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

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

Q#	E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
1	000007 Reactor Trip - Stabilization - Recovery / 1				06			Reactor trip (scram): verification that the control and safety rods are in after the trip	4.4	1
2	000008 Pressurizer Vapor Space Accident / 3		03					Controllers and positioners	2.5	1
3	000009 Small Break LOCA / 3					08		Letdown isolation valve position indication	2.9	1
4	000011 Large Break LOCA / 3	01						Natural circulation and cooling, including reflux boiling	4.1	1
	000015 RCP Malfunctions / 4 000017 RCP Malfunctions (Loss of RC Flow) / 4									0
	000022 Loss of Rx Coolant Makeup / 2									0
5	000025 Loss of RHR System / 4						01.27	Knowledge of system purpose and/or function.	3.9	1
6	000026 Loss of Component Cooling Water / 8			02				The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS	3.6	1
	000027 Pressurizer Pressure Control System Malfunction / 3									0
	000029 ATWS / 1									0
7	000038 Steam Gen. Tube Rupture / 3					04		Radiation levels (MREM/hr)	3.9	1
8	000040 Steam Line Rupture - Excessive Heat Transfer / 4		02					Sensors and detectors	2.6	1
	WE12 Uncontrolled Depressurization of all Steam Generators / 4									
9	000054 (CE/E06) Loss of Main Feedwater / 4			02				Matching of feedwater and steam flows	3.4	1
10	000055 Station Blackout / 6						04.06	Knowledge of EOP mitigation strategies.	3.7	1
11	000056 Loss of Off-site Power / 6	04						Definition of saturation conditions, implication for the systems	3.1	1
12	000057 Loss of Vital AC Inst. Bus / 6					18		The indicator, valve, breaker, or damper position which will occur on a loss of power	3.1	1
	000058 Loss of DC Power / 6									0
13	000062 Loss of Nuclear Svc Water / 4				02			Loads on the SWS in the control room	3.2	1
14	000065 Loss of Instrument Air / 8			03				Knowing effects on plant operation of isolating certain equipment from instrument air	2.9	1
15	W/E04 LOCA Outside Containment / 3	03						Annunciators and conditions indicating signals, and remedial actions associated with the LOCA Outside Containment	3.5	1
16	W/E11 Loss of Emergency Coolant Recirc. / 4						01.07	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.4	1
17	BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4		02					Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	3.9	1
18	000077 Generator Voltage and Electric Grid Disturbances / 6				04			Reactor controls	4.1	1
	K/A Category Totals:	3	3	3	3	3	3	Group Point Total:		18

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

Q#	E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
	000007 Reactor Trip - Stabilization - Recovery / 1									0
	000008 Pressurizer Vapor Space Accident / 3									0
	000009 Small Break LOCA / 3									0
	000011 Large Break LOCA / 3									0
76	000015 RCP Malfunctions / 4 000017 RCP Malfunctions (Loss of RC Flow) / 4						01.20	Ability to interpret and execute procedure steps.	4.6	1
77	000022 Loss of Rx Coolant Makeup / 2						04.50	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.0	1
	000025 Loss of RHR System / 4									0
	000026 Loss of Component Cooling Water / 8									0
78	000027 Pressurizer Pressure Control System Malfunction / 3						04.31	Knowledge of annunciator alarms, indications, or response procedures.	4.1	1
79	000029 ATWS / 1					06		Main turbine trip switch position indication	3.9	1
	000038 Steam Gen. Tube Rupture / 3									0
	000040 Steam Line Rupture - Excessive Heat Transfer / 4									1
80	WE12 Uncontrolled Depressurization of all Steam Generators / 4					02		Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	3.9	
	000054 (CE/E06) Loss of Main Feedwater / 4									0
	000055 Station Blackout / 6									0
	000056 Loss of Off-site Power / 6									0
	000057 Loss of Vital AC Inst. Bus / 6									0
81	000058 Loss of DC Power / 6					02		125V dc bus voltage, low/critical low, alarm	3.6	1
	000062 Loss of Nuclear Svc Water / 4									0
	000065 Loss of Instrument Air / 8									0
	W/E04 LOCA Outside Containment / 3									0
	W/E11 Loss of Emergency Coolant Recirc. / 4									0
	BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4									0
	000077 Generator Voltage and Electric Grid Disturbances / 6									0
	K/A Category Totals:	0	0	0	0	3	3	Group Point Total:		6

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

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

Q#	E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
	000001 Continuous Rod Withdrawal / 1									0
	000003 Dropped Control Rod / 1									0
25	000005 Inoperable/Stuck Control Rod / 1	02						Flux tilt	3.1	1
	000024 Emergency Boration / 1									0
19	000028 Pressurizer Level Malfunction / 2					01		PZR level indicators and alarms	3.4	1
	000032 Loss of Source Range NI / 7									0
	000033 Loss of Intermediate Range NI / 7									0
	000036 Fuel Handling Accident / 8									0
20	000037 Steam Generator Tube Leak / 3						02.44	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	1
	000051 Loss of Condenser Vacuum / 4									0
	000059 Accidental Liquid RadWaste Rel. / 9									0
	000060 Accidental Gaseous Radwaste Rel. / 9									0
21	000061 ARM System Alarms / 7		01					Detectors at each ARM system location	2.5	1
	000067 Plant Fire On-site / 8									0
	000068 Control Room Evac. / 8									0
	000069 Loss of CTMT Integrity / 5									1
22	W/E14 High Containment Pressure / 5				02			Operating behavior characteristics of the facility	3.3	
	000074 Inad. Core Cooling / 4									
23	W/E06 Degraded Core Cooling / 4			03				Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations	4.0	1
	W/E07 Saturated Core Cooling / 4									
	000076 High Reactor Coolant Activity / 9									0
	W/E01 Rediagnosis / 3									1
24	W/E02 SI Termination / 3		01					Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	3.4	
	W/E13 Steam Generator Over-pressure / 4									0
26	W/E15 Containment Flooding / 5					02		Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	2.9	1
	W/E16 High Containment Radiation / 9									0
	W/E03 LOCA Cooldown - Depress. / 4									0
	W/E09 Natural Circulation Operations / 4									1
27	W/E10 Natural Circulation with Steam Voide in Vessel with/without RVLIS. / 4				02			Operating behavior characteristics of the facility	3.6	
	W/E08 RCS Overcooling - PTS / 4									0
	K/A Category Totals:	1	2	1	2	2	1	Group Point Total:		9

	ES-401 PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (SRO)								Form ES-401-2	
Q#	E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
	000001 Continuous Rod Withdrawal / 1									0
	000003 Dropped Control Rod / 1									0
	000005 Inoperable/Stuck Control Rod / 1									0
	000024 Emergency Boration / 1									0
	000028 Pressurizer Level Malfunction / 2									0
	000032 Loss of Source Range NI / 7									0
	000033 Loss of Intermediate Range NI / 7									0
82	000036 Fuel Handling Accident / 8					02		Occurrence of a fuel handling incident	4.1	1
	000037 Steam Generator Tube Leak / 3									0
	000051 Loss of Condenser Vacuum / 4									0
	000059 Accidental Liquid RadWaste Rel. / 9									0
	000060 Accidental Gaseous Radwaste Rel. / 9									0
	000061 ARM System Alarms / 7									0
	000067 Plant Fire On-site / 8									0
	000068 Control Room Evac. / 8									0
	000069 Loss of CTMT Integrity / 5									0
	W/E14 High Containment Pressure / 5									
	000074 Inad. Core Cooling / 4									0
	W/E06 Degraded Core Cooling / 4									
	W/E07 Saturated Core Cooling / 4									
	000076 High Reactor Coolant Activity / 9									0
	W/E01 Rediagnosis / 3									0
	W/E02 SI Termination / 3									
	W/E13 Steam Generator Over-pressure / 4									0
	W/E15 Containment Flooding / 5									0
83	W/E16 High Containment Radiation / 9					01		Facility conditions and selection of appropriate procedures during abnormal and emergency operations	3.3	1
84	W/E03 LOCA Cooldown - Depress. / 4						04.20	Knowledge of the operational implications of EOP warnings, cautions, and notes.	4.3	1
	W/E09 Natural Circulation Operations / 4									0
	W/E10 Natural Circulation with Steam Voide in Vessel with/without RVLIS. / 4									
85	W/E08 RCS Overcooling - PTS / 4						04.18	Knowledge of the specific bases for EOPs.	4.0	1
	K/A Category Totals:	0	0	0	0	2	2	Group Point Total:		4

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

Plant Systems - Tier 2/Group 1 (RO)

Q#	System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
28	003 Reactor Coolant Pump					04							Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow	3.2	1
29	004 Chemical and Volume Control			04									RCPS	3.7	1
30	005 Residual Heat Removal								02				Pressure transient protection during cold shutdown	3.5	1
31,32	006 Emergency Core Cooling								12			04.04	Conditions requiring actuation of ECCS; Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	4.5; 4.5	2
33,34	007 Pressurizer Relief/Quench Tank	03							02				RCS; Abnormal pressure in the PRT	3; 2.6	2
35	008 Component Cooling Water							03					CCW pressure	2.7	1
36	010 Pressurizer Pressure Control		03										Indicator for PORV position	2.8	1
37,38	012 Reactor Protection						03					04.45	Trip logic circuits; Ability to prioritize and interpret the significance of each annunciator or alarm.	3.1; 4.1	2
39	013 Engineered Safety Features Actuation				13								MFW isolation/reset	3.7	1
40	022 Containment Cooling							04					Cooling water flow	3.2	1
	025 Ice Condenser														0
41,42	026 Containment Spray							03		01			Containment sump level; Pump starts and correct MOV positioning	3.5; 4.3	2
43	039 Main and Reheat Steam								01				Flow paths of steam during a LOCA	3.1	1
44,45	059 Main Feedwater			03							08		S/Gs; Feed regulating valve controller	3.5; 3	2
46	061 Auxiliary/Emergency Feedwater					01							Relationship between AFW flow and RCS heat transfer	3.6	1
47,48	062 AC Electrical Distribution			01						04			Major system loads; Operation of inverter (e.g., precharging synchronizing light, static transfer)	3.5; 2.7	2
49,50	063 DC Electrical Distribution	03										02.42	Battery charger and battery; Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	2.9; 3.9	2
51	064 Emergency Diesel Generator						08						Fuel oil storage tanks	3.2	1
52	073 Process Radiation Monitoring				01								Release termination when radiation exceeds setpoint	4.0	1
53	076 Service Water	08											RHR system	3.5	1
54	078 Instrument Air		02										Emergency air compressor	3.3	1
55	103 Containment				04								Personnel access hatch and emergency access hatch	2.5	1
															0
	K/A Category Totals:	3	2	3	3	2	2	3	4	2	1	3	Group Point Total:		28

