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R.E. GINNA
NUCLEAR POWER PLANT

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June 28, 2012

Mr. Peter Presby
U.S. NRC Region I
2100 Renaissance Boulevard
Renaissance Park
King of Prussia, PA 19406

Subject: **R.E. Ginna Nuclear Power Plant**
Docket No. 05000244

October 2012 Initial License Examination Outlines

Dear Mr. Presby:

Enclosed please find the examination outlines (written, dynamic simulator, and walk-through) for the R.E. Ginna Nuclear Power Plant initial license examination to be administered during the week of October 1, 2012. The outlines were constructed by the Ginna Operations Training exam team utilizing the guidance provided in NUREG-1021, Rev. 9, Supplement 1 and NUREG-1122, Rev. 2, Supplement 1. The examination outlines were independently reviewed and approved by a licensed SRO Operations Training supervisor.

Additional information regarding the methodology for developing the written exam outline is also included within the attachments.

In accordance with 10CFR55.49 and NUREG-1021, Section ES-201, Attachment 1, these materials shall be withheld from public disclosure until after the examinations are complete.

Sincerely,

Mark Geckle
Manager, Nuclear Training

ATTACHMENTS

Examination Outline Quality Checklist	Form ES-201-2
PWR Examination Outline (RO)	Form ES-401-2
PWR Examination Outline (SRO)	Form ES-401-2
Control Room/In-Plant Systems Outline (RO)	Form ES-301-2
Administrative Topics Outline (RO)	Form ES-301-1
Control Room/In-Plant Systems Outline (SROI)	Form ES-301-2
Administrative Topics Outline (SROI)	Form ES-301-1
Control Room/In-Plant Systems Outline (SROU)	Form ES-301-2
Administrative Topics Outline (SROU)	Form ES-301-1
Scenario # 1 Outline	Form ES-D-1
Scenario # 2 Outline	Form ES-D-1
Scenario # 3 Outline	Form ES-D-1
Scenario # 4 Outline	Form ES-D-1
Transient and Event Checklist	Form ES-301-5
Competencies Checklist	Form ES-301-6
Ginna 2012 JPM Summaries	
Ginna 2012 Written Exam Outline Methodology	
NKEG Software Random Selection Methodology (excerpt from Vendor Manual)	
Ginna Suppressed K/A List (2012)	

Facility: Robert E Ginna Printed: 06/28/2012

Date Of Exam: 10/01/2012

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18	0	0	0
	2	1	2	1				2	2			1	9	0	0	0
	Tier Totals	4	5	4				5	5			4	27	0	0	0
2. Plant Systems	1	3	2	3	3	2	2	3	3	2	2	3	28	0	0	0
	2	1	1	0	1	1	1	1	1	1	1	1	10	0 0	0	0
	Tier Totals	4	3	3	4	3	3	4	4	3	3	4	38	0	0	0

3. Generic Knowledge And Abilities Categories	RO 1	RO 2	RO 3	RO 4	RO Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	2	3	3	2	10	0	0	0	0	0

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

PWR RO Examination Outline

Printed: 06/28/2012

Facility: Robert E Ginna

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000007 Reactor Trip - Stabilization - Recovery / 1	X						EK1.02 - Shutdown margin	3.4	1
000009 Small Break LOCA / 3		X					EK2.03 - S/Gs	3.0	1
000011 Large Break LOCA / 3			X				EK3.13 - Hot-leg injection/recirculation	3.8	1
000015/000017 RCP Malfunctions / 4				X			AA1.08 - S/G LCS	3.0*	1
000022 Loss of Rx Coolant Makeup / 2				X			AA1.06 - CVCS charging pump ammeters and running indicators	2.9	1
000025 Loss of RHR System / 4						X	2.4.41 - Knowledge of the emergency action level thresholds and classifications.	2.9	1
000026 Loss of Component Cooling Water / 8			X				AK3.04 - Effect on the CCW flow header of a loss of CCW	3.5	1
000027 Pressurizer Pressure Control System Malfunction / 3		X					AK2.03 - Controllers and positioners	2.6	1
000038 Steam Gen. Tube Rupture / 3	X						EK1.01 - Use of steam tables	3.1	1
000040 Steam Line Rupture - Excessive Heat Transfer / 4		X					AK2.02 - Sensors and detectors	2.6*	1
000054 Loss of Main Feedwater / 4				X			AA1.03 - AFW auxiliaries, including oil cooling water supply	3.5	1
000056 Loss of Off-site Power / 6						X	2.2.44 - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	1
000057 Loss of Vital AC Inst. Bus / 6			X				AK3.01 - Actions contained in EOP for loss of vital ac electrical instrument bus	4.1	1
000062 Loss of Nuclear Svc Water / 4						X	2.4.11 - Knowledge of abnormal condition procedures.	4.0	1
000077 Generator Voltage and Electric Grid Disturbances / 6					X		AA2.06 - Generator frequency limitations	3.4	1
W/E04 LOCA Outside Containment / 3					X		EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations	3.4	1
W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	X						EK1.1 - Components, capacity, and function of emergency systems	3.8	1
W/E11 Loss of Emergency Coolant Recirc. / 4					X		EA2.2 - Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	3.4	1

Facility: Robert E Ginna

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
K/A Category Totals:	3	3	3	3	3	3	Group Point Total:		18

PWR RO Examination Outline

Printed: 06/28/2012

Facility: Robert E Ginna

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000003 Dropped Control Rod / 1		X					AK2.05 - Control rod drive power supplies and logic circuits	2.5	1
000005 Inoperable/Stuck Control Rod / 1					X		AA2.03 - Required actions if more than one rod is stuck or inoperable	3.5	1
000036 Fuel Handling Accident / 8			X				AK3.03 - Guidance contained in EOP for fuel handling incident	3.7	1
000037 Steam Generator Tube Leak / 3						X	2.2.38 - Knowledge of conditions and limitations in the facility license.	3.6	1
000059 Accidental Liquid RadWaste Rel. / 9		X					AK2.02 - Radioactive-gas monitors	2.7	1
W/E01 Rediagnosis / 3					X		EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations	3.2	1
W/E10 Natural Circ. / 4	X						EK1.3 - Annunciators and conditions indicating signals, and remedial actions associated with the Natural Circulation with Steam Void in Vessel with/without RVLIS	3.3	1
W/E15 Containment Flooding / 5				X			EA1.2 - Operating behavior characteristics of the facility	2.7	1
W/E16 High Containment Radiation / 9				X			EA1.1 - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	3.1	1
K/A Category Totals:	**1**	**2**	**1**	**2**	**2**	**1**	**Group Point Total:**		**9**

PWR RO Examination Outline

Facility: Robert E Ginna

ES - 401 | **Plant Systems - Tier 2 / Group 1** | **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
003 Reactor Coolant Pump						X						K6.14 - Starting requirements	2.6	1
004 Chemical and Volume Control	X											K1.07 - NIS	2.6	1
005 Residual Heat Removal				X								K4.03 - RHR heat exchanger bypass flow control	2.9	1
006 Emergency Core Cooling								X				A2.04 - Improper discharge pressure	3.4	1
007 Pressurizer Relief/Quench Tank											X	2.2.44 - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	1
007 Pressurizer Relief/Quench Tank							X					A1.03 - Monitoring quench tank temperature	2.6	1
008 Component Cooling Water											X	2.2.44 - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	1
008 Component Cooling Water							X					A1.03 - CCW pressure	2.7	1
010 Pressurizer Pressure Control		X										K2.02 - Controller for PZR spray valve	2.5	1
012 Reactor Protection						X						K6.02 - Redundant channels	2.9	1
013 Engineered Safety Features Actuation					X							K5.02 - Safety system logic and reliability	2.9	1
013 Engineered Safety Features Actuation								X				A2.04 - Loss of instrument bus	3.6	1
022 Containment Cooling											X	2.2.44 - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	1
022 Containment Cooling				X								K4.01 - Cooling of containment penetrations	2.5*	1
026 Containment Spray			X									K3.01 - CCS	3.9	1
039 Main and Reheat Steam			X									K3.06 - SDS	2.8*	1
059 Main Feedwater				X								K4.17 - Increased feedwater flow following a reactor trip	2.5*	1
061 Auxiliary/Emergency Feedwater					X							K5.03 - Pump head effects when control valve is shut	2.6	1

