Facility: **CPNPP** Date of Exam: **July 18, 2016**

Tier	Group	RO K/A Category Points K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	SRO-Only Points A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	2	3	4	N/A			3	3	N/A		3	18			
	2	2	1	2				1	2			1	9			
	Tier Totals	4	4	6				4	5			4	27			
2. Plant Systems	1	3	2	3	3	2	2	2	3	3	3	2	28			
	2	1	0	1	1	1	1	1	1	1	1	1	10			
	Tier Totals	4	2	4	4	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 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	#
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	2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	4.2	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				EK3.05 Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: Need to avoid plant transients.	3.2	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									
000054 Loss of Main Feedwater / 4			X				AK3.05 Knowledge of the reason for the following responses as they apply to the Loss of Main Feedwater (MFW): HPI/PORV cycling upon total feedwater loss.	4.6	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)

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	#
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			X				EK3.2 Knowledge of the reasons for the following responses as they apply to the LOCA Outside Containment: Normal, abnormal and emergency operating procedures associated with LOCA Outside Containment.	3.4	54 (16)
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 Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2 (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	#
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)

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
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 Form ES-401-2

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 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)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
022 Containment Cooling			X									K3.01 Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment equipment subject to damage by high or low temperature, humidity, and pressure	2.9*	13 (41)
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 the 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)

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

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

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								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)
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								K4.03 Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Securing of SAS upon loss of cooling water	3.1*	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	2	3	3	2	2	2	3	3	3	2	Group Point Total:		28

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

Plant Systems - Tier 2 / Group 2 (RO / SRO)

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														
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					X							K5.02 Knowledge of the operational implications of the following concepts as they apply to the ITM system: Saturation and subcooling of water	3.7	31 (58)
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														

Plant Systems - Tier 2 / Group 2 (**RO** / SRO)

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	1	1	1	1	1	1	1	0	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	2.2.18	Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.	2.6	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		
	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions.	3.5	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		

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				N/A					N/A				3	3	6
	2													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	RO 1	RO 2	RO 3	RO 4	RO Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
						2	1	2	2	7

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).
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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.
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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’ 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		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					X		AA2.03 Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Increasing reactor building sump level	3.8	79
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									
000054 Loss of Main Feedwater / 4						X	2.4.11 Knowledge of abnormal condition procedures.	4.1	80
000055 Station Blackout / 6									
000056 Loss of Off-site Power / 6									
000057 Loss of Vital AC Inst. Bus / 6									
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									

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	#
K/A Category Totals:					3	3	Group Point Total:		6

ES-401	PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2 (RO / **SRO**)								Form ES-401-2
E/APE # / Name / Safety Function	**K 1**	**K 2**	**K 3**	**A 1**	**A 2**	**G***	**K/A Topic(s)**	**IR**	**#**
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/EO1 & E02 Rediagnosis & SI Termination / 3									
W/E13 Steam Generator Over-pressure / 4					X		EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.	3.4	85
W/E15 Containment Flooding / 5									
W/E16 High Containment Radiation / 9									
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 (RO / SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	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 Form ES-401-2

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

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 Form ES-401-2
Plant Systems - Tier 2 / Group 2 (RO / **SRO**)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 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														
027 Containment Iodine Removal											X	2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.0	91
028 Hydrogen Recombiner and Purge Control														
029 Containment Purge														
033 Spent Fuel Pool Cooling								X				A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SFPCS	3.0	92
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														
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 IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.2	Knowledge of operator responsibilities during all modes of plant operation.			4.4	94
	2.1.30	Ability to locate and operate components, including local controls.			4.0	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	2.4.18	Knowledge of specific bases for EOPs			4.0	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	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 which 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	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.
1/2	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 trained on performing leakage calculations. 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	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.
2/2	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.
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	025 AA2.03	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 025 AA2.03.
1/1	054 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.
3	G.2.4.18	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.

Facility: **CPNPP Units 1 and 2** Date of Examination: **July 2016**

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

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations (RA1)	D,R	2.1.25 Ability to perform specific system and integrated plant procedures. (3.9) JPM: Restore Refueling Water Storage Tank Level. (RO1307)
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 (SA4)	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)

Task Summary

RA1 The applicant will restore Refueling Water Storage Tank (RWST) level in accordance with SOP-104A, Reactor Make-Up and Chemical Control System. 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. This is a direct from bank JPM. (K/A 2.1.25 - IR 3.9)

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. 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 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 ☐ SRO ☒	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) 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)

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.

Task Summary

* 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)

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	A,M,S	1
S-2	**010 – Pressurizer Pressure Control System (RO1205)** **PORV Block Valve Operability Test**	**A,D,S**	**3**
S-3	015 – Reactor Coolant Pump Malfunctions (RO1118) Respond to a Reactor Coolant Pump Seal Failure	D,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 (RO2002E)** **Transfer Containment Spray to Recirculation with Cavitation**	**A,EN,M,L,S**	**5**
S-6	062 – A.C. Electrical (RO4201A) Shift Normal Bus 1A4 Between Unit Auxiliary Transformer and Station Service Transformer	A,D,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 (RO Only)	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

@	All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.
*Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(EN)gineered safety feature	- / - / ≥ 1(control room system)
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