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

Plant Systems - Tier 2/Group 2 (RO)

Q#	System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
56	001 Control Rod Drive		03										One-line diagram of power supplies to logic circuits	2.7	1
57	002 Reactor Coolant									01			Reactor coolant leak detection system	3.7	1
58	011 Pressurizer Level Control					11							Reasons for selecting "manual" on letdown control valve controller	2.5	1
59	014 Rod Position Indication	01											CRDS	3.2	1
60	015 Nuclear Instrumentation											02.25	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	3.2	1
61	016 Non-nuclear Instrumentation				03								Input to control systems	2.8	1
	017 In-core Temperature Monitor														0
	027 Containment Iodine Removal														0
	028 Hydrogen Recombiner and Purge Control														0
	029 Containment Purge														0
	033 Spent Fuel Pool Cooling														0
62	034 Fuel Handling Equipment						02						Radiation monitoring systems	2.6	1
	035 Steam Generator														0
63	041 Steam Dump/Turbine Bypass Control							01					T-ave., verification above low/low setpoint	2.9	1
	045 Main Turbine Generator														0
	055 Condenser Air Removal														0
	056 Condensate														0
	068 Liquid Radwaste														0
64	071 Waste Gas Disposal										24		The double verification required before waste gas release	2.9	1
65	072 Area Radiation Monitoring								01				Erratic or failed power supply	2.7	1
	075 Circulating Water														0
	079 Station Air														0
	086 Fire Protection														0
	K/A Category Totals:	1	1	0	1	1	1	1	1	1	1	1	Group Point Total:		10

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

Plant Systems - Tier 2/Group 1 (SRO)

Q#	System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
	003 Reactor Coolant Pump														0
86	004 Chemical and Volume Control								27				Improper RWST boron concentration	4.2	1
87	005 Residual Heat Removal											02.40	Ability to apply Technical Specifications for a system.	4.7	1
	006 Emergency Core Cooling														0
	007 Pressurizer Relief/Quench Tank														0
	008 Component Cooling Water														0
	010 Pressurizer Pressure Control														0
	012 Reactor Protection														0
88	013 Engineered Safety Features Actuation								03				Rapid depressurization	4.7	1
	022 Containment Cooling														0
	025 Ice Condenser														0
	026 Containment Spray														0
	039 Main and Reheat Steam														0
	059 Main Feedwater														0
	061 Auxiliary/Emergency Feedwater														0
	062 AC Electrical Distribution														0
89	063 DC Electrical Distribution								01				Grounds	3.2	1
	064 Emergency Diesel Generator														0
	073 Process Radiation Monitoring														0
	076 Service Water														0
	078 Instrument Air														0
90	103 Containment											01.25	Ability to interpret reference materials, such as graphs, curves, tables, etc.	4.2	1
															0
	K/A Category Totals:	0	0	0	0	0	0	0	3	0	0	2	Group Point Total:		5

ES-401	PWR Examination Outline Plant Systems - Tier 2/Group 2 (RO)												Form ES-401-2		
Q#	System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
56	001 Control Rod Drive		03										One-line diagram of power supplies to logic circuits	2.7	1
57	002 Reactor Coolant									01			Reactor coolant leak detection system	3.7	1
58	011 Pressurizer Level Control					11							Reasons for selecting "manual" on letdown control valve controller	2.5	1
59	014 Rod Position Indication	01											CRDS	3.2	1
60	015 Nuclear Instrumentation											02.25	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	3.2	1
61	016 Non-nuclear Instrumentation				03								Input to control systems	2.8	1
	017 In-core Temperature Monitor														0
	027 Containment Iodine Removal														0
	028 Hydrogen Recombiner and Purge Control														0
	029 Containment Purge														0
	033 Spent Fuel Pool Cooling														0
62	034 Fuel Handling Equipment						02						Radiation monitoring systems	2.6	1
	035 Steam Generator														0
63	041 Steam Dump/Turbine Bypass Control							01					T-ave., verification above low/low setpoint	2.9	1
	045 Main Turbine Generator														0
	055 Condenser Air Removal														0
	056 Condensate														0
	068 Liquid Radwaste														0
64	071 Waste Gas Disposal										24		The double verification required before waste gas release	2.9	1
65	072 Area Radiation Monitoring								01				Erratic or failed power supply	2.7	1
	075 Circulating Water														0
	079 Station Air														0
	086 Fire Protection														0
	K/A Category Totals:	1	1	0	1	1	1	1	1	1	1	1	Group Point Total:		10

	ES-401											PWR Examination Outline		Form ES-401-2
												Plant Systems - Tier 2/Group 2 (SRO)		
Q#	System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
	001 Control Rod Drive														0
	002 Reactor Coolant														0
	011 Pressurizer Level Control														0
	014 Rod Position Indication														0
	015 Nuclear Instrumentation														0
	016 Non-nuclear Instrumentation														0
91	017 In-core Temperature Monitor								01				Thermocouple open and short circuits	3.5	1
	027 Containment Iodine Removal														0
	028 Hydrogen Recombiner and Purge Control														0
	029 Containment Purge														0
92	033 Spent Fuel Pool Cooling								03				Abnormal spent fuel pool water level or loss of water level	3.5	1
	034 Fuel Handling Equipment														0
93	035 Steam Generator											02.22	Knowledge of limiting conditions for operations and safety limits.	4.7	1
	041 Steam Dump/Turbine Bypass Control														0
	045 Main Turbine Generator														0
	055 Condenser Air Removal														0
	056 Condensate														0
	068 Liquid Radwaste														0
	071 Waste Gas Disposal														0
	072 Area Radiation Monitoring														0
	075 Circulating Water														0
	079 Station Air														0
	086 Fire Protection														0
	K/A Category Totals:	0	0	0	0	0	0	0	2	0	0	1	Group Point Total:		3

Facility Name: Robert E Ginna Date of Exam:12/03/14

Q#	Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
66	1. Conduct of Operations	2.1. 08	Ability to coordinate personnel activities outside the control room.	3.4	1		
67		2.1. 14	Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.	3.1	1		
68		2.1. 26	Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).	3.4	1		
94		2.1. 23	Ability to perform specific system and integrated plant procedures during all modes of plant operation.			4.4	1
		Subtotal			3		1
69	2. Equipment Control	2.2. 37	Ability to determine operability and/or availability of safety related equipment.	3.6	1		
70		2.2. 41	Ability to obtain and interpret station electrical and mechanical drawings.	3.5	1		
95		2.2. 06	Knowledge of the process for making changes to procedures.			3.6	1
96		2.2. 18	Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.			3.9	1
		Subtotal			2		2
71	3. Radiation Control	2.3. 07	Ability to comply with radiation work permit requirements during normal or abnormal conditions.	3.5	1		
72		2.3. 14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.4	1		
97		2.3. 05	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.			2.9	1
98		2.3. 11	Ability to control radiation releases.			4.3	1
		Subtotal			2		2
73	4. Emergency Procedures / Plan	2.4. 05	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.	3.7	1		
74		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.	4.0	1		
75		2.4. 41	Knowledge of the emergency action level thresholds and classifications.	2.9	1		
99		2.4. 08	Knowledge of how abnormal operating procedures are used in conjunction with EOPs.			4.5	1
100		2.4. 47	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.			4.2	1
		Subtotal			3		2
	Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
T2, G1 (RO)	**006 A2.12** (4.5/4.8)	Per 06/12/14 telecom with Chief Examiner. Original K/A **A2.09** does not reflect the fact that Ginna's RWST is *normally* vented to atmosphere. "Reselect based upon the inability to write a discriminatory question." Randomly reselected.
T2, G1 (RO)	**012 K6.03** (3.1/3.5)	Per 06/12/14 telecom with Chief Examiner. Original K/A **K6.08** refers to a system not recognized at Ginna (Core Operating Limit Support System). "Reselect if the system doesn't apply to Ginna." Randomly reselected.
T1, G1 (SRO)	**APE 022 2.4.50** (4.2/4.0)	Per 06/12/14 telecom with Chief Examiner. Original K/A **2.4.49.** Ginna does not have any Abnormal Procedures which have Immediate Operator Actions. "Two choices: (1) Reselect or (2) On Loss of Charging, what do you expect the operators to do?" Randomly reselected.
T1, G2 (RO)	**APE 037 2.2.44** (4.2/4.4)	Per 07/29/14 telecom with Chief Examiner. Original K/A **2.2.36.** Unable to write a discriminating question on this subject in this abnormal procedure. "Reselect based upon the inability to write a discriminating question."
T1, G1 (SRO)	**2.4.31** (4.2/4.1)	Per 08/18/14 telecom with Chief Examiner. Original K/A **2.4.35.** The given APE does not have any local operator actions associated with it. Randomly reselected.
T2, G1 (RO)	**007 A2.02** (2.6/)	Per 10/6/14 telecom with Chief Examiner. Original K/A **007 A4.01.** Subject K/A is not relevant to the Ginna facility. Ginna does not have a PRT Supply Spray valve as stated in the K/A. "Reselect a K/A"
T1, G1 (RO)	**057 AA2.18** (3.1/)	Per 10/6/14 telecom with Chief Examiner. Original K/A **057 AA2.11.** Subject K/A is not relevant to the Ginna facility. Ginna does not have Feed Pump controllers and are constant speed motors that use Feed Regulating valves to control flow. "Reselect K/A"
T1, G2 (RO)	**005 AK1.02** (3.1/)	Per telecom 10/31/14, discussed with Chief examiner unable to write a question to an acceptable level of difficulty (LOD) for K/A **E13 EK1.3**. "Reselected K/A"

Facility: Ginna		Date of Examination:	12/2014
Examination Level:	SRO	Operating Test Number:	2014-301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, P, R	2.1.25 (4.2) Ability to interpret reference materials, such as graphs, curves, tables, etc. JPM: Determine Time to Boil for a Loss of Shutdown Cooling
Conduct of Operations	M, R	2.1.8 (4.1) Ability to coordinate personnel activities outside the control room. JPM: Cold Weather Walkdown
Equipment Control	N, R	2.2.12 (4.1) Knowledge of Surveillance Procedures. JPM: Perform RCS Leakrate Surveillance
Radiation Control	M, R	2.3.15 (3.1) Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. JPM: Evaluate the Impact of R-14 Failure on a Planned Gaseous Decay Tank Release
Emergency Procedures/Plan	M, S	2.4.41 (4.6) Knowledge of emergency action level thresholds and classifications. JPM: Classify an Emergency Event

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

*Type Codes & Criteria:
(C)ontrol room, **(0)** (S)imulator, **(1)** or Class(R)oom **(4)**
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes) **(1)**
(N)ew or (M)odified from bank (≥ 1) **(4)**
(P)revious 2 exams (≤ 1; randomly selected) **(1)**

SRO Admin JPM Summary

A1a This is a Bank JPM randomly selected from the last two NRC Exams. The operator will be provided with two sets of shutdown conditions; one current, and one projected to exist within four days. The operator will be directed to determine the Time to Boil given a Loss of RHR for each of the two sets of conditions. The operator will be expected to use IP-OUT-2, Outage Risk Management, and determine that the Time to Boil for the present plant conditions is 2.46 hours, and for the projected conditions four days from now is .43 hours (25.96 +0.1/-0.0 minutes).

