Exelon Generation

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January 7, 2014
TMI-14-002

USNRC, Region I
2100 Renaissance Blvd, Suite 100
King of Prussia, PA 19406-2713

THREE MILE ISLAND NUCLEAR STATION, UNIT 1 (TMI-1)
RENEWED OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289

SUBJECT: SUBMITTAL OF INITIAL OPERATOR LICENSING EXAMINATION OUTLINES

Enclosed are the examination outlines, supporting the Initial License Examination scheduled for the week of April 7, 2014, at Three Mile Island Unit 1.

This submittal includes all appropriate Examination Standard forms and outlines in accordance with NUREG 1021, "Operator Licensing Examination Standards," Revision 9 Supplement 1.

In accordance with NUREG 1021, Revision 9 Supplement 1, Section ES-201, "Initial Operator Licensing Examination Process," please ensure that these materials are withheld from public disclosure until after the examinations are complete.

Should you have any questions concerning this letter, please contact Mike Fitzwater of Regulatory Assurance at (717) 948-8228. For questions concerning examination materials, please contact Rich Megill, Exam Author, at (717) 948-2093.

Respectfully,

Rick W. Libra
Site Vice President, Three Mile Island Unit 1
Exelon Generation Co., LLC

RWL/mdf

Enclosures: (Mailed to Peter Presby, Chief Examiner, NRC Region I)
- Examination Security Agreements (Form ES-201-3)
- Administrative Topics Outlines (Form ES-301-1)
- Control Room/In-Plant Systems Outline (Form ES-301-2)
- PWR Examination Outline (Form ES-401-2)
- Generic Knowledge and Abilities Outline (Tier 3) (Form ES-401-3)
- Statement detailing method of Written Outline generation
- Scenario Outlines (Form ES-D-1)
- Record of Rejected K/As (Form ES-401-4)
- Completed Checklists:
  - Examination Outline Quality Checklist (Form ES-201-2)
  - Transient and Event Checklist (Form ES-301-5)

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cc: (without attachments)
Chief, NRC Operator Licensing Branch
NRC Senior Resident Inspector -- TMI Unit 1

Facility Name: Date of Exam:

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18	3	3	6
	2	2	2	1				2	1			1	9	2	2	4
	Tier Totals	5	5	4				5	4			4	27	5	5	10
2. Plant Systems	1	3	2	3	3	3	2	2	3	2	2	3	28	3	2	5
	2	1	1	1	1	1	1	1	1	1	1	0	10	0 2	1	3
	Tier Totals	4	3	4	4	4	3	3	4	3	3	3	38	5	3	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2	3 4	7
					3		3		2		2			2 2	2 1	

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 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.

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO)		Form ES-401-2
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000007 Reactor Trip - Stabilization - Recovery / 1		02					Knowledge of the interrelations between a reactor trip and the following: Breakers, relays and disconnects	2.6	1
BW/E02 Vital System Status Verification / 1									
BW/E10 Post Trip Stabilization / 1									
000008 Pressurizer Vapor Space Accident / 3			01				Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: Why PZR level may come back on scale if RCS is saturated	3.7	1
000009 Small Break LOCA / 3					05		Ability to determine or interpret the following as they apply to a small break LOCA: The time available for action before PZR is empty, given the rate of decrease of PZR level	3.4	1
000011 Large Break LOCA / 3									0
000015 RCP Malfunctions / 4 000017 RCP Malfunctions (Loss of RC Flow) / 4	04						Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Basic steady state thermodynamic relationship between RCS loops and S/Gs resulting from unbalanced RCS flow	2.9	1
000022 Loss of Rx Coolant Makeup / 2					01		Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exists	3.2	1
000025 Loss of RHR System / 4		02					Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: LPI or Decay Heat Removal/RHR pumps	3.2	1
000026 Loss of Component Cooling Water / 8				03			Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: SWS as a backup to the CCWS	3.6	1
000027 Pressurizer Pressure Control System Malfunction / 3	02						Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases	2.8	1
000029 ATWS / 1						04.21	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	1
000038 Steam Gen. Tube Rupture / 3					09		Ability to determine or interpret the following as they apply to a SGTR: Existence of natural circulation, using plant parameters	4.2	1
000040 Steam Line Rupture - Excessive Heat Transfer / 4		02					Knowledge of the interrelations between the Steam Line Rupture and the following: Sensors and detectors	2.6	1
BW/E05 Excessive Heat Transfer / 4									
000054 Loss of Main Feedwater / 4									0
000055 Station Blackout / 6				07			Ability to operate and monitor the following as they apply to a Station Blackout: Restoration of power from offsite	4.3	1
000056 Loss of Off-site Power / 6				03			Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: Adjustment of ED/G load by selectively energizing PZR backup heaters	3.2	1
000057 Loss of Vital AC Inst. Bus / 6			01				Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus	4.1	1
000058 Loss of DC Power / 6	01						Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation	2.8	1
000062 Loss of Nuclear Svc Water / 4									0
000065 Loss of Instrument Air / 8						04.45	Ability to prioritize and interpret the significance of each annunciator or alarm.	4.1	1
BW/E04 Inadequate Heat Transfer / 4			03				Knowledge of the reasons for the following responses as they apply to the (Inadequate Heat Transfer): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	4.2	1
000077 Generator Voltage and Electric Grid Disturbances / 6						04.18	Knowledge of the specific bases for EOPs.	3.3	1
K/A Category Totals:	3	3	3	3	3	3	Group Point Total:		18