PWR RO Examination Outline

Printed: 06/28/2012

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ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
062 AC Electrical Distribution	X											K1.04 - Off-site power sources	3.7	1
063 DC Electrical Distribution									X			A3.01 - Meters, annunciators, dials, recorders, and indicating lights	2.7	1
063 DC Electrical Distribution							X					A1.01 - Battery capacity as it is affected by discharge rate	2.5	1
064 Emergency Diesel Generator		X										K2.02 - Fuel oil pumps	2.8*	1
073 Process Radiation Monitoring								X				A2.02 - Detector failure	2.7	1
076 Service Water	X											K1.17 - PRMS	3.6*	1
078 Instrument Air									X			A3.01 - Air pressure	3.1	1
078 Instrument Air										X		A4.01 - Pressure gauges	3.1	1
103 Containment										X		A4.03 - ESF slave relays	2.7*	1
103 Containment			X									K3.03 - Loss of containment integrity under refueling operations	3.7	1
K/A Category Totals:	3	2	3	3	2	2	3	3	2	2	3	**Group Point Total:**		28

PWR RO Examination Outline

Printed: 06/28/2012

Facility: Robert E Ginna

ES - 401 | **Plant Systems - Tier 2 / Group 2** | **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
001 Control Rod Drive								X				A2.14 - Urgent failure alarm, including rod-out-of-sequence and motion-inhibit alarms	3.7	1
011 Pressurizer Level Control						X						K6.05 - Function of PZR level gauges as postaccident monitors	3.1	1
014 Rod Position Indication	X											K1.01 - CRDS	3.2*	1
015 Nuclear Instrumentation		X										K2.01 - NIS channels, components, and interconnections	3.3	1
033 Spent Fuel Pool Cooling											X	2.2.38 - Knowledge of conditions and limitations in the facility license.	3.6	1
045 Main Turbine Generator							X					A1.06 - Expected response of secondary plant parameters following T/G trip	3.3	1
071 Waste Gas Disposal					X							K5.04 - Relationship of hydrogen/oxygen concentrations to flammability	2.5	1
072 Area Radiation Monitoring									X			A3.01 - Changes in ventilation alignment	2.9*	1
079 Station Air										X		A4.01 - Cross-tie valves with IAS	2.7	1
086 Fire Protection				X								K4.02 - Maintenance of fire header pressure	3.0	1
K/A Category Totals:	**1**	**1**	**0**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**Group Point Total:**		**10**

Generic Knowledge and Abilities Outline (Tier 3)

PWR RO Examination Outline

Printed: 06/28/2012

Facility: Robert E Ginna

Form ES-401-3

Generic Category	KA	KA Topic	Imp.	Points
Conduct of Operations	2.1.17	Ability to make accurate, clear, and concise verbal reports.	3.9	1
	2.1.20	Ability to interpret and execute procedure steps.	4.6	1
		Category Total:		**2**
Equipment Control	2.2.14	Knowledge of the process for controlling equipment configuration or status.	3.9	1
	2.2.20	Knowledge of the process for managing troubleshooting activities.	2.6	1
	2.2.38	Knowledge of conditions and limitations in the facility license.	3.6	1
		Category Total:		**3**
Radiation Control	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions.	3.5	1
	2.3.11	Ability to control radiation releases.	3.8	1
	2.3.13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	3.4	1
		Category Total:		**3**
Emergency Procedures/Plan	2.4.13	Knowledge of crew roles and responsibilities during EOP usage.	4.0	1
	2.4.23	Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.	3.4	1
		Category Total:		**2**

Generic Total: **10**

Facility: Robert E Ginna Printed: 06/28/2012

Date Of Exam: 10/01/2012

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	0	0	0	N/A			0	0	N/A		0	0	3	3	6
	2	0	0	0				0	0			0	0	2	2	4
	Tier Totals	0	0	0				0	0			0	0	5	5	10
2. Plant Systems	1	0	0	0	0	0	0	0	0	0	0	0	0	3	2	5
	2	0	0	0	0	0	0	0	0	0	0	0	0	0	2	1	3
	Tier Totals	0	0	0	0	0	0	0	0	0	0	0	0	5	3	8

3. Generic Knowledge And Abilities Categories	RO 1	RO 2	RO 3	RO 4	RO Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	0	0	0	0	0	1	2	2	2	7

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2 (Note #1 does not apply). Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

PWR SRO Examination Outline

Printed: 06/28/2012

Facility: Robert E Ginna

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000008 Pressurizer Vapor Space Accident / 3					X		AA2.10 - High-pressure injection valves and controllers	3.6	1
000029 ATWS / 1						X	2.1.20 - Ability to interpret and execute procedure steps.	4.6	1
000055 Station Blackout / 6						X	2.1.25 - Ability to interpret reference materials, such as graphs, curves, tables, etc.	4.2	1
000058 Loss of DC Power / 6						X	2.4.20 - Knowledge of operational implications of EOP warnings, cautions, and notes.	4.3	1
000065 Loss of Instrument Air / 8					X		AA2.07 - Whether backup nitrogen supply is controlling valve position	3.2*	1
W/E12 - Steam Line Rupture - Excessive Heat Transfer / 4					X		EA2.2 - Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	3.9	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**3**	**3**	**Group Point Total:**		**6**

PWR SRO Examination Outline

Printed: 06/28/2012

Facility: Robert E Ginna

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000068 Control Room Evac. / 8						X	2.4.6 - Knowledge of EOP mitigation strategies.	4.7	1
000076 High Reactor Coolant Activity / 9						X	2.2.25 - Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4.2	1
W/E07 Inad. Core Cooling / 4					X		EA2.2 - Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	3.9	1
W/E14 Loss of CTMT Integrity / 5					X		EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations	3.8	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**2**	**2**	**Group Point Total:**		**4**

PWR SRO Examination Outline

Printed: 06/28/2012

Facility: Robert E Ginna

ES - 401 | **Plant Systems - Tier 2 / Group 1** | **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
003 Reactor Coolant Pump								X				A2.04 - Effects of fluctuation of VCT pressure on RCP seal injection flow	2.8	1
004 Chemical and Volume Control											X	2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.4	1
062 AC Electrical Distribution								X				A2.08 - Consequences of exceeding voltage limitations	3.0*	1
064 Emergency Diesel Generator											X	2.4.9 - Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	4.2	1
073 Process Radiation Monitoring								X				A2.03 - Calibration drift	2.9*	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**0**	**0**	**0**	**3**	**0**	**0**	**2**	**Group Point Total:**		**5**

PWR SRO Examination Outline

Facility: Robert E Ginna

ES - 401 | **Plant Systems - Tier 2 / Group 2** | **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
002 Reactor Coolant								X				A2.02 - Loss of coolant pressure	4.4	1
029 Containment Purge								X				A2.04 - Health physics sampling of containment atmosphere	3.2*	1
056 Condensate											X	2.1.32 - Ability to explain and apply system limits and precautions.	4.0	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**0**	**0**	**0**	**2**	**0**	**0**	**1**	**Group Point Total:**		**3**

Generic Knowledge and Abilities Outline (Tier 3)

