<table>
<tr><td colspan="2">Facility: **CPNPP 1 & 2**</td><td colspan="17">Date of Exam: **03/29/10**</td></tr>
<tr><td rowspan="2">Tier</td><td rowspan="2">Group</td><td colspan="12">RO K/A Category Points</td><td colspan="5">SRO-Only Points</td></tr>
<tr><td>K 1</td><td>K 2</td><td>K 3</td><td>K 4</td><td>K 5</td><td>K 6</td><td>A 1</td><td>A 2</td><td>A 3</td><td>A 4</td><td>G *</td><td>Total</td><td colspan="2">A2</td><td colspan="2">G*</td><td>Total</td></tr>
<tr><td rowspan="3">1. Emergency & Abnormal Plant Evolutions</td><td>1</td><td>1</td><td>3</td><td>4</td><td></td><td></td><td></td><td>3</td><td>4</td><td></td><td></td><td>3</td><td>18</td><td colspan="2">1</td><td colspan="2">5</td><td>6</td></tr>
<tr><td>2</td><td>2</td><td>2</td><td>1</td><td></td><td></td><td></td><td>1</td><td>2</td><td></td><td></td><td>1</td><td>9</td><td colspan="2">1</td><td colspan="2">3</td><td>4</td></tr>
<tr><td>Tier Totals</td><td>3</td><td>5</td><td>5</td><td></td><td></td><td></td><td>4</td><td>6</td><td></td><td></td><td>4</td><td>27</td><td colspan="2">2</td><td colspan="2">8</td><td>10</td></tr>
<tr><td rowspan="3">2. Plant Systems</td><td>1</td><td>3</td><td>2</td><td>3</td><td>3</td><td>2</td><td>3</td><td>3</td><td>2</td><td>3</td><td>2</td><td>2</td><td>28</td><td colspan="2">3</td><td colspan="2">2</td><td>5</td></tr>
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<tr><td colspan="5" rowspan="2">3. Generic Knowledge and Abilities Categories</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">4</td><td rowspan="2">10</td><td>1</td><td>2</td><td>3</td><td>4</td><td rowspan="2">7</td></tr>
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Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems / evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems / 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 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.

NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
008 / Pressurizer Vapor Space Accident / 3						X	2.4.21	Emergency Procedures / Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.6	76
011 / Large Break LOCA / 3					X		EA2.01	Ability to determine or interpret the following as they apply to a Large Break LOCA: Actions to be taken, based on RCS temperature and pressure – saturated or superheated	4.7	77
015/17 / RCP Malfunctions / 4						X	2.2.40	Equipment Control: Ability to apply Technical Specifications for a system	4.7	78
025 / Loss of RHR System / 4						X	2.1.7	Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	4.7	79
055 / Station Blackout / 6						X	2.1.20	Conduct of Operations: Ability to interpret and execute procedure steps	4.6	80
026 / Loss of Component Cooling Water / 8						X	2.2.22	Equipment Control: Knowledge of limiting conditions for operations and safety limits.	4.7	81
065 / Loss of Instrument Air / 8					X		AA2.08	Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment	2.9	39
008 / Pressurizer Vapor Space Accident / 3			X				AK3.05	Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: ECCS termination or throttling criteria	4.0	40
057 / Loss of Vital AC Instrument Bus / 6			X				AK3.01	Knowledge of the reasons for the following responses as they apply to Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital AC instrument bus	4.1	41
058 / Loss of DC Power / 6						X	2.4.50	Emergency Procedures/Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual	4.2	42

CPNPP 1 & 2
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
W/E05 / Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4					X		EA2.2	Ability to determine and interpret the following as they apply to the Loss of Secondary Heat Sink: Adherence to appropriate procedures and operation within the limitations in the facility license and amendments	3.7	43
007 / Reactor Trip - Stabilization - Recovery / 1			X				EK3.01	Knowledge of the reasons for the following responses as they apply to the reactor trip: Actions contained in EOP for reactor trip	4.0	44
038 / Steam Generator Tube Rupture / 3		X					EK1.03	Knowledge of the operational implications of the following concepts as they apply to the SGTR: Natural circulation	3.9	45
009 / Small Break LOCA / 3				X			EA1.09	Ability to operate and/or monitor the following as they apply to a small break LOCA: RCP	3.6	46
077 / Generator Voltage and Electric Grid Disturbances / 6		X					AK2.06	Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Reactor power	3.9	47
027 / Pressurizer Pressure Control Malfunction / 3			X				AK3.03	Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunction: Actions contained in EOP for PZR PCS malfunction	3.7	48
022 / Loss of Reactor Coolant Makeup / 2					X		AA2.04	Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: How long pressurizer level can be maintained within limits	2.9	49
W/E11 / Loss of Emergency Coolant Recirculation / 4						X	2.1.28	Conduct of Operations: Knowledge of the purpose and function of major system components and controls	4.1	50
W/E04 / LOCA Outside Containment / 3		X					EK2.1	Knowledge of the interrelations between the LOCA Outside Containment and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	3.5	51
026 / 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 portion of the system causing the abnormal condition	2.6	52

CPNPP 1 & 2
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
054 / Loss of Main Feedwater / 4				X			AA1.01	Ability to operate and/or monitor the following as they apply to the Loss of Main Feedwater: AFW controls, including the use of alternate AFW sources	4.5	53
029 / ATWS / 1	X						EK1.05	Knowledge of the operational implications of the following as they apply to the ATWS: Definition of negative temperature coefficient as applied to large PWR coolant systems	2.8	54
W/E12 / Uncontrolled Depressurization of all Steam Generators / 4						X	2.4.11	Emergency Procedures/Plan: Knowledge of abnormal condition procedures	4.0	55
025 / Loss of RHR System / 4				X			AA1.12	Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: RCS temperature indicators	3.6	56
K/A Category Point Totals:	1	3	4	3	4 / 1	3 / 5	Group Point Total:			**18 / 6**