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. This is an Alternate Path JPM as the applicant must determine navigate through an Response Not Obtained in order to reset the Rod Control Urgent Failure alarm. 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 be directed to respond to primary side alarms. A malfunction will be entered for a RCP 1-03 Seal #1 Failure. The applicant will respond in accordance with ALM-0051A, Alarm Procedure 1-ALB-5A and ABN-101, Reactor Coolant Pump Trip/Malfunction. The critical steps include tripping the reactor, stopping the RCP and closing the No. 1 Seal Leakoff Valve. Closure of the No. 1 Seal Leakoff Valve is considered a Timed Significant Action. STI-214.01, Control of Timed Operator Actions, TSA-2.10 requires the effects of an RCP #1 Seal Failure be mitigated within 5 minutes of stopping the RCP by isolating RCP Seal #1 leakoff. This Time Significant Action is performed to reduce possible gross damage to the RCP seal/pump when warning of excessive leakage has occurred. This is a direct from bank JPM under Reactor Coolant Pump Malfunctions – Heat Removal from Reactor Core Primary Systems Safety Function.
(K/A 015.AA1.22 - IR 4.0 / 4.2)

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. The applicant will then be instructed to continue on with performance of the procedure and cavitation of the Train A pumps will occur as a result of sump blockage, which constitutes the modification to the bank JPM and represents an additional Alternate Path. Critical steps will include transferring Train A suction to the containment sump, securing Train B when suction cannot be realigned and stopping both Train A Containment Spray Pumps when sump blockage is encountered. 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 modified 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 be required to transfer Electrical Bus 1A4 from the Unit Auxiliary Transformer to the Station Service Transformer in accordance with SOP-603A, 6900 V Switchgear. This is an Alternate Path JPM as the incoming feeder breaker from the Unit Auxiliary Transformer does not automatically open, requiring the applicant to open the breaker. Critical steps include synchroscope operation, closing the incoming breaker from the Station Service Transformer and opening the incoming breaker from the Unit Auxiliary Transformer. This is a direct from bank JPM under the A.C. Electrical – Electrical Safety Function. (K/A 062.A4.07 - IR 3.1 / 3.1)

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 JPM will be performed by only the Reactor Operator applicants. 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.

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-905A, 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.: 1	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:

- Initiate Train A and/or Train B Containment Isolation Phase B due to Failure to Automatically Actuate Prior to Exiting FRZ-0.1A, Response to High Containment Pressure.
- Initiate Train B Chemical Addition Tank flow Prior to Exiting FRZ-0.1A, Response to High Containment Pressure.

Event No.	Malf. No.	Event Type*	Event Description
1	CC02A CC03A	C (BOP, SRO) TS (SRO)	Train A Component Cooling Water Pump 1-01 Trip Train B Component Cooling Water Pump 1-02 Auto Start Failure
2	RX15B	C (RO, SRO) TS (SRO)	Pressurizer Spray Valve PCV-455C Fails Open
3	ED07A	C (RO, BOP, SRO) TS (SRO)	Loss of Inverter (IV1PC1)
4	RC17B	M (RO, BOP, SRO)	Reactor Coolant Leak Inside Containment on Loop 2 Hot Leg of 600 GPM on 600 second ramp
5	CV01E	C (BOP)	Centrifugal Charging Pump (1-02) Auto Start Failure on Safety Injection Signal
6	RC08B2	M (RO, BOP, SRO)	Large Break Loss of Coolant Accident Inside Containment on Loop 2 Hot Leg
7	RP10A RP10B	C (BOP, SRO)	Train A and B Containment Isolation Phase B Automatic Actuation Failure
8	HS-4755 Override	C (BOP, SRO)	Train B Chemical Addition Tank Discharge Valve Fail to Auto Open

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

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

Event 1
The crew will assume the watch at 100% power with no scheduled activities per IPO-003A, Power Operations. The first event is a trip of the Train A Component Cooling Water (CCW) Pump and auto start failure of the Train B CCW Pump. The crew will enter ABN-502, Component Cooling Water System Malfunction, Section 2.0, and transfer CCW flow to Train B. The SRO will refer to Technical Specification LCO 3.7.7, Component Cooling Water System; Condition A, One CCW train inoperable. The CCW train must be restored to OPERABLE status within 72 hours.

Event 2
Once Technical Specifications have been referenced, the Loop 4 Pressurizer Spray Valve, 1-PCV-455C, will fail open due to a failure of controller 1-PK-455C to control in automatic. The valve will drift open to 50% over a 2 minute period. The crew will enter ABN-705, Section 3.0, Pressurizer Spray Valve Failure, and take manual control of 1-PCV-455C controller, 1-PK-455C, and close the valve. The SRO will reference Technical Specification LCO 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits; Condition A, One or more RCS DNB parameters not within limits. The crew must restore RCS DNB parameter(s) to within limits in 2 hours.

Event 3
The next event is a loss of Inverter IV1PC1. Actions are per ABN-603, Loss of Protection or Instrument Bus, Section 2.0, and include placing Rod Control in MANUAL, controlling Steam Generator (SG) level and RCS pressure and adjusting Charging flow due to a loss of Letdown. The SRO will refer to Technical Specification LCO 3.8.7, Inverters – Operating; Condition A, One required inverter inoperable. The inverter must be restored to OPERABLE status within 24 hours. Also, while power to the inverter is lost, the SRO will refer to Technical Specification LCO 3.8.9, Distribution Systems – Operating; Condition B, One AC vital bus subsystem inoperable. The AC vital bus subsystem must be restored to OPERABLE status within 2 hours. Technical Specification LCO 3.8.9.B will be exited when power is restored to the AC vital bus subsystem.