PWR SRO Examination Outline

Printed: 06/28/2012

Facility: Robert E Ginna

Form ES-401-3

Generic Category	KA	KA Topic	Imp.	Points
Conduct of Operations	2.1.39	Knowledge of conservative decision making practices.	4.3	1
		Category Total:		**1**
Equipment Control	2.2.5	Knowledge of the process for making design or operating changes to the facility.	3.2	1
	2.2.17	Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination woth the transmission system operator.	3.8	1
		Category Total:		**2**
Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.7	1
	2.3.6	Ability to approve release permits.	3.8	1
		Category Total:		**2**
Emergency Procedures/Plan	2.4.18	Knowledge of the specific bases for EOPs.	4.0	1
	2.4.29	Knowledge of the emergency plan.	4.4	1
		Category Total:		**2**

Generic Total: 7

Tier / Group	Randomly Selected K/A	Reason for Rejection
T1 / G1	056 2.2.44	Per 07/06/12 Ginna Outline Comments, questionable duplication of Generic K/A 2.2.44. Replaced with randomly selected 2.4.9 (Q52)
T1 / G2	037 2.2.38	Per 07/06/12 Ginna Outline Comments, questionable duplication of Generic K/A 2.2.38. Replaced with randomly selected 2.1.28 (Q36)
T2 / G1	007 2.2.44	Per 07/06/12 Ginna Outline Comments, questionable duplication of Generic K/A 2.2.44. Replaced with randomly selected 2.1.23 (Q11)
T2 / G1	008 2.2.44	Per 07/06/12 Ginna Outline Comments, questionable duplication of Generic K/A 2.2.44. Replaced with randomly selected 2.4.21 (Q33)
T2 / G1	022 2.2.44	Per 07/06/12 Ginna Outline Comments, questionable duplication of Generic K/A 2.2.44. Replaced with randomly selected 2.4.41 (Q31)
T2 / G2	033 2.2.38	Per 07/06/12 Ginna Outline Comments, questionable duplication of Generic K/A 2.2.38. Replaced with randomly selected 2.2.22 (Q39)
T2 / G2	079 A4.01	Per 07/24 telecom with Chief Examiner. He agrees that there could be too much system overlap between this Service Air system K/A and the two 078 Instrument Air K/As selected in T2/G1. He authorized the random replacement of system 079 and its previously selected K/A. Replaced with randomly selected 029 A2.01 (Q69)
T3 / G2	2.2.20	Per 7/26 telecom with Chief Examiner. Our troubleshooting procedure has no involvement with the ROs. Reselection authorized: randomly selected 2.2.42 (Q26)
T2 / G2	072 A3.01	Ginna RMS Area monitors have no automatic operation associated with them. Replaced with 002 K5.16 (Q59)
T1 / G2	059 AK2.02	Ginna only has one AB gas process monitor. Credible scenario might be possible, but plausible distractors were too difficult. Replaced with 068 AK3.09 (Q40)

Facility: Robert E. Ginna Date of Examination: October 1, 2012
Examination Level: RO X SRO ☐ Operating Test Number: 2012-N-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, N	Perform OPG-REACTIVITY-CALC K/A 2.1.37 (4.3 / 4.6) Knowledge of procedures, guidelines, or limitations associated with reactivity management.
Conduct of Operations	R, D	Determine maximum allowable Rx Vessel Head Venting time K/A 2.1.25 (3.9 / 4.2) Ability to interpret reference materials, such as graphs, curves, tables, etc.
Equipment Control	R, N	Perform 1/M plot per O-1.2.1 K/A 2.2.1 (4.5 / 4.4) Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.
Radiation Control	R, N	Utilize site survey map and the Radiation Control Manual to determine stay time for local valve operation in the RCA. K/A 2.3.4 (3.2 / 3.7) Knowledge of radiation exposure limits under normal or emergency conditions.
Emergency Procedures/Plan		N/A

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

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

Facility: Robert E. Ginna	Date of Examination: October 1, 2012
Examination Level: RO ☐ SRO X	Operating Test Number: 2012-N-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, N	Perform IV of OPG-REACTIVITY-CALC K/A 2.1.37 (4.3 / 4.6) Knowledge of procedures, guidelines, or limitations associated with reactivity management.
Conduct of Operations	R, M	Determine the Allowable Hours An Operator Can Work (JR119.001) K/A 2.1.5 (2.9* / 3.9) Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.
Equipment Control	R, N	Review STP-O-33B, SFP Pump B K/A 2.2.12 (3.7 / 4.1) Knowledge of surveillance procedures.
Radiation Control	R, N	Determine Worker Availability Based on Radiation Exposure Limits K/A 2.3.4 (3.2 / 3.7) Knowledge of radiation exposure limits under normal or emergency conditions.
Emergency Procedures/Plan	R, M	Determine PAR K/A 2.4.44 (2.4 / 4.4) Knowledge of Emergency Plan protective action recommendations.

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

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

RO Admin JPM Summary

N-RA-1 This JPM is new. The examinee will use a computer with the PCNDR and PPCX programs to perform the weekly reactivity calculation for a 50% power change from 100% power at a rate of 1% per minute, per OPG-REACTIVITY-CALC. This JPM will be accomplished in the classroom.

N-RA-2 This is bank JPM JR344.003. The examinee will perform FR-I.3, Response to Voids in Reactor Vessel, step 21 – Determine Maximum Allowable Venting Time; which will require calculation using Attacment-20.0, Attachment Vent Time, and figure FIG-12.0, Figure CNMT Hydrogen. This JPM will be accomplished in the classroom.

N-RA-3 This JPM is new. The examinee will utilize a data sheet to construct a 1/M plot for a Rx startup. The plot will include the + and – 500 pcm values, the Bank B 0% power RIL value, calculation and plotting of multiple 1/M values, and projection of whether criticality will occur within the acceptable range or not. This JPM will be accomplished in the classroom.

N-RA-4 This JPM is new. Using A-1, Radiation Control Manual, a provided current annual exposure record, and a provided survey map, the examinee will determine allowable stay time for local valve operation in the RCA. This JPM will be accomplished in the classroom.

SRO Admin JPM Summary

N-SA-1 This JPM is new. The examinee will use a computer with the PCNDR and PPCX programs to perform an independent verification of the weekly reactivity calculation for a 50% power change from 100% power at a rate of 1% per minute, per OPG-REACTIVITY-CALC. This JPM will be accomplished in the classroom.

N-SA-2 This JPM is new. The examinee will use a licensed operator's work history and determine the allowable hours that the operator can work, per CNG-SE-1.01-1002, FATIGUE MANAGEMENT AND WORK HOUR CONTROLS. This JPM will be accomplished in the classroom.

N-SA-3 This JPM is new. The examinee will be required to review a completed STP-O-33B, SPENT NFUEL POOL PUMP B, and list any deviation(s)/error(s) and applicable required action(s) on the JPM CUE SHEET. This JPM will be accomplished in the classroom.

N-SA-4 This JPM is new. Using A-1, Radiation Control Manual, a provided current annual exposure record, and a provided survey map, the examinee will determine 1) The expected dose, 2) Applicable radiation exposure limits, and 3) Whose permission is required to perform a 20 minute job that will be performed in the RCA. This JPM will be accomplished in the classroom.

N-SA-5 This JPM is modified (JS340.003). This JPM requires the examinee to make a Protective Action Recommendation to offsite authorities for an event classified as a GENERAL EMERGENCY. This JPM is considered modified because it will utilize a combination of conditions different than those previously used. This JPM will be accomplished in the classroom.