CPNPP 1 & 2
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
W/E01 & E02 / Rediagnosis & SI Termination / 3						X	2.4.18	Emergency Procedures / Plan: Knowledge of the specific bases for EOPs	4.0	82
W/E15 / Containment Flooding / 5					X		EA2.1	Ability to determine and interpret the following as they apply to the Containment Flooding: Facility conditions and selection of appropriate procedures during abnormal and emergency operations	3.2	83
W/E14 / High Containment Pressure / 5						X	2.2.38	Equipment Control: Knowledge of conditions and limitations in the facility license	4.5	84
033 / Loss of Intermediate Range NI / 7						X	2.2.22	Equipment Control: Knowledge of limiting conditions for operations and safety limits	4.7	85
061 / ARM System Alarms / 7	X						AK1.01	Knowledge of the operational implications of the following concepts as they apply to Area Radiation Monitoring System Alarms: Detector limitations	2.5	57
001 / Continuous Rod Withdrawal / 1					X		AA2.03	Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place	4.5	58
037 / Steam Generator Tube Leak / 3				X			AA1.13	Ability to operate and/or monitor the following as they apply to the Steam Generator Tube Leak: SG blowdown radiation monitors	3.9	59
076 / High Reactor Coolant Activity / 9					X		AA2.03	Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: RCS radioactivity level meter	2.5	60
W/E13 / Steam Generator Overpressure / 4		X					EK2.1	Knowledge of the interrelations between the Steam Generator Overpressure and the following: Components, and functions and control of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	3.0	61
033 / Loss of Intermediate Range NI / 7						X	2.2.44	Equipment Control: Ability to interpret control room indications to verify the status in operation of a system, and understand how operator actions and directives affect plant and system conditions	4.2	62

CPNPP 1 & 2
NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	Number	K/A Topic(s)	Imp.	Q#
024 / Emergency Boration / 1			X				AK3.02	Knowledge of the reasons for the following responses as they apply to Emergency Boration: Actions contained in EOP for emergency boration	4.2	63
067 / Plant Fire on Site / 8	X						AK1.02	Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site: Fire fighting	3.1	64
005 / Inoperable/Stuck Control Rod / 1		X					AK2.02	Knowledge of the interrelations between the Inoperable/Stuck Control Rod and the following: Breakers, relays, disconnects, and control room switches	2.5	65
K/A Category Point Totals:	2	2	1	1	2 / 1	1 / 3	Group Point Total:			**9 / 4**

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
012 / Reactor Protection								X				A2.05	Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty or erratic operation of detectors and function generators	3.2	86
026 / Containment Spray								X				A2.08	Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Safe securing of containment spray when it can be done	3.7	87
059 / Main Feedwater											X	2.4.9	Emergency Procedures/Plan: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies	4.2	88
062 / AC Electrical Distribution								X				A2.10	Ability to (a) predict the impacts of the following malfunctions or operations on the AC distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of switching power supplies on instruments and controls	3.3	89
007 / Pressurizer Relief/Quench Tank											X	2.4.47	Emergency Procedures/Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material	4.2	90
003 / Reactor Coolant Pump							X					A1.09	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: Seal flow and D/P	2.8	1
004 / Chemical and Volume Control						X						K6.20	Knowledge of the effect that a loss or malfunction of the following will have on the CVCS components: Function of demineralizer, including boron loading and temperature limits	2.5	2
005 / Residual Heat Removal		X										K2.01	Knowledge of bus power supplies to the following: RHR pumps	3.0	3

NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
006 / Emergency Core Cooling					X							K5.05	Knowledge of the operational implications of the following concepts as they apply to ECCS: Effects of pressure on a solid system	3.4	4
007 / Pressurizer Relief / Quench Tank				X								K4.01	Knowledge of the PRTS design feature(s) and/or interlock(s) that provide for the following: Quench tank cooling	2.6	5
008 / Component Cooling Water										X		A4.07	Ability to manually operate and/or monitor in the control room: Control of minimum level in the CCWS surge tank	2.9	6
010 / Pressurizer Pressure Control					X							K5.01	Knowledge of the operational implications of the following concepts as they apply to the PZR PCS: Determination of condition of fluid in PZR, using steam tables	3.5	7
010 / Pressurizer Pressure Control											X	2.1.32	Conduct of Operations: Ability to explain and apply system limits and precautions	3.8	8
012 / Reactor Protection			X									K3.01	Knowledge of the effect that a loss or malfunction of the RPS will have on the following: CRDS	3.9	9
013 / Engineered Safety Features Actuation									X			A3.02	Ability to monitor automatic operation of the ESFAS including: Operation of actuated equipment	4.1	10
013 / Engineered Safety Features Actuation						X						K6.01	Knowledge of the effect that a loss or malfunction of the following will have on the ESFAS: Sensors and detectors	2.7	11
022 / Containment Cooling			X									K3.02	Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment instrumentation readings	3.0	12
026 / Containment Spray				X								K4.01	Knowledge of CSS design feature(s) and/or interlock(s) that provide for the following: Source of water for CSS, including recirculation phase after LOCA	4.2	13

NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
039 / Main and Reheat Steam							X					A1.05	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: RCS Tave	3.2	14
039 / Main and Reheat Steam	X											K1.05	Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: Turbine generator	2.5	15
059 / Main Feedwater											X	2.2.38	Equipment Control: Knowledge of conditions and limitations in the facility license	3.6	16
061 / Auxiliary/Emergency Feedwater								X				A2.04	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: Pump failure or improper operation	3.4	17
061 / Auxiliary/Emergency Feedwater		X										K2.02	Knowledge of bus power supplies to the following: AFW electric drive pumps	3.7	18
062 / AC Electrical Distribution	X											K1.04	Knowledge of the physical connections and/or cause-effect relationships between the AC distribution system and the following systems: Offsite power sources	3.7	19
062 / AC Electrical Distribution							X					A1.01	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AC distribution system controls including: Significance of diesel generator load limits	3.4	20
063 / DC Electrical Distribution									X			A3.01	Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights	2.7	21
064 / Emergency Diesel Generator						X						K6.07	Knowledge of the effect that a loss or malfunction of the following will have on the EDG system: Air receivers	2.7	22

NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
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
076 / Service Water									X			A3.02	Ability to monitor automatic operation of the SWS, including: Emergency heat loads	3.7	24
076 / Service Water				X								K4.06	Knowledge of SWS design feature(s) and/or interlock(s) that provide for the following: Service water train separation	2.8	25
078 / Instrument Air										X		A4.01	Ability to manually operate and/or monitor in the control room: Pressure gauges	3.1	26
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	27
103 / Containment	X											K1.02	Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems: Containment isolation/containment integrity	3.9	28
K/A Category Point Totals:	3	2	3	3	2	3	3	2 / 3	3	2	2 / 2	Group Point Total:			**28 / 5**

CPNPP 1 & 2
NRC Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
068 / Liquid Radwaste								X				A2.04	Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation	3.3	91
034 / Fuel Handling Equipment										X		A4.02	Ability to manually operate and/or monitor in the control room: Neutron levels	3.9	92
086 / Fire Protection								X				A2.01	Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Manual shutdown of the FPS	3.1	93
072 / Area Radiation Monitoring					X							K5.02	Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation intensity changes with source distance	2.5	29
011 / Pressurizer Level Control		X										K2.02	Knowledge of bus power supplies to the following: PZR heaters	3.1	30
056 / Condensate	X											K1.03	Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: MFW	2.6	31
071 / Waste Gas Disposal							X					A1.06	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Waste Gas Disposal System controls including: Ventilation system	2.5	32
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	33

CPNPP 1 & 2
NRC Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Number	K/A Topics	Imp.	Q#
001 / Control Rod Drive				X								K4.03	Knowledge of CRDS design feature(s) and or interlock(s) which provide for the following: Rod control logic	3.5	34
055 / Condenser Air Removal			X									K3.01	Knowledge of the effect of a loss or malfunction of the CARS will have on the following: Main condenser	2.5	35
035 / Steam Generator								X				A2.02	Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Reactor trip/turbine trip	4.2	36
017 / In-core Temperature Monitor						X						K6.01	Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors	2.7	37
014 / Rod Position Indication											X	2.2.38	Equipment Control: Knowledge of conditions and limitations in the facility license	3.6	38
K/A Category Point Totals:	1	1	1	1	1	1	1	2 / 2	0	0 / 1	1 / 0	Group Point Total:			**10 / 3**

Facility: **CPNPP 1 & 2** Date of Exam: **03/29/10**

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.32	Ability to explain and apply system limits and precautions			4.0	94
	2.1.23	Ability to perform specific system and integrated plant procedures during all modes of plant operation			4.4	95
	2.1.39	Knowledge of conservative decision making practices	3.6	66		
	2.1.29	Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	4.1	67		
	2.1.19	Ability to use plant computers to evaluate system or component status	3.9	68		
	Subtotal			3		2
2. Equipment Control	2.2.38	Knowledge of conditions and limitations in the facility license			4.5	96
	2.2.12	Knowledge of surveillance procedures	3.7	69		
	2.2.25	Knowledge of the bases in technical specifications for limiting conditions for operations and safety limits	3.2	70		
	Subtotal			2		1
3. Radiation Control	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.			3.1	97
	2.3.6	Ability to approve release permit			3.8	98
	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions	3.2	71		
	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	72		
	Subtotal			2		2
4. Emergency Procedures / Plan	2.4.41	Knowledge of emergency action level thresholds and classifications			4.6	99
	2.4.38	Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required			4.4	100
	2.4.20	Knowledge of the operational implications of EOP warnings, cautions, and notes	3.8	73		
	2.4.13	Knowledge of crew roles and responsibilities during EOP usage	4.0	74		
	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal and emergency evolutions	3.7	75		
	Subtotal			3		2
Tier 3 Point Total				**10**		**7**

Tier / Group	Randomly Selected K/A	Reason for Rejection
1 / 2	W/E14 G 2.4.35	Q #84 – This specific K/A does not apply as there are no local operator tasks performed for the High Containment Pressure EOP at CPNPP. Randomly reselected W/E14 G 2.2.38.
1 / 2	061 AA1.01	Q #57 – This specific K/A does not apply as there are no automatic actions for Area Radiation Monitors at CPNPP. Selected 061 AK1.01 for skyscraper balance.
2 / 2	072 K3.01	Q #29 – This specific K/A does not apply as Containment Ventilation Isolation is actuated by a Process vice Area Radiation Monitor at CPNPP. Randomly reselected 072 K5.02.
2 / 2	075 A4.01	Q #33 – Unable to develop a psychometrically sound question that discriminates at the RO level. Reselected 075 A2.02.
2 / 2	055 A3.03	Q #35 – This specific K/A does not apply as there is no automatic diversion of Condenser Air Removal System exhaust at CPNPP. Randomly reselected 055 K3.01 as there are limited K/As of 2.5 or greater Importance Factor in System 055 – Condenser Air Removal.
2 / 2	072 K3.02	Q #29 – This specific K/A does not apply as there is no Fuel Handling operation that is impacted by the Area Radiation Monitoring System at CPNPP. Randomly reselected 072 K5.02 for skyscraper balance.
2 / 1	039 A1.03	Q #14 – Unable to develop a psychometrically sound question that discriminates at the RO level. Randomly reselected 039 A1.06.
1 / 1	062 AA2.02	Q #52 – Coverage of the Station Service Water System deemed adequate per Questions #24 and #25. Randomly reselected 026 AA2.03.
1 / 2	037 AA1.09	Q #59 – This specific K/A does not apply as there are no Reactor Coolant System Loop Isolation Valves at CPNPP. Randomly reselected 037 AA1.13.
1 / 1	058 G 2.2.25	Q #42 – Unable to develop a psychometrically sound question that discriminates at the RO level. Randomly reselected G 2.4.50.
3 / 1	G 2.1.14	Q #66 – Coverage of this K/A deemed adequate per NRC JPM RA3. Randomly reselected G 2.1.39.
1 / 1	011 EA2.03	Q #77 – Coverage of this K/A topic already addressed by Qs #6, #52, and #81. Reselected 011 EA2.01.
1 / 1	055 G 2.1.25	Q #80 – Unable to develop a psychometrically sound question that discriminates at the SRO level. Reselected 055 G 2.1.20.
1 / 1	026 AA2.02	Q #81 – Unable to develop a psychometrically sound question that discriminates at the SRO level. Reselected 026 G 2.2.22.
1 / 2	W/E01 & E02 G 2.4.1	Q #82 – Unable to develop a psychometrically sound question that discriminates at the SRO level. Reselected W/E01 & E02 G 2.4.18.
1 / 2	032 G 2.4.1	Q #85 – Unable to develop a psychometrically sound question that discriminates at the SRO level. Reselected 033 G 2.2.22.
2 / 1	012 A2.03	Q #86 – Unable to develop a psychometrically sound question that discriminates at the SRO level. Reselected 012 A2.05.