Event 4
When the Alternate Power Supply is aligned to IV1PC1 in ABN-603, a Reactor Coolant Leak inside Containment will commence. Once it is determined that Pressurizer level cannot be maintained, the Reactor must be manually tripped and Safety Injection manually initiated. The crew will enter EOP-0.0A, Reactor Trip or Safety Injection, and then transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

Event 5
The scenario is complicated by Train B Centrifugal Charging Pump failing to auto start upon actuation of the Safety Injection Sequencer. This will require manual action by the crew to start the Train B Centrifugal Charging Pump.

Events 6, 7 & 8
When EOP-1.0A is entered, a Large Break Loss of Coolant Accident will occur. At that point, the Unit Supervisor should recognize that entry into FRZ-0.1A, Response to High Containment Pressure, is required due to a Critical Safety Function Status Tree ORANGE path inside Containment. During performance of FRZ-0.1A, the crew will identify that Containment Isolation Phase B has failed to actuate on both trains, requiring manual actuation. Further, the Train B Chemical Addition Tank Discharge Valve has failed to automatically open requiring manual action. Opening this valve is critical as the Train A Containment Spray will need to be stopped during recirculation as a result of the earlier failure of the Train A Component Cooling Water Pump. When the actions of FRZ-0.1A are completed a transition to FRP-0.1A, Response to Imminent Pressurized Thermal Shock Condition, is required. FRP-0.1A will be exited at Step 1 RNO when it is determined that Reactor Coolant System pressure is less than 425 psig and Residual Heat Removal System flow is greater than 750 GPM. The crew will return to EOP-1.0A, Loss of Reactor or Secondary Coolant, as this is the current procedure and step in effect.

Termination Criteria
This scenario is terminated when the conditions are reached for a Transfer to Cold Leg Recirculation in accordance with EOS-1.3A.

Risk Significance Determination

Risk Significance	Event	Guidance
Failure of risk important system prior to Reactor Trip	Event 1 - Loss of Component Cooling Water Event 3 - Loss of Protection System Inverter	STI-214.01 TCA-1.6 – Stop RCPs within 10 minutes(1) of a loss of CCW Non-Safeguards Loop flow to prevent RCP Bearing Damage/Failure. (FSAR 7.1.2.5, 7.3.2.3, 9.2.2.3, 15.2.8.2, II.E.1.1) FSAR Table 7.6-2 – Bounding event is excessive feedwater flow. Hi-Hi SG level closes FW control and isolation valves, trips FW turbine and main turbine.
Risk significant core damage sequence	Events 4 and 6 - Reactor Coolant Leak then LBLOCA	STI-214.01 TCA 2.2 – Complete EOP-0.0A to the point of transition to EOP-1.0A within 11.9 minutes(1) prior to Low-Low RWST level to verify immediate and automatic actions for an SI event. (FSAR 6.3.2.8, II.B.2.2.2, DBD-ME-232, DBD-ME-261, WOG E-0 Background)
Risk significant operator actions	Restore Pressurizer Pressure Control Start Train B Centrifugal Charging Pump Actuate Phase B Containment Isolation Initiate Containment Spray Additive Flow	ODA-102 1) The operator may place a controller in the manual mode of operation when continued automatic control is inappropriate. 2) The Unit RO and/or US shall place the Reactor in a safe condition when the safety of the Reactor is in jeopardy or when operating parameters exceed any Reactor Protection System or Safeguards System setpoint without automatic protection functions occurring. FSAR 6.2.4.1.1 – Containment isolation is mandatory in the event of a LOCA. The CIS isolates the Containment to prevent or limit the escape of fission products that may result from postulated accidents. Technical Specification 3.6.7 Basis - The Spray Additive System is essential to the removal of airborne iodine within containment following a DBA.

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

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
Initiate Train A and/or Train B Containment Isolation Phase B due to Failure to Automatically Actuate Prior to Exiting FRZ-0.1A, Response to High Containment Pressure.	Recognize a failure or an incorrect automatic actuation of an ESF system or component to preclude degradation of any barrier to fission product release. FSAR 6.2.4.1.1	Procedural direction at FRZ-0.1A Step 4 to determine if Containment Spray is required. 1-ALB-2A, Window 1.8, CS ACT is LIT, Containment pressure is >18 psig and 1-ALB-2A, Window 4.11 is DARK indicating Phase B has NOT actuated.	The operator will manually actuate CS/Phase B using BOTH handswitches (simultaneously) at CB-07 or CB-02.	1-ALB-2A, Window 4.11 will light and Phase B Orange lights will be LIT on MLB for Phase B.
Initiate Train B Chemical Addition Tank flow Prior to Exiting FRZ-0.1A, Response to High Containment Pressure.	Recognize a failure or an incorrect automatic actuation of an ESF system or component to facilitate the removal of airborne iodine within containment following a DBA.	Procedural direction at FRZ-0.1A Step 4.f. to verify proper Containment Spray system valve alignment per Attachment 4, Containment Spray Alignment Injection Phase. Attachment 4 Step 1 states to verify Blue lights LIT on 1-MLB-4B3.	The operator will manually open 1-HS-4755 on CB-02.	1-HS-4755 valve position indication on CB-02 will indicate valve OPEN with RED light on. Also, operator should verify Blue Light LIT on at 1-MLB-4B3, Window 1.9.

Facility:	CPNPP 1 & 2	Scenario No.:	2	Op Test No.:	July 2016 NRC
Examiners:		Operators:			

Initial Conditions: 100% power MOL - RCS Boron is 924 ppm

Turnover: Maintain steady-state power conditions. Positive Displacement Charging Pump and 75 GPM Letdown Orifice in service per Radiation Protection request for maintenance in vicinity of letdown line.