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

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

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		01					Knowledge of the interrelations between the Inoperable / Stuck Control Rod and the following: Controllers and positioners	2.5	1
000024 Emergency Boration / 1									0
000028 Pressurizer Level Malfunction / 2						04.06	Knowledge of EOP mitigation strategies.	3.7	1
000032 Loss of Source Range NI / 7	01						Knowledge of the operational implications of the following concepts as they apply to Loss of Source Range Nuclear Instrumentation: Effects of voltage changes on performance	2.5	1
000033 Loss of Intermediate Range NI / 7									0
000036 Fuel Handling Accident / 8									0
BW/A08 Refueling Canal Level Decrease / 8									
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 / 9 8				01			Ability to operate and / or monitor the following as they apply to the Plant Fire on Site: Respirator air pack	3.6	1
000068 Control Room Evac. / 8			01				Knowledge of the reasons for the following responses as they apply to the Control Room Evacuation: System response to reactor trip	3.9	1
BW/A06 Shutdown Outside Control Room / 8									
000069 Loss of CTMT Integrity / 5									0
000074 Inad. Core Cooling / 4									0
000076 High Reactor Coolant Activity / 9									0
BW/A01 Plant Runback / 1									0
BW/A02 Loss of NNI-X / 7									1
BW/A03 Loss of NNI-Y / 7	02						Knowledge of the operational implications of the following concepts as they apply to the (Loss of NNI-Y): Normal, abnormal and emergency operating procedures associated with (Loss of NNI-Y).	3	
BW/A04 Turbine Trip / 4									0
BW/A05 Emergency Diesel Actuation / 6		01					Knowledge of the interrelations between the (Emergency Diesel Actuation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	4	1
BW/A07 Flooding / 8									0
BW/E03 Inadequate Subcooling Margin / 4									0
BW/E08 LOCA Cooldown / 4									1
BW/E09 Natural Circulation Cooldown / 4					02		Ability to determine and interpret the following as they apply to the (Natural Circulation Cooldown): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	3.5	
BW/E13 EOP Rules				02			Ability to operate and / or monitor the following as they apply to the (EOP Rules): Operating behavior characteristics of the facility.	2.8	1
BW/E14 EOP Enclosures									
K/A Category Totals:	2	2	1	2	1	1	Group Point Total:		9

ES-401	PWR Examination Outline Plant Systems - Tier 2/Group 1 (RO)												Form ES-401-2	
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
003 Reactor Coolant Pump				04		02						Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Adequate cooling of RCP motor and seals; Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: RCP seals and seal water supply	2.8; 2.7	2
004 Chemical and Volume Control				05							04.11	Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Interrelationships and design basis, including fluid flow splits in branching networks (e.g., charging and seal injection flow); Knowledge of abnormal condition procedures.	3.3; 4	2
005 Residual Heat Removal			05									Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: ECCS	3.7	1
006 Emergency Core Cooling					09							Knowledge of the operational implications of the following concepts as they apply to the ECCS: Thermodynamics of water and steam, including subcooled margin, superheat, and saturation	3.3	1
007 Pressurizer Relief/Quench Tank				01								Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: Quench tank cooling	2.6	1
008 Component Cooling Water								08				Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of shutting (automatically or otherwise) the isolation valves of the letdown cooler	2.5	1
010 Pressurizer Pressure Control			01									Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RCS	3.8	1
012 Reactor Protection					01							Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB	3.3	1
013 Engineered Safety Features Actuation		01									02.40	Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control; Ability to apply Technical Specifications for a system.	3.6; 3.4	2
022 Containment Cooling		02							01			Knowledge of power supplies to the following: Chillers; Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation	2.5; 4.1	2
025 Ice Condenser														0
026 Containment Spray	02									05		Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: Cooling water; Ability to manually operate and/or monitor in the control room: Containment spray reset	4.1; 3.5	2
039 Main and Reheat Steam					08				02			Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity; Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS	3.6; 3.1	2
059 Main Feedwater	04											Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems S/GS water level control system	3.4	1
061 Auxiliary/Emergency Feedwater			01								04.30	Knowledge of the effect that a loss or malfunction of the AFW will have on the following: RCS; Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.	4.4; 2.7	2
062 AC Electrical Distribution								03				Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of improper sequencing when transferring to or from an inverter	2.9	1
063 DC Electrical Distribution								01				Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds	2.5	1
064 Emergency Diesel Generator						08						Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks	3.2	1
073 Process Radiation Monitoring										02		Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel	3.7	1
076 Service Water							02					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures	2.6	1
078 Instrument Air	04											Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor	2.6	1
103 Containment							01					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity	3.7	1
K/A Category Totals:	3	2	3	3	3	2	2	3	2	2	3	Group Point Total:		28

ES-401												PWR Examination Outline Plant Systems - Tier 2/Group 2 (RO)		Form ES-401-2
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
001 Control Rod Drive					65							Knowledge of the following operational implications as they apply to the CRDS: CRDS circuitry, including effects of primary/secondary power mismatch on rod motion	3.2	1
002 Reactor Coolant														0
011 Pressurizer Level Control										01		Ability to manually operate and/or monitor in the control room: Charging pump and flow controls	3.5	1
014 Rod Position Indication														0
015 Nuclear Instrumentation		01										Knowledge of bus power supplies to the following: NIS channels, components, and interconnections	3.3	1
016 Non-nuclear Instrumentation														0
017 In-core Temperature Monitor														0
027 Containment Iodine Removal								01				Ability to (a) predict the impacts of the following malfunctions or operations on the CIRS; and (b) based on those predictions, use Procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High temperature in the filter system	3	1
028 Hydrogen Recombiner and Purge Control														0
029 Containment Purge														0
033 Spent Fuel Pool Cooling														0
034 Fuel Handling Equipment							02					Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Water level in the refueling canal	2.9	1
035 Steam Generator														0
041 Steam Dump/Turbine Bypass Control						03						Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS	2.7	1
045 Main Turbine Generator														0
055 Condenser Air Removal			01									Knowledge of the effect that a loss or malfunction of the CARS will have on the following: Main condenser	2.5	1
056 Condensate														0
068 Liquid Radwaste									02			Ability to monitor automatic operation of the Liquid Radwaste System including: Automatic isolation	3.6	1
071 Waste Gas Disposal				04								Knowledge of design feature(s) and/or interlock(s) which provide for the following: Isolation of waste gas release tanks	2.9	1
072 Area Radiation Monitoring	03											Knowledge of the physical connections and/or cause-effect relationships between the ARM system and the following systems: Fuel building isolation	3.6	1
075 Circulating Water														0
079 Station Air														0
086 Fire Protection														0
K/A Category Totals:	1	1	1	1	1	1	1	1	1	1	0	Group Point Total:		10