A1b This is a modified Bank JPM. The operator will be told that the weekly Cold Weather Walkdown is in progress, and the Primary, Secondary and Outside Auxiliary Operators are about ready to start their Walkdown Tours; and provided with a recently completed Attachment 1, Cold Weather Temperature Log, of O-22, Cold Weather Walkdown Procedure. They will also be told that a severe lake-effect storm is occurring with 24 inches of snow having accumulated over the last four hours. The operator will be directed to review Attachment 1 and perform section 6.1, General Instructions, of O-22, and prioritize any areas of concern for the Auxiliary Operator Walkdown Tours; and then to document discrepancies, if any, and any required action(s). The operator will be expected to review the most recent plant area temperature data and then use O-22 to determine that two areas of the plant are too low in temperature. Then the operator will prioritize the actions of the Auxiliary Operators to tour these areas first, identify discrepancies and actions required in accordance with the attached KEY.

A2 This is a new JPM. The operator will be told that the plant is at 100% power with 40 gpm Letdown flow, that the VCT Level point on the PPCS is OOS, that at 1020 the VCT level was 25% when the HCO started the Day Shift RCS Leakage Monitoring Surveillance, intending to monitor for about 4 hours, and that at 1125 a VCT Level instrument failure occurred which I&C corrected and restored to operation within 10 minutes. The operator will be provided with a current Attachment 3, RCS Leakage Surveillance Record, with recent Auto Makeup information, and directed to record the data for the recent Auto Makeup on Attachment 3 AND to determine the current RCS Leakrate. The operator will also be directed to identify any Tech Spec LCOs that have been exceeded, and if any LCOs are exceeded, identify any required Tech Spec ACTION. The operator will be expected to complete Attachment 3 in accordance with the KEY provided, and identify the LCO 3.4.13, RCS Operational Leakage, is NOT met, and that the required ACTION is to reduce leakage to within limits within 4 hours (A.1).

A3 This is a modified Bank JPM. The operator will be told that the plant is at 100% power, that an A Gas Decay Tank release is being planned, that the HCO reports that R-14A, Plant Vent Gas, is de-energized for preventative maintenance that will require 2 more hours, and must be completed before it can be returned to service, and that R-14, Plant Vent Gas, has just failed LOW and I&C has indicated that it will take at least four hours to troubleshoot and correct the failure. The operator will be directed to determine the required actions based on the failure of R-14, and to determine the earliest time at which the A Gas Decay Tank release can start, and to identify any actions that will be required to start the release. The operator will be expected to determine that (1) on-going releases (i.e. ventilation) may continue provided grab samples are taken and analyzed for isotopic activity at least once per 8 hours, that (2) at least one of either R-14 or R-14A must be restored to OPERABLE status within 30 days or it must be explained in the next

Annual Radioactive Effluent Release Report, and that (3) the earliest time that the A Gas Decay Tank release can be initiated is 2 hours from now when the maintenance on R-14A is completed, and ONLY if at least two independent samples of the tank's contents are analyzed, and at least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

A4 This is a modified Bank JPM. This JPM will be given immediately following the associated Simulator Scenario. Consequently, there are five separate versions of this JPM, one each associated with the submitted Simulator Scenarios. A portion of this JPM is Time Critical. After each Simulator Scenario, the operator will be told that the sequence of events that they have just performed on the recently completed Simulator Scenario must be evaluated for any Emergency Classifications. The operator will be directed to determine the Emergency Classification that would have been made first in the recently completed scenario, and Identify the basis for this classification. The operator will be given a time that all the indications needed to classify the first event requiring an Emergency Classification were apparent in the Control Room and then directed to identify the latest time at which the Emergency Classification must have been made. Finally, the operator will be directed to identify any subsequent emergency classifications that would have been made in the Scenario, and the basis for this classification. The operator will be expected to make the appropriate classifications and identify the time frames as appropriate.

Facility: Ginna		Date of Examination:	12/2014
Exam Level (circle one): SRO(U)		Operating Test No.:	2014-301

Control Room Systems@ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U, including 1 ESF)		
System / JPM Title	Type Code*	Safety Function
A. 005 Residual Heat Removal System [005 A2.03 (2.9/3.1)] Lineup RCDT Pump for Core Cooling	S, D, L	4P
B. EPE E14 High Containment Pressure [EPE E14 EA1.1 (3.7/3.7)] Verify Containment Isolation and Heat Removal	S, N, A, EN	5
C. 004 Chemical and Volume Control System [004 A4.08 (3.8/3.4)] CVCS Leak Isolation	S, N, A	2
In-Plant Systems@ (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)		
I. APE 068 Control Room Evacuation [APE 068 AA1.01 (4.3/4.5)] Locally Operate the ARVs	P, D, E	8
J. EPE 055 Station Blackout [EPE 055 EA2.03 (3.9/4.7)] Isolate RCP Seals	D, R, E	6

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

JPM Summary

JPM A This is a Bank JPM. The operator will be told that the plant was performing a normal cooldown when both RHR Pumps tripped, that the S/G's are unavailable, that AP-RHR.1, Loss of RHR, was entered and attempts to restart the pumps were unsuccessful, and that the crew is in progress of establishing Containment Integrity and all personnel are clear of Containment. The operator will be directed to lineup the RCDT pumps to provide core cooling per ER-RHR.1, Section 6.2, Closed Loop RCS to RHR Cooling. The operator will be expected to line up the RCDT pumps to provide core cooling per Section 6.2 of ER-RHR.1.

JPM B This is a new JPM. The operator will be told that the plant tripped from 100% power and safety injection has actuated, that the crew entered E-0, Reactor Trip or Safety Injection, and then transitioned to E-2, Faulted Steam Generator Isolation, and that due to a degrading transient, an Orange Path now exists on the Containment Critical Safety Function Status Tree. The operator will be directed to verify Containment Isolation and Heat Removal systems are operating as expected by performing FR-Z.1, Response to High Containment Pressure, starting from Step 1. During the course of this action, the operator will recognize that two Containment Isolation Valves have failed to automatically close, and that the Containment Spray ESFAS Signal has failed to function **(Alternate Path)**. The operator will be expected to verify Containment Isolation and Heat Removal systems are operating in accordance with Steps 1-3 of FR-Z.1. When the operator discovers that two Containment Isolation Valves have failed to close as expected the operator will close or direct that alternative valves be closed in accordance with ATT-3.0, Attachment CI/CVI. When the operator discovers that the Containment Spray ESFAS Signal has failed to function, the operator will take action to initiate Containment Spray. The operator will also start the D CNMT Recirc Fan.

JPM C This is a new JPM. The operator will be told that the Plant is at 100% power, that the crew has entered AP-CVCS.1, "CVCS Leak," completing steps 1-5, that the Pressurizer level is stable, and that the leak is believed to be in the Containment. The operator will be directed to take appropriate actions to isolate the leak starting with Step 6 of AP-CVCS.1, CVCS Leak, and control RCS inventory to maintain Pressurizer level stable. During the course of this action, the operator will recognize that both operating Charging

Pumps have tripped **(Alternate Path)**. The operator will be expected to isolate Normal Letdown and Charging, recognize that both operating Charging Pumps have tripped, use AP-CVCS.3 to restore Charging flow, and then place Excess Letdown in service.

JPM I This is a Bank JPM randomly selected from the last two NRC JPM Exams. The operator will be told that the plant has experienced an uncontrollable Control Room Complex fire, that the crew performed the required actions of AP-CR.1, Control Room Inaccessibility, and transitioned to ER-FIRE.1, Alternate Shutdown For Control Complex Fire, that they are the Head Control Operator (HCO) and are equipped with a radio, that the Shift Manager is stabilizing the plant in MODE 3, and that all the required Appendix R equipment has been retrieved from the Appendix R locker outside the Control Room. The operator will be directed to locally open the A Atmospheric Relief Valve, V-3411, three turns per P-15.2, Dump Steam Through ARVs Locally. The operator will be expected to locally open the A Atmospheric Relief Valve, V-3411, three turns per P-15.2, and then isolate this valve when the handwheel will not rotate in the close direction.

JPM J This is a Bank JPM. The operator will be told that the plant was operating at 100% power when it suffered a loss of all AC power, and that the crew entered ECA-0.0, Loss of All AC Power, and is at the step that directs local isolation of RCP Seals. The operator will be directed to isolate RCP seals in accordance with the applicable portions of ATT-21.0, Attachment RCS Isolation. The operator will be expected to locally isolate the RCP Seals per ATT-21.0.

Facility: Ginna		Date of Examination:	12/2014
Examination Level:	RO	Operating Test Number:	2014-301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed	
Conduct of Operations	D, R	2.1.25 (3.9)	Ability to interpret reference materials, such as graphs, curves, tables, etc.
		JPM:	Calculate SDM for an Operating Reactor with an Untripable Rod
Conduct of Operations	N, R	2.1.43 (4.1)	Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.
		JPM:	Perform an Independent Verification of a Post-Trip Xenon Reactivity Calculation
Equipment Control	N, R	2.2.12 (3.7)	Knowledge of Surveillance Procedures.
		JPM:	Perform RCS Leakrate Surveillance
Radiation Control	N, R	2.3.5 (2.9)	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
		JPM:	Evaluate Steam Generator Tube Leakage from R-47 Reading

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

*Type Codes & Criteria:

(C)ontrol room, **(0)** (S)imulator, **(0)** or Class(R)oom **(4)**
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes) **(1)**
(N)ew or (M)odified from bank (≥ 1) **(3)**
(P)revious 2 exams (≤ 1; randomly selected) **(0)**

RO Admin JPM Summary

A1a This is a Bank JPM. The operator will be provided with a set of data relating to normal power plant operation, and told that Rod K-7 has been determined to be untripable. The operator will be directed to calculate Shutdown Margin (SDM), and determine if sufficient SDM exists for the current plant conditions. The operator will be expected to calculate Shutdown Margin in accordance with the O-3.2, Shutdown Margin for an Operating Reactor, and determine that sufficient SDM does NOT exist.