Critical Tasks:

- Trip Reactor Coolant Pumps within 5 minutes upon a Loss of Subcooling per EOP-0.0A, Reactor Trip or Safety Injection, Foldout Page.
- Identify and Isolate the Ruptured Steam Generator prior to commencing ECCS flow reduction.
- Initiate Cooldown of Reactor Coolant System prior to commencing ECCS flow reduction.

Event No.	Malf. No.	Event Type*	Event Description
1		N (RO)	Start Centrifugal Charging Pump 1-01 and Secure Positive Displacement Charging Pump
2	RX18	I (BOP, SRO)	Feed Header Pressure Transmitter (PT-508) Fails High.
3	RX09A	I (RO, SRO) TS (SRO)	Main Turbine 1st Stage Pressure (PT-505A) Fails Low
4	SW01A	C (BOP, SRO) TS (SRO)	Station Service Water Pump 1-01 Trip
5	SG01A	M (RO, BOP, SRO)	Steam Generator (1-01) Tube Rupture at 400 GPM (300 second ramp).
6	RX16A	C (RO, SRO)	Power Operated Relief Valve (PCV-455A) Fails Open Upon Reactor Trip.
7	MS10A1 MS10A2	NOTE 1	Main Steam Safety Valves (MS-021 & MS-022) on Steam Generator (1-01) Fail Open Upon Reactor Trip.
8	RH01C	C (BOP)	Residual Heat Removal Pump (1-01) Safety Injection Sequencer Start Failure.

* (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)
3	Abnormal events (2-4)
1	Major transients (1-2)
3	EOPs entered/requiring substantive actions (1-2)
1	EOP contingencies requiring substantive actions (0-2)
3	Critical tasks (2-3)

NOTE 1: Event 7 is part of the Major (Faulted/Ruptured SG), but is listed separately for clarity.

SCENARIO SUMMARY NRC 2

Event 1
The crew will assume the watch at 100% power in IPO-003A, Power Operations. The Positive Displacement Charging Pump (PDP) is in service and Letdown flow is 75 GPM per Radiation Protection while maintenance is performed. The scenario begins by placing Centrifugal Charging Pump 1-01 in service and securing the Positive Displacement Charging Pump (PDP) per SOP-103A, Chemical and Volume Control System.

Event 2
When the PDP is secured, a Main Feedwater (MFW) Header Pressure Transmitter 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 during the subsequent down power.

Event 3
When plant parameters are restored to normal, a Turbine 1st Stage Pressure Transmitter 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; Condition T, One or more required channels inoperable. The crew must verify the P-13, TURB ≤ 10% PWR, interlock is in its required state for existing unit conditions within one (1) hour, OR be in MODE 2 within 7 hours.

Event 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 and will be to perform an Initial Operator Action to place Emergency Diesel Generator 1-01 in Pull-out. Other actions will include starting Centrifugal Charging Pump 1-02 and securing Train A equipment cooled by Station Service Water. The SRO will refer to Technical Specification LCO 3.7.8, Station Service Water System; Condition B, One SSWS train inoperable. The SSWS train must be restored to OPERABLE status within 72 hours. The SRO will also refer to Technical Specification LCO 3.8.1, AC Sources – Operating; Condition B, One DG inoperable. SR 3.8.1.1 for the required offsite circuits must be performed within 1 hour AND once per 8 hours thereafter.

Event 5
The next event is the major event as a 400 GPM Steam Generator Tube Rupture will commence on a 300 second ramp. When control of the plant is no longer feasible, the crew will trip the Reactor, initiate a Safety Injection, and enter EOP-0.0A, Reactor Trip or Safety Injection.

Events 6 & 7
The scenario is complicated by a Power Operated Relief Valve (PORV) and Two Main Steam Safety Valves failing open upon Reactor Trip. The PORV should be isolated once identified at Step 10 of EOP-0.0A if not already recognized. When SG 1-01 is identified as faulted, the crew will transition from EOP-0.0A to EOP-2.0A, Faulted Steam Generator Isolation. When SG 1-01 is isolated in EOP-2.0A, a transition to EOP-3.0A, Steam Generator Tube Rupture, will be made. When it is determined that Ruptured Steam Generator 1-01 pressure is less than 420 psig in EOP-3.0A, a transition to ECA-3.1A, SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired, will be made.

Event 8
The scenario includes Train A Residual Heat Removal Pump that fails to start upon initiation of the Safety Injection Sequencer.

Termination Criteria
This scenario is terminated when a cooldown is commenced in ECA-3.1A.