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

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

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000007 Reactor Trip - Stabilization - Recovery / 1									
BW/E02 Vital System Status Verification / 1					01		Ability to determine and interpret the following as they apply to the (Vital System Status Verification): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	4	1
BW/E10 Post Trip Stabilization / 1									
000008 Pressurizer Vapor Space Accident / 3									0
000009 Small Break LOCA / 3									0
000011 Large Break LOCA / 3						04.20	Knowledge of the operational implications of EOP warnings, cautions, and notes.	4.3	1
000015 RCP Malfunctions / 4 000017 RCP Malfunctions (Loss of RC Flow) / 4									0
000022 Loss of Rx Coolant Makeup / 2									0
000025 Loss of RHR System / 4									0
000026 Loss of Component Cooling Water / 8									0
000027 Pressurizer Pressure Control System Malfunction / 3									0
000029 ATWS / 1									0
000038 Steam Gen. Tube Rupture / 3									0
000040 Steam Line Rupture - Excessive Heat Transfer / 4									
BW/E05 Excessive Heat Transfer / 4						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.	4.4	1
000054 Loss of Main Feedwater / 4					05		Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Status of MFW pumps, regulating and stop valves	3.7	1
000055 Station Blackout / 6									0
000056 Loss of Off-site Power / 6						04.18	Knowledge of the specific bases for EOPs.	4	1
000057 Loss of Vital AC Inst. Bus / 6									0
000058 Loss of DC Power / 6									0
000062 Loss of Nuclear Svc Water / 4					05		Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The normal values for SWS-header flow rate and the flow rates to the components cooled by the SWS	2.5	1
000065 Loss of Instrument Air / 8									0
BW/E04 Inadequate Heat Transfer / 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 (SRO)

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
000036 Fuel Handling Accident / 8									1
BW/A08 Refueling Canal Level Decrease / 8					01		Ability to determine and interpret the following as they apply to the (Refueling Canal Level Decrease): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	4	
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						04. 47	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	4.2	1
000061 ARM System Alarms / 7									0
000067 Plant Fire On-site / 9 8									0
000068 Control Room Evac. / 8						01. 23	Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.4	1
BW/A06 Shutdown Outside Control Room / 8									
000069 Loss of CTMT Integrity / 5									0
000074 Inad. Core Cooling / 4									0
000076 High Reactor Coolant Activity / 9									0
BW/A01 Plant Runback / 1									0
BW/A02 Loss of NNI-X / 7									0
BW/A03 Loss of NNI-Y / 7									
BW/A04 Turbine Trip / 4									0
BW/A05 Emergency Diesel Actuation / 6									0
BW/A07 Flooding / 8									0
BW/E03 Inadequate Subcooling Margin / 4									0
BW/E08 LOCA Cooldown / 4									0
BW/E09 Natural Circulation Cooldown / 4									
BW/E13 EOP Rules					01		Ability to determine and interpret the following as they apply to the (EOP Rules): Facility conditions and selection of appropriate procedures during abnormal and emergency	4	1
BW/E14 EOP Enclosures									
K/A Category Totals:	0	0	0	0	2	2	Group Point Total:		4

ES-401												PWR Examination Outline Plant Systems - Tier 2/Group 1 (SRO)		Form ES-401-2
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
003 Reactor Coolant Pump														0
004 Chemical and Volume Control														0
005 Residual Heat Removal														0
006 Emergency Core Cooling														0
007 Pressurizer Relief/Quench Tank											02.36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	4.2	1
008 Component Cooling Water														0
010 Pressurizer Pressure Control														0
012 Reactor Protection								03				Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Incorrect channel bypassing	3.7	1
013 Engineered Safety Features Actuation														0
022 Containment Cooling														0
025 Ice Condenser														0
026 Containment Spray														0
039 Main and Reheat Steam								05				Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Increasing steam demand, its relationship to increases in reactor power	3.6	1
059 Main Feedwater														0
061 Auxiliary/Emergency Feedwater														0
062 AC Electrical Distribution														0
063 DC Electrical Distribution											02.25	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4.2	1
064 Emergency Diesel Generator														0
073 Process Radiation Monitoring														0
076 Service Water								02				Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Service water header pressure	3.1	1
078 Instrument Air														0
103 Containment														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 Form ES-401-2

Plant Systems - Tier 2/Group 2 (SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	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								02				Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty or erratic operation of detectors or compensating components	3.5	1
016 Non-nuclear Instrumentation								01				Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure	3.1	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
034 Fuel Handling Equipment														0
035 Steam Generator														0
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											04.47	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	4.2	1
K/A Category Totals:	0	0	0	0	0	0	0	2	0	0	1	Group Point Total:		3

Facility Name: Date of Exam:

