Form 4.1-BWR Pressurized-Water Reactor Examination Outline

Facility: Cooper Nuclear Station Date of Exam: June 18, 2024

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Total	A2	G	Total
1. Emergency and Abnormal Plant Evolutions	1	3	4	3	N/A			4	3	N/A		3	20			7
	2	2	1	1				1	1			0	6			3
	Tier Totals	5	5	4				5	4			3	26			10
2. Plant Systems	1	2	2	2	3	3	3	3	2	2	2	2	26			5
	2	1	1	1	1	1	1	1	1	1	1	1	11			3
	Tier Totals	3	3	3	4	4	4	4	3	3	3	3	37			8
3. Generic Knowledge and Abilities Categories	CO		EC				RC		EM				6	CO / EC / RC / EM		7
	2		2				1		1							
4. Theory	Reactor Theory				Thermodynamics								6			
	3				3											

Notes: CO = Conduct of Operations; EC = Equipment Control; RC = Radiation Control; EM = Emergency Procedures/Plan

* These systems/evolutions may be eliminated from the sample when Revision 2 of the K/A catalog is used to develop the sample plan

** These systems/evolutions are only included as part of the sample (as applicable to the facility) when Revision 2 of the K/A catalog is used to develop the sample plan

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Partial or Complete Loss of Forced Core Flow Circulation: (CFR: 41.8 to 41.10) AK1.04 Thermal-hydraulic instabilities	4.3	9
295003 (APE 3) Partial or Complete Loss of AC Power			X				Knowledge of the reasons for the following responses or actions as they apply to Partial or Complete Loss of AC Power: (CFR: 41.5 / 45.6) AK3.01 Manual and automatic bus transfer	3.7	11
295004 (APE 4) Partial or Total Loss of DC Power					X		Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of DC Power: (CFR: 41.10 / 43.5 / 45.13) AA2.01 Partial or complete loss of DC power	4.0	20
295005 (APE 5) Main Turbine Generator Trip		X					Knowledge of the relationship between Main Turbine Generator Trip and the following systems or components: (CFR: 41.7 / 45.8) AK2.03 Recirculation system	3.5	4
295006 (APE 6) Scram	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to SCRAM: (CFR: 41.8 to 41.10) AK1.03 Reactivity control	3.9	2
295016 (APE 16) Control Room Abandonment / 7						X	2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12)	4.0	14
295018 (APE 18) Partial or Complete Loss of CCW		X					Knowledge of the relationship between Partial or Complete Loss of Component Cooling Water and the following systems or components: (CFR: 41.7 / 45.8) AK2.05 RHR/LPCI	3.3	13
295019 (APE 19) Partial or Complete Loss of Instrument Air					X		Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Instrument Air: (CFR: 41.10 / 43.5 / 45.13) AA2.01 Instrument air pressure	4.1	15
295021 (APE 21) Loss of Shutdown Cooling			X				Knowledge of the reasons for the following responses or actions as they apply to Loss of Shutdown Cooling: (CFR: 41.5 / 45.6) AK3.05 Establishing alternate heat removal flow paths	4.0	6
295023 (APE 23) Refueling Accidents				X			Ability to operate and/or monitor the following as they apply to Refueling Accidents: (CFR: 41.7 / 45.6) AA1.10 Alternate fuel pool makeup systems	3.5	18
295024 (EPE 1) High Drywell Pressure				X			Ability to operate and/or monitor the following as they apply to High Drywell Pressure: (CFR: 41.7 / 45.6) EA1.13 Suppression pool cooling	3.8	5
295025 (EPE 2) High Reactor Pressure			X				Knowledge of the reasons for the following responses or actions as they apply to High Reactor Pressure: (CFR: 41.5 / 45.6) EK3.03 HPCI operation	3.5	1

295026 (EPE 3) Suppression Pool High Water Temperature				X			Ability to operate and/or monitor the following as they apply to Suppression Pool High Water Temperature: (CFR: 41.7 / 45.6) EA1.09 Safety/relief valves	4.1	17
295027 (EPE 4) High Containment Temperature (Mark III Containment Only)									
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only)						X	2.1.20 Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)	4.6	19
295030 (EPE 7) Low Suppression Pool Water Level		X					Knowledge of the relationship between Low Suppression Pool Water Level and the following systems or components: (CFR: 41.7 / 45.8) EK2.04 RHR/LPCI	3.8	16
295031 (EPE 8) Reactor Low Water Level				X			Ability to operate and/or monitor the following as they apply to Reactor Low Water Level: (CFR: 41.7 / 45.6) EA1.06 Automatic depressurization system	4.3	7
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown		X					Knowledge of the relationship between SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown and the following systems or components: (CFR: 41.7 41.8 / 45.8) EK2.18 Reactor feedwater system	3.8	12
295038 (EPE 15) High Offsite Radioactivity Release Rate					X		Ability to determine and/or interpret the following as they apply to High Offsite Radioactivity Release Rate: (CFR: 41.10 / 43.5 / 45.13) EA2.05 Emergency plan implementation	3.6	10
600000 (APE 24) Plant Fire on Site						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. (CFR: 41.5 / 43.5 / 45.12)	4.2	8
700000 (APE 25) Generator Voltage and Electric Grid Disturbances	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8) AK1.04 Frequency changes	3.2	3
K/A Category Totals:	3	4	3	4	3	3	Group Point Total:		20

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	#
295002 (APE 2) Loss of Main Condenser Vacuum									
295007 (APE 7) High Reactor Pressure									
295008 (APE 8) High Reactor Water Level	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to High Reactor Water Level: (CFR: 41.8 to 41.10) AK1.03 Feed flow/steam flow mismatch	3.6	23
295009 (APE 9) Low Reactor Water Level									
295010 (APE 10) High Drywell Pressure									
295011 (APE 11) High Containment Temperature (Mark III Containment only)									
295012 (APE 12) High Drywell Temperature	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to High Drywell Temperature: (CFR: 41.8 to 41.10) AK1.01 Drywell pressure	4.0	25
295013 (APE 13) High Suppression Pool Water Temperature/ 5									
295014 (APE 14) Inadvertent Reactivity Addition									
295017 (APE 17) High Offsite Release Rate									
295020 (APE 20) Inadvertent Containment Isolation		X					Knowledge of the relationship between Inadvertent Containment Isolation and the following systems or components: (CFR: 41.7 / 45.8) AK2.13 Containment atmosphere control system	3.0	26
295022 (APE 22) Loss of Control Rod Drive Pumps			X				Knowledge of the reasons for the following responses or actions as they apply to Loss of Control Rod Drive Pumps: (CFR: 41.5 / 45.6) AK3.02 Restoring CRDM cooling/drive water flow	3.6	24
295029 (EPE 6) High Suppression Pool Water Level					X		Ability to determine and/or interpret the following as they apply to High Suppression Pool Water Level: (CFR: 41.10 / 43.5 / 45.13) EA2.03 Drywell/containment water level	3.8	22
295032 (EPE 9) High Secondary Containment Area Temperature				X			Ability to operate and/or monitor the following as they apply to High Secondary Containment Area Temperature: (CFR: 41.7 / 45.6) EA1.02 Leak detection system	3.8	21
295033 (EPE 10) High Secondary Containment Area Radiation Levels									
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									
295035 (EPE 12) Secondary Containment High Differential Pressure									
295036 (EPE 13) Secondary Containment High Sump/Area Water Level									
500000 (EPE 16) High Containment Hydrogen Concentration									
K/A Category Point Totals:	2	1	1	1	1	0	Group Point Total:		6

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	#
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.02 Valves	3.7	49
205000 (SF4 SCS) Shutdown Cooling			X									Knowledge of the effect that a loss or malfunction of the Shutdown Cooling System (RHR Shutdown Cooling Mode) will have on the following systems or system parameters: (CFR: 41.7 / 45.4) K3.05 Fuel pool cooling and cleanup	3.0	43
206000 (SF2, SF4 HPCI) High-Pressure Coolant Injection										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.02 Flow controller	4.3	47
207000 (SF4 IC) Isolation (Emergency) Condenser														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray					X							Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Low-Pressure Core Spray System: (CFR: 41.5 / 45.3) K5.07 Adequate core cooling	4.5	48
209002 (SF2, SF4 HPCS) High-Pressure Core Spray														
211000 (SF1 SLCS) Standby Liquid Control						X						Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Standby Liquid Control System: (CFR: 41.7 / 45.7) K6.01 Plant air systems	2.5	36
212000 (SF7 RPS) Reactor Protection					X							Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Reactor Protection System: (CFR: 41.5 / 45.3) K5.02 Logic channel arrangements	4.1	39
215003 (SF7 IRM) Intermediate-Range Monitor				X								Knowledge of Intermediate Range Monitor System design features and/or interlocks that provide for the following: (CFR: 41.7) K4.06 Alarm seal-in	2.9	44
215004 (SF7 SRMS) Source-Range Monitor								X				Ability to (a) predict the impacts of the following on the Source Range Monitor System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) A2.03 Stuck detector	3.1	45
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor	X											Knowledge of the physical connections and/or cause and effect relationships between the Average Power Range Monitor/Local Power Range Monitor System and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.03 RBMS	3.9	30

217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling	X											Knowledge of the physical connections and/or cause and effect relationships between the Reactor Core Isolation Cooling System and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.14 Primary containment isolation system	3.9	31
218000 (SF3 ADS) Automatic Depressurization											X	2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (CFR: 41.10 / 43.5 / 45.3)	4.2	51
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff								X				Ability to (a) predict the impacts of the following on the Primary Containment Isolation System/Nuclear Steam Supply Shutoff and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 45.6) A2.09 Inadvertent system initiation	4.0	41
239002 (SF3 SRV) Safety Relief Valves										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.05 Reactor pressure	4.4	52
259002 (SF2 RWLCS) Reactor Water Level Control									X			Ability to monitor automatic features of the Reactor Water Level Control System, including: (CFR: 41.7 / 45.7) A3.12 Transfer from three-element to one-element control	3.4	42
261000 (SF9 SGTS) Standby Gas Treatment							X					Ability to predict and/or monitor changes in parameters associated with operation of the Standby Gas Treatment System, including: (CFR: 41.5 / 45.5) A1.05 Primary containment oxygen level (Mark I and II)	3.0	50
262001 (SF6 AC) AC Electrical Distribution				X								Knowledge of AC Electrical Distribution design features and/or interlocks that provide for the following: (CFR: 41.7) K4.07 Breaker closure permissives	3.6	33
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)									X			Ability to monitor automatic operation of the Uninterruptable Power Supply (AC/DC), including: (CFR: 41.7 / 45.7) A3.01 Transfer of power sources	3.4	27
263000 (SF6 DC) DC Electrical Distribution											X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)	4.4	35
264000 (SF6 EGE) Emergency Generators (Diesel/Jet)			X									Knowledge of the effect that a loss or malfunction of the Emergency Generators will have on the following systems or system parameters: (CFR: 41.7 / 45.4) K3.03 Operationally significant loads	4.3	40
300000 (SF8 IA) Instrument Air						X						Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Instrument Air System: (CFR: 41.8 / 45.7) K6.15 Low instrument air pressure	3.6	28
400000 (SF8 CCW) Component Cooling Water		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.02 CCW valves	2.8	34

510000 (SF4 SWS) Service Water							X					Ability to predict and/or monitor changes in parameters associated with operation of the Service Water System, including: (CFR: 41.5 / 45.5) A1.06 CCW heat exchanger outlet temperature	3.4	46
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray						X						Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Low-Pressure Core Spray System: (CFR: 41.7 / 45.7) K6.05 ECCS room cooler(s)	3.4	32
218000 (SF3 ADS) Automatic Depressurization					X							Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Automatic Depressurization System: (CFR: 41.5 / 45.3) K5.01 ADS logic operation	4.3	38
215003 (SF7 IRM) Intermediate-Range Monitor							X					Ability to predict and/or monitor changes in parameters associated with operation of the Intermediate Range Monitor System, including: (CFR: 41.5 / 45.5) A1.06 Lights and alarms	3.6	37
261000 (SF9 SGTS) Standby Gas Treatment				X								Knowledge of Standby Gas Treatment System design features and/or interlocks that provide for the following: (CFR: 41.7) K4.04 Radioactive particulate filtration	3.4	29
K/A Category Point Totals:	2	2	2	3	3	3	3	2	2	2	2	Group Point Total:		26