A1b This is a new JPM. The operator will be told that the plant has tripped from 50% power, that the reactor had been operating at this power for the last 3 days prior to the trip, that the XENON PREDICT program on the PPCS is NOT available, that the Reactor Engineer has completed the Xenon Reactivity Calculation, and that it is currently 0900 on 12/3/14. The operator will be directed to independently verify the Xenon Reactivity Calculation in accordance with Step 6.1.3 of O-3, Hot Shutdown With Xenon Present. The operator will be expected to complete an independent verification of the Xenon Reactivity Calculation and determine that Xenon will equal the value at time of shutdown at 2200 on 12/3/14.

A2 This is a new JPM. The operator will be told that the plant is at 100% power with 40 gpm Letdown flow, that the VCT Level point on the PPCS is OOS, that at 1020 the VCT level was 25% when the HCO started the Day Shift RCS Leakage Monitoring Surveillance, intending to monitor for about 4 hours, and that at 1125 a VCT Level instrument failure occurred which I&C corrected and restored to operation within 10 minutes. The operator will be provided with a current Attachment 3, RCS Leakage Surveillance Record, with recent Auto Makeup information, and directed to record the data for the recent Auto Makeup on Attachment 3 AND to determine the current RCS Leakrate. The operator will also be directed to identify any Tech Spec LCOs that have been exceeded. The operator will be expected to complete Attachment 3 in accordance with the KEY provided, and identify the LCO 3.4.13, RCS Operational Leakage, is NOT met.

A3 This is a new JPM. The operator will be told that the plant is operating at 100% power, that the crew has been notified that RMS-R47, R-47 AIR EJECTOR NOBLE GAS MONITOR, has alarmed in the TSC, provided with current readings, and told that the crew is evaluating the need to enter AP-SG.1, Steam Generator Tube Leak. The operator will be directed to determine if entry into AP-SG.1 is appropriate by (1) determining if the R-47 alarm setpoint is set properly and (2) comparing the local reading to Curve #06-004 in accordance with PPCS-R47AR. The operator will be expected to determine that the R-47 alarm setpoint is set lower than identified by P-9, and that the local R-47 reading is too low to support entry into AP.SG.1.

Facility: Ginna		Date of Examination:	12/2014
Exam Level: RO		Operating Test No.:	2014-301

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

	System / JPM Title	Type Code*	Safety Function
A.	005 Residual Heat Removal System [005 A2.03 (2.9/3.1)] Lineup RCDT Pump for Core Cooling	S, D, L	4P
B.	EPE E14 High Containment Pressure [EPE E14 EA1.1 (3.7/3.7)] Verify Containment Isolation and Heat Removal	S, N, A, EN	5
C.	004 Chemical and Volume Control System [004 A4.08 (3.8/3.4)] CVCS Leak Isolation	S, N, A	2
D.	EPE 038 Steam Generator Tube Rupture [EPE 038 EA1.04 (4.3/4.1)] Depressurize the RCS During a SGTR	S, N, A	3
E.	045 Main Turbine Generator System [045 A2.17 (2.7*/2.9*)] Synchronize Generator On-Line with Improper Load Pickup	S, P, D, A	4S
F.	001 Control Rod Drive System [001 A4.06 (2.9/3.2)] Continuous Rod Motion During Re-alignment of Misaligned Control Rod	S, N, A	1
G.	062 AC Electrical Distribution [062 A1.03 (2.5/2.8)] Transfer Instrument Bus B to Normal Supply	S, N, A	6
H.	016 Non-Nuclear Instrumentation System [016 A4.01 (2.9/2.8)] Defeat Failed RCS Temperature Channel	S, D	7

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

	System / JPM Title	Type Code*	Safety Function
I.	APE 068 Control Room Evacuation [APE 068 AA1.01 (4.3/4.5)] Locally Operate the ARVs	P, D, E	8
J.	EPE 055 Station Blackout [EPE 055 EA2.03 (3.9/4.7)] Isolate RCP Seals	D, R, E	6
K.	APE 062 Loss of Nuclear Service Water [APE 062 AA1.07 (2.9/3.0)] Fire Water Cooling to CCW Heat Exchanger A	D, R, E	4P

@	All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for RO
(A)lternate path	4-6 (6)
(C)ontrol room	
(D)irect from bank	≤ 9 (6)
(E)mergency or abnormal in-plant	≥ 1 (3)
(EN)gineered Safety Feature	-
(L)ow-Power / Shutdown	≥ 1 (1)
(N)ew or (M)odified from bank including 1(A)	≥ 2 (5)
(P)revious 2 exams	≤ 3 (2) (Randomly Selected)
(R)CA	≥ 1 (2)
(S)imulator	

JPM Summary

JPM A This is a Bank JPM. The operator will be told that the plant was performing a normal cooldown when both RHR Pumps tripped, that the S/G's are unavailable, that AP-RHR.1, Loss of RHR, was entered and attempts to restart the pumps were unsuccessful, and that the crew is in progress of establishing Containment Integrity and all personnel are clear of Containment. The operator will be directed to lineup the RCDT pumps to provide core cooling per ER-RHR.1, Section 6.2, Closed Loop RCS to RHR Cooling. The operator will be expected to line up the RCDT pumps to provide core cooling per Section 6.2 of ER-RHR.1.

JPM B This is a new JPM. The operator will be told that the plant tripped from 100% power and safety injection has actuated, that the crew entered E-0, Reactor Trip or Safety Injection, and then transitioned to E-2, Faulted Steam Generator Isolation, and that due to a degrading transient, an Orange Path now exists on the Containment Critical Safety Function Status Tree. The operator will be directed to verify Containment Isolation and Heat Removal systems are operating as expected by performing FR-Z.1, Response to High Containment Pressure, starting from Step 1. During the course of this action, the operator will recognize that two Containment Isolation Valves have failed to automatically close, and that the Containment Spray ESFAS Signal has failed to function **(Alternate Path)**. The operator will be expected to verify Containment Isolation and Heat Removal systems are operating in accordance with Steps 1-3 of FR-Z.1. When the operator discovers that two Containment Isolation Valves have failed to close as expected the operator will close or direct that alternative valves be closed in accordance with ATT-3.0, Attachment CI/CVI. When the operator discovers that the Containment Spray ESFAS Signal has failed to function, the operator will take action to initiate Containment Spray. The operator will also start the D CNMT Recirc Fan.

JPM C This is a new JPM. The operator will be told that the Plant is at 100% power, that the crew has entered AP-CVCS.1, "CVCS Leak," completing steps 1-5, that the Pressurizer level is stable, and that the leak is believed to be in the Containment. The operator will be directed to take appropriate actions to isolate the leak starting with Step 6 of AP-CVCS.1, CVCS Leak, and control RCS inventory to maintain Pressurizer level stable. During the course of this action, the operator will recognize that both operating Charging

Pumps have tripped **(Alternate Path)**. The operator will be expected to isolate Normal Letdown and Charging, recognize that both operating Charging Pumps have tripped, use AP-CVCS.3 to restore Charging flow, and then place Excess Letdown in service.

JPM D This is a new JPM. The operator will be told that the plant has experienced a Steam Generator Tube Rupture, that the crew has completed E-3, Steam Generator Tube Rupture, through Step 17, however an attempt to re-establish Instrument Air to the Containment has been unsuccessful, that the RCS has been cooled to target temperature, and that the crew is ready to commence RCS depressurization. The operator will be directed to depressurize the RCS to minimize break flow starting with Step 18 of E-3. During the course of this action, the operator will recognize that ATT-12.0 must be used to open the Pressurizer PORV and that when the operator attempts to close the PORV following RCS depressurization, the valve fails OPEN **(Alternate Path)**. The operator will be expected to determine that Pressurizer Spray and the normal use of the Pressurizer PORVs is not possible, and use ATT-12.0 to open one Pressurizer PORV; and then close the PORV Block Valve when it is recognized that the previously opened Pressurizer PORV has failed OPEN.

JPM E This is a Bank JPM randomly selected from the last two NRC JPM Exams. The operator will be told that the generator is being started following a refueling outage, that the generator is at 1800 rpm and the turbine is fully warmed up, and that the generator output voltage is 19 KV. The operator will be directed to synchronize the generator on-line per O-1.2, Plant Startup From Hot Shutdown to Full Load, steps 6.13.1 through 6.13.10. During the course of this action, the operator will recognize that Automatic Load Pickup feature of the Turbine Control System has failed **(Alternate Path)**. The operator will be expected to synchronize the Main Generator to the Electrical Grid, and when it is recognized that the Automatic Load Pickup has failed to function, the operator will manually load the Turbine to 40 to 60 MW, without reverse powering the Main Generator.

JPM F This is a new JPM. The operator will be told that while performing control rod testing, Shutdown Bank Rod I11 failed to move inward with the rest of the Shutdown Bank rods, that I&C has successfully repaired the problem, and that ER-RCC.2, Restoring a Misaligned RCC, has been entered to restore the misaligned rod. The operator will be directed to continue rod restoration per Section 6.2 of ER-RCC.2. During the course of this action, the operator will recognize that a continuous rod insertion event is on-going **(Alternate Path)**. The operator will be expected to move the misaligned rod to a position in the core lower than its associated bank, and then realign the Shutdown Bank with the misaligned rod. While aligning the Shutdown Bank with the misaligned rod, the operator will diagnose a continuous rod insertion accident and manually trip the reactor. The operator will perform the Immediate Actions of E-0 and determine that Safety Injection is NOT required.

JPM G This is a new JPM. The operator will be told that the B Instrument Bus is on Maintenance Power supply, that maintenance has been completed and the Hold Clearance has been removed, that It is desired to restore the B Instrument Bus to its normal power supply, and that there are no failed or defeated protection channels. The operator will be directed to transfer the "B" Instrument Bus from the Maintenance power supply to the normal power supply per Section 6.3 of ER-INST.3, Instrument Bus Power Restoration. During the course of this action, the operator will recognize that Pressurizer Pressure Controller has failed to control in AUTO **(Alternate Path)**. The operator will be

expected to restore Instrument Bus B to its normal power supply in accordance with ER-INST.3, and take manual control of the Pressurizer Pressure Controller (431K) to stabilize Pressurizer pressure, when it is recognized that the controller has failed in AUTO.

JPM H This is a Bank JPM. The operator will be told that the plant was operating at 100% power when TI-402 failed high, and that the crew took all the appropriate actions and the plant is stable at current plant conditions. The operator will be directed to defeat the affected RCS temperature channel per Attachment 2 of ER-INST.1, Reactor Protection Bistable Defeat After Instrumentation Loop Failure. The operator will be expected to defeat the failed channel in accordance with ER-INST.1.