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1. 20	Ability to interpret and execute procedure steps.	4.6	1	4.6	
	2.1. 36	Knowledge of procedures and limitations involved in core alterations.	3	1	4.1	
	2.1. 42	Knowledge of new and spent fuel movement procedures.	2.5	1	3.4	
	2.1. 34	Knowledge of primary and secondary plant chemistry limits.	2.7		3.5	1
	2.1. 41	Knowledge of the refueling process.	2.8		3.7	1
	2.1.					
	Subtotal			3		2
2. Equipment Control	2.2. 01	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	4.5	1	4.4	
	2.2. 13	Knowledge of tagging and clearance procedures.	4.1	1	4.3	
	2.2. 22	Knowledge of limiting conditions for operations and safety limits.	4	1	4.7	
	2.2. 36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	3.1		4.2	1
	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		4.4	1
	2.2.					
	Subtotal			3		2
3. Radiation Control	2.3. 11	Ability to control radiation releases.	3.8	1	4.3	
	2.3. 15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	1	3.1	
	2.3. 04	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2		3.7	1
	2.3. 14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.4		3.8	1
	2.3.					
	2.3.					
	Subtotal			2		2
4. Emergency Procedures / Plan	2.4. 03	Ability to identify post-accident instrumentation.	3.7	1	3.9	
	2.4. 17	Knowledge of EOP terms and definitions.	3.9	1	4.3	
	2.4. 40	Knowledge of SRO responsibilities in emergency plan implementation.	2.7		4.5	1
	2.4.					
	2.4.					
	2.4.					
	Subtotal			2		1
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected KA	Reason for Rejection
1 / 2	001 / AK1.19 replaced by A03 / AK1.2	The subject K/A isn't relevant at the subject facility.
2 / 1	006 / K6.19 replaced by 064 / K6.08	Topic oversampled. 006/K6.19 overlaps with K/A 005 / K3.05 on the Written Exam.
2 / 1	073 / A4.01 replaced by 004 / K4.05	Topic oversampled. 073/A4.01 overlaps with K/A 057 / AK3.03 and 068 / A3.02 on the Written Exam.
2 / 1	078 / A3.01 replaced by 039 / A3.02	Topic oversampled. 078 / A3.01 overlaps with K/A 065 / 2.4.45 and 078 / K1.04 on the Written Exam.
2 / 1	003 / 2.2.37 replaced by 061 / 2.4.30	The subject K/A isn't relevant at the subject facility.
2 / 2	017 / K3.01 replaced by 055 / K3.01	Topic oversampled. 017 / K3.01 overlaps with 038 / EA2.09 on the Written Exam.
2 / 2	079 / 2.2.39 replaced by 015 / K2.01	The subject K/A isn't relevant at the subject facility.
2 / 1 SRO	008 / 2.4.9 replaced by 063 / 2.2.25	Topic oversampled. 008 / 2.4.9 overlaps with 026 / AA1.03, 008 / A2.08, 013 / K2.01 and 026 / K1.02 on the Written Exam.
2 / 1 SRO	010 / A2.01 replaced by 012 / A2.03	Topic oversampled. 010 / A2.01 overlaps with 056 / AA1.03, 016 / A2.01, and 028 / 2.4.6 on the Written Exam.
2 / 2 SRO	034 / K5.03 replaced by 016 / A2.01	Topic oversampled. 034 / K5.03 overlaps with 025 / AK2.02, and G 2.1.36 on the Written Exam.

Three Mile Island
2014 NRC Initial License Written Examination
Written Examination Outline Methodology

The written examination outline was developed using NKEG Version 1.1 software by Westinghouse. Generation of RO and SRO outlines was done randomly from the database containing NUREG 1122 Rev 2 Supp 1 catalog, as pre-suppressed for TMI system design.

Pre-suppression was done by reviewing previously approved, suppressed knowledges and abilities and suppressing them against the appropriate Tier 1 and Tier 2 categories. In accordance with NUREG 1021 Revision 9 supp 1 ES-401 D.1.b Generic 2.2.3 and 2.2.4 were suppressed, also the Generic not specifically listed for sampling of Tiers 1 and 2 were suppressed, except for the Tier 3 sampling.

K/A rejections are listed on ES-401-4 as required by NUREG 1021.

Facility: Three Mile Island	Date of Examination: April 2014
Examination Level: RO ☒ SRO ☐	Operating Test Number: 289-2014-301

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M/R	Calculate an Estimated Critical Boron Concentration in accordance with 1103-15B, ESTIMATED CRITICAL CONDITIONS. 2.1.25 (3.9): Ability to interpret station reference materials such as graphs, curves, tables, etc.
Conduct of Operations	M/R	Perform OP-TM-300-202, QUADRANT POWER TILT AND CORE IMBALANCE USING THE OUT-OF-CORE DETECTOR SYSTEM, and compare to COLR limits. 2.1.37 (4.3): Knowledge of procedures, guidelines, or limitations associated with reactivity management.
Equipment Control	N/R	Reactor Coolant Pump electrical print reading to determine pump operation. 2.2.41 (3.9): Ability to obtain and interpret station electrical and mechanical drawings.
Radiation Control		Category not selected for RO applicants.
Emergency Procedures/Plan	M/S	ERO Notification – Alt path 2.4.39 (3.9): Knowledge of RO responsibilities in emergency plan implementation.

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

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

2014 TMI RO NRC EXAMINATION

CONDUCT OF OPERATIONS: Given a set of plant conditions, calculate the Estimated Critical Boron Concentration for reactor startup. Modified Bank JPM.

CONDUCT OF OPERATIONS: Given a failed computer, perform OP-TM-300-202, QUADRANT POWER TILT AND CORE IMBALANCE USING THE OUT-OF-CORE DETECTOR SYSTEM, and compare to COLR limits. Modified Bank JPM. Calculation is RO/SRO common.

EQUIPMENT CONTROL: Given a set of conditions and Reactor Coolant Pump electrical prints, determine if and where the Reactor Coolant Pump can be operated. New JPM.

RADIATION CONTROL: Given a General Emergency declaration and a failure of the ERO Notification using the World Wide Web, initiate activation of the ERO using the Live Everbridge Agent. Modified Bank JPM.

Facility: Three Mile Island Date of Examination: April 2014

Examination Level: RO ☐ SRO ☒ Operating Test Number: 289-2014-301

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M/R	Review and Approve an Estimated Critical Boron Concentration. 2.1.25 (4.2): Ability to interpret station reference materials such as graphs, curves, tables, etc.
Conduct of Operations	M/R	Review and Approve OP-TM-300-202, QUADRANT POWER TILT AND CORE IMBALANCE USING THE OUT-OF-CORE DETECTOR SYSTEM, and compare to COLR limits. 2.1.37 (4.6): Knowledge of procedures, guidelines, or limitations associated with reactivity management.
Equipment Control	N/R	Reactor Coolant Pump electrical print reading to determine pump operation and Tech Spec implications. 2.2.41 (3.9): Ability to obtain and interpret station electrical and mechanical drawings.
Radiation Control	M/R	Implement the Requirements of ODCM for RMS Operability. 2.3.15 (3.1) Knowedge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
Emergency Procedures/Plan	D/R	Emergency Action Level Identification, Event Declaration, and Protective Action Recommendation. 2.4.44 (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 5 are required.