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 (SF1 CRDH) CRD Hydraulic		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.04 SCRAM discharge volume vent and drain valve solenoids	3.8	55
201002 (SF1 RMCS) Reactor Manual Control														
201003 (SF1 CRDM) Control Rod and Drive Mechanism														
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer														
202001 (SF1, SF4 RS) Recirculation														
202002 (SF1 RSCTL) Recirculation Flow Control														
204000 (SF2 RWCU) Reactor Water Cleanup					X							Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Reactor Water Cleanup System: (CFR: 41.5 / 45.3) K5.09 System flow	3.1	54
214000 (SF7 RPIS) Rod Position Information														
215001 (SF7 TIP) Traversing In-Core Probe														
215002 (SF7 RBMS) Rod Block Monitor	X											Knowledge of the physical connections and/or cause and effect relationships between the Rod Block Monitor System and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.04 Reactor recirculation system	3.4	57
216000 (SF7 NBI) Nuclear Boiler Instrumentation			X									Knowledge of the effect that a loss or malfunction of the Nuclear Boiler Instrumentation will have on the following systems or system parameters: (CFR: 41.7 / 45.4) K3.29 Jet pump flow	3.3	58
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode									X			Ability to monitor automatic operation of the RHR/LPCI: Torus/Suppression Pool Cooling Mode, including: (CFR: 41.7 / 45.7) A3.01 Valve operation	3.9	61
223001 (SF5 PCS) Primary Containment and Auxiliaries														
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Cooling Mode														
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup						X						Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Fuel Pool Cooling and Clean- Up: (CFR: 41.7 / 45.7) K6.07 Component cooling water systems	3.2	63

234000 (SF8 FH) Fuel Handling Equipment														
239001 (SF3, SF4 MRSS) Main and Reheat Steam														
239003 (SF9 MSIVLC) Main Steam Isolation Valve Leakage Control														
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating														
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary														
256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater														
268000 (SF9 RW) Radwaste							X					Ability to predict and/or monitor changes in parameters associated with operation of the Radwaste System, including: (CFR: 41.5 / 45.5) A1.02 Offsite release (liquid/gaseous release)	3.2	56
271000 (SF9 OG) Offgas														
272000 (SF7, SF9 RMS) Radiation Monitoring														
286000 (SF8 FPS) Fire Protection										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.05 Motor-driven fire pump start	3.2	60
288000 (SF9 PVS) Plant Ventilation														
290001 (SF5 SC) Secondary Containment														
290003 (SF9 CRV) Control Room Ventilation				X								Knowledge of Control Room Ventilation design features and/or interlocks that provide for the following: (CFR: 41.7) K4.03 Differential pressure control	3.1	62
290002 (SF4 RVI) Reactor Vessel Internals								X				Ability to (a) predict the impacts of the following on the Reactor Vessel and Internals and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) A2.03 Control rod drop accident	4.4	59
510001 (SF8 CWS) Circulating Water											X	2.1.27 Knowledge of system purpose and/or function. (CFR: 41.7)	3.9	53
K/A Category Point Totals:	1	1	1	1	1	1	1	1	1	1	1	Group Point Total:		11

Form 4.1-COMMON Common Examination Outline

Facility: Cooper Nuclear Station		Date of Exam: June 18, 2024				
Generic Knowledge and Abilities—Tier 3 (RO/SRO)						
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.18	Ability to make accurate, clear, and concise logs, records, status boards, and reports (CFR: 41.10 / 45.12 / 45.13)	3.6	64		
	2.1.38	Knowledge of the station's requirements for verbal communications when implementing procedures (CFR: 41.10 / 45.13)	3.7	69		
	Subtotal		N/A		N/A	
2. Equipment Control	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity (CFR: 41.5 / 41.10 / 43.5 / 43.6 / 45.1)	4.5	67		
	2.2.41	Ability to obtain and interpret station electrical and mechanical drawings (reference potential) (CFR: 41.10 / 45.12 / 45.13)	3.5	66		
	Subtotal		N/A		N/A	
3. Radiation Control	2.3.12	Knowledge of radiological safety principles and 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, or alignment of filters (CFR: 41.12 / 43.4 / 45.9 / 45.10)	3.2	68		
	Subtotal		N/A		N/A	
4. Emergency Procedures/ Plan	2.4.45	Ability to prioritize and interpret the significance of each annunciator or alarm (CFR: 41.10 / 43.5 / 45.3 / 45.12)	4.1	65		
	Subtotal		N/A		N/A	
Tier 3 Point Total				**6**		**7**

Theory—Tier 4 (RO)				
Category	K/A #	Topic	RO	
			IR	#
Reactor Theory	292003 K1.07	Reactor Kinetics and Neutron Sources: Explain prompt critical, prompt jump, and prompt drop	3.3	74
	292006 K1.10	Fission Product Poisons: Plot the curve and explain the reasoning for the reactivity insertion by Xenon-135 versus time for the following: Reactor startup with xenon-135 already present in the core	2.9	73
	292008 K1.06	Reactor Operational Physics: List parameters that should be monitored and controlled upon reaching initial criticality	4.2	71
	Subtotal		N/A	
Thermodynamics	293007 K1.12	Heat Transfer: Define percent reactor power	2.7	72
	293009 K1.30	Thermal Limits: Relate thermal time constant to transient operating condition	2.7	70
	293010 K1.01	Brittle Fracture and Vessel Thermal Stress: State the brittle fracture mode of failure	2.8	75
	Subtotal		N/A	
Tier 4 Point Total				**6**

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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 and Abnormal Plant Evolutions	1												20	4	3	7
	2				N/A					N/A			6	2	1	3
	Tier Totals												26	6	4	10
2. Plant Systems	1												26	3	2	5
	2												11	1 1	1	3
	Tier Totals												37	5	3	8

Tier	RO: CO	EC	RC	EM	RO Total	SRO: CO	EC	RC	EM	SRO Total
3. Generic Knowledge and Abilities Categories	2	2	1	1	6	2	2	1	2	7

Tier	Reactor Theory	Thermodynamics	Total
4. Theory	3	3	6

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation									
295003 (APE 3) Partial or Complete Loss of AC Power									
295004 (APE 4) Partial or Total Loss of DC Power						X	2.4.18 Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)	4.0	79
295005 (APE 5) Main Turbine Generator Trip						X	2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (CFR: 41.10 / 43.2 / 45.6)	4.7	82
295006 (APE 6) Scram									
295016 (APE 16) Control Room Abandonment / 7									
295018 (APE 18) Partial or Complete Loss of CCW									
295019 (APE 19) Partial or Complete Loss of Instrument Air									
295021 (APE 21) Loss of Shutdown Cooling					X		Ability to determine and/or interpret the following as they apply to Loss of Shutdown Cooling: (CFR: 41.10 / 43.5 / 45.13) AA2.04 Reactor water temperature	4.3	76
295023 (APE 23) Refueling Accidents						X	2.4.31 Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)	4.1	80
295024 (EPE 1) High Drywell Pressure									
295025 (EPE 2) High Reactor Pressure					X		Ability to determine and/or interpret the following as they apply to High Reactor Pressure: (CFR: 41.10 / 43.5 / 45.13) EA2.04 Suppression pool level	3.4	78
295026 (EPE 3) Suppression Pool High Water Temperature									
295027 (EPE 4) High Containment Temperature (Mark III Containment Only)									
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only)									
295030 (EPE 7) Low Suppression Pool Water Level									
295031 (EPE 8) Reactor Low Water Level									
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown									
295038 (EPE 15) High Offsite Radioactivity Release Rate									
600000 (APE 24) Plant Fire on Site					X		Ability to determine and/or interpret the following as they apply to Plant Fire on Site: (CFR: 41.10 / 43.5 / 45.13) AA2.15 Requirements for establishing a fire watch (SRO Only)	3.3	77
700000 (APE 25) Generator Voltage and Electric Grid Disturbances					X		Ability to determine and/or interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 43.5 / 45.5 / 45.7 / 45.8) AA2.07 Operations status of safety-related (vital) buses	4.0	81
K/A Category Totals:					4	3	Group Point Total:		7

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	#
295002 (APE 2) Loss of Main Condenser Vacuum									
295007 (APE 7) High Reactor Pressure									
295008 (APE 8) High Reactor Water Level									
295009 (APE 9) Low Reactor Water Level					X		Ability to determine and/or interpret the following as they apply to Low Reactor Water Level: (CFR: 41.10 / 43.5 / 45.13) AA2.01 Reactor water level	4.5	83
295010 (APE 10) High Drywell Pressure									
295011 (APE 11) High Containment Temperature (Mark III Containment only)									
295012 (APE 12) High Drywell Temperature									
295013 (APE 13) High Suppression Pool Water Temperature/ 5									
295014 (APE 14) Inadvertent Reactivity Addition									
295017 (APE 17) High Offsite Release Rate									
295020 (APE 20) Inadvertent Containment Isolation									
295022 (APE 22) Loss of Control Rod Drive Pumps									
295029 (EPE 6) High Suppression Pool Water Level									
295032 (EPE 9) High Secondary Containment Area Temperature									
295033 (EPE 10) High Secondary Containment Area Radiation Levels									
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9					X		Ability to determine and/or interpret the following as they apply to Secondary Containment Ventilation High Radiation: (CFR: 41.10 / 43.5 / 45.13) EA2.01 Ventilation radiation levels	4.0	85
295035 (EPE 12) Secondary Containment High Differential Pressure									
295036 (EPE 13) Secondary Containment High Sump/Area Water Level									
500000 (EPE 16) High Containment Hydrogen Concentration						X	2.2.38 Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)	4.5	84
K/A Category Point Totals:					2	1	Group Point Total:		3

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	#
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode														
205000 (SF4 SCS) Shutdown Cooling														
206000 (SF2, SF4 HPCI) High-Pressure Coolant Injection														
207000 (SF4 IC) Isolation (Emergency) Condenser														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray														
209002 (SF2, SF4 HPCS) High-Pressure Core Spray														
211000 (SF1 SLCS) Standby Liquid Control														
212000 (SF7 RPS) Reactor Protection														
215003 (SF7 IRM) Intermediate-Range Monitor														
215004 (SF7 SRMS) Source-Range Monitor														
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor														
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling								X				Ability to (a) predict the impacts of the following on the Reactor Core Isolation Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR 41.5 / 43.5 / 45.6) A2.01 Inadvertent system initiation signal	4.2	89
218000 (SF3 ADS) Automatic Depressurization														
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff								X				Ability to (a) predict the impacts of the following on the Primary Containment Isolation System/Nuclear Steam Supply Shutoff and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 45.6) A2.14 Reactor protection system failures	4.0	86
239002 (SF3 SRV) Safety Relief Valves								X				Ability to (a) predict the impacts of the following on the Safety Relief Valves and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) A2.03 Stuck-open SRV	4.4	87
259002 (SF2 RWLCS) Reactor Water Level Control											X	2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13)	4.2	90
261000 (SF9 SGTS) Standby Gas Treatment														
262001 (SF6 AC) AC Electrical Distribution														
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)														

263000 (SF6 DC) DC Electrical Distribution											X	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 43.2)	4.2	88
264000 (SF6 EGE) Emergency Generators (Diesel/Jet)														
300000 (SF8 IA) Instrument Air														
400000 (SF8 CCW) Component Cooling Water														
510000 (SF4 SWS) Service Water														
K/A Category Point Totals:								3			2	Group Point Total:		5

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 (SF1 CRDH) CRD Hydraulic														
201002 (SF1 RMCS) Reactor Manual Control														
201003 (SF1 CRDM) Control Rod and Drive Mechanism														
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer														
202001 (SF1, SF4 RS) Recirculation														
202002 (SF1 RSCTL) Recirculation Flow Control														
204000 (SF2 RWCU) Reactor Water Cleanup														
214000 (SF7 RPIS) Rod Position Information														
215001 (SF7 TIP) Traversing In-Core Probe								X				Ability to (a) predict the impacts of the following on the Traversing In-Core Probe and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 45.6) A2.07 Failure to retract during accident conditions (BWR 2, 3, 4, 5)	3.5	91
215002 (SF7 RBMS) Rod Block Monitor														
216000 (SF7 NBI) Nuclear Boiler Instrumentation														
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode														
223001 (SF5 PCS) Primary Containment and Auxiliaries														
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Cooling Mode														
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup														
234000 (SF8 FH) Fuel Handling Equipment							X					Ability to predict and/or monitor changes in parameters associated with operation of the Fuel Handling System, including: (CFR: 41.5 / 45.5) A1.05 Refueling machine position, speed, or direction	2.8	93
239001 (SF3, SF4 MRSS) Main and Reheat Steam														
239003 (SF9 MSIVLC) Main Steam Isolation Valve Leakage Control														
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating														
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary														