Risk Significance Determination

Risk Significance	Event	Guidance
Risk significant core damage sequence	Event 5 - Steam Generator Tube Rupture	STI-214.01 TCA-1.9 – Manual Actions to Mitigate Effects of a SGTR. 1) TDAFW flow stopped within 3 mins(1) of Rx Trip (excessive flow). 2) Isolate ruptured SG within 13 mins after SGTR initiation. 3) Initiate Max Rate C/D within 5 mins after isolation of ruptured SG. 4) Initiate RCS Depress w/ PORVs within 2 mins after completion of RCS C/D. 5) Secure ECCS within 2 mins after completion of RCS C/D. These time critical actions are completed to prevent SG overfill during a SGTR event and minimize contamination of secondary systems. (FSAR 15.6.3.1.1; WCAP-16871- P, Section 6.4; DBD-ME-027)
	Event 6 – Failed Open PORV on Rx Trip	STI-214.01 TSA-2.11 – Initiate RCP Trip When SBLOCA Criteria Met within 5 mins from time SBLOCA RCP Trip Criteria is satisfied; to prevent peak cladding temperature from exceeding 2200°F. (FSAR II.K.3.5; WCAP-9584; WOG ERG Generic Issue for RCP Trip/Restart)
Risk significant operator actions	Restore Steam Generator Level Control	ODA-102 1) The operator may place a controller in the manual mode of operation when continued automatic control is inappropriate.
	Start RHR Pump	2) The Unit RO and/or US shall place the Reactor in a safe condition when the safety of the Reactor is in jeopardy or when operating parameters exceed any Reactor Protection System or Safeguards System setpoint without automatic protection functions occurring.
	Isolate Failed Open Pressurizer PORV	FSAR 15.6 – Inadvertent opening of a pressurizer relief valve is analyzed in the "Decrease in Reactor Coolant Inventory" section of FSAR Chapter 15.
	Identify and Isolate Faulted/Ruptured SG Initiate RCS Cooldown	Identification and Isolation of Faulted/Ruptured SG and Initiation of RCS C/D are discussed in the "Risk significant core damage sequence" above.

(1) Time validation does not consider the existence of other failures such as the Faulted SG concurrent with the SGTR.

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
Trip Reactor Coolant Pumps 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 to plant safety to preclude a degradation of any barrier to fission product release. FSAR II.K.3.5; WCAP-9584; WOG ERG Generic Issue for RCP Trip / Restart	Procedurally driven from EOP-0.0A and EOP-1.0A Attachment 1A. Availability of Subcooling indication both on meters and plant computer.	The operator will secure ALL RCPs using the handswitches on CB-05.	Indication of pump stop including light indication, flow and motor current.
Identify and Isolate the Ruptured Steam Generator prior to commencing ECCS flow reduction.	Take one or more actions that would prevent a challenge to plant safety to preclude a degradation of any barrier to fission product release. STI-214.01 TCA-1.9; FSAR 15.6.3.1.1; WCAP-16871-P Section 6.4; DBD-ME-027	Procedurally driven from EOP-2.0A and EOP-3.0A to isolate the faulted and ruptured SG to prevent further RCS cooldown and to minimize contamination of secondary systems.	The operator will close the AFW flow control valve to SG 1-01 and TDAFW Pump steam admission valve.	Valve positions will change and AFW flow to SG 1-01 will reduce to zero.
Initiate Cooldown of Reactor Coolant System prior to commencing ECCS flow reduction.	Take one or more actions that would prevent a challenge to plant safety to preclude degradation of any barrier to fission product releases. STI-214.01 TCA-1.9; FSAR 15.6.3.1.1; WCAP-16871-P Section 6.4; DBD-ME-027	Procedurally driven from ECA-3.1A to commence cooldown to reduce the overall temperature of the RCS.	The operator will increase dumping steam from the SGs via the Steam Generator ARVs to reduce RCS temperature.	Lowering SG pressures and lowering RCS temperatures beginning with the cold leg temperatures.

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:
- Manually Initiate Safety Injection Prior to exiting EOP-0.0A, Reactor Trip or Safety Injection.
- Identify and Isolate the Faulted Steam Generator prior to commencing ECCS flow reduction.

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	CV31A	C (RO, SRO) TS (SRO)	Centrifugal Charging Pump (1-01) Sheared Shaft.
4	FW22	R (BOP) N (RO, SRO)	Low Pressure Feedwater Heater Bypass Valve (PV-2286) Fails Open.
5	FW25A	M (RO, BOP, SRO)	Feedwater Line Leak to Steam Generator (1-01) Outside Containment After Feedwater Isolation Valve (600 second ramp).
6	RP07A RP07B	C (RO)	Train A Safety Injection Fails to Automatically Actuate. Train B Safety Injection Fails to Automatically Actuate
7	CS02F CS02H	C (BOP)	Train B Containment Spray Pumps 1-02 & 1-04 Safety Injection Sequencer Start Failure.
8	FW38A	C (BOP)	Steam Generator 1-01 Feedwater Isolation Valve (HS-2134) Actuation Failure.

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

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

SCENARIO SUMMARY NRC 3

Event 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 element. The crew enters ABN-704, T_c/N-16 Instrumentation Malfunction Section 2.0, places the Control Rods in MANUAL and defeats the failed channel. Control Rods will be restored in Manual to their pre-failure position. The SRO will refer to Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation (Functions 6 & 7); Condition E, One channel inoperable. The channel must be placed in trip within 72 hours.

Event 2
The next event is a failure low of Main Steam Line 1 Pressure Instrument, PT-514. Operator 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. The channel must be placed in trip within 72 hours.

Event 3
The next event is a sheared shaft of the running Centrifugal Charging Pump (CCP). When low flow alarms are received, the Initial Operator Actions of ABN-105, Chemical and Volume Control System Malfunction, Section 3.0, will be performed (start the standby CCP). The SRO will refer to Technical Specification LCO 3.5.2, ECCS – Operating; Condition A, One train inoperable because of the inoperability of a Centrifugal Charging Pump. The pump must be restored to OPERABLE status within 7 days. Technical Requirement Manual 13.1.31, Boration Injection System - Operating; Condition B, One charging pump inoperable also with a 7 day Completion Time will be entered as well.