* Type Codes & Criteria:

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

(D)irect from bank ($\leq$ 3 for ROs; $\leq$ 4 for SROs & RO retakes)

(N)ew or (M)odified from bank ($\geq$ 1)

(P)revious 2 exams ($\leq$ 1; randomly selected)

2014 TMI SRO NRC EXAMINATION

CONDUCT OF OPERATIONS: Review and approve an Estimated Critical Boron Concentration calculation with multiple errors. Modified Bank JPM.

CONDUCT OF OPERATIONS: Given a failed computer, perform OP-TM-300-202, QUADRANT POWER TILT AND CORE IMBALANCE USING THE OUT-OF-CORE DETECTOR SYSTEM, and compare to COLR limits and make final approval recommendation. Modified Bank JPM. Calculation is RO/SRO common.

EQUIPMENT CONTROL: Given a set of conditions and Reactor Coolant Pump electrical prints, determine if and where the Reactor Coolant Pump can be operated and any Tech Spec implications. New JPM.

EQUIPMENT CONTROL: Given a set of conditions with a gas release in progress and various RMS components out of service, determine the actions to be taken IAW facility procedures. Modified Bank JPM

RADIATION CONTROL: Given a set of conditions, determine and declare the appropriate Emergency Action Level and make a Protective Action Recommendation IAW with the TMI Emergency Plan. Bank JPM.

Facility: Three Mile Island Date of Examination: APR 2014

Exam Level: RO ☒ SRO-I ☐ SRO-U ☐ Operating Test Number: 289-2014-301

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. Manually Initiate ESAS (006 A2.12)	L/D/A/S	2
b. Restore Seal Injection with a Loss of ICCW (003 K6.02)	N/A/S	4P
c. Transfer Feedwater Pump From ICS to the Motor Speed Changer (Sys 059 A2.11	D/S	4S
d. Perform Emergency Operations of Reactor Building Emergency Cooling Water (Sys 022) A4.04	M/A/S	5
e. Energize 1E Bus from SBO (Sys 064 A4.01)	L/D/A/S	6
f. Respond IAW OP-TM-MAP-C0101 with Failure (Sys 072) A3.01	A/P/S	7
g. Initiate and Isolate a Reactor Building Purge (Sys 029 K1.01)	D/L/A/S	8
h. Feed from the "C" RCBT and the BAMT (001 A4.02)	N/S	1

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

i. Return a Battery Charger to Service (AP 058) AA1.03	D	6
j. Perform a Cooldown Outside the Control Room (AP 068) G2.1.30	D/E	8
k. Init EB IP (Sys 004) G2.1.30	D/E/R	1

@ 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	

THREE MILE ISLAND 2014 NRC RO EXAMINATION

JPM A – Manually Initiate ESAS. Alternate Path due to no ESAS component response when manual pushbuttons are operated requires the candidate to navigate out of the normal section and into an alternate section of OP-TM-211-901.
Safety significance: Failure to manually initiate ESAS will result in a loss of Reactor Coolant System Inventory.

JPM B – Restore Seal Injection with a Loss of ICCW. New JPM. Candidate will be directed to restore Seal Injection after a trip of the "B" Makeup Pump. Alternate Path 1 due to "A" Makeup Pump not starting, candidate will be required to navigate to a different section of OP-TM-AOP-041. Alternate Path 2 will occur with a loss of ICCW. With no Seal Injection and no ICCW, there is a navigation point (IAAT) to trip the reactor IAW OP-TM-EOP-001 and trip RCP's.
Safety significance: Failure to complete this JPM will result in an extended loss of RCP Seal cooling.

JPM C – Transfer Feedwater Pump From ICS to the Motor Speed Changer. Bank JPM.
Safety significance: Failure to complete this JPM successfully will lead to a Feedwater Pump Trip and a Plant Runback/possible Reactor Trip.

JPM D – Perform Emergency Operations of Reactor Building Emergency Cooling Water. Modified Bank JPM. Alternate path. While Initiating RB Emergency Cooling, one valve will not open and cooler oulet pressures are too high, so the candidate will be required to navigate out of the normal section and into an alternate section of OP-TM-534-901.
Safety significance: Failure to properly control RB Emergency Cooling will result in elevated RB Pressures and Temperatures.

JPM E – Energize the 1E Bus from the SBO. Bank JPM. Involves restoring the "B" train ES Bus from Station Blackout Diesel power to normal Off-Site power.
Safety significance: Improper operation could lead to loss of 1 train of ES components through loss of ES bus, causing significant degradation in coping capability.

JPM F – Respond IAW OP-TM-MAP-C0101 Alarm Response with Failure. Previous NRC. Alternate Path JPM. Automatic actions do not occur IAW the alarm response, requiring navigation to another procedure.
Safety significance Failure to place control tower on "recirculation" following high airborne contamination in the Control Room may result in unnecessary dose for the personnel that must remain to operate the plant.

JPM G – Initiate and Isolate a Reactor Building Purge. Bank JPM. Alternate Path. RB Purge is initiated and is followed by an alarm, requiring navigation to an alarm response and a RB purge isolation.
Safety significance: Failure to complete this JPM will result in an uncontrolled release to the public to occur.

JPM H – Feed from the "C" RCBT and the BAMT. New JPM. A Feed is calculated, lined up, and performed from multiple sources simultaneously.
Safety significance: Failure to complete this JPM correctly will result in the wrong amount of Boron being inserted into the RCS, which will cause a reactivity event.

