Line 1 Pressure Instrument (PT-514). Crew actions are per ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st-Stage Pressure and Feed Header Pressure Instrument Malfunction, Section 2.0. Section 2.0 is designated for Steam Line Pressure Malfunction. The crew must manually control Steam Generator level, transfer to an Alternate Steam Flow Channel, and restore Steam Generator (SG) Feedwater Flow Control to AUTO. The SRO will refer to Technical Specification LCO 3.3.2, Engineered Safety Feature System (ESFAS) Instrumentation (Functions 1.e & 4.d); Condition D, One channel inoperable.

Event 3 (Key 3)

The next event is a failure of the Letdown Heat Exchanger Outlet Flow Controller, TK-130. The controller output will fail to zero demand and cause TCV-4646, LTDN HX OUT TEMP CTRL valve to close. This will result in Letdown Heat Exchanger High temperature alarms and Letdown flow to divert to the VCT on high temperature. The crew will respond per the ALM, take manual control of TK-130 and raise demand to establish Letdown Heat Exchanger Outlet temperature to approximately 95°F.

Event 4 (Key 4)

The next event is a trip of Station Service Water Pump 1-01. Crew actions are per ABN-501, Station Service Water System Malfunction, Section 2.0. Section 2.0 is designated for Station Service Water Pump Trip. Various equipment controls, as directed by ABN-501, are placed in PULL-OUT to prevent starting with no cooling water available. The SRO will refer to Technical Specification LCO 3.7.8, Station Service Water System; Condition B, One SSWS Train inoperable. The SRO will also refer to Technical Specification LCO 3.8.1, AC Sources – Operating; Condition B, One DG inoperable as DG 1-01 must be placed in PULL-OUT upon the loss of Train A Station Service Water.

Event 5 (Key 5)

The next event is High Pressure Turbine Stop Valve #3 fails closed. The crew will enter ABN-401, Main Turbine Malfunction, Section 9.0. Section 9.0 is designated for Inadvertent Closure of an HP or LP Stop or Control Valve. Actions include placing rod control in Auto to allow the rod control system to respond to the plant transient and reducing turbine load to allow all operable HP Control Valves to come off their full open seat.

Event 6 – (Key 6)

The major event is a Tube Rupture on SG 1-03. The Crew will diagnose the Tube Rupture due to multiple Radiation alarms and lowering Pressurizer Pressure and Level. The crew will enter EOP-0.0A, Reactor Trip or Safety Injection and transition to EOP-3.0A, Steam Generator Tube Rupture. A maximum rate RCS cooldown to a target CET temperature as determined in EOP-3.0A will be conducted.

Event 7 – CT-1 (Auto)

The Reactor will be manually tripped and Safety Injection manually initiated. The Reactor will fail to trip from both handswitches at CB-07 and CB-10. The crew will then perform CT-1, Manually Trip Reactor due to Failure to Automatically Trip prior to exiting EOP-0.0A, Reactor Trip or Safety Injection. The Reactor must be manually tripped by momentarily de-energizing 480V Normal Switchgear 1B3 and 1B4 to de-energize the Rod Drive MG Sets. The critical task is considered not met if the crew is not successful in tripping the reactor during EOP-0.0A and transitions to FRS-0.1A.

Scenario Event Description
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Event 8 (Auto) CT-2

During performance of CT-2, Identify and Isolate the Ruptured Steam Generator Prior to Commencing an Operator Induced Cooldown per EOP-3.0A, Steam Generator Tube Rupture. SG 1-03 MSIV will fail to close. The crew will close all remaining MSIVs, disable the Steam Dumps, and close the Main Steam to Auxiliary Steam Supply Valve. The RCS cooldown will then be conducted via the intact SG ARVs to atmosphere.

Termination Criteria

This scenario is terminated when the target CET Temperature is reached during the RCS cooldown in accordance with EOP-3.0A, Steam Generator Tube Rupture.

Scenario Event Description
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Risk Significance Determination

Risk Significance	Event	Guidance
Failure of risk important system prior to Reactor Trip	Event 4 – Station Service Water Pump Trip	ABN-501; DBD-ME-011 – Initial operator action to place the affected DG Emergency Stop/Start handswitch in PULL-OUT to remove the DG from service as it will ONLY operate for 1 minute under load, without service water cooling flow, before damage will occur.
Risk significant operator actions	<p>Event 7 – Manually tripping the Reactor by momentarily de-energizing 1B3 and 1B4</p> <p>Event 8 – Closing all intact SG MSIVs upon failure of the ruptured SG MSIV to close</p>	<p>FSAR 15.8 – The worst common mode failure which is postulated to occur is the failure to trip the reactor after an anticipated transient has occurred.</p> <p>FSAR 15.6.3.2 – The closing of all intact SG MSIVs falls in line with the conservative analysis of the postulated SGTR which assumes a loss of offsite power. Thus, a release of steam from the secondary system occurs due to the loss of steam dump capability and the subsequent venting to the atmosphere through the ARVs.</p>
Risk significant core damage sequence	Events 5 – Steam Generator Tube Rupture	(1) STI-214.01 TCA-1.9 – Manual Actions to Mitigate Effects of a Steam Generator Tube Rupture: 1) TDAFWP flow stopped (excessive AFW flow) within 3 minutes of reactor trip. 2) Identify and Isolate ruptured SG within 13 minutes after initiation of SGTR. 3) Initiate maximum rate cooldown within 5 minutes after isolation of ruptured SG. 4) Initiate RCS depressurization with PORVs within 2 minutes after completion of RCS cooldown. 5) Secure ECCS within 2 minutes after completion of RCS depressurization.

(1) Crew manning for Initial License Examination less than Timed Operator Action validation constraints

Scenario Event Description
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Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
<p>CT-1 – Manually Trip Reactor due to Failure to Automatically Trip prior to exiting EOP-0.0A, Reactor Trip or Safety Injection</p>	<p>Recognize a failure or an incorrect automatic actuation of an ESF system or component. FSAR 7.1.2.1</p>	<p>Procedural direction at EOP-0.0A Step 1 to determine if a reactor trip has occurred. Position indication of the Reactor Trip breakers and Reactor Power, Annunciator First out alarms.</p>	<p>The operator will attempt to manually trip the Reactor with the handswitches on both CB-07 and CB-10; however, the Reactor will fail to trip. The operator will then momentarily deenergize the 480V normal switchgear 1B3 and 1B4 to secure power to the Rod Drive MG sets.</p>	<p>De-energizing the Rod Drive MG sets will result in a loss of power to the Rod Drive Mechanisms and the Control Rods will insert into the core. Reactor Trip Breakers will remain closed, neutron flux will lower and rod bottom lights will be lit.</p>
<p>CT-2 – Identify and Isolate the Ruptured Steam Generator Prior to Commencing an Operator Induced Cooldown per EOP-3.0A, Steam Generator Tube Rupture.</p>	<p>Take one or more actions that would prevent a challenge to plant safety. STI-214.01, TCA-1.9; FSAR 15.6.3.1.1; WCAP-16871-P, Section 6.4; DBD-ME-027</p>	<p>Procedurally driven from EOP-3.0A, to identify and isolate a ruptured SG. Indications include MSL Radiation alarms and SG level.</p>	<p>The operator will attempt close the ruptured SG MSIV from the control room, however, the MSIV will fail to close and all other MSIVs must be closed. The MSIV will be locally closed in the field. The operator will stop feeding the SG once sufficient level to cover the tubes is available.</p>	<p>SG pressure increasing, AFW flow reduced to zero and valve position indications.</p>