Event 4
The next event is the Low Pressure Heater Bypass Valve (1-PV-2286) fails open. Entry into ABN-302, Feedwater, Condensate, Heater Drain System Malfunction Section 7.0, is required. Rod Control will be returned to AUTO and a Manual Turbine Runback to 900 MWe performed. Plant stabilization will be the end point of this event as restoration is a prolonged process when done properly.

Event 5
When plant conditions are stable, a Feed Line Break will commence on a 600 second ramp outside Containment, downstream of Steam Generator (SG) 1-01 Feed Line Isolation Valve HS-2134. The crew will manually initiate a Reactor Trip and Safety Injection. EOP-0.0A, Reactor Trip or Safety Injection, will be entered and actions implemented until it is determined that SG 1-01 pressure is lower than the other Steam Generators. A transition to EOP-2.0A, Faulted Steam Generator Isolation, will be required.

Events 6, 7 & 8
The scenario includes a failure of Safety Injection to automatically actuate and a failure of HS-2134, Main Feedwater Line Isolation Valve, to close in automatic when the P-4 interlock is satisfied. The Train B Containment Spray Pumps must be manually started due to a Safety Injection Sequencer failure.

Termination Criteria
This scenario is terminated when the Faulted Steam Generator is identified and isolated per EOP-2.0A and a transition to EOS-1.1A, Safety Injection Termination, is required.

Risk Significance Determination

Risk Significance	Event	Guidance
Failure of risk important system prior to Reactor Trip	Event 3 - Centrifugal Charging Pump Sheared Shaft	FSAR 9.3.4.1.2.1 – Standby charging pump must be manually started to maintain Pressurizer Water level and RCP seal injection flow.
Risk significant core damage sequence	Events 5 - Feed Line Break Outside Containment	STI-214.01 TCA-1.1 – Isolate AFW flow to Faulted SG following a Feed Line Break within 30 minutes after event initiation Reactor trip on SG low-low level. This will limit/terminate the excessive cooldown associated with faulted SG and terminate break flow from faulted SG. Also ensures 10CFR100 dose limits and containment design pressure not exceeded.
Risk significant operator actions	Manually Initiate Turbine Runback	FSAR 10.4.7.2 – Under transient conditions, such as a 40% steam dump, fluid in the HDTs may flash, causing low water level and tripping the HDPs resulting in their loss of feedwater flow to the MFPs. This in turn would result in a loss of MFW flow and then a turbine trip. To avoid a turbine trip the low pressure feedwater heater bypass valve is sized to 96% of full flow capacity to supply the MFPs during a load drop transient. However, when the valve opens all Low Pressure Feedwater Heaters are bypassed which necessitates a manual turbine runback to ≤ 900 MW due to the power increase that will occur.
	Manually Initiate a Train of Safety Injection	FSAR 6.3.2 – ECCS components are designed such that a minimum of three Accumulators, one Charging Pump, one Safety Injection Pump, and one RHR Pump together with their associated valves and piping will assure adequate core cooling, therefore, at least one train of ECCS must be initiated.
	Start Train B Containment Spray Pumps	FSAR 6.2.2 – The CSS is designed to remove heat from Containment following an accident. There are two redundant CS Trains; this system along with ECCS removes post-accident thermal energy from containment.
	Isolate Faulted Steam Generator	STI-214.01 TCA-1.1 – Isolate AFW flow to Faulted SG following a Feed Line Break within 30 minutes after event initiation Reactor trip on SG low-low level.

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
Manually Initiate Safety Injection Prior to exiting EOP-0.0A, Reactor Trip or Safety Injection.	Recognize a failure or an incorrect automatic actuation of an ESF system or component to preclude degraded ECCS capacity. FSAR 6.3.2	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.
Identify and Isolate the Faulted Steam Generator prior to commencing ECCS flow reduction.	Take one or more actions that would prevent a challenge to plant safety to preclude degradation of any barrier to fission product release. STI-214.01 TCA-1.1; FSAR 15.2.8 & II.E.1.1; DBD-ME-206.	Procedurally driven from EOP-2.0A to limit/terminate excessive cooldown associated with faulted SG and terminate break flow from faulted SG.	The operator will close the AFW flow control valve to SG 1-01 and TDAFW Pump steam admission valve.	Valve position will change and AFW flow to SG 1-01 will reduce to zero.

Facility:	CPNPP 1 & 2	Scenario No.:	4	Op Test No.:	July 2016 NRC
Examiners:		Operators:			

Initial Conditions: ~3% power BOL - RCS Boron is 1669 ppm by Chemistry sample. Steam Dump System in service for Reactor Coolant System Temperature Control.

Turnover: Recirculate the Refueling Water Storage Tank prior to MODE 1 entry then raise Reactor Power from 3% to 8% in preparation for Turbine Startup.