Facility: Robert E. Ginna	Date of Examination: October 1, 2012
Exam Level: RO X SRO-I ☐ SRO-U ☐	Operating Test No.: 2012-N-1

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. 004 Chemical and Volume Control System Borate for two stuck rods per ES-0.1 step 4, and ER-CVCS.1 004 A4.07 (3.9/3.7): Ability to manually operate and/or monitor in the control room: Boration/dilution	S, N, A	1
B. 006 Emergency Core Cooling System Secure SIPs as part of Termination of Bleed and Feed, per FR-H.1, Response To Loss Of Secondary Heat Sink 006 A2.02 (3.9/4.3): Ability to (a) predict the impact of the following operations on the RCS; and (b) based on those predictions, use procedures to control the consequences of those operations: Loss of flow path.	S, N, A	3
C. 005 Heat Removal – Primary System, RHR System Align RHR and SI Systems For High Head Recirculation 002 A2.04 (4.3/4.6): Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of heat sinks.	S, D,	4P
D. 045 Heat Removal – Secondary System, Main Turbine Generator Synchronize Generator On-Line (Improper Load Pickup) 045 A2.17 (2.7*/2.9*): Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunction of electrohydraulic control.	S, D, A	4S
E. 015 Nuclear Instrumentation System Perform STP-O-6.1 on Source Range N-31 015 A4.02 (3.9/3.9): Ability to manually operate and/or monitor in the control room: NIS indicators.	S, L, N	7

F. 064 Electrical Shutdown the "A" Emergency Diesel Generator (2008 NRC ILT) 064 A4.06 (3.9/3.9): Manual start, loading, and stopping of the ED/G.	S, P	6
G. 012 Reactor Protection System Respond to a Spurious CVI While at Full Power 012 A2.01 (3.1/3.6): Ability to (a) predict the impacts of the following malfunctions or operation on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty bistable operation.	S, EN, N	7
H. 008 Plant Service Systems Respond to rapidly lowering CCW Surge Tank level 008 A2.02 (3.2/3.5): Ability to (a) predict the impacts of the following malfunctions or operation on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low surge tank level.	S, A, N	8
In-Plant Systems[@] (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
I. 006 Emergency Core Cooling System Makeup to "A" SI Accumulator Using the SI Accumulator Makeup Pump 006 A1.13 (3.5/3.7): Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration).	R, N	2
J. 071 Waste Gas Disposal System (2010 NRC ILT) Release 'A' GDT 071 A4.27 (3.0* / 2.7*): Ability to manually operate and/or monitor in the control room: Opening and closing of the decay tank discharge control valve.	R, P, A	9
K. 103 Containment System Perform Attachment CI/CVI (Intermediate Bldg Cold Side) 103 A2.03 (3.5* / 3.8*): Ability to (a) predict the impacts of the following malfunctions or operation on the containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation.	E, EN, D	5

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

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

Facility: Robert E. Ginna Exam Level: RO ☐ SRO-I X SRO-U ☐	Date of Examination: October 1, 2012 Operating Test No.: 2012-N-1

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. 004 Chemical and Volume Control System Borate for two stuck rods per ES-0.1 step 4, and ER-CVCS.1 004 A4.07 (3.9/3.7): Ability to manually operate and/or monitor in the control room: Boration/dilution	S, N, A	1
B. 006 Emergency Core Cooling System Secure SIPs as part of Termination of Bleed and Feed, per FR-H.1, Response To Loss Of Secondary Heat Sink 006 A2.02 (3.9/4.3): Ability to (a) predict the impact of the following operations on the RCS; and (b) based on those predictions, use procedures to control the consequences of those operations: Loss of flow path.	S, N, A	3
C. 005 Heat Removal – Primary System, RHR System Align RHR and SI Systems For High Head Recirculation 002 A2.04 (4.3/4.6): Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of heat sinks.	S, D	4P
D. 045 Heat Removal – Secondary System, Main Turbine Generator Synchronize Generator On-Line (Improper Load Pickup) 045 A2.17 (2.7*/2.9*): Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunction of electrohydraulic control.	S, D, A	4S
E. 015 Nuclear Instrumentation System Perform STP-O-6.1 on Source Range N-31 015 A4.02 (3.9/3.9): Ability to manually operate and/or monitor in the control room: NIS indicators.	S, L, N	7

F.	064 Electrical Shutdown the "A" Emergency Diesel Generator (2008 NRC ILT) 064 A4.06 (3.9/3.9): Manual start, loading, and stopping of the ED/G.	S, P	6
H.	008 Plant Service Systems Respond to rapidly lowering CCW Surge Tank level 008 A2.02 (3.2/3.5): Ability to (a) predict the impacts of the following malfunctions or operation on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/low surge tank level.	S, N, A	8
In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)			
I.	006 Emergency Core Cooling System Makeup to "A" SI Accumulator Using the SI Accumulator Makeup Pump 006 A1.13 (3.5/3.7): Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Accumulator pressure (level, boron concentration).	R, N	2
J.	071 Waste Gas Disposal System (2010 NRC ILT) Release 'A' GDT 071 A4.27 (3.0* / 2.7*): Ability to manually operate and/or monitor in the control room: Opening and closing of the decay tank discharge control valve.	R, P, A	9
K.	103 Containment System Perform Attachment CI/CVI (Intermediate Bldg Cold Side) 103 A2.03 (3.5* / 3.8*): Ability to (a) predict the impacts of the following malfunctions or operation on the containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation.	E, EN, D	5

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

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

Facility: Robert E. Ginna   Date of Examination: October 1, 2012
Exam Level: RO ☐ SRO-I ☐ SRO-U X   Operating Test No.: 2012-N-1

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. 004 Chemical and Volume Control System Borate for two stuck rods per ES-0.1 step 4, and ER-CVCS.1 004 A4.07 (3.9/3.7): Ability to manually operate and/or monitor in the control room: Boration/dilution	S, N, A	1
B. 006 Emergency Core Cooling System Secure SIPs as part of Termination of Bleed and Feed, per FR-H.1, Response To Loss Of Secondary Heat Sink 006 A2.02 (3.9/4.3): Ability to (a) predict the impact of the following operations on the RCS; and (b) based on those predictions, use procedures to control the consequences of those operations: Loss of flow path.	S, N, A	3
E. 015 Nuclear Instrumentation System Perform STP-O-6.1 on Source Range N-31 015 A4.02 (3.9/3.9): Ability to manually operate and/or monitor in the control room: NIS indicators.	S, L, N	7

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
J. 071 Waste Gas Disposal System (2010 NRC ILT) Release 'A' GDT 071 A4.27 (3.0* / 2.7*): Ability to manually operate and/or monitor in the control room: Opening and closing of the decay tank discharge control valve.	R, P, A	9
K. 103 Containment System Perform Attachment CI/CVI (Intermediate Bldg Cold Side) 103 A2.03 (3.5* / 3.8*): Ability to (a) predict the impacts of the following malfunctions or operation on the containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation.	E, EN, D	5

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

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

Control Room/In-Plant JPM Summary

JPM A This is a new JPM. The examinee will be directed to perform the boration step of ES-0.1, REACTOR TRIP RESPONSE. He will have to determine that there are two control rods not fully inserted – one on each of two different MRPI screens. This is an alternate path JPM because the normal boration method will not work, and the examinee will have to refer to ER-CVCS.1, REACTOR MAKEUP CONTROL MALFUNCTION, to ultimately initiate emergency boration. This JPM is performed on the simulator.

JPM B This is a new JPM. The examinee will be placed in a Loss of Secondary Heat Sink where bleed and feed has been established and is in the process of being terminated. The examinee will secure the running SI pumps per the procedure beginning at step 34. The JPM is considered alternate path because it involves multiple RNO actions. This JPM is performed on the simulator.