256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater														
268000 (SF9 RW) Radwaste														
271000 (SF9 OG) Offgas											X	2.4.12 Knowledge of operating crew responsibilities during emergency and abnormal operations (CFR: 41.10 / 45.12)	4.3	92
272000 (SF7, SF9 RMS) Radiation Monitoring														
286000 (SF8 FPS) Fire Protection														
288000 (SF9 PVS) Plant Ventilation														
290001 (SF5 SC) Secondary Containment														
290003 (SF9 CRV) Control Room Ventilation														
290002 (SF4 RVI) Reactor Vessel Internals														
510001 (SF8 CWS) Circulating Water														
K/A Category Point Totals:							1	1			1	Group Point Total:		3

Form 4.1-COMMON Common Examination Outline

Facility: Cooper Nuclear Station   Date of Exam: June 18, 2024

Generic Knowledge and Abilities—Tier 3 (RO/SRO)

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.5	Ability to use procedures related to shift staffing, such as minimum crew complement or overtime limitations (reference potential) (CFR: 41.10 / 43.5 / 45.12)			3.9	95
	2.1.15	Knowledge of administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda (CFR: 41.10 / 45.12)			3.4	94
	Subtotal		N/A		N/A	
2. Equipment Control	2.2.20	Knowledge of the process for managing troubleshooting activities (CFR: 41.10 / 43.5 / 45.13)			3.8	97
	2.2.35	Ability to determine technical specification mode of operation (CFR: 41.7 / 41.10 / 43.2 / 45.13)			4.5	100
	Subtotal		N/A		N/A	
3. Radiation Control	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only) (CFR: 43.4 / 45.10)			3.8	99
	Subtotal		N/A		N/A	
4. Emergency Procedures/ Plan	2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations (CFR: 41.7 / 41.10 / 43.5 / 45.12)			4.4	96
	2.4.32	Knowledge of operator response to loss of annunciators (CFR: 41.10 / 43.5 / 45.13)			4.0	98
	Subtotal		N/A		N/A	
Tier 3 Point Total				**6**		**7**

Form 4.1-1 Record of Rejected Knowledge and Abilities

Refer to Examination Standard (ES)-4.2, “Developing Written Examinations,” Section B.3, for deviations from the approved written examination outline.

Tier/Group	Randomly Selected K/A	Reason for Rejection
		None

Form 3.2-1 Administrative Topics Outline

Facility: Cooper Nuclear Station		Date of Examination: 06/10/2024
Examination Level: RO ■	SRO ☐	Operating Test Number: CNS-2024-06

Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
Conduct of Operations	A.1 Maintain RO License Active Status K/A 2.1.4 (3.3)	(R) (N)
Conduct of Operations	A.2 Determine Actions for a Mispositioned Control Rod K/A 2.1.37 (4.3) (Used from NRC 2020-09)	(R) (D)
Equipment Control	A.3 Verify RCIC Standby Status K/A 2.2.44 (4.2)	(R) (N)
Radiation Control	A.4 Calculate Dose for Planned Special Exposure K/A 2.3.12 (3.2)	(R) (N)
Emergency Plan	N/A	N/A

RO

A1 **Maintaining RO License Active Status**

This JPM evaluates the Examinee's ability to use procedure 2.0.7 to document license requirements and maintain an Active Status.

1. Fill in Attachment 1 using an operator's hours for all watch stations
2. Glasses verification
3. Determine if operators will maintain proficiency

New

A2 **Determine Actions for a Mispositioned Control Rod**

This JPM evaluates the examinee's ability to identify a mispositioned Control Rod and determine recovery actions. Examinee determined that Control Rod 02-31 was mispositioned and identified actions required per section 8 of procedure 10.13, Control Rod Sequence and Movement Control.

Bank

A3 **Verify RCIC Standby Status**

This JPM evaluates the Examinee's ability to use procedure 2.0.4, RELIEF PERSONNEL AND SHIFT TURNOVER, to verify RCIC in a Standby Status. Examinee determined the following components were not in a Normal Lineup; RCIC-MO-18, RCIC-MO-41, RCIC-MO-16. The Applicant determined that with RCIC-MO-16 closed, RCIC is not in a standby line-up.

New

A4 **Determine Planned Special Exposure Dose**

This JPM evaluates the Examinee's ability to calculate estimated dose for a PSE and if work is allowed. Examinee determined will determine the estimated dose workers will receive, determined out of the 3 candidates that only one is able to perform the PSE.

New

Instructions for completing Form 3.2-1, “Administrative Topics Outline”

1. For each license level, determine the number of administrative job performance measures (JPMs) and topic areas as follows:

Topic	Number of JPMs	
	RO*	SRO and RO Retakes
Conduct of Operations	1 (or 2)	2
Equipment Control	1 (or 0)	1
Radiation Control	1 (or 0)	1
Emergency Plan	1 (or 0)	1
Total	**4**	**5**

* Reactor operator (RO) applicants do not need to be evaluated on every topic (i.e., “Equipment Control,” “Radiation Control,” or “Emergency Plan” can be omitted by doubling up on “Conduct of Operations”), unless the applicant is taking only the administrative topics portion of the operating test (with a waiver or excusal of the other portions).

2. Enter the associated knowledge and abilities (K/A) statement and summarize the administrative activities for each JPM.

3. For each JPM, specify the type codes for location and source as follows:
   Location:
   (C)ontrol room, (S)imulator, or Class(R)oom
   Source and Source Criteria:
   (P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)
   (D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)
   (N)ew or Significantly (M)odified from bank (no fewer than one)

Form 3.2-1 Administrative Topics Outline

Facility: Cooper Nuclear Station		Date of Examination: 6/10/2024
Examination Level: RO ☐ SRO ☒		Operating Test Number: CNS-2024-06
Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
Conduct of Operations	A.5 Direct Actions for Chemistry Out of Limits K/A 2.1.34 (3.5) (Modified from NRC 2020-09 A6)	(R) (M)
Conduct of Operations	A.6 Direct Actions for Unexplained Decrease in Reactor Power (NRC 2021 Retake which was not used) K/A 2.1.7 (4.7), 2.1.25 (4.2)	(R) (D)
Equipment Control	A.7 Determine Whether Reactor Mode Change is Allowed NRC 2021 Retake which was not used) K/A 2.2.22 (4.7)	(R) (D)
Radiation Control	A.8 Approve Radioactive Discharge Release Permit (NRC 2021 Retake which was not used) K/A 2.3.6 (3.8)	(R) (D)
Emergency Plan	A.9 Determine Emergency Classification EAL (HU4.2). K/A 2.4.41 (4.6)	(R) (N)

Instructions for completing Form 3.2-1, “Administrative Topics Outline”

1. For each license level, determine the number of administrative job performance measures (JPMs) and topic areas as follows:

Topic	Number of JPMs	
	RO*	**SRO and RO Retakes**
Conduct of Operations	1 (or 2)	2
Equipment Control	1 (or 0)	1
Radiation Control	1 (or 0)	1
Emergency Plan	1 (or 0)	1
Total	**4**	**5**

* Reactor operator (RO) applicants do not need to be evaluated on every topic (i.e., “Equipment Control,” “Radiation Control,” or “Emergency Plan” can be omitted by doubling up on “Conduct of Operations”), unless the applicant is taking only the administrative topics portion of the operating test (with a waiver or excusal of the other portions).

2. Enter the associated knowledge and abilities (K/A) statement and summarize the administrative activities for each JPM.

3. For each JPM, specify the type codes for location and source as follows:

   Location:

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

   Source and Source Criteria:

   (P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)
   (D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)
   (N)ew or Significantly (M)odified from bank (no fewer than one)

CN 2024 NRC Exam
NRC SRO Admin JPM Description

SRO

A5 **<u>Determine Actions for Chemistry Out of Limits</u>**

Examinee determines isolating Condenser Water Box B2, commencing an immediate plant shutdown, and establishing Mode 4 as rapidly as conditions permit is required IAW Procedure 2.4CHEM, Chemistry Parameter Out of Limit.

A6 **<u>Interpret plant conditions to diagnose jet pump failure and determine required action (Procedure 2.4RXPWR, Attachment 2)</u>**

IAW Procedure 2.4RXPWR, Reactor Power Anomalies, examinee determines a jet pump failure has occurred, and entry into TS LCO 3.4.2 Condition A and plant shutdown per Procedure 2.1.4, Normal Shutdown, or 2.1.4.1, Rapid Shutdown, are required.

A7 **<u>Determine whether mode change is allowed based on plant conditions.</u>**

The applicant determines a Reactor Mode change to Mode 2 is not permitted with DG2 inoperable in accordance with Technical Specifications LCO 3.8.1, LCO 3.0.4, and Bases.

A8 **<u>Approve Radioactive Discharge Release Permit</u>**

The applicant will identify errors that prevent SM approval of a LRW discharge. Examinee reviews entries in Section 5 of Procedure 8.8.11 Attachment 1, Liquid Radioactive Waste Discharge Form, identifies errors IAW the answer key, and does not sign the Shift Manager approval for release.

A9 **<u>Determine Emergency Classification EAL (HU4.2)</u>**

The applicant will have to determine the correct EAL for a Fire located in a Table H-1 area (Control Building MCC fire, the alarm does not directly give the location of the fire unless the candidate understands the plant locations, the candidate will have to understand that the cable spreading room is part of the Control Building.

The plant will be at 100% rated power, when a single fire alarm is received in the MCR. There is no other evidence of a fire, so the building watches are sent out to investigate. At T-35 minutes, the report comes in that light smoke is seen in a air conditioning unit in the location. The fire brigade posts a fire watch to unsure a fire does not start at T-65.

The candidate will have to determine the correct EAL (HU4.2):

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.1 Unusual Event

A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table H-1 Fire Areas
• PC (Drywell and Torus) • Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

Mode Applicability:

All

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., **no** other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **any** Table H-1 area

AND

The existence of a FIRE is **not** verified (proved or disproved) within 30 min. of alarm receipt (Note 1)

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Table H-1 Fire Areas
• PC (Drywell and Torus) • Reactor Building • Control Building • Service Water Pump Room • Diesel Generator Building • Cable Expansion Room

Mode Applicability:

All

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.3	**Unusual Event**
A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

Note 1: The Emergency Director should declare the event promptly upon determining that the time limit has been exceeded, or will likely be exceeded. The Emergency Director is not allowed additional time to declare after the time limit is exceeded.

Mode Applicability:

All

Form 3.2-2 Control Room/In-Plant Systems Outline

Facility: Cooper Nuclear Station	Date of Examination: 6/10/24 Operating Test Number: CNS-2024-06
Exam Level: ☒ RO ☒ SRO-I ☒ SRO-U	

System/JPM Title	Type Code	Safety Function
Control Room Systems		
S1. Shift REC pumps and Respond to Low Pressure New K/A 295018 AA1.03 (3.3) RO, SROI, SROU	A, N, S	8
S2. Transfer RPSPP Bus A From MG Set to CDP1B 2014 NRC K/A: 212000 A2.02 (4.0) RO	D, S	7
S3. Start second RR pump and respond to overload/ground New K/A 202001 A4.01 (4.0) RO, SROI	A, N, S	1
S4. Alternate pressure control with reactor feed pumps 2017 NRC K/A 259001 A4.02 (4.0) RO, SROI	D, L, S	2
S5. Vent Primary Containment 2011 NRC K/A 223001 A2.07 (4.4) RO, SROI, SROU	D, S	5

S6. Conduct alternate emergency depressurization with main steam line drains 2012 NRC K/A 239001 A4.02 (3.4) RO, SROI	D, L, S	3
S7. Place AC Exciter in Service New K/A 262001.A2.05 (3.8) RO, SROI	A, N, S	6
S8. Manually Initiate HPCI (Place HPCI in Pressure Control Mode). New K/A 206000 A4.04 (3.8) RO, SROI, SROU	A, EN, L, N, S	4
In-Plant Systems		
P1. Startup HPCI in the ASD Room New K/A 295016 AA1.07 (4.0) RO, SROI, SROU	A, N, E, L, R	2
P2. Transfer CREFs to its alternate power supply New K/A 288000 A2.07 (3.1) RO, SROI, SROU	N	9
P3. Startup RPS MG Set A 2020-09 NRC K/A 212000 A2.01 (3.9) RO, SROI	A, D	7

1. Determine the number of control room system and in-plant system job performance measures (JPMs) to develop using the following table:

License Level	Control Room	In-Plant	Total
Reactor Operator (RO)	8	3	11
Senior Reactor Operator-Instant (SRO-I)	7	3	10
Senior Reactor Operator-Upgrade (SRO-U)	2 or 3	3 or 2	5

2. Select safety functions and systems for each JPM as follows:

   Refer to Section 1.9 of the applicable knowledge and abilities (K/A) catalog for the plant systems organized by safety function. For pressurized-water reactor operating tests, the primary and secondary systems listed under Safety Function 4, "Heat Removal from Reactor Core," in Section 1.9 of the applicable K/A catalog, may be treated as separate safety functions (i.e., two systems, one primary and one secondary, may be selected from Safety Function 4). From the safety function groupings identified in the K/A catalog, select the appropriate number of plant systems by safety functions to be evaluated based on the applicant's license level (see the table in step 1).