Critical Tasks:

- Manually Trip Reactor Due to Reactor Protection System Failure Prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.
- Trip Reactor Coolant Pumps within 5 minutes upon a Loss of Subcooling per EOP-1.0A, Loss of Reactor or Secondary Coolant, Foldout Page.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP)	Recirculate the Refueling Water Storage Tank with Containment Spray Pump 1-01
2	RX12	C (RO)	Main Steam Header Pressure (PT-507) Fails High on 360 second ramp
3	TP06A TP07B	C (BOP)	Turbine Plant Cooling Water Pump Trip with Failure of Standby Pump to Start
4		R (RO) N (BOP, SRO)	Raise Power to 6% to 8% in Preparation for Synchronizing Main Generator to Electrical Grid
5	RX04A	I (BOP, SRO) TS (SRO)	Steam Generator (1-01) Level Channel (LT-551) Fails Low
6	CS02A	TS (SRO)	Containment Spray Pump (1-01) Trip
7	RP14B	M (RO, BOP, SRO)	Spurious Train B Safety Injection Actuation Signal
8	RP01	C (RO)	Automatic Reactor Trip Failure Reactor Protection System Failure requires Manual Reactor Trip
9	RP09B	C (BOP)	Train B Containment Isolation Phase A Automatic Actuation Failure
10	RC08C1	M (RO, BOP, SRO)	Small Break Loss of Coolant Accident Inside Containment When Safety Injection Pump 1-02 is Secured in EOS-1.1A

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

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

SCENARIO SUMMARY NRC 4

Event 1
The crew will assume the watch with power at approximately 3% per IPO-003A, Power Operations, with direction to raise Reactor Power from 3% to 8% in preparation for Turbine startup. Prior to raising power, the crew will begin recirculating the Refueling Water Storage Tank (RWST) using Containment Spray Pump 1-01 per SOP-204A, Containment Spray System, Section 5.1.3, Recirculation through the Recirculation Header.

Event 2
The next event is a failure high of Main Steam Header Pressure Transmitter (PT-507) on a 360 second ramp. The crew enters ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st-Stage Pressure and Feed Header Pressure Instrument Malfunction, Section 3.0. Section 3.0 is designated for Main Steam header pressure Malfunction. Actions include placing the Steam Dump System in MANUAL to regain control of Reactor Coolant System (RCS) temperature. The controller will remain in MANUAL for the duration of the scenario.

Event 3
When RCS temperature control is restored, a trip of the running TPCW Pump will occur. The standby pump will fail to start and manual operator action will be required to start the standby pump. Crew response will be per ABN-306, Turbine Plant Cooling Water System Malfunction.

Event 4
The crew will continue with IPO-003A, Section 5.1, Warmup and Synchronization of the Turbine Generator, Step 5.1.16, and perform a power ascension using the Rod Control and Steam Dump Systems.

Event 5
The next event is a Steam Generator Level Transmitter failure. Actions are per ABN-710, Steam Generator Level Instrumentation Malfunction. The BOP will be required to take manual control of the Feedwater Bypass Control Valve and select an alternate controlling channel to return the Feedwater System to automatic control. The SRO will refer to Technical Specification LCO 3.3.1, Reactor Trip System Instrumentation (Function 14); Condition E, One channel inoperable. The channel must be placed in trip within 72 hours. The SRO will also refer to Technical Specification LCO 3.3.2, ESFAS Instrumentation (Functions 6.c. and 5.b.); Conditions D and I, One channel inoperable (for both conditions). For both Conditions, the channel must be placed in trip within 72 hours.

Event 6
When conditions are stable, Containment Spray Pump 1-01 will trip. Actions are per ALM-0022A, 1-ALB-2B, Window 1.3 – ANY CSP OVRLD TRIP. The SRO will refer to Technical Specification LCO 3.6.6, Containment Spray System; Condition A, One Containment Spray Train inoperable. The Containment Spray Train must be restored to OPERABLE status within 72 hours.

Events 7, 8 & 9
When Technical Specifications have been referenced, a spurious Train B Safety Injection Signal will actuate. The crew will determine that the Reactor did not automatically trip and initiate a Reactor Trip and enter EOP-0.0A, Reactor Trip or Safety Injection. This scenario is complicated by Train B Containment Isolation Phase A Automatic actuation failure.

Event 10
The crew will exit EOP-0.0A at Step 15 and enter EOS-1.1A, Safety Injection Termination. While in EOS-1.1A, Safeguards Signals are reset, the Charging flow path is realigned, and Safety Injection (SI) Pumps are stopped. When SI Pump 1-02 is stopped, a Small Break Loss of Coolant Accident will initiate. At this point, the crew will follow guidance of EOS-1.1A, Attachment 1A, that requires a transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

Termination Criteria
The scenario is terminated when it is determined in EOP-1.0A that Pressurizer pressure continues to slowly lower, and a transition to EOS-1.2 A, Post LOCA Cooldown and Depressurization, is required.

Risk Significance Determination

Risk Significance	Event	Guidance
Failure of risk important system prior to trip	Event 5 - Containment Spray Pump Trip	FSAR 6.2.1.4.8 – The failure of one of the two redundant Containment Spray trains following certain accident conditions may result in reduced ability to remove energy from the Containment atmosphere.
Risk significant core damage sequence	Event 7 - Automatic Reactor Trip Failure	FSAR 7.2.1.1.1 – The Reactor Trip System automatically initiates a reactor trip whenever necessary to prevent fuel damage for an anticipated operational transient (Condition II), to limit core damage for infrequent faults (Condition III), and so that the energy generated in the core is compatible with the design provisions to protect the RCS pressure boundary for limiting fault conditions (Condition IV).
	Event 9 - Small Break Loss of Coolant Accident	STI-214.01 TSA-2.11 – Initiate RCP Trip When SBLOCA Criteria Met within 5 mins from time SBLOCA RCP Trip Criteria is satisfied; to prevent peak cladding temperature from exceeding 2200°F. (FSAR II.K.3.5; WCAP-9584; WOG ERG Generic Issue for RCP Trip/Restart)
Risk significant operator actions	Manually Trip Reactor	FSAR 7.2.1.1 – Whenever a direct process or calculated variable exceeds a setpoint the reactor will be shut down in order to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into containment.
	Terminate ECCS Following a Spurious Safety Injection	STI-214.01 TSA-1.7 – ECCS injection terminated within 14 minutes(1) from event initiation.
	Initiate Train B Containment Isolation	FSAR 6.2.4.1.1 – Containment isolation is mandatory in the event of a LOCA. The CIS isolates the Containment to prevent or limit the escape of fission products that may result from postulated accidents.