JPM C This is Bank JPM JR006.013. The examinee will be placed in ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, with RHR pump suction aligned to the sump. However, RCS pressure will be greater than RHR pump shutoff head, requiring the examinee to realign the SI pump suction to the RHR pump discharge. This is a time critical JPM – an SI pump must be restarted within 10 minutes of the time that the last SI pump was secured. This JPM is performed on the simulator.

JPM D This is Bank JPM JR045.005. The examinee will synchronize the main generator on-line per O-1.2, PLANT STARTUP FROM HOT SHUTDOWN TO FULL LOAD. The JPM is alternate path because the automatic load pickup will fail to work properly, and the examinee will be required to adjust load manually. This JPM is performed on the simulator.

JPM E This is a new JPM. The examinee will perform STP-O-6.1, SOURCE RANGE NUCLEAR INSTRUMENTATION SYSTEM CHANNELS N-31 AND N-32, for channel N-31. The JPM will include preparing for the test, defeat of the N-31 channel, verifying response to test signals, and test of the HIGH FLUX AT SHUTDOWN ALARM (including calculation of bistable loop width). This is a low power JPM - the plant is shutdown with the Source Ranges energized. This JPM is performed on the simulator.

JPM F This JPM was previously performed on the 2008 ILT NRC Exam. The monthly surveillance of the A D/G is in progress per STP-O-12.1, EMERGENCY DEISEL GENERATOR A. The examinee will shutdown the DG per the procedure. This JPM is performed on the simulator.

JPM G This is a new JPM that involves an Engineered Safety Feature. The examinee will be the HCO assigned the Control Room Monitoring function with the plant at steady-state full power. A spurious Containment Ventilation Isolation (CVI) will occur, and the examinee will respond per the associated Alarm Response procedure. The examinee will ensure CVI has occurred, and, after the cue from the examiner, reset the CVI, including verifying the repositioning of appropriate valves and re-starting the pump for the R10A/R11/R12 radiation monitors. This JPM is performed on the simulator.

JPM H This is a new JPM. The examinee will be the HCO assigned the Control Room Monitoring function with the plant is at steady state full power. A large CCW supply line break will occur. The examinee will respond per AP-CCW.2, LOSS OF CCW DURING POWER OPERATION. This will involve providing makeup to the CCW surge tank, determining that surge tank level cannot be maintained greater than 10%, closing the letdown isolation valve, performing E-0, REACTOR TRIP OR SAFETY INJECTION immediate actions from memory, stopping both RCPs and placing both CCW pumps in pull stop. This is an alternate path JPM because the examinee will perform E-0 immediate actions and then return to AP-CCW.2 for further actions. This JPM is performed on the simulator.

JPM I This is a new JPM. The examinee will perform makeup to the “A” SI Accumulator using the SI Accumulator Makeup pump per S-16.13, RWST WATER MAKEUP TO ACCUMULATORS. The examinee will simulate performing the valve lineup, starting the pump and throttling the discharge dump valve to maintain the proper pressure. This JPM is performed in the RCA.

JPM J This JPM was previously performed on the 2010 ILT NRC Exam. The examinee will simulate performing the lineup and release of the “A” GDT per S-4.2.5, RELEASE OF GAS DECAY TANK, CH-703 Attachment 1, Gaseous Waste Release Form. The JPM is alternate path because the associated radiation monitor exceeds the alarm value and the gas decay tank release AOV (RCV-014) fails to close automatically. This JPM is performed in the RCA.

JPM K This is Bank JPM JC103.004. This is an Emergency in-plant JPM that involves an Engineered Safety Feature. While performing E-0, REACTOR TRIP OR SAFETY INJECTION, it was found that some Containment Isolation valves were not properly aligned. The examinee is directed to simulate isolating the valves in the Intermediate Building clean side using the alternate isolation valves per ATT-3.0, Attachment CI/CVI.

Facility:	**Robert E. Ginna**	Scenario No.: **1**	Op Test No.: **2012-N-1**
Examiners:	____________	Operators:	____________ (SRO)
	____________		____________ (RO)
	____________		____________ (BOP)

Initial Conditions:	The plant is at 75% power (MOL). Several days ago, the plant was taken to 50% due to a failure of the B MFW pump. Corrective maintenance was performed, and the plant is returning to full power per O-5.2, Load Ascension, Step 6.2.32, Placing Condensate Booster Pumps In Service. The trim valves are closed and MFP suction pressure is approaching 200 psig.
Turnover:	The following equipment is Out-Of-Service: B EDG has been OOS for fuel oil pump replacement since yesterday. Expected return to service is tomorrow. A-52.4 submitted for ITS 3.8.1.B, 7 Day Action. SR-3.8.1.1 and 3.8.1.2 completed successfully; SR-3.8.1.1 due again in 6 hrs. Continue load ascension.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N-BOP N-SRO	Start the first Condensate Booster Pump per O-5.2, Steps 6.2.32.1 thru 6.2.32.7
2	A-MIS07 IND-MIS44	(TS)-SRO	CNMT Recirc Fan Vibration Alarm Fan 1B Vibration
3	RCS20 (630) ROD12 (A+M)	R-RO R-SRO	Average-Tavg fails high Rod Stop Failure
4	ROD02-J10 (STA)	R-RO R-BOP R (TS)-SRO	Dropped rod J10
5	RCS05A	C-RO C-BOP C-SRO	'A' RCP trip results in reactor trip
6	ROD03-K7 ROD03-K9 ROD03-L8 (Untrippable)	R-RO R-SRO	Multiple (3) rods not fully inserted
7	EDS01A EDS01B GEN04A (All conditions)	M-BOP M-RO M-SRO	Loss of Offsite Circuit 7T Loss of Offsite Circuit 767 'A' EDG fails to start

8	RPS07M RPS07N	C-BOP C-SRO	3504A fails to open in Auto 3505A fails to open in Auto
9	RPS07O RPS07Q	C-BOP C-SRO	'A' SW pump Auto failure to start 'C' SW pump Auto failure to start
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

Ginna 2012 NRC Scenario #1

The plant is at 75% power (MOL). Several days ago, the plant was taken to 50% due to a failure of the B MFW pump. Corrective maintenance was performed, and the plant is returning to full power per O-5.2, Load Ascension, Step 6.2.32, Placing Condensate Booster Pumps In Service.

The following equipment is Out-Of-Service: B EDG is out of service for replacement of a fuel oil pump. ITS entry for 3.8.1.B, a 7-day LCO, has been entered, and all required surveillances are current.

Shortly after taking the watch, the operator will meet the conditions for starting the Condensate Booster Pumps (trim valves are closed and MFP suction pressure is approaching 200 psig), and will start the first CBP. After the first CBP is started the B-32, CNMT RECIRC FAN VIB, alarm will occur indicating a problem with the CNMT Recirc fans. The AR procedure will have a field operator attempt to reset the alarm (it won't), and directs that a non-running CRF be started. CRS will reference ITS 3.6.6 (7 day LCO).

After the field operator reports all conditions normal after the CBP start, the Average-Tavg signal will fail high, resulting in auto rod motion inward. The crew should respond per AR-F-15, RCS TAVG DEV and AP-RCC.1, Continuous Rod Withdrawal/Insertion.

After the actions of AR-F-15 are complete, Control Rod J10 (Control Bank 'C') will drop. Crew should respond per AR-C-14, ROD BOTTOM, and AP-RCC.3, Dropped Rod Recovery. The CRS will address ITS 3.1.4 (Rod Group Alignment) and 3.2.4 (QPTR). When the CRS refers to ER-RCC.1, Dropped Rod Recovery, per Step 8 of AP-RCC.3, the 'A' RCP will trip.