2 / 1	062 A2.12	Q #89 – Unable to develop a psychometrically sound question that discriminates at the SRO level. Reselected 062 A2.10.
3 / 1	G 2.1.39	Q #95 – Coverage of this K/A topic already addressed by Q #66. Reselected G 2.1.23.
3 / 4	G 2.4.28	Q #99 – Unable to develop a psychometrically sound question that discriminates at the SRO level. Reselected G 2.4.41.
3 / 1	G 2.1.15	Q #67 – Unable to develop a psychometrically sound question that discriminates at the RO level. Reselected G 2.1.19.

Facility: CPNPP Units 1 & 2		Date of Examination:	03/29/10
Examination Level	RO ☐	Operating Test Number:	NRC

Administrative Topic (see Note)	Type Code*	Describe Activity to be Performed
Conduct of Operations	M, R	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. (3.9) JPM: Perform a Manual Quadrant Power Tilt Ratio Calculation (RO1803A).
Conduct of Operations	M, R	2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. (4.1) JPM: Perform a Power Change Worksheet Calculation (RO1302).
Equipment Control	M, R	2.2.6 Knowledge of the process for making changes to procedures. (3.0) JPM: Initiate a Procedure Change (RO5004).
Radiation Control	D, S	2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (3.4) JPM: Perform Actions for an Accident Involving Spent Fuel (RO4504).
Emergency Plan		

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

*Type Codes & Criteria:

(C)ontrol room, (S)imulator, or Class(R)oom

(D)irect from bank (≤ 3 for ROs; ≤ for 4 for SROs & RO retakes)

(N)ew or (M)odified from bank (≥ 1)

(P)revious 2 exams (≤ 1; randomly selected)

RA1 The candidate will perform a manual Quadrant Power Tilt Ratio calculation per OPT-302, Calculating Power Tilt Ratio, and determine whether Acceptance Criteria are met. The critical steps include recording data, accurately performing calculations and applying Acceptance Criteria. This is a modified bank JPM.

RA2 The candidate will perform a Power Change Calculation Worksheet per IPO-003A, Power Operations, Attachment 3, Power Change Worksheet, for a Unit downpower. The critical steps include making reactivity determinations based on plant conditions. This is a modified bank JPM.

RA3 The candidate will initiate a Procedure Change Notice per STA-202, Procedure Change Notice for a mislabeled step in ABN-501, Station Service Water System Malfunction. The critical steps include proper identification of the required level of review, proper form completion and correctly performing the mark-up of the affected page. This is a modified bank JPM.

RA4 The candidate will implement radiological emergency actions per ABN-908, Fuel Handling Accident, for an accident involving spent fuel in the Fuel Handling Building. The critical steps include initiating local evacuation, Site notification, and ensuring proper ventilation alignment. This is a bank JPM.

Facility: CPNPP Units 1 and 2		Date of Examination:	03/29/10
Examination Level	SRO ☐	Operating Test Number:	NRC

Administrative Topic (see Note)	Type Code*	Describe Activity to be Performed
Conduct of Operations	M, R	2.1.1 Knowledge of conduct of operations requirements. (4.2) JPM: Determine Technical Specification and Event Reportability (SO1005).
Conduct of Operations	D, R	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (4.4) JPM: Manually Perform Critical Safety Function Status Checks (SO1135).
Equipment Control	N, 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 (New).
Radiation Control	M, R	2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. (3.7) JPM: Select Personnel for Emergency Exposure (SO1142).
Emergency Plan	M, R	2.4.44 Knowledge of emergency plan protective action recommendations. (4.4) JPM: Determine Protective Action Requirements (SO1136).

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

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

Administrative Topics Outline
Task Summary

SA1 The applicant will identify impacted Technical Specification Limiting Conditions for Operations and determine Event Reportability per STA-501, Non-Routine Reporting and CPNPP Technical Specifications. The critical steps include identifying the Technical Specification and determining the oral and written Reporting Requirements. This is a modified bank JPM.

SA2 The applicant will manually determine Critical Safety Function Status during a LOCA scenario. The critical tasks include accurately determining the status for each Critical Safety Function. This is a bank JPM.

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 new JPM.

SA4 The applicant will be required to choose a volunteer for an Emergency Exposure per EPP-305, Emergency Exposure Guidelines and Personnel Dosimetry. The critical steps require the applicant to choose the appropriate volunteer for a lifesaving activity. This is a modified bank JPM.

SA5 The applicant will determine Protective Actions per EPP-304, Protective Action Recommendations. The critical steps include determining the proper Protective Actions, Pasquill Stability Class, and Zones to be evacuated or sheltered. This is a modified bank JPM.

Facility:	CPNPP Units 1 and 2	Date of Examination:	03/29/10
Exam Level:	RO ☐ SRO(I) ☐ **SRO (U)** ☐	Operating Test No.:	NRC

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

	System / JPM Title	Type Code*	Safety Function
S-1	001 – Control Rod Drive System (RO1008) Perform Control Rod Exercises (RO ONLY)	D, S	1
S-2	**006 – Emergency Core Cooling System (New)** **Align Cold Leg Injection During a Loss of Inventory**	**A, EN, L, N, S**	**2**
S-3	010 – Pressurizer Pressure Control System (RO1209B) Control Pressurizer Pressure During Cooldown	D, EN, L, S	3
S-4	**003 – Reactor Coolant Pump System (RO1118)** **Respond to Reactor Coolant Pump Seal Malfunction**	**A, D, S**	**4-P**
S-5	061 – Auxiliary / Emergency Feedwater System (RO3504) Test the Turbine Driven Auxiliary Feedwater Pump	A, M, S	4-S
S-6	**022 – Containment Cooling System (New)** **Respond to Containment High Temperature Alarm**	**A, N, S**	**5**
S-7	062 – AC Electrical Distribution System (New) Transfer 480 VAC Bus from Normal to Alternate Source	N, S	6
S-8	016 – Non-Nuclear Instrumentation System (RO1829) Respond to Steam Flow Instrument Failure	A, D, S	7

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

P-1	**064 – Emergency Diesel Generator System (AO6311A)** **Perform Local Restoration of EDG**	**D, E, R**	**6**
P-2	004 – Chemical & Volume Control System (AO5202) Restore Charging Flow with PD Charging Pump	D, E, R	2
P-3	**086 – Fire Protection System (New)** **Perform Actions for Fire In Containment**	**E, N, 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 The candidate will perform Control Rod Exercises for Control Bank D rods per OPT-106A, Control Rods Exercise. This is a bank JPM under Control Rod Drive System – Reactivity Control Safety Function.