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

Critical Task Determination

Critical Task	Safety Significance	Cueing	Measurable Performance Indicators	Performance Feedback
Manually Trip Reactor Due to Reactor Protection System Failure Prior to Exiting EOP-0.0A, Reactor Trip or Safety Injection.	Recognize a failure or an incorrect automatic actuation of an ESF system or component to preclude degradation of any barrier to fission product release. FSAR 7.2.1.1.1	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.	The operator will manually trip the Reactor with the handswitch on CB-07 or CB-10 placed to trip.	Reactor Trip Breakers open, flux lowering, rod bottom lights lit.
Trip Reactor Coolant Pumps within 5 minutes upon a Loss of Subcooling per EOP-1.0A, Loss of Reactor or Secondary Coolant.	Take one or more actions that would prevent a challenge to plant safety to preclude a degradation of any barrier to fission product release. FSAR II.K.3.5; WCAP-9584; WOG ERG Generic Issue for RCP Trip / Restart	Procedurally driven from EOP-1.0A. 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.

Facility: CPNPP 1 and 2 Date of Exam: 07/11/16 Operating Test No.: July NRC

APPLICANT	EVENT TYPE	SCENARIOS												TOTAL	MINIMUM(*)		
		CPNPP #1			CPNPP #3												
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
SRO-U1	RX	-					4							**1**	**1**	**1**	**0**
	NOR	-					-							**0**	**1**	**1**	**1**
	I/C	1,2,3, 7,8					2,7,8							**8**	**4**	**4**	**2**
	MAJ	4,6					5							**3**	**2**	**2**	**1**
	TS	1,2,3					-							**3**	**0**	**2**	**2**
SRO-U2	RX	-			-									**0**	**1**	**1**	**0**
	NOR	-			4									**1**	**1**	**1**	**1**
	I/C	1,2,3, 7,8			1,2,3									**8**	**4**	**4**	**2**
	MAJ	4,6			5									**3**	**2**	**2**	**1**
	TS	1,2,3			1,2,3									**6**	**0**	**2**	**2**
SRO-U3	RX	-			-									**0**	**1**	**1**	**0**
	NOR	-			4									**1**	**1**	**1**	**1**
	I/C	1,2,3, 7,8			1,2,3									**8**	**4**	**4**	**2**
	MAJ	4,6			5									**3**	**2**	**2**	**1**
	TS	1,2,3			1,2,3									**6**	**0**	**2**	**2**
SRO-I1	RX		-		-									**0**	**1**	**1**	**0**
	NOR		-		4									**1**	**1**	**1**	**1**
	I/C		2,3		1,2,3									**5**	**4**	**4**	**2**
	MAJ		4,6		5									**3**	**2**	**2**	**1**
	TS		-		1,2,3									**3**	**0**	**2**	**2**

Facility: CPNPP 1 and 2		Date of Exam: 07/11/16										Operating Test No.: July NRC					
APPLICANT	EVENT TYPE	SCENARIOS															
		CPNPP #1			CPNPP #3										TOTAL	MINIMUM(*)		
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
RO1	RX			-		-								**0**	**1**	**1**	**0**
	NOR			-		4								**1**	**1**	**1**	**1**
	I/C			1,3,5,7,8		1,3,6								**8**	**4**	**4**	**2**
	MAJ			4,6		5								**3**	**2**	**2**	**1**
	TS			-		-								**0**	**0**	**2**	**2**
RO2	RX		-				4							**1**	**1**	**1**	**0**
	NOR		-				-							**0**	**1**	**1**	**1**
	I/C		2,3				2,7,8							**5**	**4**	**4**	**2**
	MAJ		4,6				5							**3**	**2**	**2**	**1**
	TS		-				-							**0**	**0**	**2**	**2**
RO3	RX			-		-								**0**	**1**	**1**	**0**
	NOR			-		4								**1**	**1**	**1**	**1**
	I/C			1,3,5,7,8		1,3,6								**8**	**4**	**4**	**2**
	MAJ			4,6		5								**3**	**2**	**2**	**1**
	TS			-		-								**0**	**0**	**2**	**2**
RO4	RX		-				4							**1**	**1**	**1**	**0**
	NOR		-				-							**0**	**1**	**1**	**1**
	I/C		2,3				2,7,8							**5**	**4**	**4**	**2**
	MAJ		4,6				5							**3**	**2**	**2**	**1**
	TS		-				-							**0**	**0**	**2**	**2**
RO5	RX			-		-								**0**	**1**	**1**	**0**
	NOR			-		4								**1**	**1**	**1**	**1**
	I/C			1,3,5,7,8		1,3,6								**8**	**4**	**4**	**2**
	MAJ			4,6		5								**3**	**2**	**2**	**1**
	TS			-		-								**0**	**0**	**2**	**2**

Instructions:	
1.	Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the "at-the-controls (ATC)" and "balance-of-plant (BOP)" positions; Instant SROs must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an Instant SRO *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.
2.	Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional instrument or component malfunctions on a 1-for-1 basis.
3.	Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirements specified for the applicant's license level in the right-hand columns.