The RCP trip results in a reactor trip and the crew will respond per E-0, Reactor Trip or Safety Injection. Both FRVs will fail to close due to the earlier Average-Tavg failure. Depending upon the timing of Immediate Action completion and the crew response to recognize/close the FRVs, the crew may remain in E-0 (SI due to excessive feed water cooling) or transition to ES-0.1, Reactor Trip Response. If the crew remains in E-0, the LOAAC event will occur when the HCO is directed to perform Att.27.0, Automatic Action Verification. If the crew successfully isolated MFW and transitioned to ES-0.1, the LOAAC event will occur at Step 4, after the crew has initiated emergency boration in response to the 3 control rods not inserted. In either path, transition to ECA-0.0, Loss of All AC Power will be required when the 'A' EDG failed to start following the loss of offsite power.

At Step 18 of ECA-0.0 (after RCP seal injection and thermal barriers are isolated), the AO will report the 'A' EDG is now available for start. The crew should manually start the 'A' EDG from the main control board, go to Step 28 (per the CAUTION prior to Step 8), recognize that no SW pumps are running, and start one SW pump and manually perform SW isolation per Step 29.

The scenario will end after SW has been established in Step 29 or when the correct transition to either ECA-0.1 or ECA-0.2 is determined in Step 31 (examiner's discretion).

Critical Tasks:

ECA-0.0--B

Establish greater than 200 GPM AFW flow before both S/G levels decrease to < 120"wide range level [160" Adverse CNMT] .

Safety Significance: Failure to establish the minimum required AFW flow rate, under the postulated accident conditions, is a violation of the basic objective of ECA-0.0 and of the assumptions of the analyses upon which ECA-0.0 is based. Both intend to mitigate deterioration of RCS conditions while AC emergency power is not available. Without AFW flow, the S/Gs could not support any significant plant cooldown. Thus, the crew would lose the ability to delay the adverse consequences of core uncover. Also, without AFW flow, decay heat would still open the safety valves and would rapidly deplete the S/G inventory, leading to a loss of secondary heat sink or S/G dryout. Decay heat would then increase RCS temperature and pressure until the pressurizer PORVs open, imposing a larger LOCA than RCP seal leakage. Both of these examples violate the basic assumptions of the analyses on which ECA-0.0 is based, complicating the mitigation actions.

Required Plant Conditions:

1. A loss of all AC condition exists
2. Less than 200 GPM of AFW flow
3. The TDAFW pump is available, but must be manually started
4. S/G NR levels are less than 5%

Scenario Conditions:

1. A loss of all AC condition exists (no offsite power, 'B' EDG OOS, 'A' EDG fails)
2. Both MDAFW pumps have no power, TDAFW pump steam admission valves have not auto opened
3. Manual opening of TDAFW pump steam admission valves is available
4. Normal post-trip S/G level shrink in conjunction with no AFW availability will result in NR levels <5%.

ECA-0.0--F

Manually start the SW pump such that the EDG does not fail because of damage caused by engine overheating.

Safety Significance: Failure to manually start the SW pump under the postulated plant conditions means that the EDG is running without SW cooling. Running without SW cooling leads to a high-temperature condition that can result in EDG failure due to damage caused by engine overheating. Under the postulated plant conditions, the running EDG is the only operable EDG. Thus, failure to perform the critical task constitutes "mis-operation or incorrect crew performance that leads to degraded emergency power capacity."

Required Plant Conditions:

1. Loss of all AC initial condition with subsequent restoration of AC power from one EDG.
2. SW pump(s) fail to start automatically when the associated emergency bus is re-energized by the EDG.
3. A SW pump aligned to provide cooling for the running EDG can be manually started from the MCR.
4. The other EDG is inoperable.

Scenario Conditions:

1. AC power will be restored by 'A' EDG.
2. Auto start of 'A' and 'C' SWPs is failed.
3. Start of either 'A' or 'C' SW pump will provide cooling for the 'A' EDG
4. 'B' EDG is OOS.

ECA-0.0--H

Isolate RCP seal injection before a charging pump is started and isolate RCP thermal barriers before a CCW pump is started.

Safety Significance: Failure to isolate RCP seal injection before starting a charging pump or the thermal barrier before starting a CCW pump, under the postulated conditions can result in unnecessary and avoidable degradation of the RCS fission product barrier – specifically at the point of the RCP seals, especially if the RCPs are subsequently started. Additionally, failure to perform the critical task results in "significant degradation in their mitigative capability of the plant" in that the RCPs are not available for subsequent recovery actions (except for the red-path condition on the Core Cooling CSF that persists despite secondary depressurization).

Required Plant Conditions:

1. Station Blackout with subsequent restoration of power to at least one AC emergency bus.
2. Blackout condition persists long enough for the crew to reach Step 8 of ECA-0.0, and RCP #1 seal outlet temperature is >235°F.

Scenario Conditions:

1. Loss of all AC existed. 'A' EDG has started and is providing power to one emergency train (buses 14 and 18).
2. The 'A' EDG will not be reported as available until Step 18 of ECA-0.0 (after RCP seal injection and thermal barriers are isolated, if the crew correctly applies the procedure). With no seal injection flow, RCP #1 seal outlet temperatures will rise to >235°F.

Facility:	**Robert E. Ginna**	Scenario No.: **2**	Op Test No.: **2012-N-1**
Examiners:	____________	Operators:	____________ (SRO)
	____________		____________ (RO)
	____________		____________ (BOP)

Initial Conditions:	The plant is at ~3% power (BOL). Plant startup is in progress per O-1.2, Plant Startup From Hot Shutdown To Full Load. The team is currently at step 6.7 (page 39). The "A" Main Feed Pump has just been placed in-service. The Shift Manager (SM) has directed that 50 gpm Blowdown flow from each SG is desired.
Turnover:	The following equipment is Out-Of-Service: B EDG has been OOS for fuel oil pump replacement since yesterday. Expected return to service is tomorrow. A-52.4 submitted for ITS 3.8.1.B, 7 Day Action. SR-3.8.1.1 and 3.8.1.2 completed successfully; SR-3.8.1.1 due again in 6 hrs.

Event	Malf. No.	Event Type*	Event Description
1	NA	N-BOP N-SRO	Transition from AFW to MFW per O-1.2 steps 6.7.1 through 6.7.5.
2	RCS11F (980)	I-RO I-BOP I(TS)-SRO	Th RTD (Channel 4) fails high
3	RCS01B (15)	C-RO C(TS)-SRO	"B" RCP Thermal Barrier leak (CLIC)
4	SIS01	NA	Spurious SI ("A" Train)
5#	EDS1A EDS1B	NA	Loss of Offsite Power
6#	RCS19D (1500)	M-BOP M-RO M-SRO	LOCA Outside Containment
7	RPS05A RPS05B (man avail)	C-RO C-SRO	Reactor Trip Breakers fail to open automatically (manual available)
8	GEN08 (A DG)	C-RO C-SRO	"A" DG fails to start automatically (manual start available)
9	RPS12-F1 RPS11-E3 (isol sig only)	C-RO C-SRO	AOV-1597 (R10A, 11, 12 Suction) fails to close automatically AOV-5735 ("A" SG Sample) fails to close automatically
10	RPS07K	C-BOP C-SRO	"A" MDAFW Pump fails to start automatically
# = will be initiated by the Spurious SI event			
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

Ginna 2010 NRC Scenario #2

The plant is at ~3% power (BOL). Plant startup is in progress per O-1.2, Plant Startup From Hot Shutdown To Full Load. The team is currently at step 6.7 (page 39). The "A" Main Feed Pump has just been placed in-service. The Shift Manager (SM) has directed that 50 gpm Blowdown flow from each SG is desired.

The following equipment is Out-Of-Service: B EDG has been OOS for fuel oil pump replacement since yesterday. Expected return to service is tomorrow. A-52.4 submitted for ITS 3.8.1.B, 7 Day Action. SR-3.8.1.1 and 3.8.1.2 completed successfully; SR-3.8.1.1 due again in 6 hrs.