S-2 The candidate will initiate Hot Leg Injection per ABN-104, Residual Heat Removal System, Section 8.0, Complete Loss of Decay Heat Removal Capability - RCS Not Filled after a loss of inventory event while in MODE 5. The alternate path occurs when Safety Injection Pumps cannot be aligned and a flowpath is established to initiate Cold Leg Injection. This is a new, low power JPM under the Emergency Core Cooling System – Reactor Coolant System Inventory Control Safety Function.

S-3 The candidate will reduce Reactor Coolant System Pressure and Block Safety Injection per IPO-005A, Plant Cooldown from Hot Standby to Cold Shutdown, Section 5.1, Cooldown from MODE 3 to MODE 5, Step 5.1.5. This is a bank JPM under the Pressurizer Pressure Control System – Reactor Pressure Control Safety Function.

S-4 The candidate will recognize the indications and perform the actions for Reactor Coolant Pump Seal Abnormalities per ABN-101, Reactor Coolant Pump Trip / Malfunction. The alternate path occurs when it is recognized that excessive leakoff requiring a Unit and Reactor Coolant Pump Trip exists. This is a bank JPM under the Reactor Coolant Pump System – Primary System Heat Removal from Reactor Core Safety Function.

S-5 The candidate will perform Post-Maintenance Testing of the Turbine Driven Auxiliary Feedwater Pump with flow to the Steam Generators per SOP-304A, Auxiliary Feedwater System. The alternate path occurs when a high bearing temperature requires a trip of the pump. This is a modified bank JPM under Auxiliary / Emergency Feedwater System – Secondary System Heat Removal from Reactor Core Safety Function. This is a PRA significant action.

S-6 The candidate will recognize the indications and perform the actions for a high Containment temperature condition per ALM-0031A, 1-ALB-3A, Window 1.1, CTMNT TEMP HI. The alternate path occurs when less than three Containment Fan Coolers are available and the non-operating Control Rod Drive Mechanism Cooling Fan is aligned to cool Containment. This is a new JPM under the Containment Cooling System – Containment Integrity Safety Function.

S-7 The candidate will align 480 VAC Bus 1EB1 to the Alternate Power Source per SOP-604A, 480 VAC Switchgear and MCCs, Step 5.3.2, 480 V Safeguards Bus Transfer from the Normal to the Alternate Power Source. This is a new JPM under the AC Electrical Distribution System – Electrical Safety Function.

S-8 The candidate will respond to a Steam Flow Instrument failure per ALM-0081A, 1-ALB-8A, Window 1.8, SG 1 STM FLO & FW FLO MISMATCH and ABN-707, Steam Flow Instrument Malfunction. The alternate path occurs when Steam Generator level is not being adequately controlled and the operator must take manual control. This is a bank JPM under the Non-Nuclear Instrumentation System – Instrumentation Safety Function.

P-1 The candidate will perform local restoration of the Emergency Diesel Generator per ABN-601, Response to a 138/345 KV System Malfunction, Attachment 1, Restoration of a Diesel Generator following a Station Blackout. This is a bank JPM under the Emergency Diesel Generators System – Electrical Safety Function.

P-2 The candidate will perform the actions to restart the Positive Displacement Charging Pump and restore Charging flow per ABN-301, Instrument Air System Malfunction and SOP-103A, Chemical and Volume Control. This is a bank JPM under the Chemical and Volume Control System – Reactor Coolant System Inventory Control Safety Function.

P-3 The candidate will perform actions during a fire in Containment per ABN-807A/B, Response to a Fire in the Containment Building, Attachment 1, Actions to be Taken by the Nuclear Equipment Operator. This is a new, time critical JPM under the Fire Protection System – Plant Service Systems Safety Function.

Facility: CPNPP 1 and 2							Date of Exam: 03/29/10					Operating Test No.: NRC					
APPLICANT	EVENT TYPE	SCENARIOS															
		CPNPP #1			CPNPP #2									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
SROU	RX	-			-			-			-			**-**	**1**	**1**	**0**
	NOR	2			1			-			1,2			**-**	**1**	**1**	**1**
	I/C	1,2,3,4,5			2,3,4			1,2,3,4			3,4,5,7			**-**	**4**	**4**	**2**
	MAJ	6,8			5			5			8			**-**	**2**	**2**	**1**
	TS	1,3			2,3			2,4			3,4			**-**	**0**	**2**	**2**
SROI	RX	-	-		-	1		-	-		-	2		**-**	**1**	**1**	**0**
	NOR	2	-		1	-		-	-		1,2	-		**-**	**1**	**1**	**1**
	I/C	1,2,3,4,5	1,3,4,9		2,3,4	4,6		1,2,3,4	2,4,7		3,4,5,7	7,9,10		**-**	**4**	**4**	**2**
	MAJ	6,8	6,8		5	5		5	5		8	8		**-**	**2**	**2**	**1**
	TS	1,3	-		2,3	-		2,4	-		3,4	-		**-**	**0**	**2**	**2**
RO	RX		-	-		1	-		-	-		2	-	**-**	**1**	**1**	**0**
	NOR		-	2		-	1		-	-		-	1,2	**-**	**1**	**1**	**1**
	I/C		1,3,4,9	5,7		4,6	2,3,7		2,4,7	1,3,8		7,9,10	3,5	**-**	**4**	**4**	**2**
	MAJ		6,8	6,8		5	5		5	5		8	8	**-**	**2**	**2**	**1**
	TS		-	-		-	-		-	-		-	-	**-**	**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.