JPM I This is a Bank JPM randomly selected from the last two NRC JPM Exams. The operator will be told that the plant has experienced an uncontrollable Control Room Complex fire, that the crew performed the required actions of AP-CR.1, Control Room Inaccessibility, and transitioned to ER-FIRE.1, Alternate Shutdown For Control Complex Fire, that they are the Head Control Operator (HCO) and are equipped with a radio, that the Shift Manager is stabilizing the plant in MODE 3, and that all the required Appendix R equipment has been retrieved from the Appendix R locker outside the Control Room. The operator will be directed to locally open the A Atmospheric Relief Valve, V-3411, three turns per P-15.2, Dump Steam Through ARVs Locally. The operator will be expected to locally open the A Atmospheric Relief Valve, V-3411, three turns per P-15.2, and then isolate this valve when the handwheel will not rotate in the close direction.

JPM J This is a Bank JPM. The operator will be told that the plant was operating at 100% power when it suffered a loss of all AC power, and that the crew entered ECA-0.0, Loss of All AC Power, and is at the step that directs local isolation of RCP Seals. The operator will be directed to isolate RCP seals in accordance with the applicable portions of ATT-21.0, Attachment RCS Isolation. The operator will be expected to locally isolate the RCP Seals per ATT-21.0.

JPM K This is a Bank JPM. The operator will be told that the plant was operating a power when a loss of all Service Water pumps occurred, that the crew responded using the emergency operating procedures and Attachment 2.4, Attachment No Service Water Pumps, as required, that the crew is ready to align alternate cooling to one CCW heat exchanger, that a normal SW discharge alignment is desired, and that SW isolations MOV-4616 and MOV-4735 have been closed from the Control Room. The operator will be directed to line up Fire Water Cooling from the plant fire header to the A CCW Heat Exchanger per ER-CCW.1, Fire Water Cooling to CCW and A SFP Heat Exchangers. The operator will be expected to locally align fire water to CCW HX A in accordance with ER-CCW.1.

Facility: Ginna		Date of Examination:	12/2014
Examination Level:	SRO	Operating Test Number:	2014-301

Administrative Topic (see Note)	Type Code*	Describe activity to be performed	
Conduct of Operations	D, P, R	2.1.25 (4.2)	Ability to interpret reference materials, such as graphs, curves, tables, etc.
		JPM:	Determine Time to Boil for a Loss of Shutdown Cooling
Conduct of Operations	M, R	2.1.8 (4.1)	Ability to coordinate personnel activities outside the control room.
		JPM:	Cold Weather Walkdown
Equipment Control	N, R	2.2.12 (4.1)	Knowledge of Surveillance Procedures.
		JPM:	Perform RCS Leakrate Surveillance
Radiation Control	M, R	2.3.15 (3.1)	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
		JPM:	Evaluate the Impact of R-14 Failure on a Planned Gaseous Decay Tank Release
Emergency Procedures/Plan	M, S	2.4.41 (4.6)	Knowledge of emergency action level thresholds and classifications.
		JPM:	Classify an Emergency Event

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

*Type Codes & Criteria:

(C)ontrol room, **(0)** (S)imulator, **(1)** or Class(R)oom **(4)**
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes) **(1)**
(N)ew or (M)odified from bank (≥ 1) **(4)**
(P)revious 2 exams (≤ 1; randomly selected) **(1)**

SRO Admin JPM Summary

A1a This is a Bank JPM randomly selected from the last two NRC Exams. The operator will be provided with two sets of shutdown conditions; one current, and one projected to exist within four days. The operator will be directed to determine the Time to Boil given a Loss of RHR for each of the two sets of conditions. The operator will be expected to use IP-OUT-2, Outage Risk Management, and determine that the Time to Boil for the present plant conditions is 2.46 hours, and for the projected conditions four days from now is .43 hours (25.96 +0.1/-0.0 minutes).

A1b This is a modified Bank JPM. The operator will be told that the weekly Cold Weather Walkdown is in progress, and the Primary, Secondary and Outside Auxiliary Operators are about ready to start their Walkdown Tours; and provided with a recently completed Attachment 1, Cold Weather Temperature Log, of O-22, Cold Weather Walkdown Procedure. They will also be told that a severe lake-effect storm is occurring with 24 inches of snow having accumulated over the last four hours. The operator will be directed to review Attachment 1 and perform section 6.1, General Instructions, of O-22, and prioritize any areas of concern for the Auxiliary Operator Walkdown Tours; and then to document discrepancies, if any, and any required action(s). The operator will be expected to review the most recent plant area temperature data and then use O-22 to determine that two areas of the plant are too low in temperature. Then the operator will prioritize the actions of the Auxiliary Operators to tour these areas first, identify discrepancies and actions required in accordance with the attached KEY.

A2 This is a new JPM. The operator will be told that the plant is at 100% power with 40 gpm Letdown flow, that the VCT Level point on the PPCS is OOS, that at 1020 the VCT level was 25% when the HCO started the Day Shift RCS Leakage Monitoring Surveillance, intending to monitor for about 4 hours, and that at 1125 a VCT Level instrument failure occurred which I&C corrected and restored to operation within 10 minutes. The operator will be provided with a current Attachment 3, RCS Leakage Surveillance Record, with recent Auto Makeup information, and directed to record the data for the recent Auto Makeup on Attachment 3 AND to determine the current RCS Leakrate. The operator will also be directed to identify any Tech Spec LCOs that have been exceeded, and if any LCOs are exceeded, identify any required Tech Spec ACTION. The operator will be expected to complete Attachment 3 in accordance with the KEY provided, and identify the LCO 3.4.13, RCS Operational Leakage, is NOT met, and that the required ACTION is to reduce leakage to within limits within 4 hours (A.1).

A3 This is a modified Bank JPM. The operator will be told that the plant is at 100% power, that an A Gas Decay Tank release is being planned, that the HCO reports that R-14A, Plant Vent Gas, is de-energized for preventative maintenance that will require 2 more hours, and must be completed before it can be returned to service, and that R-14, Plant Vent Gas, has just failed LOW and I&C has indicated that it will take at least four hours to troubleshoot and correct the failure. The operator will be directed to determine the required actions based on the failure of R-14, and to determine the earliest time at which the A Gas Decay Tank release can start, and to identify any actions that will be required to start the release. The operator will be expected to determine that (1) on-going releases (i.e. ventilation) may continue provided grab samples are taken and analyzed for isotopic activity at least once per 8 hours, that (2) at least one of either R-14 or R-14A must be restored to OPERABLE status within 30 days or it must be explained in the next

Annual Radioactive Effluent Release Report, and that (3) the earliest time that the A Gas Decay Tank release can be initiated is 2 hours from now when the maintenance on R-14A is completed, and ONLY if at least two independent samples of the tank's contents are analyzed, and at least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

A4 This is a modified Bank JPM. This JPM will be given immediately following the associated Simulator Scenario. Consequently, there are five separate versions of this JPM, one each associated with the submitted Simulator Scenarios. A portion of this JPM is Time Critical. After each Simulator Scenario, the operator will be told that the sequence of events that they have just performed on the recently completed Simulator Scenario must be evaluated for any Emergency Classifications. The operator will be directed to determine the Emergency Classification that would have been made first in the recently completed scenario, and Identify the basis for this classification. The operator will be given a time that all the indications needed to classify the first event requiring an Emergency Classification were apparent in the Control Room and then directed to identify the latest time at which the Emergency Classification must have been made. Finally, the operator will be directed to identify any subsequent emergency classifications that would have been made in the Scenario, and the basis for this classification. The operator will be expected to make the appropriate classifications and identify the time frames as appropriate.

Facility: Ginna		Date of Examination:	12/2014
Exam Level (circle one): SRO(I)		Operating Test No.:	2014-301

Control Room Systems@ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U, including 1 ESF)		
System / JPM Title	Type Code*	Safety Function
A. 005 Residual Heat Removal System [005 A2.03 (2.9/3.1)] Lineup RCDT Pump for Core Cooling	S, D, L	4P
B. EPE E14 High Containment Pressure [EPE E14 EA1.1 (3.7/3.7)] Verify Containment Isolation and Heat Removal	S, N, A, EN	5
C. 004 Chemical and Volume Control System [004 A4.08 (3.8/3.4)] CVCS Leak Isolation	S, N, A	2
D. EPE 038 Steam Generator Tube Rupture [EPE 038 EA1.04 (4.3/4.1)] Depressurize the RCS During a SGTR	S, N, A	3
E. 045 Main Turbine Generator System [045 A2.17 (2.7*/2.9*)] Synchronize Generator On-Line with Improper Load Pickup	S, P, D, A	4S
F. 001 Control Rod Drive System [001 A4.06 (2.9/3.2)] Continuous Rod Motion During Re-alignment of Misaligned Control Rod	S, N, A	1
G. 062 AC Electrical Distribution [062 A1.03 (2.5/2.8)] Transfer Instrument Bus B to Normal Supply	S, N, A	6
In-Plant Systems@ (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)		
I. APE 068 Control Room Evacuation [APE 068 AA1.01 (4.3/4.5)] Locally Operate the ARVs	P, D, E	8
J. EPE 055 Station Blackout [EPE 055 EA2.03 (3.9/4.7)] Isolate RCP Seals	D, R, E	6
K. APE 062 Loss of Nuclear Service Water [APE 062 AA1.07 (2.9/3.0)] Fire Water Cooling to CCW Heat Exchanger A	D, R, E	4P

@	All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for SRO(I)
(A)lternate path	4-6 (6)
(C)ontrol room	
(D)irect from bank	≤ 8 (5)
(E)mergency or abnormal in-plant	≥ 1 (3)
(EN)gineered Safety Feature	-
(L)ow-Power / Shutdown	≥ 1 (1)
(N)ew or (M)odified from bank including 1(A)	≥ 2 (5)
(P)revious 2 exams	≤ 3 (2) (Randomly Selected)
(R)CA	≥ 1 (2)
(S)imulator	

JPM Summary

JPM A This is a Bank JPM. The operator will be told that the plant was performing a normal cooldown when both RHR Pumps tripped, that the S/G's are unavailable, that AP-RHR.1, Loss of RHR, was entered and attempts to restart the pumps were unsuccessful, and that the crew is in progress of establishing Containment Integrity and all personnel are clear of Containment. The operator will be directed to lineup the RCDT pumps to provide core cooling per ER-RHR.1, Section 6.2, Closed Loop RCS to RHR Cooling. The operator will be expected to line up the RCDT pumps to provide core cooling per Section 6.2 of ER-RHR.1.