After taking the watch, the team will transition from Auxiliary Feed Water (AFW) to Main Feed Water (MFW) per O-1.2 steps 6.7.1 through 6.7.5. This will involve the BOP verifying operation of the Feed Regulating Valves (FRVs) and the FRV Bypass valves, placing the FRV Bypass Valves in Auto, securing the AFW pumps, and placing the AFW pumps in safeguards alignment.

Next a channel four Thot RTD will fail high. The team will respond per:
AR-F-15, RCS Tavg Dev 4°F, or
AR-F-16, Avg Tavg – Tref Dev +/- 5 °F, and
ER-INST.1, Reactor Protection Bistable Defeat After Instrumentation Loop Failure.
The RO will transfer the charging pump that is in automatic to manual and manually adjust its speed to control Pressurizer (PRZR) level. The BOP will defeat the failed channel per ER-INST.1. The SRO will refer to ITS 3.1.6, the COLR, and ITS 3.3.1 and 3.3.2.

Next, a thermal barrier leak will occur on the "B" RCP. The team will respond per: AR-RMS-17, R-17, Component Cooling, and AP-CCW.1, Leakage Into The Component Cooling Loop. Per the AR, the RO will verify RCV-017 closed, and the SRO will direct the RP Tech to perform CH-PRI-CCW-LEAK to determine CCW leakage. Per AP-CCW.1, the RO will isolate the leak by closing AOV-754B. *Note that later in the scenario, when SI occurs, instrument air to containment will be isolated and AOV-754B will fail open.* The CCW leak will need to be re-isolated when the team restores Instrument Air to containment per EOP direction. (Due to electrical power availability, power will not be available to close MOV-749B and MOV-759B – the alternate isolation method.)

Next, a spurious "A" train SI signal will occur. At the same time there will be a loss of offsite power, the "A" DG will fail to start automatically, and a LOCA outside containment will be initiated. The team will respond per E-0, Reactor Trip or Safety Injection. At immediate action step 1, the RO will recognize that the reactor failed to trip automatically and will trip the reactor manually. At immediate action step 3 the BOP will inform the team of the status of Safeguard buses (no safeguard buses energized), and the RO will depress the "A" DG start pushbutton (front left section of the MCB) which will restore the "A" train of safeguard buses (buses 14 and 18). When the RO performs Attachment 27.0, Attachment Automatic Action Verification, (step 6 of E-0) he will discover that AOV-5735, "A" SG Sample Valve, and AOV-1597, R-10A/11/12 Suction Isolation failed to isolate automatically, and will close them manually. At step 7 no MDAFW pumps will be running. The BOP will manually start the "A" MDAFW pump and align the TDAFW pump as necessary.

The LOCA outside containment will be due to failure of check valve 853B. When the SI occurs RCS will flow backwards through MOV-852B (Core Deluge Valve) and cause an RHR system piping break inside the Auxiliary building (RHR system piping is designed for a maximum of 600 psi – not RCS pressure). RCS will also flow from the RHR system to the CVCS system (via a check valve around HCV-133), lift relief valve 203, and flow to the PRT. Within a few minutes (depending on the size of the RHR piping break) the PRT rupture disc will blow and cause a LOCA inside containment. Because of this, the team is expected to transition to E-1, Loss of Reactor or Secondary Coolant, at E-0 step 17, rather than transitioning to ECA-1.2, LOCA Outside Containment, at E-0 step 26. (Transition from E-0 step 26 is possible depending on break size and how fast the team progresses through E-0, but is unlikely.)

Instrument air to containment will be re-established at step 9 of E-1. This will be the team's earliest opportunity to re-isolate the CCW system leak. (Unable to close MOV-749B and 759B due to bus 16 de-energized.) The team will transition to ECA-1.2 at E-1 step 17.

At ECA-1.2 step 3 the RO will close MOV-852A and the team will check for an RCS pressure rise. When pressure doesn't rise they will reopen MOV-852A and close MOV-852B. This time RCS pressure will rise, indicating isolation of the LOCA outside containment. (The pressure rise may be delayed or slow depending on RCS inventory. If a sizable bubble exists in the reactor vessel head, injection flow has to refill the head to some extent before RCS pressure will rise significantly. The team should realize this.)

Terminate the scenario upon transition back to E-1.

Additional Information:

When the Aux Bldg sump reaches 15,000 gallons the RHR pumps, Aux Bldg sump pumps, and RCDT pumps will fail due to submergence (2 to 4 minute time delay after submergence).

Critical Tasks:

E-0--C

Energize at least one emergency bus before transition out of E-0.

Safety Significance:

Failure to re-energize an emergency train leads to degraded emergency power capacity, needless degradation of a fission release barrier (RCS through RCP Seals), and results in a continuing loss of RCS inventory.

Required Plant Conditions:

1. Reactor trip.
2. No emergency bus energized.
3. At least one EDG can be connected to an emergency bus.

Scenario Conditions:

1. Spurious SI will require reactor trip.
2. Offsite power is lost, "B" DG is OOS, and "A" DG fails to start automatically.
3. "A" DG can be started manually.

ECA-1.2--A

Isolate LOCA outside containment prior to transition out of ECA-1.2.

Safety Significance:
Failure to isolate this leak degrades containment integrity resulting in continuing radioactive release.

Required Plant Conditions:

1. Isolable LOCA outside containment.

Scenario Conditions:

1. The LOCA outside containment may be isolated by closing MOV-852B.

Facility: **Robert E. Ginna**	Scenario No.: **3**	Op Test No.: **2012-N-1**	
Examiners:		Operators:	(SRO)
			(RO)
			(BOP)

Initial Conditions:	The plant is at 69% power (MOL). Several days ago, the plant was taken to 50% due to a failure of the "B" MFW pump. Corrective maintenance was performed, and the plant was raised to 69% four days ago. It is intended to observe the "B" MFP operation for two more days at this power level and then raise power to 100%. RG&E Energy Control Center has requested transfer to 100 – 0 lineup in preparation for circuit 7T inspection next shift. The WCC is standing by to place protected equipment signs per OPG-Protected Equipment, Attachment 4, page 16. Attachment 4 is at the HCO's desk. The SM has directed that you complete the transfer when you have the shift.
Turnover:	The following equipment is Out-Of-Service: "B" MDAFW Pump has been out of service for bearing replacement since yesterday. Expected return to service is tomorrow. A-52.4 submitted for ITS 3.7.5, 7 Day Action.

Event No.	Malf. No.	Event Type*	Event Description
1	NA	N-BOP N-SRO	Swap to 100/0 electrical lineup on Circuit 767 per O-6.9.2, Establishing And/Or Transferring Offsite Power To Bus 12A / 12B
2	PZR02D (1500) IND-RPS07BK (off)	I-BOP I-RO I(ITS)-SRO	Pressurizer pressure channel 449 fails low (ITS 3.3.1: Trip channel within 6 hrs, TRM: TR 3.4.3 Immediately declare ATWS mitigating capability inoperable – was applicable until channel 429 was selected as the controlling channel in the PLP) TC408C Bistable status light failed
3	CVC02 (25 gpm)	C-RO C(TS)-SRO	CVCS letdown line leak
4	REM-CND50 (0.95?)	R-BOP R-RO R-SRO	Loss of Condenser vacuum (Ramp to turbine trip value over 12 minutes) Turbine Trip causes Reactor Trip
5	OVR-DI-TUR05BD	C-BOP C-SRO	EH "Go" pushbutton does not work
6	STM09A STM09B (100)	NA	"B" SG Safety Valve 3508 failed open (close when B&F criteria met) "A" SG Safety Valve 3509 failed open (close when B&F criteria met) [NOTE – fail enough Safeties open to preclude transition to ES-0.1]
7	EDS08 (both)	M-BOP M-RO M-SRO	4KV Auto Bus Transfer fails to occur automatically