   For RO/SRO-I applicants: Each of the control room system JPMs and, separately, each of the in-plant system JPMs must evaluate a different safety function, and the same system or evolution cannot be used to evaluate more than one safety function in each location. One of the control room system JPMs must be an engineered safety feature.

   For SRO-U applicants: Evaluate SRO-U applicants on five different safety functions. One of the control room system JPMs must be an engineered safety feature, and the same system or evolution cannot be used to evaluate more than one safety function.

3. Select a task for each JPM that supports, either directly or indirectly and in a meaningful way, the successful fulfillment of the associated safety function. Select the task from the applicable K/A catalog (K/As for plant systems or emergency and abnormal plant evolutions) or the facility licensee's site-specific task list. If this task has an associated K/A, the K/A should have an importance rating of at least 2.5 in the RO column. K/As that have importance ratings of less than 2.5 may be used if justified based on plant priorities; inform the NRC chief examiner if selecting K/As with an importance rating less than 2.5. The selected tasks must be different from the events and evolutions conducted during the simulator operating test and tasks tested on the written examination. A task that is similar to a simulator scenario event may be acceptable if the actions required to complete the task are significantly different from those required in response to the scenario event.

   Apply the following specific task selection criteria:

- At least one of the tasks shall be related to a shutdown or low-power condition.

- Four to six of the tasks for RO and SRO-I applicants shall require execution of alternative paths within the facility licensee's operating procedures. Two to three of the tasks for SRO-U applicants shall require execution of alternative paths within the facility licensee's operating procedures.

- At least one alternate path JPM must be new or modified from the bank.

- At least one of the tasks conducted in the plant shall evaluate the applicant's ability to implement actions required during an emergency or abnormal condition.

- At least one of the tasks conducted in the plant shall require the applicant to enter the radiologically controlled area. This provides an excellent opportunity for the applicant to discuss or demonstrate radiation control administrative subjects.

If it is not possible to develop or locate a suitable task for a selected system, return to step 2 and select a different system.

4. For each JPM, specify the codes for type, source, and location:

Code	**License Level Criteria**		
	RO	**SRO-I**	**SRO-U**
(A)lternate path	4–6 [~~5~~6]	4–6 [~~5~~6]	2–3 [3]
(C)ontrol room			
(D)irect from bank	≤ 9 [5]	≤ 8 [4]	≤ 4 [1]
(E)mergency or abnormal in-plant	≥ 1 [1]	≥ 1 [1]	≥ 1 [1]
(EN)gineered safety feature (for control room system)	≥ 1 [~~2~~1]	≥ 1 [~~2~~1]	≥ 1 [1]
(L)ow power/shutdown	≥ 1 [4]	≥ 1 [4]	≥ 1 [2]
(N)ew or (M)odified from bank (must apply to at least one alternate path JPM)	≥ 2 [6]	≥ 2 [6]	≥ 1 [4]
(P)revious two exams (randomly selected)	≤ 3 [0]	≤ 3 [0]	≤ 2 [0]
(R)adiologically controlled area	≥ 1 [1]	≥ 1 [1]	≥ 1 [1]
(S)imulator			

(S1) **Shifting REC pumps – REC pipe break occurs**
Shift REC pumps using Procedure 2.2.65.1. After the second pump is started a large enough leak initiated to enter Emergency Procedure 5.2REC, Loss of REC, and perform IOAs and scram the plant.

Alternate path.

New

(S2) **Transfer RPS PP1B from RPS MG set B to CDP-1A**
Transfer RPSPP1B from RPS MG Set B to CDP-1A in accordance with Procedure 2.2.22, Vital Instrument Power System.

2014 NRC

(S3) **Startup of "A" Reactor Recirculation Pump from Single Loop Operations**
Start the "A" Reactor Recirculation Pump with the "B" Reactor Recirculation Pump running and the reactor is in Mode 1 (Single Loop Ops) IAW [Reactor Recirculation System Operations] Procedure 2.2.68.1 starting at step 6.14 with Attachment 1 already completed and signed and ending at step 6.19.7.2. When the operator is changing position of the Jog bypass switch it will cause an overload alarm to come in requiring tripping the RR pump.

Alternate path.

New

(S4) **Alternate pressure control with reactor feed pumps**
Perform Alternate Pressure Control using Reactor Feed Pump IAW Procedure 5.8.1, RPV Pressure Control Systems, to control RPV pressure.

2017 NRC

(S5) **Vent Primary Containment per 2.4PC**
Due to a primary containment pressure rise, 2.4PC is entered and venting of the primary containment is required. IAW Procedure 2.2.60 [Primary Containment Cooling and Nitrogen Inerting System] "Hard Card" the primary containment is vented to control the pressure rise.

2011 NRC

(S6) **Establish cooldown using MSL drains**
Following a reactor scram and MSIV closure, perform alternate pressure control IAW Procedure 5.8.1, RPV Pressure Control Systems, using main steam line drains and establish a cooldown of < 100°F/hr.

2012 NRC

(S7) **Place AC Exciter in Service – Alternate Path**
This JPM evaluates the Examinee's ability to place the AC Exciter in-service during startup. Examinee closed the GEN EXCITER FIELD BKR, raised Main Generator

Voltage to ~22 kV using the GEN BASE ADJUST switch, performed a Ground Test on the AC Exciter System, opened the GEN EXCITER FIELD BKR after failing Ground Test.

Alternate Path

New

(S8) **<u>Manually initiate HPCI for injection – Suction swap due to high Torus level.</u>**
The plant is shut down due to a scram from MSIV closure that requires HPCI placed in pressure control/injection. When HPCI is being lined up for pressure control the suction valves will swap causing the test valves to go closed. This requires PTMs to installed and starting over to place HPCI in-service.

Alternate path.

New

In Plant Systems JPMs

(P1) **<u>Control RPV Water Level from ASD Panel</u>**
Place HPCI in-service per procedure 5.1ASD, Alternate Shutdown, Attachment 2. HPCI will be running when the applicant gets there and when placing switches to isolate it will cause the controller to fail in auto requiring the applicant to take manual control of the HPCI controller.

Alternate Path

New

(P2) **<u>Transfer normal power to alternate power for the Emergency Booster Fan, BF-C-1A/Exhaust Booster Fan, BF-C-1B.</u>**
With a rad release in progress and a loss of normal power to the Emergency Booster Fan, BF-C-1A/Exhaust Booster Fan, BF-C-1B transfer to alternate power IAW Procedure 2.2.84 [HVAC Main Control Room and Cable Spreading Room] Section 12.

New

(P3) **<u>Startup RPS MG Set B</u>**
Start then shut down RPS MG Set B after determining voltage cannot be maintained within the required operating band.

Alternate Path

2020-09 NRC

NUREG-1021, Form 3.3-1

A. Scenario Outline

Facility: Cooper Nuclear Station — Scenario No.: 1

Scenario Source: IC 20 — Operating-Test No.: CNS-2024-06

Examiners: ______________ Applicants/Operators: ______________

______________ ______________

______________ ______________

Turnover:

The plant is at 94% power at the end of life (EOL).

1. Raise reactor power to 95% following power reduction per a Reactor Engineering request using Reactor Recirculation IAW Procedure 2.1.10.
2. Shift Turbine High Pressure Fluid (DEH) pumps IAW Procedure 2.2.80, Section 4.

Critical Tasks:

- **CT#1 –** Commence Emergency Depressurization (ED) prior to RPV level sustained below -183 inches Corrected Fuel Zone (CFZ) during a Loss of Coolant Accident (LOCA).
- **CT#2 –** Manually close all Main Steam Isolation Valves (MSIVs) prior to exceeding 15 minutes with RPV level below -113 inches Wide Range (WR).
- **CT#3 –** Manually restore RPV injection using Core Spray (CS) Loop B discharge valve, CS-MO-12B, prior to exceeding 15 minutes with RPV level below -183 inches CFZ.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N (BOP, CRS)	**DEH Pump Shift** • BOP shifts DEH pumps per 2.2.80
2 (Auto Trigger)	Annunciator B-1/A-6 TGF-PI-85 lowering	C (BOP, CRS) MC (BOP)	**DEH Pump Trip and Standby DEH Pump Auto-Start Failure** • BOP starts standby DEH pump per B-1/A-6
3	N/A	R (ATC, CRS)	**Reactor Power Ascension** • ATC raises reactor power per 2.1.10 using RR to ~95% power

4 (Trigger)	HV02A to 10 RR14A/C to 52000	I (ATC, CRS) TS (CRS)	**Reactor Recirculation NMF-FM-81A Upscale Failure** • CRS determines LCO 3.3.1.1 and TLCO 3.3.1 are not met and enters LCO 3.3.1.1, Conditions A and C for APRMs A, C, and E, and TLCO 3.3.1, Condition A ATC inserts a half scram to comply with LCO 3.3.1.1, Condition C
5 (Trigger)	HV02A to 10 RM02D to 0	TS (CRS)	**Main Steam Line 'D' Radiation Monitor Downscale Failure Due to Earthquake** • CRS determines LCO 3.3.6.1 is not met and enters Condition A (24 hours)
6 (Trigger)	HV02A to 10 ZAICRDFC301(2) to 1 Annunciator 9-5-2/F-6 to ON (momentarily)	C (ATC, CRS) MC (ATC)	**CRD Flow Controller Balance Setpoint Upscale Failure Due to Earthquake** • ATC takes manual control of CRD flow control valve
7 (Trigger)	HV02A to 10 SW07A SW07B SW07C	C (ATC, BOP, CRS)	**TEC Pump Trip (Total Loss) Due to Aftershock** • CRS directs reactor scram and rapid pressure reduction per 2.4TEC • CRS directs transfer of RPV level control to HPCI/RCIC per 2.4TEC • CRS directs securing all Feedwater injection per 2.4TEC • ATC scrams the reactor per 2.4TEC • BOP trips the Main Turbine per 2.4TEC • BOP lowers RPV pressure to 500 – 600 psig per 2.4TEC
8	HP12	C (ATC, CRS)	**High Pressure Coolant Injection (HPCI) Auxiliary Oil Pump (AOP) Leak** • ATC places HPCI AOP in Pull-To-Lock (PTL)
9	RP04	C (BOP, CRS) MC (BOP)	**Group 1 Isolation Failure** • CRS directs insertion of manual Group 1 isolation • BOP inserts manual Group 1 isolation **(CT#1)**
10 (Auto Trigger)	RR20A to 20	M (ATC, BOP, CRS)	**Loss of Coolant Accident (LOCA)** • Crew performs Emergency Depressurization when it is determined that RPV level cannot be restored and maintained greater than -183 inches CFZ **(CT#2)**
11 (Auto Trigger)	HV02A to 60 RH17A to DE-	C (ATC, CRS)	**Power Loss to Residual Heat Removal (RHR) Inboard Injection Valves (RHR-MO-25A/B), Trip of CS Pump A,**

	ENER RH18A to DE-ENER CS01A CS02B	MC (ATC)	**Failure of CS Loop B Injection Valve (CS-MO-12B) to Auto-Open** • CRS directs lineup of all available injection systems before, during, and following Emergency Depressurization • ATC opens CS-MO-12B **(CT#3)**
*** (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control**			