JPM B This is a new JPM. The operator will be told that the plant tripped from 100% power and safety injection has actuated, that the crew entered E-0, Reactor Trip or Safety Injection, and then transitioned to E-2, Faulted Steam Generator Isolation, and that due to a degrading transient, an Orange Path now exists on the Containment Critical Safety Function Status Tree. The operator will be directed to verify Containment Isolation and Heat Removal systems are operating as expected by performing FR-Z.1, Response to High Containment Pressure, starting from Step 1. During the course of this action, the operator will recognize that two Containment Isolation Valves have failed to automatically close, and that the Containment Spray ESFAS Signal has failed to function **(Alternate Path)**. The operator will be expected to verify Containment Isolation and Heat Removal systems are operating in accordance with Steps 1-3 of FR-Z.1. When the operator discovers that two Containment Isolation Valves have failed to close as expected the operator will close or direct that alternative valves be closed in accordance with ATT-3.0, Attachment CI/CVI. When the operator discovers that the Containment Spray ESFAS Signal has failed to function, the operator will take action to initiate Containment Spray. The operator will also start the D CNMT Recirc Fan.

JPM C This is a new JPM. The operator will be told that the Plant is at 100% power, that the crew has entered AP-CVCS.1, "CVCS Leak," completing steps 1-5, that the Pressurizer level is stable, and that the leak is believed to be in the Containment. The operator will be directed to take appropriate actions to isolate the leak starting with Step 6 of AP-CVCS.1, CVCS Leak, and control RCS inventory to maintain Pressurizer level stable. During the course of this action, the operator will recognize that both operating Charging

Pumps have tripped **(Alternate Path)**. The operator will be expected to isolate Normal Letdown and Charging, recognize that both operating Charging Pumps have tripped, use AP-CVCS.3 to restore Charging flow, and then place Excess Letdown in service.

JPM D This is a new JPM. The operator will be told that the plant has experienced a Steam Generator Tube Rupture, that the crew has completed E-3, Steam Generator Tube Rupture, through Step 17, however an attempt to re-establish Instrument Air to the Containment has been unsuccessful, that the RCS has been cooled to target temperature, and that the crew is ready to commence RCS depressurization. The operator will be directed to depressurize the RCS to minimize break flow starting with Step 18 of E-3. During the course of this action, the operator will recognize that ATT-12.0 must be used to open the Pressurizer PORV and that when the operator attempts to close the PORV following RCS depressurization, the valve fails OPEN **(Alternate Path)**. The operator will be expected to determine that Pressurizer Spray and the normal use of the Pressurizer PORVs is not possible, and use ATT-12.0 to open one Pressurizer PORV; and then close the PORV Block Valve when it is recognized that the previously opened Pressurizer PORV has failed OPEN.

JPM E This is a Bank JPM randomly selected from the last two NRC JPM Exams. The operator will be told that the generator is being started following a refueling outage, that the generator is at 1800 rpm and the turbine is fully warmed up, and that the generator output voltage is 19 KV. The operator will be directed to synchronize the generator on-line per O-1.2, Plant Startup From Hot Shutdown to Full Load, steps 6.13.1 through 6.13.10. During the course of this action, the operator will recognize that Automatic Load Pickup feature of the Turbine Control System has failed **(Alternate Path)**. The operator will be expected to synchronize the Main Generator to the Electrical Grid, and when it is recognized that the Automatic Load Pickup has failed to function, the operator will manually load the Turbine to 40 to 60 MW, without reverse powering the Main Generator.

JPM F This is a new JPM. The operator will be told that while performing control rod testing, Shutdown Bank Rod I11 failed to move inward with the rest of the Shutdown Bank rods, that I&C has successfully repaired the problem, and that ER-RCC.2, Restoring a Misaligned RCC, has been entered to restore the misaligned rod. The operator will be directed to continue rod restoration per Section 6.2 of ER-RCC.2. During the course of this action, the operator will recognize that a continuous rod insertion event is on-going **(Alternate Path)**. The operator will be expected to move the misaligned rod to a position in the core lower than its associated bank, and then realign the Shutdown Bank with the misaligned rod. While aligning the Shutdown Bank with the misaligned rod, the operator will diagnose a continuous rod insertion accident and manually trip the reactor. The operator will perform the Immediate Actions of E-0 and determine that Safety Injection is NOT required.

JPM G This is a new JPM. The operator will be told that the B Instrument Bus is on Maintenance Power supply, that maintenance has been completed and the Hold Clearance has been removed, that It is desired to restore the B Instrument Bus to its normal power supply, and that there are no failed or defeated protection channels. The operator will be directed to transfer the "B" Instrument Bus from the Maintenance power supply to the normal power supply per Section 6.3 of ER-INST.3, Instrument Bus Power Restoration. During the course of this action, the operator will recognize that Pressurizer Pressure Controller has failed to control in AUTO **(Alternate Path)**. The operator will be

expected to restore Instrument Bus B to its normal power supply in accordance with ER-INST.3, and take manual control of the Pressurizer Pressure Controller (431K) to stabilize Pressurizer pressure, when it is recognized that the controller has failed in AUTO.

JPM I This is a Bank JPM randomly selected from the last two NRC JPM Exams. The operator will be told that the plant has experienced an uncontrollable Control Room Complex fire, that the crew performed the required actions of AP-CR.1, Control Room Inaccessibility, and transitioned to ER-FIRE.1, Alternate Shutdown For Control Complex Fire, that they are the Head Control Operator (HCO) and are equipped with a radio, that the Shift Manager is stabilizing the plant in MODE 3, and that all the required Appendix R equipment has been retrieved from the Appendix R locker outside the Control Room. The operator will be directed to locally open the A Atmospheric Relief Valve, V-3411, three turns per P-15.2, Dump Steam Through ARVs Locally. The operator will be expected to locally open the A Atmospheric Relief Valve, V-3411, three turns per P-15.2, and then isolate this valve when the handwheel will not rotate in the close direction.

JPM J This is a Bank JPM. The operator will be told that the plant was operating at 100% power when it suffered a loss of all AC power, and that the crew entered ECA-0.0, Loss of All AC Power, and is at the step that directs local isolation of RCP Seals. The operator will be directed to isolate RCP seals in accordance with the applicable portions of ATT-21.0, Attachment RCS Isolation. The operator will be expected to locally isolate the RCP Seals per ATT-21.0.

JPM K This is a Bank JPM. The operator will be told that the plant was operating a power when a loss of all Service Water pumps occurred, that the crew responded using the emergency operating procedures and Attachment 2.4, Attachment No Service Water Pumps, as required, that the crew is ready to align alternate cooling to one CCW heat exchanger, that a normal SW discharge alignment is desired, and that SW isolations MOV-4616 and MOV-4735 have been closed from the Control Room. The operator will be directed to line up Fire Water Cooling from the plant fire header to the A CCW Heat Exchanger per ER-CCW.1, Fire Water Cooling to CCW and A SFP Heat Exchangers. The operator will be expected to locally align fire water to CCW HX A in accordance with ER-CCW.1.

Facility:	**Ginna**	Scenario No.: **2**	Op Test No.: **N2014-301**
Examiners:		Operators:	(SRO)
			(RO)
			(BOP)

Initial Conditions:	The plant is at 100% power (EOL), and has been at full power for 145 days. The area has experienced thunderstorms over the last 6 hours, and this is expected to continue for the next 6 hours.
Turnover:	The following equipment is Out-Of-Service: The B CNMT Spray Pump has been declared INOPERABLE due to excessive inboard vertical vibration during STP-O-3-COMP-B surveillance test yesterday. Engineering is evaluating the data. A-52.4 submitted for ITS 3.6.6, 72 hour Action.

Event No.	Malf. No.	Event Type*	Event Description
1	CLG05 CLG02A	C-RO C(TS)-SRO	Leak on the CCW System/B CCW Pump Trips
2	PZR03B	I-BOP I-RO I(TS)-SRO	PZR Level Channel 427 fails LOW
3	TUR11B	R-RO N-BOP N-SRO	Turbine Control Valve CV-L4 Drifts Closed
4	FDW07C	C-BOP C(TS)-SRO	B FRV fails AS-IS (Manual Control Available)
5	STM03 STM05A STM05B	M-RO M-BOP M-SRO	MSLB Downstream of MSIVs with MSIVs Stuck OPEN
6	RPS05A RPS05B	C-RO C-SRO	RPS fails to Trip Rx in AUTO/Manual Pushbutton
7	STM03	NA	Steam Break Degrades Creating PTS Concern
8	EDS01A EDS01B	C-BOP C-SRO	Loss of Offsite Power

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

Ginna 2014 NRC Scenario #2

The plant is at 100% power (EOL), and has been at full power for 145 days. The area has experienced thunderstorms over the last 6 hours, and this is expected to continue for the next 6 hours.

The following equipment is Out-Of-Service: The B CNMT Spray Pump has been declared INOPERABLE due to excessive inboard vertical vibration during STP-O-3-COMP-B surveillance test yesterday. Engineering is evaluating the data. A-52.4 submitted for ITS 3.6.6, 72 hour Action.

Shortly after taking the watch, a CCW System Supply Relief Valve will lift and fail to reseat causing a 30 gpm CCW System leak. Approximately two minutes afterwards the B CCW Pump will trip, and the B CCW Pump will automatically start. The operator will respond in accordance with AR-A-17, MOTOR OFF RCP CCWP, and enter AP-CCW.2, Loss of CCW During Power Operation. The operator will address Technical Specification 3.7.7, Component Cooling Water System.

After this, Pressurizer Level Channel 427 will fail LOW, resulting in letdown isolation and de-energizing the pressurizer heaters. The crew will respond per AR-F-11, PZR LOW LEVEL 13%, and ER-INST.1, Reactor Protection Bistable Defeat After Instrumentation Loop Failure. They will defeat the failed channel, reset PZR heaters, reduce charging to a single charging pump, and re-establish letdown per S-3.2.E, Placing In or Removing From Service Normal Letdown/Excess Letdown. The crew will start a second charging pump and slowly restore PZR level to program (52%). The operator will address Technical Specification 3.3.1, Reactor Trip System (RTS) Instrumentation and 3.4.9, Pressurizer.

Then, turbine control valve CVL-4 will drift closed. The crew will respond per AP-TURB.2, Turbine Load Rejection, begin a load reduction to less than 50% power using AP-TURB.5, Rapid Load Reduction.

During the load reduction, a failure of the B FRV to control in AUTO. The operator will respond per AR-G-5, S/G/ B LEVEL DEVIATION ±7%, or upon observing an abnormally high level in the B Steam Generator and control the B FRV manually. The operator will address Technical Specification 3.7.3, Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), and Associated Bypass Valves.

Subsequently, a major steamline break will occur in the Turbine Building downstream of the MSIVs. Both MSIVs will fail to close automatically in response to manual attempts from the MCB. The reactor will fail to trip automatically and manual pushbutton will not work, requiring opening of normal supply breakers for Bus 13 and Bus 15 to de-energize rod drive MG sets to trip the reactor. The crew will enter E-0, Reactor Trip or Safety Injection.