8	EDS04A	NA	Loss of Bus 14 (conditional on reactor trip)
9	FDW12 (0)	NA	TDAFW Pump speed control failed to 0
10	FDW15B	C-BOP C-SRO	SAFW Pump "D" fails to start
11	SIS02A SIS02B (man avail)	NA	SI fails to actuate automatically
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

After the SRO has had an opportunity to identify the ITS / TRM requirements associated with the 449 failure, and determine that there are no ITS concerns associated with the CVCS leak; a loss of condenser vacuum will occur. (The loss of vacuum will be ramped over 12 minutes so that a turbine trip will occur at the end of that time period.) The team will respond per:
AR-H-7, Condenser Hi Pressure 25.5" HG, and
AP-TURB.4, Loss Of Condenser Vacuum, and refer to
AP-TURB.5, Rapid Load Reduction.
They will reduce turbine load in an attempt to stabilize condenser vacuum, and get out of the "Avoid" region in condenser backpressure. When the BOP initiates load reduction he will discover the EH control panel will not respond in OPER PAN, and will reduce turbine load manually. The team will be unable to stabilize condenser vacuum. A turbine trip and reactor trip will occur.

(Note – it may be necessary to initially use a quick ramp on REM-CND50 to reach the conditions where the AP will be applicable, and then use a very gradual ramp to allow the team to respond, and then use a quick ramp again to cause the turbine / reactor trips. This will be determined during validation.)

Upon reactor trip a failure of the 4KV Auto Bus Transfer and a loss of Bus 14 will occur. Additionally, the TDAFW Pump speed control is failed to zero. The team will respond per E-0, Reactor Trip or Safety Injection. With steam dump unavailable, SG pressure will rise rapidly causing SG ARVs and SG Safeties to open. One SG safety associated with each SG will fail full open (more if necessary – to be determined during preparation / validation), causing a significant cooldown, RCS depressurization, and SI signal. The team will recognize that SI failed to actuate automatically and will manually actuate SI. At step 9 the team will transition to FR-H.1, Response To Loss of Secondary Heat Sink.

The MDAFW and TDAFW pumps are unavailable. "C" SBAFP has no power. Depending on SG level, the team will either go to step 13 for Bleed & Feed, or attempt to align and start "D" SBAFP. If there is time for the SBAFP start, the SRO will determine feed requirements per Attachment 22.0, Attachment Restoring Feed Flow (attempt to feed at max rate [225 gpm]). "D" SBAFP will trip when they try to start it. When less than or equal to120 inches wide range SG level is reached in both SGs, the team will go to step 13 for Bleed & Feed. The team will perform Attachment 12.0, Attachment N2 PORVs to operate the PORVs. Subsequently, Bus 15 will be energized by cross-tie to Bus 16, and instrument air will be aligned to the PORVs.

After Bleed and Feed is established (PORVs open and SI flow verified), at the Lead Examiner's discretion, the SRO will be notified from the WCC that the "B" MDAFP is available. The team will establish AFW flow per the guidance of step 3.

Terminate the scenario when auxiliary feed flow is established.

Critical Tasks:

E-0--D

Manually actuate at least one train of SI before transition to FR-H.1.

Safety Significance:

The acceptable results obtained in the FSAR analyses are predicated on the assumption that, at the very least, one train of safeguards actuates. If SI is not actuated, the FSAR assumptions and results are invalid. Because compliance with the assumptions of the FSAR is part of the facility license condition, failure to manually actuate at least one train of SI (when it is possible to do so) constitutes a violation of the license condition.

Required Plant Conditions:

1. Reactor trip with valid SI required but not automatically actuated.
2. SI can be manually actuated from the control room.

Scenario Conditions:

1. Loss of Condenser vacuum causes turbine trip which causes a reactor trip (P-9 present). Upon reactor trip, steam flow through the SG safeties will be sufficient to cause a valid SI signal prior to the team completing E-0 step 4.
2. Manual SI is available.

FR-H.1--F

Initiate RCS Bleed so that the RCS depressurizes sufficiently for RCS Feed (SI injection) to occur.

Safety Significance:

Failure to initiate RCS Bleed & Feed before the RCS saturates at a pressure above the shutoff head of the SI pumps results in significant and sustained core uncovery.

Required Plant Conditions:

1. Reactor trip.
2. No AFW available.
3. Secondary heat sink required.
4. RCS pressure below PRZR PORV setpoint.

Scenario Conditions:

1. Loss of Condenser vacuum causes turbine trip which causes a reactor trip (P-9 present).
2. Failure of 4KV Auto Bus Transfer de-energizes the condensate pumps and MFPs. "B" MDAFP is OOS for bearing replacement. Loss of Bus 14 de-energizes "A" MDAFP. The TDAFP speed controller is failed to zero. SB
3. Decay heat is adequate to require a heat sink.
4. The team must follow the guidance of FR-H.1 to initiate RCS Bleed & Feed while RCS pressure is below PRZR PORV setpoint.

Ginna 2012 NRC Scenario #3

The plant is at 69% power (MOL). Several days ago, the plant was taken to 50% due to a failure of the "B" MFW pump. Corrective maintenance was performed, and the plant was raised to 69% four days ago. It is intended to observe the "B" MFP operation for two more days at this power level and then raise power to 100%. RG&E Energy Control Center has requested transfer to 100 – 0 lineup in preparation for circuit 7T inspection next shift. The WCC is standing by to place protected equipment signs per OPG-Protected Equipment, Attachment 4, page 16. Attachment 4 is at the HCO's desk. The SM has directed that you complete the transfer when you have the shift.

The following equipment is Out-Of-Service: "B" MDAFW Pump has been out of service for bearing replacement since yesterday. Expected return to service is tomorrow. A-52.4 submitted for ITS 3.7.5, 7 Day Action.

Shortly after taking the watch the team will swap to 100/0 electrical lineup per O-6.9.2.

Following the electrical lineup swap, Pressurizer (PRZR) controlling pressure channel 449 will fail low, causing PRZR control and backup heaters to energize, and disabling the automatic opening of PORV-431C. The RO will take manual control of PRZR pressure (by either taking manual control of the 431K controller, or by taking manual control of the PRZR heaters). The team will respond per:
AR-F-10, Pressurizer Lo Press 2205 psi, or
AR-F-27, Pressurizer Lo Press Channel Alert 1873 psi; and
AP-PRZR.1, Abnormal Pressurizer Pressure.
When the BOP defeats the failed channel, bistable status light TC408C will fail to light, which indicates that the channel may not be in the tripped condition. The SRO will notify the WCC, Operations Management, and possibly I/C to investigate further to ensure ITS requirements are met. (Sometimes the SRO will contact I/C directly, and sometimes he will request that the WCC contact I/C. Either communication is acceptable.) The SRO will refer to ITS 3.3.1 and TRM TR 3.4.3 (he will determine ITS 3.3.2 and 3.3.3 are not applicable).

While the PT-449 defeat is in progress, a 25 gpm letdown leak will occur. The team will respond per:
AR-E-16, RMS Process Monitor High Activity,
EPIP 1-13, Local Radiation Emergency, and
AP-CVCS.1, CVCS Leak.
The SRO will direct the RO to announce a local radiation emergency in the Auxiliary building basement and direct all personnel to evacuate the Auxiliary building basement. The RO will isolate normal letdown and establish excess letdown per Attachment 9.1, Attachment Excess L/D. When it is identified as CVCS leakage there will be no ITS implications. (Also not a CLOC [closed loop outside containment] concern per A-3.3, Containment Integrity Program.)