Facility:	CPNPP 1 & 2	Scenario No.: 1	Op Test No.: March 2010 NRC
Examiners:		Operators:	

Initial Conditions:	• 100% power MOL - RCS Boron is 910 ppm by Chemistry sample. • Motor Driven Auxiliary Feedwater Pump 1-01 OOS for coupling repair.
Turnover:	Maintain steady-state 100% power conditions.
Critical Tasks:	• Manually Trip the Main Turbine when Automatic Reactor Protection Trip Fails. • Establish Heat Removal using Reactor Coolant System Bleed and Feed.

Event No.	Malf. No.	Event Type*	Event Description
1 +10 min	RX17A	C (RO, SRO) TS (SRO)	Power Operated Relief Valve (PCV-455A) seat leakage.
2 +20 min	TC08A	N (BOP, SRO)	High Pressure Turbine Stop Valve #4 fails closed. Manual Turbine Runback required.
3 +30 min	RX05A	I (RO, SRO) TS (SRO)	Pressurizer Level Transmitter (LT-459A) fails high.
4 +35 min	TU04	C (RO, SRO)	Main Turbine vibration @ 15 mils on a 300 second ramp requiring manual Reactor trip.
5 +35 min	TC07A	C (BOP, SRO)	Main Turbine fails to trip on Reactor trip requiring manual Turbine trip.
6 +36 min	ED01	M (RO, BOP, SRO)	Loss of Offsite Power 30 seconds after Reactor trip.
7 +36 min	EG06B	C (BOP)	Emergency Diesel Generator (1-02) start failure.
8 +40 min	FW09A	M (RO, BOP, SRO)	Turbine Driven Auxiliary Feedwater Pump trips on overspeed 300 seconds after Reactor trip. Total Loss of Feedwater.
9 +50 min	DIRCV 8000A	C (RO)	Power Operated Relief Valve (PCV-455A) Block Valve (1/1-8000A) fails closed.

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

SCENARIO SUMMARY NRC #1

The crew will assume the watch and maintain steady-state conditions per IPO-003A, Power Operations. Auxiliary Feedwater Pump 1-01 is out-of-service for coupling repair.

The first event is a leaking Power Operated Relief Valve (PORV). The crew responds per Alarm Procedure ALM-0052A, Window 3.1, Pressurizer PORV Outlet Temperature High. Actions include cycling the PORV after its associated PORV Block Valve is closed. The PORV Block Valve will later fail to open complicating the scenario. The SRO will refer to Technical Specifications.

When ALM-0052A actions are complete, a High Pressure Turbine Stop Valve fails closed. Electrical output of the Generator will drop from approximately 1265 MWe to 950 MWe and require an immediate Turbine Runback to 900 MWe. Actions are performed per ABN-401, Main Turbine Malfunction, Section 9.0, Inadvertent Closure of an HP or LP Stop or Control Valve. When the Turbine Runback is initiated, the crew will monitor for proper Rod Control and Steam Dump System response.

When plant conditions are stable, a Pressurizer level instrument will fail high. The crew will respond per ABN-706, Pressurizer Level Instrumentation Malfunction. The RO will take manual control of Pressurizer level or Charging flow to maintain Pressurizer level on program. Once the faulty instrument is identified and an alternate controlling channel is selected, Charging flow and Pressurizer level control will be returned to AUTO. The SRO will refer to Technical Specifications.

The next event is an increasing vibration on the Main Turbine caused by the High Pressure Stop Valve closure. The Unit Supervisor will enter ABN-401, Main Turbine Malfunction, Section 2.0, Turbine Shaft or Frame Vibration High and the crew will monitor the condition of the Main Turbine and Generator and vibration will continue to increase requiring a manual Reactor and Turbine trip. When the Reactor is tripped, the Turbine will fail to trip and require a manual actuation.

At this point, the crew will enter EOP-0.0A, Reactor Trip or Safety Injection and transition to EOS-0.1A, Reactor Trip Response at Step 4. The scenario is complicated by a Loss of Offsite Power and the failure of the Train B Emergency Diesel Generator to start. While in EOS-0.1A, the crew will determine that a Total Loss of Feedwater has occurred when the Turbine Driven Auxiliary Feedwater Pump trips and a transition to FRH-0.1A, Response to Loss of Secondary Heat Sink is required. An immediate transition in FRH-0.1A to Step 12 will be identified and the RO must perform alternate actions when only one PORV will open.

The scenario is terminated when adequate Reactor Coolant System Bleed and Feed is verified.

Risk Significance:

• Risk important components out of service:	Auxiliary Feedwater Pump 1-01
• Risk significant core damage sequence:	Turbine Trip Failure
	Loss of Feedwater Flow
• Risk significant operator actions:	Manually Trip Main Turbine
	Initiate RCS Bleed and Feed
	PORV Block Valve Fails to Open

Facility: CPNPP 1 & 2	Scenario No.: 2	Op Test No.: March 2010 NRC	
Examiners:		Operators:	

Initial Conditions:
- ~1X10^{-8} amps BOL - RCS Boron is 1545 ppm by Chemistry sample.
- Steam Dump System in service for RCS Temperature Control.

Turnover: Raise Power to 2% in preparation for plant startup to 100% power.

Critical Tasks:
- Restore Feedwater Flow to any Affected Steam Generator.
- Manually Initiate Safety Injection Upon Failure to Automatically Actuate.
- Manually Initiate Containment Isolation Phase A Upon Failure to Automatically Actuate.
- Perform Actions to Identify and Isolate Faulted Steam Generator.

Event No.	Malf. No.	Event Type*	Event Description
1 +20 min		R (RO) N (BOP, SRO)	Raise power to 2%.
2 +30 min	Override	C (BOP, SRO) TS (SRO)	Safety Injection Accumulator (1-01) nitrogen leak.
3 +40 min	FW24B	I (BOP, SRO) TS (SRO)	Motor Driven Auxiliary Feedwater Pump (1-02) trip.
4 +60 min	ED05H	C (RO, BOP, SRO) TS (SRO)	Loss of 6.9 KV Safeguards Bus 1EA1.
5 +65 min	MS03A	M (RO, BOP, SRO)	Steam Generator (1-01) Steam Line Break outside Containment.
6 +65 min	RP07A RP07B	I (RO)	Safety Injection Train A and Train B fail to automatically actuate.
7 +65 min	RP09A RP09B	C (BOP)	Containment Isolation Phase A Train A and Train B fail to automatically actuate.