The crew will ultimately transition to E-2, Faulted S/G Isolation. Since both MSIVs are failed open, and with a downstream break, both SGs will be affected and their pressures will lower concurrently. Eventually, the crew will transition to ECA-2.1, Uncontrolled Depressurization of Both Steam Generators. After the operator throttles AFW flow in ECA-2.1, the leak will increase in magnitude, resulting in an Orange Path on RCS

Integrity. Simultaneously the event will be complicated by a Loss of Offsite Power. Both Emergency Diesel Generators will start and automatically repower the respective Emergency Busses, and the operator will need to re-establish AFW flow of 50 gpm to each Steam Generator.

The crew will implement FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, when RCS cold leg temperatures reach 311°F. In FR-P.1 the crew will stop the SI pumps and the RHR pumps.

The scenario will be terminated after the SI and RHR pumps are secured.

Critical Tasks:

1. Manually trip the reactor from the control room before transition to FR-S.1

Safety Significance: Failure to manually trip the reactor causes a challenge to the subcriticality CSF beyond that irreparably introduced by the postulated conditions. Additionally, it constitutes an "incorrect performance that necessitates the crew taking action which complicates the event mitigation strategy that demonstrates the inability by the crew to recognize a failure of the automatic actuation of the RPS.

2. Control the AFW flowrate to 50 gpm per SG in order to minimize the RCS Cooldown rate

Safety Significance: Failure to control the AFW flow rate to the SGs when able to do so causes a challenge to the subcriticality and RCS Integrity CSFs beyond that irreparably introduced by the postulated conditions. Additionally, it constitutes an "incorrect performance that may necessitate the crew taking action which complicates the event mitigation strategy (Transition to FR-S.1 on inability to maintain a negative SUR).

Facility: **Ginna**		Scenario No.: **3**	Op Test No.: **N2014-301**
Examiners:	______	Operators:	______ (SRO)
	______		______ (RO)
	______		______ (BOP)

Initial Conditions:	The plant is at 48% power (MOL). The plant power was reduced several days ago due to a malfunction on the A MFW Pump. Corrective Maintenance has been completed, and the pump is running for retest. RG&E Energy Control Center has requested that the electric plant be aligned to a 0/100 configuration on circuit 7T to allow the RG&E personnel to perform an insulator inspection on the 767 Line. The 767 line will remain OPERABLE throughout the inspection.
Turnover:	The following equipment is Out-Of-Service: The B AFW Pump is OOS for Bearing Replacement. A-52.4 submitted for ITS 3.7.5, 7 day Action. MRPI for Rod G3 indicates on the bottom, however, this is a confirmed instrumentation problem (I&C is working).

Event No.	Malf. No.	Event Type*	Event Description
1	NA	N-BOP N-SRO	Shift Electric Plant
2	EDS07B	C-RO C-BOP C(TS)-SRO	Loss of B Instrument Bus
3	PZR02D	I-RO I-BOP I(TS)-SRO	Pressurizer Pressure (PT-449) fails HIGH
4	TUR05E	R-RO N-BOP N-SRO	Main Turbine High Vibration/Downpower
5	CND03A	C-BOP C-SRO	Hotwell Level Transmitter fails HIGH
6	FDW09A	M-RO M-BOP M-SRO	Feed Line Rupture Inside Containment/Turbine fails to Auto Trip
7	SIS02A SIS02B	C-RO C-SRO	Safety Injection fails to actuate Automatically
8	RPS07K REM FDW32 OVR FDW42A FDW15B	C-BOP C-SRO	A AFW Pump to Auto Start/Trips after Manual Start/TDAFW Pump trips on Overspeed/ Standby AFW fails to function

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

Ginna 2014 NRC Scenario #3

The plant is at 48% power (MOL). The plant power was reduced several days ago due to a malfunction on the A MFW Pump. Corrective Maintenance has been completed, and the pump is running for retest. RG&E Energy Control Center has requested that the electric plant be aligned to a 0/100 configuration on circuit 7T to allow the RG&E personnel to perform an insulator inspection on the 767 Line. The 767 line will remain OPERABLE throughout the inspection.

The following equipment is Out-Of-Service: The B AFW Pump is OOS for Bearing Replacement. A-52.4 submitted for ITS 3.7.5, 7 day Action. MRPI for Rod G3 indicates on the bottom, however, this is a confirmed instrumentation problem (I&C is working).

Shortly after taking the watch, the operator will shift the Electric Line-up from 50/50 to 0/100 in accordance with O-6.9.2, Establishing and/or Transferring Offsite Power to Bus 12A/12B.

Shortly afterwards, a loss of the B Instrument Bus will occur. The operator will respond in accordance with AR-E-14, LOSS B INSTR. BUS. Power will be restored to the bus per guidance in ER-INST.3, Instrument Bus Power Restoration, which will include the isolation and re-establishment of Normal Letdown in accordance with S-3.2E, Placing In or Removing From Service Normal Letdown/Excess Letdown. The operator will address AR-K-32, CV AIR DRYER LOW PRESS SG B/D TANK HIGH LEVEL, while restoring from the transient. The operator will address Technical Specification 3.8.7, AC Instrument Bus Sources Modes 1-4, and 3.8.9, Distribution Systems – Modes 1, 2, 3 and 4.

After this, the controlling Pressurizer Pressure Transmitter will fail HIGH, causing the Spray Valves to open. The operator will respond in accordance with AR-F-2, PRESSURIZER HIGH PRESS 2310 PSI and AR-F-10, PRESSURIZER LO PRESS 2205 PSI, and enter AP-PZR.1, Abnormal PZR Pressure. AP-PZR.1 will refer the operator to ER-INST.1, Reactor Protection Bistable Defeat After Instrumentation Loop Failure, for the defeat of PT-449. The operator will address Technical Specification 3.4.1, RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits; 3.3.1, Reactor Trip System (RTS) Instrumentation; and TRM-3.4.3, Anticipated Transient Without Scram (ATWS) Mitigation.

Following this, a turbine high vibration condition on Bearing #5 will develop within about 3 minutes. The operator will respond in accordance with AR-I-27, ROTOR ECCENTRICITY OR VIBRATION; and enter AP-TURB.3, Turbine Vibration; and then AP-TURB.5, Rapid Downpower. The operator will need to lower the Turbine Load to stabilize the vibrations. During the load reduction Hotwell Level Transmitter will fail HIGH. The operator will need to manually control the Hotwell Level Controller.

Subsequently, a feed line rupture inside Containment will occur. The Reactor will trip however, the Turbine will fail to trip automatically and Safety Injection will fail to actuate automatically. The operator will enter E-0, Reactor Trip or Safety Injection. On the reactor trip the A AFW Pump will fail to Autostart, then trip after it is manually started; and the TDAFW Pump will trip on overspeed. The operator will transition from E-0 to FR-H.1, Response to a Loss of Secondary Heat Sink.

The operator will unsuccessfully attempt to place the Standby AFW System in service, and then attempt to restore a Secondary Heat Sink using the MFW System. Once the Secondary Heat Sink is re-established using MFW, the operator will transition back to E-0, and then transition to E-2, Faulted Steam Generator Isolation.

The scenario will terminate at Step 9 of E-2, after the crew has determined that a transition to E-1, Loss of Reactor or Secondary Coolant, is required.

Critical Tasks:

1. **Manually actuate Safety Injection before transition out of E-0 into ES-0.1**

Safety Significance: Failure to actuate Safety Injection when it is required to be actuated, and can be actuated, violates the assumptions of the Safety Analysis and constitutes incorrect performance that could lead to misdiagnosis of the event, implementation of an incorrect mitigation strategy and ultimately degradation of the RCS and/or fuel cladding fission product barriers.

2. **Establish feedwater flow into at least one Steam Generator before RCS Bleed and Feed is required.**

Safety Significance: Failure to establish feedwater flow into at least one Steam Generator results in the crew having to rely upon the lower-priority action of having to initiate RCS Bleed and Feed to minimize the possibility of core uncovery. Failure to perform this task, when able to do so, constitutes incorrect performance that leads to degradation of the RCS and/or fuel cladding fission product barriers.

Facility:	**Ginna**	Scenario No.: **4**	Op Test No.: **N2014-301**
Examiners:	____________	Operators:	____________ (SRO)
	____________		____________ (RO)
	____________		____________ (BOP)

Initial Conditions:	The plant is at 48% power (MOL). Power has been held at this level for approximately four days while corrective maintenance is performed on the A MFW Pump. Maintenance has just been completed. Chemistry has requested that Letdown flow be raised from 40 gpm to 60 gpm. Also, it is expected to swap EHC Pumps per P-17 for routine equipment rotation. The area has experienced severe weather over the last 6 hours, and this is expected to continue for the next 6 hours.
Turnover:	The following equipment is Out-Of-Service: The B AFW Pump is OOS for Bearing Replacement. Steam Flow channel FT-475 is OOS. The channel has been defeated per ER-INST.1.

Event No.	Malf. No.	Event Type*	Event Description
1	NA	N-RO N-SRO	Swap Letdown Orifice (40 gpm to 60 gpm)
2	NA	N-BOP N-SRO	Alternate EHC Supply Pumps
3	NIS06A CVC07A CVC23	C-RO I-BOP I(TS)-SRO	Power Range N41 Upper Detector fails LOW/Letdown Pressure Controller Failure
4	PZR05B	C-RO C(TS)-SRO	PORV Leak/Block Valve Failure
5	TUR09D	R-RO C-BOP C-SRO	Downpower/Failure of Turbine Control/EHC
6	PZR07	M-RO M-BOP M-SRO	Pzr Steam Space Break
7	RPS11-A3	C-RO C-SRO	Seal Water Return Isolation CIV Fails in AUTO

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

Ginna 2014 NRC Scenario #4

The plant is at 48% power (MOL). Power has been held at this level for approximately four days while corrective maintenance is performed on the A MFW Pump. Maintenance has just been completed. Chemistry has requested that Letdown flow be raised from 40 gpm to 60 gpm. Also, it is expected to swap EHC Pumps per P-17 for routine equipment rotation. The area has experienced severe weather over the last 6 hours, and this is expected to continue for the next 6 hours.

The following equipment is Out-Of-Service: The B AFW Pump is OOS for Bearing Replacement. Steam Flow channel FT-475 is OOS. The channel has been defeated per ER-INST.1.

Shortly after taking the watch, the operator will swap Letdown Orifice Control valves in accordance with S-3.2P, Swapping CVCS Letdown Orifices.

After this, the operator will alternate the EHC Supply Pump alignment in accordance with P-17, Operations Control Room Operating Instructions, Attachment 12, CROI-12 Swapping Electro-Hydraulic Pumps.