B. NUREG-1021, Rev 12, Criteria for Evaluation Scenarios, Table 3.4-1 and Table 3.4-2

Quantitative Attributes Table			
Attribute	**ES-3.3-1 Target**	**Actual**	**Description**
Events after EOP entry	1-2	2	1. Failure of CS-MO-12B to auto open 2. HPCI oil leak
Abnormal Events	2-4	4	1. Sheared DEH pump (B-1/A-6 TURB EH FLUID LOW PRESSURE) 2. Loss of TEC (2.4TEC) 3. CRD Flow Controller Fail high in Auto (9-5-2/F-6 CRD DISCHARGE FILTER HIGH D/P) 4. Earthquake (5.1QUAKE)
Major Transients	1-2	1	1. LOCA
EOP entries requiring substantive action	1-2	3	1. EOP-2A 2. EOP-3A 3. EOP-5A
EOP contingencies requiring substantive action	≥1 per set	1	1. EOP-2A Contingency #2 – Emergency RPV Depressurization

Pre-identified Critical Tasks	≥2	3	**<u>CT#1</u>** **Given a condition where a LOCA has occurred, and a failed Group 1 isolation signal is present, the crew will take action to insert a manual Group 1 isolation prior to exceeding 15 minutes from the time that RPV has lowered to -113 inches (WR) level.** **<u>CT#2</u>** **Given a condition where RPV level is lowering to -158" CFZ (TAF) and cannot be maintained above -183" CFZ (MSCWL) and it is apparent to the crew that insufficient high pressure injection systems will be available to restore level, the crew will Emergency Depressurize by opening 4 SRVs before RPV level lowers below -183" CFZ. (Momentary shrink below -183" CFZ due to automatic SRV operation in Low-Low Set mode does not constitute failure of this CT.)** **<u>CT#3</u>** **Given a condition where a LOCA is in progress, CS Pump A trips, RHR-MO-25A and RHR-MO-25B lose power, an ED is in progress, and CS-MO-12B fails to automatically open, the crew will manually open CS-MO-12B to restore RPV level within 15 minutes from RPV level lowering below -183 inches (CFZ).**
Normal Events	N/A	1	1. Shift DEH pumps. (BOP)
Reactivity Manipulations	N/A	1	1. Raise RX power with RR to 100%. (ATC)
Manual Control of Automatic Function	≥1	4	1. Standby DEH pump failed to auto start (BOP) 2. CRD Flow Controller Fail high in Auto (ATC) 3. Failure of CS-MO-12B (ATC) 4. Failure of Group 1 Isolation (BOP)

Instrument/ Component Failures	N/A	7	1. Sheared DEH pump (B-1/A-6 TURB EH FLUID LOW PRESSURE) 2. Loss of TEC (2.4TEC) 3. CRD Flow Controller Fail high in Auto (9-5-2/F-6 CRD DISCHARGE FILTER HIGH D/P) 4. Failure of CS-MO-12B to auto open 5. HPCI oil leak 6. Failure of Group 1 Isolation 7. RRFT-110A and C fail high
Total Malfunctions	N/A	11	1. Sheared DEH pump (B-1/A-6 TURB EH FLUID LOW PRESSURE) 2. Loss of TEC (2.4TEC) 3. CRD Flow Controller Fail high in Auto (9-5-2/F-6 CRD DISCHARGE FILTER HIGH D/P) 4. Failure of CS-MO-12B to auto open 5. HPCI oil leak 6. Failure of Group 1 Isolation 7. MSL D Rad Monitor failed upscale 8. NMF-FM-81A fail upscale 9. LOCA 10. CS-MO-12A fail to open 11. RHR-MO-25A/B loss of power
TS Evaluation	2	2	1. LCO 3.3.1.1 Condition A and C, TLCO 3.3.1 Condition A 2. LCO 3.3.6.1 Condition A

C. NUREG-1021, Rev. 12, Criteria for Evaluation Scenarios, Table 3.4-1 and Table 3.4-2

Critical Task # 1	Given a condition where a LOCA has occurred, and a failed Group 1 isolation signal is present, the crew will take action to insert a manual Group 1 isolation prior to exceeding 15 minutes from the time that RPV has lowered to -113 inches (WR) level.
Safety Significance	Failure to recognize the failure of the isolation of the MSL at the isolation setpoint of – 113 inches could cause an increase of inventory loss leading to lower water level and failure to take manual action per the PSTGs Step RC/L-1 and Conduct of Ops will result in ability of the plant to provide adequate core cooling, otherwise resulting in core damage and a large offsite release which would cause an elevated EAL
Cues	ADS Timers Actuated alarm 9-3-1/A-1. Wide Range and Fuel Zone/CFZ RPV level indications approaching or exceeding Level 1 (-113”).
Measurable Performance Indicators	Closure of the MSLs and MSL Drains
Performance Feedback	On the MSIVs and the MSL Drains (Green light ON-Red light OFF)
Applicability	LOCA in which level cannot maintain Adequate Core Cooling with a Failed Group 1 signal present
Justification for the chosen performance limit	Per PSTGs Step RC/L-1 Initiate each of the following which should have initiated but did not: • Isolations 15 minutes was chosen as time limit for this CT as an alternative boundary condition (ES-3.3, C.2) due to being a reasonable amount of time, as agreed upon by the NRC chief examiner and the facility licensee.
BWR Owners Group Appendix	App. B, Step RC/L-1
Scenario Guide Requirements	The scenario must be designed that RPV level will not be maintained above – 113 inches (WR)

Critical Task # 2	Given a condition where RPV level is lowering to -158" CFZ (TAF) and cannot be maintained above -183" CFZ (MSCWL) and it is apparent to the crew that insufficient high pressure injection systems will be available to restore level, the crew will Emergency Depressurize by opening 4 SRVs before RPV level lowers below -183" CFZ. (Momentary shrink below -183" CFZ due to automatic SRV operation in Low-Low Set mode does not constitute failure of this CT.)
Safety Significance	The MSCWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. When water level decreases below MSCWL with injection, clad temperatures may exceed 1500°F.
Cues	Corrected Fuel Zone indication (SPDS) falls to -158" and lowering trend continues, and, before -158" CFZ is reached, initial conditions, field reports, and control room indications convey that adequate high-pressure injection cannot be restored before level falls below -183" CFZ.
Measurable Performance Indicators	Manipulation of any four SRV controls on panel 9-3: SRV-71A, SRV-71B, SRV-71E, SRV-71G, SRV-71H, SRV-71C, SRV-71D, SRV-71F
Performance Feedback	Crew will observe SRV light indication go from green to red, amber pressure switch lights illuminate, reactor pressure lowering on SPDS and panel 9-3 and 9-5 meters and recorders, and SRV tailpipe temperatures rise on recorder MS-TR-166.
Applicability	EOP-1A conditions with RPV pressure above the shutoff head of available low pressure injection systems or subsystems and any system injecting to the RPV (i.e. not in steam cooling).
Justification for the chosen performance limit	The MSCWL (-183" CFZ) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Emergency depressurization is allowed when level cannot be restored and maintained above TAF (-158" CFZ) and should be performed, if in the judgment of the CRS, level cannot be maintained above -183" CFZ. Since it is intended for the scenario supporting this CT to, early in the event, clearly indicate no high pressure injection systems can be made available to reverse the lowering level trend, the crew will have time to communicate and open 4 SRVs before -183" CFZ.
BWR Owners Group Appendix	App. B, Contingency#1
Scenario Guide Requirements	LOCA severity should result in a near linear RPV level reduction that causes level to fall to TAF over approximately 20 minutes from the time the initial LOCA signal is received. It is very important to design the scenario such that the crew has information early during the LOCA event to determine high pressure injection systems cannot be recovered or optimized in order to stabilize level before -183" CFZ is

	reached. The crew should know this within approximately 10 minutes from the start of the LOCA and by the time level lowers to -100” CFZ to allow time to align/realign low pressure systems for injection before level reaches -158” CFZ, so that the only remaining action when TAF is reached will be to conduct emergency depressurization. (e.g. As an initial condition, HPCI turbine is disassembled for maintenance. A field report for a RCIC valve malfunction states a valve has mechanical binding in the gearbox, cannot be manually opened and will take 4 hours to repair.

Critical Task # 3	Given a condition where a LOCA is in progress, CS Pump A trips, RHR-MO-25A and RHR-MO-25B lose power, an ED is in progress, and CS-MO-12B fails to automatically open, the crew will manually open CS-MO-12B to restore RPV level within 15 minutes from RPV level lowering below -183 inches (CFZ).
Safety Significance	Failure to recognize the only available low pressure injection system did not auto align to commence injection when ED lowers pressure to allow injection, and failure to take manual action per Conduct of Ops will result in unavailability of safety-related equipment necessary to provide adequate core cooling, otherwise resulting in core damage and a large offsite release.
Cues	Indication CS-MO-12B failed to open with the following conditions present: • Green light on and Red lamp extinguished at respective pump hand switch on panel 9-3 • Indication of Drywell Pressure ≥ 1.83 psig • Indication of RPV water level ≤ -113" • RX pressure 291 to 436 psig
Measurable Performance Indicators	Manipulation of controls as required to open CS-MO-12B from Panel 9-3: Operator places CS-MO-12B control switch to OPEN on Panel 9-3
Performance Feedback	Crew will observe Red light illuminate and Green light extinguish for CS-MO-12B on Panel 9-3
Applicability	LOCAs with high pressure makeup capability insufficient to maintain RPV level. This is only applicable if manual action from the Control Room would be effective in aligning the affected LP ECCS pump(s) to provide injection.
Justification for the chosen performance limit	Attempting to align the ECCS systems must be performed to within 15 minutes from the time a loss of adequate core cooling had been met. This would be within the time period to prevent escalating to a SAE for a potential loss of the Fuel in the Fission Product Barrier. From the time ED is complete if the crew properly implements EOP-1A decision steps regarding restoring and maintaining RPV level they should have to perform this task.
BWR Owners Group Appendix	App. B, Contingency#1
Scenario Guide Requirements	LOCA severity should result in a near linear RPV level reduction that causes level to fall to TAF over approximately 15 minutes from the time the initial LOCA signal is received. (The LOCA malfunction severity may be ramped initially, but it should reach its final severity within approximately the first 3 minutes.)

Scenario Summary

The plant is operating at 94% power at End-of-Life (EOL) in the operating cycle when the crew takes the watch. Reactor power was lowered for a control rod exchange. The crew is directed to raise power to 95% at 1% per hour.

Scenario Event 1 begins when the CRS orders the BOP to start DEH pump B and secure DEH pump A **(BOP ACTION)** **(NORMAL)**

Scenario Event 2 begins 5 seconds following the stopping of DEH pump A. DEH header pressure will lower and the DEH low pressure alarm will actuate. DEH pump A will *not* automatically start. The crew will dispatch a non-licensed operator (NLO) to investigate DEH pressure at the DEH pumps. The report will be that DEH pump B is running but it has no discharge pressure. The CRS will direct starting DEH pump A. The BOP will start DEH pump A and secure DEH pump B. **(BOP ACTION)** **(MC)**

Scenario Event 3 begins when crew commences raising reactor power with Reactor Recirculation (RR) to 95% **(ATC ACTION)** **(RX MANIPULATION)** per 2.1.10. The current 'S' value on both RRMG controllers is 87%. When the crew raises each RRMG controller to 89%, reactor power will be ~95%. This event ends when reactor power has been raised to ~95%.