JPM I – Return a Battery Charger to Service. Bank JPM.
Safety significance: Failure to complete the JPM properly will result in a degraded condition of the Station Batteries.

JPM J – Perform a Cooldown Outside of the Control Room. Bank JPM. Candidate will perform IMA's of OP-TM-EOP-020.
Safety significance: Failure to complete the JPM properly will result in an unstable critical condition for the Reactor and/or the Secondary Plant.

JPM K – Initiate Emergency Boration InPlant. Bank JPM. RCA entry required. Candidates will initiate an Emergency Boration from outside of the Control Room.
Safety significance: Failure to complete the task will result in insufficient shutdown boron concentration.

Facility: Three Mile Island Date of Examination: APR 2014

Exam Level: RO ☐ SRO-I ☒ SRO-U ☐ Operating Test Number: 289-2014-301

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. Manually Initiate ESAS (006 A2.12)	L/D/A/S	2
b. Restore Seal Injection with a Loss of ICCW (003 K6.02)	N/A/S	4P
c. Transfer Feedwater Pump From ICS to the Motor Speed Changer (Sys 059 A2.11	D/S	4S
d. Perform Emergency Operations of Reactor Building Emergency Cooling Water (Sys 022) A4.04	M/A/S	5
e. Energize 1E Bus from SBO (Sys 064 A4.01)	L/D/A/S	6
f. Respond IAW OP-TM-MAP-C0101 with Failure (Sys 072) A3.01	A/P/S	7
g. Initiate and Isolate a Reactor Building Purge (Sys 029 K1.01)	D/L/A/S	8
h.		

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

i. Return a Battery Charger to Service (AP 058) AA1.03	D	6
j. Perform a Cooldown Outside the Control Room (AP 068) G2.1.30	D/E	8
k. Init EB IP (Sys 004) G2.1.30	D/E/R	1

@ 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	

THREE MILE ISLAND 2014 NRC RO EXAMINATION

JPM A – Manually Initiate ESAS. Alternate Path due to no ESAS component response when manual pushbuttons are operated requires the candidate to navigate out of the normal section and into an alternate section of OP-TM-211-901.
Safety significance: Failure to manually initiate ESAS will result in a loss of Reactor Coolant System Inventory.

JPM B – Restore Seal Injection with a Loss of ICCW. New JPM. Candidate will be directed to restore Seal Injection after a trip of the "B" Makeup Pump. Alternate Path 1 due to "A" Makeup Pump not starting, candidate will be required to navigate to a different section of OP-TM-AOP-041. Alternate Path 2 will occur with a loss of ICCW. With no Seal Injection and no ICCW, there is a navigation point (IAAT) to trip the reactor IAW OP-TM-EOP-001 and trip RCP's.
Safety significance: Failure to complete this JPM will result in an extended loss of RCP Seal cooling.

JPM C – Transfer Feedwater Pump From ICS to the Motor Speed Changer. Bank JPM.
Safety significance: Failure to complete this JPM successfully will lead to a Feedwater Pump Trip and a Plant Runback/possible Reactor Trip.

JPM D – Perform Emergency Operations of Reactor Building Emergency Cooling Water. Modified Bank JPM. Alternate path. While Initiating RB Emergency Cooling, one valve will not open and cooler oulet pressures are too high, so the candidate will be required to navigate out of the normal section and into an alternate section of OP-TM-534-901.
Safety significance: Failure to properly control RB Emergency Cooling will result in elevated RB Pressures and Temperatures.

JPM E – Energize the 1E Bus from the SBO. Bank JPM. Involves restoring the "B" train ES Bus from Station Blackout Diesel power to normal Off-Site power.
Safety significance: Improper operation could lead to loss of 1 train of ES components through loss of ES bus, causing significant degradation in coping capability.

JPM F – Respond IAW OP-TM-MAP-C0101 Alarm Response with Failure. Previous NRC. Alternate Path JPM. Automatic actions do not occur IAW the alarm response, requiring navigation to another procedure.
Safety significance Failure to place control tower on "recirculation" following high airborne contamination in the Control Room may result in unnecessary dose for the personnel that must remain to operate the plant.

JPM G – Initiate and Isolate a Reactor Building Purge. Bank JPM. Alternate Path. RB Purge is initiated and is followed by an alarm, requiring navigation to an alarm response and a RB purge isolation.
Safety significance: Failure to complete this JPM will result in an uncontrolled release to the public to occur.

JPM H – Not selected for SROs.

JPM I – Return a Battery Charger to Service. Bank JPM.
Safety significance: Failure to complete the JPM properly will result in a degraded condition of the Station Batteries.

JPM J – Perform a Cooldown Outside of the Control Room. Bank JPM. Candidate will perform IMA's of OP-TM-EOP-020.
Safety significance: Failure to complete the JPM properly will result in an unstable critical condition for the Reactor and/or the Secondary Plant.

JPM K – Initiate Emergency Boration InPlant. Bank JPM. RCA entry required. Candidates will initiate an Emergency Boration from outside of the Control Room.
Safety significance: Failure to complete the task will result in insufficient shutdown boron concentration.