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

SCENARIO SUMMARY NRC #2

The crew will assume the watch with a Plant Startup in progress and will continue raising power to approximately 2% per IPO-002A, Plant Startup from Hot Standby.

When conditions are stable, a Safety Injection Accumulator nitrogen leak will occur. Actions are performed per the Alarm Response Procedure and SOP-201A, Safety Injection Accumulators. The SRO will refer to Technical Specifications.

When conditions are stable, Motor Driven Auxiliary Feedwater Pump 1-02 will trip. The crew will refer to ABN-305, Auxiliary Feedwater System Malfunction, Section 3.0 and determine that Steam Generator levels are slowly decreasing and start the Turbine Driven Auxiliary Feedwater Pump. The SRO will refer to Technical Specifications.

The next event is a loss of 6.9 KV Safeguards Bus 1EA1. The crew will respond per ABN-602, Response to a 6900/480V System Malfunction. Actions include starting a Centrifugal Charging Pump and stopping the Emergency Diesel Generator without Station Service Water flow. With the loss of the 2nd Motor Driven Auxiliary Feedwater Pump, the BOP will align feedwater flow to Steam Generators 1-01 and 1-02. Additionally, the crew will perform actions per ABN-602 to ensure necessary plant equipment is operating and affected equipment is placed in PULL OUT. The SRO will refer to Technical Specifications.

When ABN-602 actions are complete, a Steam Line Break outside Containment will occur on Steam Generator 1-01. The crew will enter EOP-0.0A, Reactor Trip or Safety Injection and then transition to EOP-2.0A, Faulted Steam Generator Isolation at Step 12. While performing the actions of EOP-0.0A, the RO will be required to manually initiate both Trains of Safety Injection and the BOP will be required to manually initiate both Trains of Containment Isolation Phase A.

Once the faulted Steam Generator is isolated, the Unit Supervisor will transition to EOS-1.1A, Safety Injection Termination.

The scenario is terminated after EOS-1.1A, Safety Injection Termination is entered and the actions to secure Safety Injection flow and establish Letdown flow are performed.

Risk Significance:

• Failure of risk important system prior to trip:	Loss of MDAFW Pump 1-01
	Loss of a 6.9 KV Safeguards Bus
• Risk significant core damage sequence:	Steam Line Break Outside Containment
• Risk significant operator actions:	Restore Auxiliary Feedwater Flow
	Manually Initiate Safety Injection
	Manually Initiate Containment Isolation
	Identify & Isolate Faulted Steam Generator

Facility: CPNPP 1 & 2 Scenario No.: 3 Op Test No.: March 2010 NRC

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions:
- 72% power MOL - RCS Boron is 916 ppm by Chemistry sample.
- Motor Driven Auxiliary Feedwater Pump 1-01 OOS for coupling repair.

Turnover: Maintaining 72% power per Load Controller direction. Rod Control in AUTO.

Critical Tasks:
- Emergency Borate due to Loss of Digital Rod Position Indication.
- Identify and Isolate the Ruptured Steam Generator.
- Cooldown the Reactor Coolant System.

Event No.	Malf. No.	Event Type*	Event Description
1 +15 min	FW16	C (BOP, SRO)	Lowering Condenser vacuum requiring power reduction.
2 +25 min	RP05A	I (RO, SRO) TS (SRO)	Reactor Coolant System Loop (1-01) Narrow Range Cold Leg Temperature Instrument (TI-411B) fails low.
3 +30 min	RX01G	I (BOP, SRO)	Steam Generator (1-04) Feed Flow Instrument (FT-540) fails high.
4 +40 min	CV01B	C (RO, SRO) TS (SRO)	Centrifugal Charging Pump (1-01) trip.
5 +45 min	SG01D	M (RO, BOP, SRO)	Steam Generator (1-04) Tube Rupture ramping to 650 gpm over 5 minutes.
6 +45 min	MS07D		Steam Generator (1-04) Main Steam Isolation Valve (HV-2336A) fails closed upon initial Radiation Monitor alarm.
7 +45 min	RD12C	I (RO)	Digital Rod Position Indication power failure upon manual or auto Reactor trip actuation. Emergency boration required.
8 +55 min	Override	C (BOP)	Steam Generator (1-04) Turbine Driven Auxiliary Feedwater Pump Steam Supply Valve (HV-2452-1) fails to close.

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

SCENARIO SUMMARY NRC #3

The crew will assume the watch at 72% power with no scheduled activities per IPO-003A, Power Operations. The Grid Controller has requested that power remain at this level due to transmission line overload until further notice. Auxiliary Feedwater Pump 1-01 is out-of-service for coupling repair.

The first event is a loss of Condenser vacuum due to a drained loop seal. The crew will respond per ABN-304, Main Condenser and Circulating Water System Malfunction, Section 3.0. Actions include lowering of Main Turbine load until Condenser vacuum is maintained above 24.5" of vacuum.

The next event it is a low failure of T_{COLD} transmitter, TI-411B. Operator actions are per ABN-704, T_{COLD} / N-16 Instrumentation Malfunction, Section 2.0 and require stopping the withdrawal of Control Rods and restoring Reactor Coolant System temperature and Pressurizer level to normal. The SRO will refer to Technical Specifications.

When conditions are stable, a Steam Generator Flow Transmitter fails high. Operator response is per ABN-708, Feedwater Flow Instrument Malfunction, Section 2.0. The operator must take manual control of the affected Feed Control Valve to prevent a Unit trip on low Steam Generator water level. After manual control is established, an Alternate Channel is selected and Automatic control restored.

When the Steam Generator level control has been returned to Automatic, a loss of the running Centrifugal Charging Pump will occur. The crew will enter ABN-105, Chemical and Volume Control System Malfunction and perform actions to immediately restored Charging flow. The SRO will refer to Technical Specifications.