3.1 RR Loop A flow instruments, RR-FT-110A and RR-FT-110C, fail upscale causing NMF-FM-81A to fail upscale. The crew will recognize the flow comparator mismatch and that the APRMs on the "A" side are now non-conservative with respect to APRM flow bias scram setpoints. Technical Specification LCO 3.3.1.1 2(b) and TLCO 3.3.1 Functions (4) (5) are no longer met. Conditions A and C for LCO 3.3.1.1 2(b) will be entered for this event with a required action to restore RPS trip capability. This is satisfied by inserting a manual half scram on RPS Channel A in accordance with Procedure 4.5, Reactor Protection/Alternate Rod Insertion System, section 4. **(CRS TS ACTION)** The CRS will direct the ATC to insert a half scram to comply with LCO 3.3.1.1, Condition C. The ATC will insert a half scram on RPS Channel A. **(ATC ACTION)** Condition A for TLCO 3.3.1 (4) (5) will be entered requiring a rod block inserted within 1 hour. **(CRS TS ACTION)**

Scenario Event 4 begins when an earthquake occurs which is identified by ground shaking, noise, and annunciator B-3/B-1, SEISMIC EVENT, alarming. The crew responds per 5.1QUAKE and addresses the following conditions:

4.1 MSL D Radiation Monitor fails downscale. This is indicated by Annunciator 9-4-1/B-4 and VID alarm point 1757, MAIN STM LINE CHAN D. The CRS will determine that LCO 3.3.6.1 is not met and will enter Condition A. **(CRS TS ACTION)**

4.2 CRD Flow Controller setpoint fails upscale. The increased flow will cause a CRD Discharge Filter High D/P alarm to come in and reset. The crew will identify the failed CRD flow controller and take manual control of the CRD flow Controller. This action will restore CRD flow to pre-event values. **(ATC ACTION)** **(MC)**

Scenario Event 5 begins when an earthquake aftershock occurs causing all TEC pumps to trip. Manual attempts to start any TEC pump will fail. TEC pressure will remain less than 55 psig and unable to be restored. Per Abnormal Procedure 2.4TEC, TEC Abnormal, the crew will scram the reactor **(ATC ACTION)**, trip the main turbine **(BOP ACTION)**, and rapidly reduce reactor pressure to 500 to 600 psig with BPVs. While taking action per 2.4TEC and EOP-5A, the crew will identify the following conditions:

5.1 While transferring level and pressure control to HPCI/RCIC/SRVs, when the crew attempts to place HPCI in-service, HPCI will not start due to the Aux Oil Pump failure. When sent to investigate, the NLO will report that oil is all over the floor in the HPCI room. HPCI AOP will be placed in PTL. **(ATC ACTION)**

Scenario Event 6 begins when the crew has taken action to trip the RFPs, CBPs, and Condensate pumps per 2.4TEC. A reactor coolant leak will initiate inside the drywell from the RR discharge line. **(MAJOR)**. CS pump A will trip following automatic ECCS initiation and cannot be restarted. The crew will be required to enter EOP-1A and EOP-3A due to lowering RPV level and high drywell pressure. The crew will take action to maximize CRD and inject SLC for level control. When RPV level lowers to -113 inches (WR), the crew will identify that Group 1 Isolation failed and take action to insert a Group 1 Isolation manually. **(BOP MC)** **(CT#1)** Once it has been determined that RPV level cannot be restored and maintained above -158 inches (CFZ) and prior to -183 inches (CFZ), the crew will enter EOP-2A **(CT#2)** and perform an Emergency Depressurization by opening 6 SRVs (minimum required 4 SRVs). When the fourth SRV is opened an aftershock will occur causing power to be lost to both RHR-MO-25A and RHR-MO-25B.

6.1 During the ED, when RPV pressure has lowered to between 291-436 psig, the crew will assess all low pressure ECCS failures and that the only available injection path is with CS Loop B. CS-MO-12B will fail to auto open and must be manually opened to restore RPV water level. **(CT#3)** **(ATC ACTION)** **(MC)**

The scenario can be terminated when ED is complete and RPV water level is being restored to post ED water level band.

NUREG-1021, Form 3.3-1

A. Scenario Outline

Facility: Cooper Nuclear Station Scenario No.: 2

Scenario Source: IC 278 Operating-Test No.: CNS-2024-06

Examiners: ____________ Applicants/Operators: ____________

Turnover:

The plant is at 60% power. Reactor Feed Pump (RFP) 'A' is in the process of being started up IAW Procedure 2.2.28.1, Feedwater System Operation, Section 5. The procedure is complete through Step 5.6.

1. Continue the RFP 'A' startup IAW Procedure 2.2.28.1, Feedwater System Operation, Section 5, starting at Step 5.7.

Critical Tasks:

- **CT#1 –** Isolate RCIC steam line prior to exceeding Max Safe Temperature in two areas of Secondary Containment
- **CT#2 –** Inhibit ADS prior to Tech Spec cooldown rate violation during an ATWS.
- **CT#3 –** Initiate SLC injection or insert all control rods to 02 prior to exceeding BIIT during an ATWS.
- **CT#4 –** Stop and prevent injection to lower RPV level and suppress neutronic oscillations before exceeding 25% peak to peak on any APRM.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N (BOP, CRS)	**RFP 'A' Startup** • Place "A" RFP in service feeding the vessel IAW 2.2.28.1, Feedwater System Operation
2 (Trigger)	RR17B from 68 to 75	C (ATC, BOP, CRS) MC (ATC) TS (CRS)	**RR Pump 'B' Uncontrolled Speed Rise** • Single recirc pump runaway on the "B" Recirc pump requiring entry into 2.4RR (Scoop tube lock out). BOP transfers power supplies for Recirc pumps. CRS will enter LCO 3.4.1, Condition B.

3	N/A	R (ATC, CRS)	**Restore RR Loop Flow Mismatch** • Power ascension to restore Recirculation loop flow mismatch to within LCO 3.4.1 limits will commence using Reactor Recirculation System IAW 2.1.10, Station Power Changes.
4 (Trigger)	RC05	I (BOP, CRS) MC (BOP) TS (CRS)	**RCIC Inadvertent Initiation** • RCIC initiates. The crew will enter 2.4CSCS. The BOP will secure RCIC. The CRS will enter LCO 3.5.3, Condition A.
5 (Trigger)	RC06 to 7, RC07	C (BOP, CRS) MC (BOP)	**RCIC Steam Leak, Failure of Group 5 Auto-Isolation** • RCIC steam leak. The BOP will manually isolate RCIC steam line prior to 2 areas above max safe. **(CT#1)**
6 (Trigger)	RD18 from 0 to 100	C (ATC, CRS) MC (ATC)	**Scram Air Header Rupture & Failure of RPS Automatic Scram** • The scram air header will experience a rupture. Control Rod HCU scram valves will open, requiring a reactor scram. • Upon any RPS automatic scram signal as a result of plant transient when multiple rods drift, RPS fails to initiate scram and ATC will manually scram the reactor.
7	RD02A/B, RP01A/B/C/D, RD27, RD26, TC07A/B/C, CR05, CR04B	M (ATC, BOP, CRS)	**ATWS, Failure of ATWS-RPT Trip of RR Pumps** • Scram results in a hydraulic ATWS >3%. Bypass valves will fail closed and the Main Turbine will inadvertently trip, resulting in heat addition to Torus through SRVs. • Reactor power oscillations will occur **(CT#2, CT#3, and CT#4)**
8	RR24A/B	C (ATC, CRS) MC (ATC)	**Reactor Recirculation ATWS-RPT Trip Failure** • Both RR pumps will fail to automatically trip on ATWS-RPT (< 1072 psig or > -42") and the ATC will manually trip these pumps during the ATWS
*** (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control**			

B. NUREG-1021, Rev 12, Criteria for Evaluation Scenarios, Table 3.4-1 and Table 3.4-2

Quantitative Attributes Table			
Attribute	**ES-3.3-1 Target**	**Actual**	**Description**
Events after EOP entry	1-2	3	1. Scram Air Header Rupture 2. Failure of Automatic RPS Scram 3. Failure of RR ATWS-RPT Automatic Trip
Abnormal Events	2-4	4	1. Reactor Recirc. Uncontrolled Speed Rise (2.4RR) 2. RCIC Inadvertent Initiation (2.4CSCS) 3. RCIC Steam Line Break (9-3-1/E-10, AREA HIGH TEMP) 4. Scram Air Header Rupture (9-5-2/F-5, SCRAM VALVE PILOT AIR LOW PRESSURE)
Major Transients	1-2	1	1. ATWS
EOP entries requiring substantive action	1-2	3	1. EOP-5A 2. EOP-6A 3. EOP-7A
EOP contingencies requiring substantive action	≥1 per set	1	1. EOP-6A/7A Contingency #5 – ATWS RPV Control

Pre-identified Critical Tasks	≥2	4	**<u>CT#1</u>** **Given a condition where a primary system is discharging into the Secondary Containment through a steam line break and the automatic isolation fails, the crew takes manual action to isolate the steam leak prior to exceeding Maximum Safe Operating (MSO) Temperature in two areas (For this scenario, Torus 890' ENE is the first area to exceed MSO after ~9 minutes and Torus 896' W is the second area to exceed MSO after ~15 minutes).** **<u>CT#2</u>** **Given a condition where an ATWS has occurred, the crew inhibits ADS prior to uncontrolled injection from high-volume, low-pressure systems and before exceeding the Tech Spec cool down rate limit during a failure to scram.** **<u>CT#3</u>** **Given a condition where control rods fail to scram and energy is discharging to the primary containment (e.g., SRVs, LOCA), the crew initiates SLC injection or inserts all control rods to at least position 02 before exceeding the Boron Injection Initiation Temperature (BIIT) curve (For this scenario, with no operator action post scram, BIIT is exceeded ~15 minutes after initial scram).** **<u>CT#4</u>** **Given a condition where an ATWS has occurred and reactor power is above 3%, the crew stops and prevents injection from all sources (except boron, CRD, RCIC) as necessary to lower RPV level to below -60" (or LL, as applicable), prior to neutronic oscillations exceeding 25% peak-to-peak indicated on any APRM (For this scenario, this will occur 15 minutes after the initial scram).**
Normal Events	N/A	1	1. Startup RFP 'A' (BOP)
Reactivity Manipulations	N/A	1	1. Restore RR Loop Flow Mismatch within LCO 3.4.1 (ATC)
Manual Control of Automatic Function	≥1	5	1. Reactor Recirc Controller Failure (ATC) 2. RCIC Inadvertent Initiation (BOP) 3. RCIC Group 5 Auto Failure (BOP) 4. Failure of Auto RPS Scram (ATC) 5. Failure of RR ATWS-RPT Auto Trip (ATC)

Instrument/ Component Failures	N/A	5	1. Reactor Recirc Controller Failure (ATC/BOP) 2. RCIC Inadvertent Initiation (BOP) 3. RCIC Group 5 Auto Failure (BOP) 4. Failure of Auto RPS Scram (ATC) 5. Failure of RR ATWS-RPT Auto Trip (ATC)
Total Malfunctions	N/A	7	1. Reactor Recirc Controller Failure 2. RCIC Inadvertent Initiation 3. RCIC Group 5 Auto Failure 4. Scram Air Header Rupture 5. Failure of Auto RPS Scram 6. Failure of RR ATWS-RPT Auto Trip 7. ATWS
TS Evaluation	2	2	1. LCO 3.4.1, Condition B 2. LCO 3.5.3, Condition A

C. NUREG-1021, Rev. 12, Criteria for Evaluation Scenarios, Table 3.4-1 and Table 3.4-2

Critical Task # 1	Given a condition where a primary system is discharging into the Secondary Containment through a steam line break and the automatic isolation fails, the crew takes manual action to isolate the steam leak prior to exceeding Maximum Safe Operating (MSO) Temperature in two areas (For this scenario, Torus 890' ENE is the first area to exceed MSO after ~9 minutes and Torus 896' W is the second area to exceed MSO after ~15 minutes).
Safety Significance	EOP-5A directs isolating primary system leaks into secondary containment when a maximum normal operating value is exceeded. Failing to do so can result in an unnecessary offsite release and endanger plant personnel. Isolating the leak terminates the RCS discharge into secondary containment.
Cues	Indication of rising or Maximum Operating values in an area of a system which is connected to the RCS, combined with abnormal system parameters (e.g. such as levels, pressures, and flow rates). Field reports of visible/audible leaks into secondary containment.
Measurable Performance Indicators	Crew places the control switch for the applicable isolation valve(s) to CLOSE.
Performance Feedback	Indication for applicable isolation valve(s) Green light illuminates and Red light extinguishes. Secondary Containment parameter(s) eventually stabilizes and lowers. RPV and/or associated system parameters indicate leak has been isolated.
Applicability	EOP-5A conditions where a system (primary or non-primary) is discharging into the secondary containment and manual isolation capability from the control room is possible. This includes manipulation of valve control switches and valve power supply control switches, as applicable. *<u>If the leaking system is required for adequate core cooling this task is not applicable.</u>*
Justification for the chosen performance limit	Per PSTGs Step SC/T-3 When an area temperature exceeds its maximum normal operating temperature (Table SC-1), isolate all systems that are discharging into the area except systems required for damage control and systems required to be operated by the Emergency Procedure Guidelines for Hot Conditions.
BWR Owners Group Appendix	App. B, Step SC/T-3
Scenario Guide Requirements	The scenario must be able to drive at least one secondary containment parameter to its Max Safe value in two plant areas if the crew does not take action to isolate the leak. The crew scramming and reducing RPV pressure to reduce the driving head of the leak should not prevent reaching the Max Safe value for a parameter in two plant areas.