Facility:	Three Mile Island	Scenario No.: 1	Op Test No.: 289-2014-301
Examiners:		Operators:	
Initial Conditions:	• (Temporary IC-xxx)		
	• 100% Power, MOL.		
	• SBO OOS For Maintenance, expected to return to service in 10 hours.		
	• Crane work is occurring on the West side of the Plant to stage new piping.		
Turnover:	Maintain 100% Reactor Power		
Critical Tasks:	• Isolate Possible RCS Leak Paths (CT-3)		
	• Establish Natural Circulation RC Flow (CT-12)		
	• Turbine Trip (CT-18)		

Event No.	Malf. No.	Event Type*	Event Description
1	DHR32	TS CRS	BWST Level Lowers, entry into OP-TM-MAP-E0204.
2	MS03A	C CRS C ARO	Isolable Steam Leak in Turbine Bldg, entry into OP-TM-AOP-051. (ARO: Isolate Steam Leak)
3	IC23	I CRS I URO I ARO	SG/RX Demand Fails to Zero Volts, entry into OP-TM-AOP-070. (URO/ARO: Coordinate to stabilize plant in ICS HAND control)
4	TU01D	N CRS R URO N ARO	High Vibrations on Main Turbine, entry into OP-TM-MAP-K0201 and 1102-4. (URO: Lower Reactor Power, ARO: Manually Adjust Feedwater)
5	ES08A	I CRS I URO I ARO	Inadvertent 1600# ESAS Signal, entry into OP-TM-AOP-046. (URO: Immediate Manual Actions, ARO: Restores Letdown)
6	EDR14 EG07A	TS CRS C ARO	Loss of 1D 4k Bus, entry into OP-TM-AOP-013. (ARO: Places LO-P-6 in PTL)
7	ED12	I CRS I URO	Loss of Stator Coolant Pumps, Main Turbine fails to automatically trip. (URO: Trip Reactor, ARO: Adjust Main Feedwater)
8	ED01	M CRS M URO M ARO	Loss of Offsite Power, entry into OP-TM-AOP-020.
9	EG07B	C CRS C URO	EG-Y-1B Trips, Station Blackout. (URO: Isolates Cooling paths to RCP's)

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

Three Mile Island NRC Scenario #1

Event #1: When the crew has accepted the watch, the Lead Examiner will cue the lowering level of the BWST due to a crane piercing the tank at the 53.5 ft level.

The crew will diagnose the low level in the BWST by lowering level on the BWST level indicators (CC and CR) and annunciator E-3-4 in alarm. Although it initially drops rapidly, the crew will identify it as steady at approximately 54.5 ft. The CRS will review and declare the following Tech Spec: 3.3.1.1.a.

Once the Tech Spec has been declared, the scenario can continue.

Event #2: The Lead Examiner will cue the Isolable Steam Leak in the Turbine Building.

The operators will diagnose a Secondary Side Steam Leak based on a lowered efficiency of the Secondary Plant (Megawatts, Header Pressure, OTSG pressures, etc.)

Upon a Steam Leak in the Turbine Building, the following Critical Safety Functions are affected:

> **CSF 1**, Reactivity and Reactor Power Control: Maintain control of the fission process, maintain the capability to shutdown the reactor and the capability to maintain the reactor in a shutdown condition. Control energy production and reactor power distribution based on design limits and current core heat removal capability. A secondary side steam leak will bypass steam away from the Turbine. If the steam leak is large enough, electrical generation will lower. ICS will raise reactor power to raise MWe back up to ICS demanded MW. If the steam leak causes Tave to lower, ICS will also try to raise reactor power to recover Tave.
>
> **CSF 7**, Electrical Power: Provide electrical power as required to accomplish the other Critical Safety Functions. Provide AC and DC power for emergency equipment operation and instrumentation systems. A Steam Leak may affect power to non-safety related equipment in the Turbine Building.
>
> **CSF 9**, Fire Protection and Remote Shutdown Capability: Maintain means to prevent detect and suppress fires, as well as the capability to perform a plant shutdown without access to the Control Room. A Steam Leak may cause fire alarms to actuate. This could be the first indication of a Steam Leak to the Control Room. May cause sprinkler systems to actuate or fire dampers to close.

The crew will diagnose the Steam Leak and the CRS will enter OP-TM-AOP-051, Secondary Side High Energy Leak. OP-TM-AOP-051 is entered for Steam Leaks that affect large portions of the plant and therefore it is not obvious to the operator what needs to be done initially to isolate the leak.

The affected Building is determined and from there, OP-TM-AOP-051 systematically attempts to isolate the leak remotely from the Control Room while taking steps to minimize the adverse effects of a steam environment on safety related equipment. The OP-TM-AOP-051 mitigation strategy for a Steam Leak in the Turbine Building is as follows:

- Attempt to isolate the leak from the Control Room.
- Shutdown and Cooldown the plant in a controlled manner to minimize pressure surges that could make the leak worse. Shutdown may have to be done quickly or the plant may have to be tripped depending on the circumstances.

The ARO will isolate Steam Leak by closing the appropriate valve, MS-V-5B. This can be performed because the steam supplies to the Main Feedwater Pumps are as follows:

- Below 25% power, Main Steam, only, supplies the Main Feedwater Pumps.
- Between 25% and 40% power, Main Steam supplements Extraction Steam as supplies to the Main Feedwater Pumps.
- Above 40% power, Extraction Steam, only, supplies the Main Feedwater Pumps.

Once the Steam Leak has been isolated, the scenario can continue.

Event #3: The Lead Examiner will cue the failure of the SG/RX Demand Station to Zero Volts. This will cause an ICS transient, which if not responded to swiftly, will cause a Reactor Trip.

The crew will diagnose the ICS failure by a rapid change in RCS pressure, Reactor Power rising, multiple annunciator alarms, and/or changes in indications at multiple ICS stations. Entry into OP-TM-AOP-070, PRIMARY TO SECONDARY HEAT TRANSFER UPSET will be required based on RCS pressure not being controlled in ICS AUTO.

> "RCS pressure is not being controlled" requires the operator to make a subjective determination, based on their skills, training, and experience. A determination that RCS pressure is being controlled should include the following elements: 1) The reason for the transient is understood 2) RCS pressure response is consistent with the expected response for the event 3) Automatic or manual control in accordance with normal operating procedures is effectively controlling RCS pressure. A conservative assessment (i.e. concluding that RCS pressure is not being controlled) is appropriate when the three conditions above cannot be satisfied. ICS failures are one class of events that can lead to an upset in primary to secondary heat transfer. Most ICS failures can be mitigated by use of the appropriate manual control normal operating procedures.