Then, the N41 Power Ranger Upper Detector will fail Low. The operator will respond in accordance with AR-E-26, POWER RANGE CHANNEL DEV ±2%, and/or AR-E-28, POWER RANGE ROD DROP ROD STOP -5%/5SEC, and then enter ER-NIS.3, PR Malfunction. The operator will address Technical Specification 3.3.1, Reactor Trip Instrumentation. While the CO is defeating the failed Power Range Channel, a failure of the Letdown Pressure Controller will cause PCV-135 to fail OPEN. The operator will need to take manual control of the failed controller, and control Letdown pressure manually.

Subsequently, Pressurizer PORV-431C will fail partially open. The operator will respond in accordance with AR-F-18, PRZR PORV OUTLET HI TEMP 145°F, and enter AP-PZR.1, Abnormal Pressurizer Pressure. When the operator attempts to isolate the PORV, the Block Valve will fail to shut fully resulting in a 2-5 gpm leak into the PRT. The crew may implement AP-RCS.1, Reactor Coolant Leak, and prepare to make a Containment entry. Ultimately, the crew will be directed to take the unit off-line. The operator will address Technical Specification 3.4.11, Pressurizer PORVs, 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, as well as TRM 3.4.3, Anticipated Transients Without Scram (ATWS) Mitigation. Although the crew will address Technical Specification 3.4.13, RCS Leakage, it is likely that at the current leakrate, this specification will be met.

The operator will take the unit off line in accordance with AP-TURB.5, Rapid Load Reduction During the downpower, the Main Turbine will fail in automatic control, and shift to manual control. The operator will identify that the load reduction has been stopped, and use manual control of the turbine to restart and continue the downpower. The remainder of the downpower will need to be accomplished using manual control of the Turbine.

When the plant is at approximately 40% power, the piping leading to the failed PORV will fail so that a vapor space Small Break LOCA occurs. The plant will trip and Safety

Injection will be actuated, and the operator will enter E-0, Reactor Trip or Safety Injection. On the plant trip Containment Isolation Valve MOV-313, the Seal Water Return Line CIV, will fail to close automatically, requiring that the operator manually close this valve.

The operator will transition from E-0 to E-1, Loss of Reactor or Secondary Coolant, and then into ES-1.2 Post-LOCA Cooldown and Depressurization. While in ES-1.2, an Orange Path may occur on the RCS Integrity Critical Safety Function Status Tree.

The scenario will terminate at Step 14 of ES-1.2, after the crew has demonstrated the ability to evaluate/perform the SI flow reduction sequence, or upon transition to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition.

Critical Tasks:

1. Trip all RCPs within 5 minutes of reaching trip criteria

Safety Significance: Failure to trip all RCPs when required can lead to core uncovery and to fuel temperatures in excess of 2200°F. Analyses have shown that if the RCPs are tripped within 5 minutes of the trip criteria being met, PCT will remain below 2200 °F, and if this action is delayed beyond 5 minutes, this PCT will be exceeded. It is a management expectation that the RCPs be tripped as quickly as possible, but within 5 minutes when the trip criteria is met. Failure to take this action represents mis-operation by the operator which leads to degradation of the fuel cladding fission produce barrier, and a violation of a license condition.

2. Close the Seal Water Return Containment Isolation Valve before transition out of E-0.

Safety Significance: Failure to close at least one Containment Isolation Valve on each critical penetration under the postulated conditions when it is possible to do so, constitutes mis-operation leading to degradation of the Containment Barrier. Failure to take this action leads to an unnecessary release of fission products to the auxiliary building, increasing the potential for release to the environment, and reducing accessibility to vital equipment within the Auxiliary Building. Higher radiation levels within the Auxiliary Building will result in a degradation of ALARA principles.

Facility: **Ginna**		Scenario No.: **5**	Op Test No.: **N2014-301**
Examiners:		Operators:	(SRO)
			(RO)
			(BOP)

Initial Conditions:	The plant is at 1 x 10-8 amps (BOL). The plant ran at 100% power for 12 days, and then tripped four days ago due to a MFW Pump failure. The repairs have been made and the plant is ready to be started back up. The crew will be directed to pull rods to the point of adding heat (POAH), and start the A MFW Pump in accordance with O-1.2, Plant Startup From Hot Shutdown to Full Power Load, Step 6.3.4 and beyond. The area has experienced thunderstorms over the last 6 hours, and this is expected to continue for the next 6 hours.
Turnover:	The following equipment is Out-Of-Service: The B Condensate Pump is OOS for Bearing Replacement.

Event No.	Malf. No.	Event Type*	Event Description
1	NA	R-RO N-SRO	Raise Power to POAH
2	NA	N-BOP N-SRO	Testing of the MFW Oil Pumps
3	CLG10 CLG02A A-EDS33	C-RO C(TS)-SRO	480VAC Ground/A CCW Pump trips w/B CCW Pump failure to start in AUTO
4	EDS04D CLG01D	C-BOP C(TS)-SRO	Fault on 480V Bus 18/SW Pump D fails to start
5	CVC07A	C-RO C-SRO	PCV-135 fails CLOSED
6	NIS05A	C-BOP C(TS)-SRO	Loss of Compensating Voltage to Intermediate Range N35
7	STM04D STM09A	M-RO M-BOP M-SRO	B ARV fails CLOSED/B SG Safety Valve Lifts and fails OPEN
8	RPS05A RPS05B	C-RO C-SRO	Failure of Reactor to Trip in AUTO only
9	SGN04B RCS16	C-BOP C-SRO	B Steam Generator Tube Rupture/1% Failed Fuel

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

Ginna 2014 NRC Scenario #5

The plant is at 1 x 10-8 amps (BOL). The plant ran at 100% power for 12 days, and then tripped four days ago due to a MFW Pump failure. The repairs have been made and the plant is ready to be started back up. The crew will be directed to pull rods to the point of adding heat (POAH), and start the B MFW Pump in accordance with O-1.2, Plant Startup From Hot Shutdown to Full Power Load, Step 6.3.4 and beyond. The area has experienced thunderstorms over the last 6 hours, and this is expected to continue for the next 6 hours.

The following equipment is Out-Of-Service: The B Condensate Pump is OOS for Bearing Replacement.

Shortly after taking the watch, the operator will raise power to the POAH in accordance with Step 6.3.4 of O-1.2, Plant Startup From Hot Shutdown to Full Load. The operator will raise reactor power and stabilize reactor power at 0.5-1.0%; and then control Tavg at 547°F.

After this, the operator will conduct the oil pump checks on the A Main Feedwater (MFW) Pump accordance with Attachment 5, Main Feed Pump A Startup of O-1.2, Plant Startup From Hot Shutdown to Full Load, in preparation for starting the A MFW Pump.

Subsequently, a 480VAC ground will occur on Bus 14 and the A CCW Pump will trip, and the B CCW Pump will fail to automatically start. The operator will either start the A CCW pump manually per A-503.1, Emergency and Abnormal Operating Procedures Users Guide; or respond in accordance with AR-A-22, CCW PUMP DISCHARGE LO PRESS 60 PSI, and enter AP-CCW.2, Loss of CCW During Power Operation. The operator will evaluate Technical Specification 3.7.7, CCW System.

Shortly after this, a fault on 480V Bus 18 will occur, resulting in Bus 18 de-energizing. The operator will respond in accordance with AR-L-23, BUS18 UNDER VOLTAGE SAFEGUARDS and/or AR-L-5, SAFEGUARD BUS MAIN BREAKER OVERCURRENT TRIP and enter AP-ELECT.17/18, Loss of Safeguards Bus 17/18. The D Service Water Pump has failed to start, leaving only the B SW Pump running. The operator may leave the A EDG running or trip it within AP-ELECT.17/18, but in either case align Alternate Cooling to the EDG. The operator will enter AP-SW.2, Loss of Service Water, and take actions to isolate non-essential SW loads. The operator will address Technical Specification 3.8.1, AC Sources – Modes 1, 2, 3, and 4; 3.8.9, Distribution Systems – Modes 1, 2, 3,and 4; and 3.7.8, Service Water System.

During the recovery, Letdown Pressure Control Valve PCV-135 will fail closed causing the Letdown Line Relief valve to lift to the PRT. The operator will respond in accordance with AR-A-11, LETDOWN LINE HI PRESS 400 PSI, and take manual control of the valve.

Shortly afterwards, the compensating voltage power supply for the Intermediate Range Nuclear Instrument N35 will fail Low. The operator will respond in accordance with AR-E-9, IR N-35 LOSS OF COMPENSATING VOLTAGE and enter ER-NIS.2, IR MALFUNTION. The operator will address Technical Specification 3.3.1, Reactor Trip Instrumentation.

After this, the B ARV will inadvertently close, and one SG Safety valve will open and stick open. The reactor will fail to trip automatically, and the operator will need to manually trip the reactor and enter E-0, Reactor Trip or Safety Injection. On the trip approximately 1% failed fuel will occur. The operator will proceed through E-0, and then transition to E-2, Faulted Steam Generator Isolation.

While in E-2 a large Steam Generator Tube Rupture will develop on the B Steam Generator, and the operator will transition to E-3, Steam Generator Tube Rupture. Because the B Steam Generator is both Ruptured and Faulted, the operator will transition to ECA-3.1, SGTR with Loss of Reactor Coolant, Subcooled Recovery Desired.

Also, while in E-2, it is expected that high temperatures will occur on the RCPs due to low service water flow, requiring the crew to stop these pumps. Because of this the NC condition of the RCS will tend to lower loop Tcold temperatures causing an Orange/Red condition on RCS Integrity. If so, the operator may need to transition to FR-P.1, Response to Imminent Pressurized Thermal Shock Condition.

The scenario will terminate at Step 15 of ECA-3.1, after the crew has monitored conditions for continuing with the Subcooled Recovery procedure, or upon transition to FR-P.1.

Critical Tasks:

1. **Manually trip the reactor prior to transition to FR-S.1, "Response to Nuclear Generation/ATWS."**

Safety Significance: Failure to trip the reactor when required causes a challenge to the Subcriticality Critical Safety Function that otherwise would not exist. This mis-operation by the operator necessitates the crew taking compensating action which complicates the event mitigation strategy and demonstrates an inability by the operator to recognize a failure of the automatic actuation of the RPS.

2. **Isolate the Faulted Steam Generator before transitioning out of E-2.**

Safety Significance: Failure to isolate a Faulted SG that can be isolated causes challenges to the Critical Safety Functions that would not otherwise occur. Failure to isolate flow could result in an unwarranted Orange or Red Path condition on RCS Integrity, Subcriticality (if cooldown is allowed to continue uncontrollably).