Critical Task # 2	Given a condition where an ATWS has occurred, the crew inhibits ADS prior to uncontrolled injection from high-volume, low-pressure systems and before exceeding the Tech Spec cool down rate limit during a failure to scram.
Safety Significance	Inhibiting ADS before injection from high volume, cold water systems occur ensures a related power excursion will not be experienced that could challenge to the fuel barrier. Inhibiting ADS before the Tech Spec cooldown limit is exceeded ensures the RPV fission product barrier is not challenged by a significant thermal transient.
Cues	ADS Timers Actuated alarm 9-3-1/A-1. Wide Range and Fuel Zone/CFZ RPV level indications approaching or exceeding Level 1 (-113”). ADS valve control switch red and amber indicating lights on Panel 9-3 ON.
Measurable Performance Indicators	Manipulation of ADS A and ADS B Inhibit switches on Panel 9-3 vertical section.
Performance Feedback	Inhibit switches click into the vertical, inhibit position on Panel 9-3 prior to breaking the tech spec required cooldown rate. 9-3-1/D-1 ADS INHIBITED alarm comes in
Applicability	ATWS condition that requires intentionally lowering RPV level to suppress neutronic oscillations
Justification for the chosen performance limit	Per PSTGs Step C5/L-1 Monitor and control RPV water level. Inhibit ADS
BWR Owners Group Appendix	App. B, Step C5/L-1
Scenario Guide Requirements	The scenario must be designed to include a high power ATWS (>3%) that requires lowering RPV level to between -60” and -183” (WR/CFZ)

Critical Task # 3	Given a condition where control rods fail to scram and energy is discharging to the primary containment (e.g., SRVs, LOCA), the crew initiates SLC injection or inserts all control rods to at least position 02 before exceeding the Boron Injection Initiation Temperature (BIIT) curve (For this scenario, with no operator action post scram, BIIT is exceeded ~15 minutes after initial scram).
Safety Significance	If boron injection is initiated or all control rods are inserted to position 02 before suppression pool temperature reaches the BIIT, emergency RPV depressurization may be precluded at lower reactor power levels. At higher reactor power levels, however, the suppression pool heat up rate may become so high that the Hot Shutdown Boron Weight of boron cannot be injected before suppression pool temperature reaches the Heat Capacity Temperature Limit even if boron injection is initiated early in the event. Since failure-to-scram conditions may present severe plant safety consequences, the requirement to initiate boron injection is independent of any anticipated success of control rod insertion. If the failure to scram EOP were to be exited, other procedures would not provide the guidance for control rod insertion necessary to achieve reactor shutdown. Before exiting EOP-6A ensures guidance to effect reactor shutdown is not removed.
Cues	Manual scram is initiated, numerous control rods indicate beyond position 00, and reactor power not downscale on panel 9-5 indications. Suppression Pool temperature rising on PMIS and panel indications.
Measurable Performance Indicators	Operator manipulates key locked switch for SLC pumps to START. Operator selects individual control rods by depressing the respective pushbutton on the panel 9-5 matrix and inserts the rod by manipulating the emergency in switch on panel 9-5.
Performance Feedback	The SLC pumps red lights illuminated, SLC discharge pressure rising, SLC tank level lowering by 26% on panel 9-5. Operator selecting and inserting control rods indicated by rod position decreasing to 00 for selected rod on panel 9-5.
Applicability	ATWS condition that heats the Suppression Pool due to constant energy discharge from core
Justification for the chosen performance limit	Per PSTGs Step C5/Q-7 Perform the following: • When periodic neutron flux oscillations in excess of 25% (Large Oscillation Threshold) peak-to-peak commence and continue or before average torus water temperature reaches 195°F (Boron Injection Initiation Temperature), continue at Step. C5/Q-7.1. ○ BORON INJECTION IS REQUIRED; inject boron into the RPV using SLC and inhibit ADS.
BWR Owners Group Appendix	App. B, Step C5/Q-7
Scenario Guide Requirements	The scenario is designed so reactor power remains high enough during the ATWS to cause continuous SRV lifting to the torus until operator action (i.e., SLC injection, rod insertion) taken to lower power.

Critical Task # 4	Given a condition where an ATWS has occurred and reactor power is above 3%, the crew stops and prevents injection from all sources (except boron, CRD, RCIC) as necessary to lower RPV level to below -60" (or LL, as applicable), prior to neutronic oscillations exceeding 25% peak-to-peak indicated on any APRM (For this scenario, this will occur 15 minutes after the initial scram).
Safety Significance	Applicability for this CT is during EOP-7A conditions where it is necessary to lower reactor water level to -60" WR before 25% peak-to-peak neutron flux oscillations potentially could occur with localized fuel power peaking. This value was chosen because it establishes margin to conditions where fuel damaging power oscillations may theoretically occur per the PSTGs.
Cues	Manual scram is initiated and RPS fails to de-energize and reactor power remains >3% on Panel 9-5 indications and SPDS and RPV level is > -60" WR on SPDS.
Measurable Performance Indicators	Operator manipulates Feedwater HMIs on Panel 9-5 or Panel A as necessary to stop FW injection. Operator manipulates HPCI controls on panel 9-3 to stop HPCI injection.
Performance Feedback	Feedwater flow indication on panel 9-5 indicate zero. HPCI flow indication on panel 9-3 indicates zero and/or HPCI injection MOV indicates closed.
Applicability	EOP-7A conditions where power remains above 3% following completion of migrating tasks of procedure 2.1.5 REACTOR SCRAM. If, due to scenario design and dynamics, level lowers to below -60" WR without crew action, this should not be selected as a critical task.
Justification for the chosen performance limit	Per PSTGs Step C5/L-3 If while executing the following step: • RPV water level is above -60 in. (24 in. below the feedwater sparger nozzles), Deliberately lower RPV water level by terminating and preventing all injection into the RPV except from boron injection systems, CRD and RCIC, defeating interlocks if necessary, until RPV water level drops below -60 in. (24 in. below the feedwater sparger nozzles).
BWR Owners Group Appendix	App. B, Step C5/L-3
Scenario Guide Requirements	Initial conditions, ATWS with greater than 3% following 2.1.5 mitigating actions for a reactor scram. The scenario should be designed such that level will remain above -60" WR, apart from crew action. Scenario designed to cause APRM oscillations following a time delay after initial scram attempt.

Scenario Summary

The crew will assume the watch at 60% power. Reactor Feed Pump (RFP) ‘A’ is in the process of being started up.

Scenario Event 1 begins when the crew continues RFP ‘A’ startup IAW 2.2.28.1, Section 5, Placing Second RFP In Service, starting at step 5.7. RFP ‘A’ discharge pressure is slightly below discharge header pressure. The crew will open RFP ‘A’ discharge valve, shift RFP ‘A’ (and Min Flow Valve) to AUTO **(BOP ACTION)** **(NORMAL)**.

Scenario Event 2 begins when Reactor Recirculation (RR) Pump ‘B’ experiences an electrical failure such that the scoop tube position moves uncontrollably. The crew will have indications of a step rise in recirculation flow, reactor power, and Main Generator output. The crew will lock out RR Pump ‘B’ Scoop Tube IAW 2.4RR, Reactor Recirculation Abnormal, Step 3.3.1.1 **(ATC ACTION)**. Recirculation flow will stabilize and a trip of the ‘B’ RR Pump will not be required. The CRS will evaluate TS LCO 3.4.1, Condition B **(CRS TS ACTION)**. RR flow mismatch will be outside the TS required deviation. The crew will transfer RR Pump ‘A’ to the SSST IAW alarm card 9-4-3/C-6, Step 2.3 **(BOP ACTION)**.

Scenario Event 3 begins when the Shift Manager directs the crew to match RR loop flows by raising RR Pump ‘A’ speed IAW Procedure 2.1.10 to comply with TS LCO 3.4.1 **(ATC ACTION)** **(RX MANIPULATION)**.

Scenario Event 4 begins when a failure in the RCIC circuitry causes an inadvertent initiation. The crew will respond to alarm 9-4-1/A-1. RCIC must be secured IAW 2.4CSCS (Attachment 4) or 9-4-1/A-1 **(BOP ACTION)** **(MC)**. The CRS will evaluate TS LCO 3.5.3, Condition A for RCIC being INOPERABLE and verify HPCI is OPERABLE **(CRS TS ACTION)**.

Scenario Event 5 begins when the perturbation from the RCIC initiation causes a steam leak to develop downstream of RCIC-MO-16. The crew will recognize that Secondary Containment radiation levels and temperatures are rising. An evaluation of the Control Room indications (e.g., steam supply pressure for RCIC compared to the RPV/HPCI, PMIS area temperatures/radiation values, etc.) will alert the crew to the location of the leak. The crew will recognize that the auto isolation failed and will insert a manual isolation **(CT#1)** **(BOP ACTION)** **(MC)**.

Scenario Event 6 begins when the Scram Air Header ruptures. Annunciator 9-5-2/F-5, Scram Valve Pilot Air Low Pressure, will alert the crew to the condition. The crew will monitor the full core display. When one or more scram valves fail open, as indicated by the blue lights for the respective HCU scram valves energizing, the crew will scram the reactor and enter Procedure 2.1.5, Reactor Scram. All automatic scrams will fail and a manual scram will be required for this condition **(ATC ACTION)** **(MC)**.

Scenario Event 7 begins when the reactor is scrammed. Control rods will fail to insert due to blockages in both scram discharge volumes **(MAJOR)**. The Main Turbine will trip and all three bypass valves will fail closed. SRVs will control reactor pressure. Reactor power will be approximately 17%. The crew will enter EOP-1A, RPV Control (1-3), and transition to EOP-6A, Reactor Power/Pressure (Failure-To-Scram), and EOP-7A, RPV Level (Failure-To-Scram). The crew will inject SLC and install the necessary EOP PTMs to bypass RPS interlocks and insert control rods individually via RMCS IAW Procedure 5.8.3, Attachment 1 **(CT#3)**. The crew will inhibit ADS in preparation to lower reactor water level **(CT#2)**. The crew will perform Stop and Prevent for all injection sources except CRD and SLC (RCIC is isolated) due to reactor power being above 3% **(CT#4)**. RPV level will be intentionally lowered below -60" WR to lower core inlet subcooling and lower reactor power **(CT#4)**.

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7

7.1 RR Pumps 'A' and 'B' will not trip on the ATWS-RPT trip signal (≤ 1072 psig or ≥ -42") and the crew will be required to manually trip RR Pumps **(ATC ACTION)** **(MC)**.

The scenario can be terminated when Hot Shutdown Boron Weight has been injected or all Control Rods have been inserted to at least position 02, the CRS has ordered a final RPV level band or +3" to +54", or at the discretion of the Lead Examiner.

Op-Test No.: CNS 2024-06
Scenario No.: 3

NUREG 1021 FORM 3.3-1

A. Scenario Outline

Facility: Cooper Nuclear Station Scenario No.: 3

Scenario Source: IC 16 Operating-Test No.: CNS-2024-06

Examiners: ______________________ Applicants/Operators: ______________________

______________________ ______________________

______________________ ______________________

Turnover:
The plant is at 71% power BOL.

1. Shift TEC pumps per 2.2.76 Section 6.
2. Raise Rx power to 74% using Reactor Recirculation IAW 2.1.10, to commence power increase following quarterly down power.