When Technical Specifications have been addressed, a Steam Generator Tube Rupture will ramp in over five minutes on Steam Generator 1-04. With increasing Main Steam Line radiation levels and lowering Pressurizer pressure, the Unit Supervisor will direct a Reactor and Turbine Trip.

The crew enters EOP-0.0A, Reactor Trip or Safety Injection and performs actions through Step 13 and then transitions to EOP-3.0A, Steam Generator Tube Rupture. Once it has been determined that Steam Generator 1-04 is the source of the tube rupture, the Main Steam Isolation Valve for that Steam Generator will fail closed. Isolation of Steam Generator 1-04 is complicated when its associated Main Steam Supply Valve to the Turbine Driven Auxiliary Feedwater (TDAFW) Pump will not close. The Response Not Obtained actions include tripping the TDAFW Pump. Additionally a Loss of Digital Rod Position Indication System will require an Emergency Boration.

This scenario is terminated when the ruptured Steam Generator is isolated and the crew is commencing a cooldown and depressurization of the Reactor Coolant System.

Risk Significance:

- Risk important components out of service: Auxiliary Feedwater Pump 1-01
- Failure of risk important system prior to trip: Centrifugal Charging Pump 1-01
- Risk significant core damage sequence: Steam Generator Tube Rupture
- Risk significant operator actions: Emergency Borate Due to Loss of DRPI

  Isolate Ruptured Steam Generator

Facility: CPNPP 1 & 2	Scenario No.: 4	Op Test No.: March 2010 NRC	
Examiners:		Operators:	

Initial Conditions:
- ~3% power BOL - RCS Boron is 1545 ppm by Chemistry sample.
- Steam Dump System in service for RCS Temperature Control.

Turnover: Transfer from Auxiliary Feedwater System to Main Feedwater System.

Critical Tasks:
- Identify Excess Reactor Coolant System leakage and Manually Trip Reactor.
- Trip Reactor Coolant Pumps on Loss of Subcooling.

Event No.	Malf. No.	Event Type*	Event Description
1 +15 min		N (BOP, SRO)	Transfer from Auxiliary Feedwater System to Main Feedwater System and place Feedwater Bypass Control Valves in AUTO.
2 +30 min		R (RO, BOP) N (SRO)	Raise power to 8% in preparation for synchronizing the Main Generator to the electrical grid.
3 +40 min	RX04C	I (BOP, SRO) TS (SRO)	Steam Generator (1-03) Level Transmitter (LT-553) fails low.
4 +50 min	NI03A	TS (SRO)	Power Range Nuclear Instrument (N-41) detector fails high.
5 +55 min	TP02A	C (BOP, SRO)	Turbine Plant Cooling Water Pump (1-01) sheared shaft.
6 +60 min	AN2A_02.1 AN2A_03.1		ALB-02A-2.1, Seismic Monitoring System Activation. ALB-02A-3.1, Operating Basis Earthquake Exceedance.
7 +70 min	CV02	C (RO, SRO)	Charging Line leak inside Containment.
8 +70 min	RC08B1	M (RO, BOP, SRO)	Small Break Loss of Coolant Accident inside Containment.
9 +70 min	RP01	I (RO)	Automatic Reactor Trip failure.
10 +70 min	Override	C (RO)	Reactor Coolant Pump (1-02) fails to manually trip. Manually open feeder breaker to 6.9 kV Bus 1A2.

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

SCENARIO SUMMARY NRC #4

The crew will assume the watch with power at approximately 3% per IPO-002A, Plant Startup from Hot Standby. The crew will transfer Feedwater flow from the Auxiliary Feedwater System to the Main Feedwater System in preparation for raising power to 8%. This is followed by entry into SOP-304A, Auxiliary Feedwater System, Section 5.2, Shutdown and Standby of the Auxiliary Feedwater System.

When transfer of Feedwater has been completed, the crew will enter IPO-003A, Power Operations, Section 5.1, Warmup and Synchronization of the Turbine Generator and perform a power ascension using the Rod Control and Steam Dump Systems.

When power has been raised 3% to 5%, a Steam Generator Level Transmitter will fail low. 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 then select an alternate controlling channel to return the Feedwater System to automatic control. The SRO will refer to Technical Specifications.

When Technical Specifications are addressed, a Power Range Nuclear Instrument will fail high. The crew will enter ABN-703, Power Range Instrument Malfunction. The crew will perform actions to defeat inputs from the failed channel. The SRO will refer to Technical Specifications.

The next event is a sheared shaft of the running Turbine Plant Cooling Water (TPCW) Pump. The crew will enter ABN-306, Turbine Plant Cooling Water System Malfunction and recognize that the TPCW Pump is running without discharge flow or pressure indications and start the standby TPCW Pump.

When TPCW flow is restored, a seismic event will occur. The crew will enter ABN-907, Acts of Nature, Section 2.0, Earthquake and perform actions as required by the ABN. This is the initiating event for the Charging Line Leak inside Containment. The crew will enter ABN-103, Excessive Reactor Coolant Leakage and perform actions in an attempt to locate the source of the leakage. While performing actions in ABN-103 the crew will isolate Letdown and Charging and determine that the source of leakage is in the Charging Line. When actions to place Excess Letdown in service are reached a Small Break Loss of Coolant Accident will occur.

With the automatic Reactor Trip function disabled, the crew will determine that a manual Reactor Trip must be performed with entry into EOP-0.0A, Reactor Trip or Safety Injection. While performing EOP-0.0A actions the Reactor Coolant Pumps (RCP) must be stopped due to a loss of subcooling. RCP 1-02 will not trip from its normal location and require deenergizing of the associated 6900 V Bus or local trip of the breaker by an operator in the field. At Step 14, a transition to EOP-1.0A, Loss of a Reactor or Secondary Coolant will occur.

The scenario is terminated when an evaluation of plant status is performed to verify Cold Leg Recirculation capability.

Risk Significance:

- Risk important components out of service: None
- Risk significant core damage sequence: Small Break Loss of Coolant Accident
- Risk significant operator actions: Manually Trip Reactor Due to SBLOCA

  Manually Trip Reactor Coolant Pumps