Critical Tasks:

- **CT#1 –** Manually close EG2 prior to RPV level falling below -158" during a loss of offsite power (LOOP) and loss of coolant accident (LOCA).
- **CT#2 –** Commence Emergency Depressurization (ED) during an unisolable HPCI steam leak prior to two areas in Secondary Containment exceeding Maximum Safe Operating (MSO) Temperature.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP, CRS)	**Shift TEC pumps per 2.2.76 Section 6.** BOP Shifts TEC pumps
2		R (ATC, CRS)	**Raise RX Power** • ATC Raises power per 2.1.10 using RR to ~74% power for power ascension following downpower for control rod adjustment.
3 Trigger	NM09C to 0	I (ATC, CRS)	**APRM C Fails Downscale** • ATC will respond to 9-5-1/C-8, APRM downscale and take alarm card actions to bypass APRM C.
4 Trigger	RP03C	C (BOP, ATC, CRS) TS (CRS)	**RPS B Loss of Power** • BOP transfers RPS B to the alternate source. • BOP will verify and reset group isolations IAW 2.1.22. • ATC resets the "B" half scram IAW 2.1.5 • CRS determines that LCO 3.4.5 is not met and enters Condition B.
5 Trigger	HP05	MC (BOP) C (BOP, CRS) TS (CRS)	**Spurious HPCI Initiation** • BOP will take immediate operator actions to secure HPCI. • CRS will determine LCO 3.5.1 is not met and enter Condition C.

6 Trigger	ED04 ZDIDGSWCSDG1 to NASP	C (ATC, BOP, CRS) MC (BOP)	**LOOP/Failure of DG1 to start/ Failure of DG2 to auto start** • BOP closes breaker EG2 when EG2 failed to AUTO close when DG2 got to rated power. • ATC takes scram actions and places the RX Mode switch to Refuel/Shutdown.
7 Trigger	HP06 HP09 ZDIHPCISWS1 to open ZDIHPCISWS2 to open ZDIHPCISWS32 to OFF	M (ATC, BOP, CRS)	**HPCI Steam Line Break/Failure to isolate HPCI line/ED** • BOP/ATC attempts to isolate HPCI by closing HPCI-MO-15 and HPCI-MO-16. Reports that all actions from the control room failed to isolate HPCI steam line. • CRS will enter EOP-2A when 2nd area reaches maximum safe operating temperature and direct emergency depressurization. **CT#1** Given a condition when high pressure injection systems cannot maintain RPV level and low pressure ECCS systems fail to automatically start due to loss of AC power, crew manually starts DG1(2) and/or closes DG-1(2) output breaker to energize LP ECCS systems prior to RPV water level falling below -158” CFZ (TAF) **CT#2** Given a condition where two areas exceed their maximum safe operating temperature, the crew will enter EOP-2A and emergency depressurize the RPV prior to a third area reaching its maximum safe operating temperature.
*** (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control**			

B. NUREG 1021 Rev 12 Criteria for Evaluation Scenarios, Table 3.4-1 and Table 3.4-2

Quantitative Attributes Table			
Attribute	**ES-3.3-1 Target**	**Actual**	**Description**
Events after EOP entry	1-2	3	1. DG1failure to AUTO start 2. EG2 failure to Auto Close 3. HPCI failure to isolate
Abnormal Events	2-4	4	1. Loss of all AC (5.3SBO/5.3EMPWR) 2. Inadvertent HPCI Initiation (2.4CSCS) 3. APRM downscale (9-5-1/C-8) 4. Loss of RPS B (2.1.22)
Major Transients	1-2	1	1. HPCI Steam Line Break
EOP entries requiring substantive action	1-2	2	1. EOP-5A 2. EOP-2A
EOP contingencies requiring substantive action	≥1 per set	1	1. EOP-2A Contingency #2 – Emergency RPV Depressurization
Pre-identified Critical Tasks	≥2	2	**CT#1** Given a condition when high pressure injection systems cannot maintain RPV level and low pressure ECCS systems fail to automatically start due to loss of AC power, crew manually closes EG2 to energize LP ECCS systems prior to RPV water level falling below -158” CFZ (TAF) **CT#2** Given a condition where two areas exceed their maximum safe operating temperature, the crew will enter EOP-2A and commence emergency depressurization of the RPV by opening at least 4 SRVs prior to a third area reaching its maximum safe operating temperature. (MSOT). (The first area to exceed MSOT is the SW Quad with TS-99G/TS 105B. The second area to exceed the MSOT is the NW Quad with TS-99C. The third area to exceed the MSOT is the SE Quad with TS-117F)
Normal Events	N/A	1	1. Shifting TEC pumps per 2.2.76 Section 6
Reactivity Manipulations	N/A	1	1. Raise RX power with RR to ~74%. (ATC)
Manual Control of Automatic Function	≥1	2	1. Inadvertent HPCI Initiation (BOP) 2. Failure of EG2 to Auto close (BOP)

Instrument/ Component Failures	N/A	4	1. APRM C Downscale (ATC) 2. RPS B power failure (ATC) (BOP) 3. HPCI spurious initiation (BOP) 4. LOOP (ATC) (BOP)
Total Malfunctions	N/A	8	1. APRM C downscale 2. RPS B power failure 3. LOOP 4. EG2 fails to Auto close 5. DG1 fails to start 6. HPCI spurious initiation 7. HPCI steam leak 8. HPCI isolation valves fail to isolate
TS Evaluation	2	2	1. LCO 3.4.5 Condition B 2. LCO 3.5.1 Condition C

C. Critical Task Tables

Critical Task #1	Given a condition when high pressure injection systems cannot maintain RPV level and low pressure ECCS systems fail to automatically start due to loss of AC power, crew manually closes EG2 to energize LP ECCS systems prior to RPV water level falling below -158” CFZ (TAF)
Safety Significance	Failure to recognize the auto start not occurring and energizing of the safety bus, and failure to take manual action per Procedure 5.3EMPWR will result in unavailability of safety-related equipment necessary to provide adequate core cooling, otherwise resulting in core damage and a large offsite release.
Cues	Indication and/or annunciation that all ac emergency buses are de-energized ·Bus energized lamps extinguished ·Circuit breaker position ·Bus voltage ·EDG status Control room lighting dimmed
Measurable Performance Indicators	Manipulation of controls as required to energize Div 2 AC emergency bus from panel C: The crew places DIESEL GENERATOR SYNC Switch to EG2 and DIESEL GENERATOR # 2 Breaker EG-2 control switch to CLOSE and energizes 4160G Bus prior to level lowering to TAF.
Performance Feedback	Crew will observe light indication for equipment powered by Division 2 AC illuminate on panel 9-3 and bus voltage ~4200V on panel C.
Applicability	Loss of off-site power events when all sources of off-site power are lost and a diesel generator fails to auto start or energize its bus. This is only applicable if manual action from the Control Room would be effective in energizing the bus.
Justification for the chosen performance limit	Attempting to start ECCS systems must be performed to determine their availability by the time TAF is reached in order to properly implement EOP-1A decision steps regarding restoring and maintaining RPV level.
BWR Owners Group Appendix	App. B, Contingency #1
Scenario Guide Requirements	LOCA severity should result in a near linear RPV level reduction that causes level to fall to TAF over approximately 15 - 20 minutes from the time the initial LOCA signal is received. (The LOCA malfunction severity may be ramped initially, but it should reach its final severity within approximately the first 3 minutes.)

Critical Task # 2	Given a condition where two areas exceed their maximum safe operating temperature, the crew will enter EOP-2A and commence emergency depressurization of the RPV by opening at least 4 SRVs prior to a third area reaching its maximum safe operating temperature. (MSOT). (The first area to exceed MSOT is the Steam Tunnel area. The second area to exceed MSOT is the SW Quad with TS-99G/TS-105B. The third area to exceed the MSOT is the NW Quad with TS-99C.)
Safety Significance	Should secondary containment temperatures exceed their maximum safe operating values in more than one area, the RPV must be depressurized to preclude further temperature increases. RPV depressurization places the primary system in its lowest possible energy state, rejects heat to the suppression pool in preference to outside the containment, and reduces the driving head and flow of primary systems that are unisolated and discharging into the secondary containment. The criteria of "more than one area" specified in this step identifies the rise in secondary containment temperature as a wide-spread problem which may pose a direct and immediate threat to secondary containment integrity, equipment located in the secondary containment, and continued safe operation of the plant.
Cues	SPDS indication for secondary containment parameters indicate area radiation, area temperature, or area water level has exceeded its maximum safe operating value in two areas.
Measurable Performance Indicators	Opening of at least 4 SRV controls on panel 9-3: SRV-71A, SRV-71B, SRV-71E, SRV-71G, SRV-71H, SRV-71C, SRV-71D, SRV-71F
Performance Feedback	Crew will observe SRV light indication go from green to red, amber pressure switch lights illuminate, reactor pressure lowering on SPDS and panel 9-3 and 9-5 meters and recorders, and SRV tailpipe temperatures rise on recorder MS-TR-166.
Applicability	EOP-5A conditions, RCS leaks into secondary containment with the RPV pressurized.
Justification for the chosen performance limit	Emergency Depressurization is required due to effects of a break spreading into and potentially affecting safety equipment and operations in more than one area; however, emergency depressurization is not allowed until the second area exceeds its Max Safe limit. Before the Max Safe limit is exceeded in a third area gives reasonable time for the crew to perform emergency depressurization before the leak hampers equipment or operations in an even more widespread area.
BWR Owners Group Appendix	App. B, steps SC/T-4.2, SC/r-2.2, SC/L-2.2. Contingency #2
Scenario Guide Requirements	The scenario must be able to drive the selected parameter to its Max Safe value in three plant areas. If temperature is chosen, a failure to scram event, where RPV pressure is not allowed to be lowered, is well suited. Also, ensure the leak severity itself, or subsequent cold water injection, does not deplete RPV pressure (driving head) so low that Max Safe in a third area cannot be reached. The crew should be driven to ED, versus

	just reducing pressure, to provide a consistent, measurable performance indicator. The CT listed in the scenario should list which instruments/areas will exceed their MSO limit first, second, and third.

Scenario Summary

The plant is operating at ~ 71% power at BOL in the operating cycle when the crew takes the watch. The crew is directed to shift TEC pumps and to raise power to ~74% to power ascension following a quarterly down power.

Event 1 begins when the BOP shifts TEC pumps per 2.2.76 Section 6. He will start TEC-P-B and secure TEC-P-C. **(BOP-NORMAL)**

Event 2 begins when crew commences raising power with Reactor Recirculation to 74% power **(ATC - RX Manipulation)** per 2.1.10.

Event 3 begins when APRM ‘C’ fails downscale. The ATCO will respond to Annunciator 9-5-1/C-8, APRM Downscale, and take alarm card actions to bypass APRM ‘C’. **(ATC ACTION)** The CRS will address technical specifications and determine that LCO 3.3.1.1 is a potential LCO only.

Event 4 begins when B RPS MG set EPAs trip causing a loss of B RPS. The BOP will respond to Annunciator C-1/F-2, RPS PWR PANEL 1B VOLTAGE FAILURE and transfers RPS B to the alternate source. **(BOP ACTION)** The BOP will verify and reset Group Isolations IAW 2.1.22. The ATCO will monitor 9-5 and reset the ‘B’ half scram IAW 2.1.5. **(ATC ACTION)** The CRS will determine that LCO 3.4.5 Condition B and DLCO 3.2.4 Condition A are not met. **(CRS TS ACTION)** The CRS will determine that LCO 3.3.8.2 is a potential LCO only since the RPS MG set is not in –service.

Event 5 begins when HPCI spuriously initiates. The BOP will respond to 9-3-2/A-1 HPCI LOGIC ACTUATED, and take immediate operator actions to secure HPCI. **(BOP ACTION) (MC)**The CRS will enter 2.4CSCS and direction actions. The CRS will determine that LCO 3.5.1 Condition is not met. **(CRS TS ACTION)**

Event 6 begins when a loss of all AC occurs. DG-1 fails to start and DG2 starts but EG2 fails to automatically tie to the Bus. The BOP will take DG2 sync switch to EG2 and then close breaker EG2 restoring power to the G critical bus. **(MC BOP ACTION)** **(CT#1)** The crew will enter 5.3EMPWR and take subsequent operator actions. The ATCO will perform scram actions and place the RX mode switch to Refuel **(ATC ACTIONS)** and take 2.1.5 actions to place the RX Mode switch to SHUTDOWN.

Event 7 (MAJOR) begins when rupture of the HPCI Steam Line occurs. CRS will enter EOP-5A and direct isolating the HPCI steam line. HPCI will fail to auto isolate and cannot be isolated from the Control Room or the ASD Room. Temperatures will continue to rise. The CRS will transition to EOP-2A when two areas reach Maximum Safe Operating Temperature (MSOT). The crew will emergency depressurize prior to a third area MSOT:

- When 2 areas exceed MSOT, commence ED prior to a third area reaching MSOT **(CT#2)**

The scenario can be terminated when ED is complete and RPV water level band is ordered and is being recovered towards the prescribed band.