This entry into AOP-070 is unique from the other scenarios because Reactor Power will lower while RCS pressure rises. The URO will have to place the Diamond Panel in manual to control Reactor Power or RPS will trip the reactor on high RCS pressure.

Once the plant is stabilized in ICS HAND control, the scenario can continue.

Event #4: The Lead Examiner will cue the High Vibrations on Main Turbine.

The crew will diagnose the High Vibrations on Main Turbine by Annunciator K-2-1 in alarm, and multiple PPC points in alarm. The crew will commence a power reduction to < 45% with ICS in manual to trip the Main Turbine. This is the reactivity manipulation for the scenario.

Once sufficient reactivity manipulation has occurred, the scenario can continue.

Event #5: The Lead Examiner will cue the Inadvertent 1600# ESAS Signal. The crew must quickly recognize the condition and perform the required Immediate Manual Actions to minimize the RCS pressure transient and pressurizer in-surge due to HPI. Additionally, while at power, immediately reducing HPI also minimizes the possibility of a reactor trip on high RCS pressure. The following Critical Safety Functions are affected by an inadvertent ESAS signal:

CSF 1, Reactivity & Reactor Power Control: Maintain control of the fission process, maintain the capability to shutdown the reactor and the capability to maintain the reactor in a shutdown condition. Control energy production and reactor power distribution based on design limits and current core heat removal capability**. HPI Actuation** will insert negative reactivity as borated water from the BWST will be injected into the RCS. ICS will pull control rods to maintain ULD demand and RCS Tave. Potential all rods out condition. As primary side power decreases, ICS cross limits will lower feedwater flow in an attempt to match primary to secondary heat removal.

CSF 2, Reactor Vessel Inventory Control: Provide the means to maintain the core covered with sub cooled water. **HPI Actuation** will isolate RCS letdown and normal makeup. HPI will cause pressurizer level to rise. MU tank level will lower (MUT level/pressure requirements). BWST level will lower (BWST TS level).

CSF 3, RCS Integrity: Maintain the capability to control heatup and cooldown rates and control RCS pressure to prevent reactor vessel brittle fracture or LTOP events. Maintain RCP seal cooling to prevent excessive loss of RCS inventory through RCP seals. **HPI Actuation** will cause RCS pressure to rise. Seal Injection is not affected by an ES actuation. AOP-046 has the operator secure the remaining MU pump if the MU-V-16s cannot be closed from the control room. This action immediately terminates HPI and seal injection. The thermal barrier heat exchangers provide adequate seal cooling when the SI is secured. AOP-046 ensures adequate thermal barrier cooling prior to terminating seal injection. Once MU-V-36 and 37 are opened locally and the appropriate MU-V-16s are closed, a makeup pump is restarted and SI is re-established.

CSF 4, Core Heat Removal: Provide the capability to remove core heat production at all times. **HPI Actuation** will inject cold BWST into the RCS. The RCS will also cool due to the negative reactivity insertion from the injected BWST water lowering core power.

CSF 5, Containment Integrity: Provide means to prevent or minimize fission product release to the environment. (1) Maintain containment pressure below design and (2) Provide capability to isolate the containment when required. An inadvertent **HPI Actuation** will start the Reactor River system, and the RB cooling fans will be running in slow speed. Normal cooling to the RB cooling fans will be isolated. RB Temperature and pressure will lower due to the actuation of the RR system. The degree of temperature reduction would depend of river temperature. Building Spray System valves will open to align the BWST to the RB, but the BS pumps would not start unless a 30# signal is present. Closure of the containment isolation valves under the HPI signal would not adversely affect their associated systems.

CSF 7, Electrical Power: Provide electrical power as required to accomplish the other Critical Safety Functions. Provide AC and DC power for emergency equipment operation and instrumentation systems. An inadvertent **HPI Actuation** will start the emergency diesel generators unloaded.

CSF 8, Auxiliary Emergency Systems: Provide equipment cooling (closed cooling & ventilation), and other support requirements to accomplish the other Critical Safety Functions. Provide Instrument Air for operation of EFW, ADVs, RCP Support Systems and some containment isolation valves. An inadvertent **HPI Actuation** will start support systems to support ECCS and RB cooling systems. The DC and DR pumps will start to support MUP and DHP cooling which would be running during an inadvertent actuation. ES selected NS and NR pumps will start. Two NR and NS pumps are normally running. There is a potential for three NS pumps running which would start an overcooling of the NS system.

CSF 9, Fire Protection & Remote Shutdown Capability: Maintain means to prevent, detect, and suppress fires, as well as the capability to perform a plant shutdown without access to the Control Room. An inadvertent **HPI Actuation** would trip FS-P-2.

The crew will diagnose the Inadvertent “A” 500# ESAS Signal by multiple annunciators in alarm, “A” Train components in their ES actuated state, and/or “A” EDG running, while all primary indications appear steady or rising (RCS pressure not at 500#).

The URO will perform the Immediate Manual Actions of OP-TM-AOP-046, INADVERTANT ESAS.
The ARO will restore letdown IAW OP-TM-211-950 (performing the appropriate portion of the procedure when restoring from isolation following an ESAS signal).

Once the plant is stabilized and Letdown is restored, the scenario can continue.

Event #6: The Lead Examiner will cue the Loss of the 1D 4Kv Bus.

Upon a Loss of the 1D 4Kv Bus, the following Critical Safety Functions are affected:

CSF 1, Reactivity & Reactor Power Control: Maintain control of the fission process, maintain the capability to shutdown the reactor and the capability to maintain the reactor in a shutdown condition. Control energy production and reactor power distribution based on design limits and current core heat removal capability. A loss of the 1D 4Kv bus will not affect reactor shutdown capability. Emergency boration must be performed using "B" Train as MU-V-14A, MU Pump Suction Valve from BWST, will be unavailable.
CSF 2, Reactor Vessel Inventory Control: Provide the means to maintain the core covered with sub cooled water. A loss of the 1D 4Kv bus will cause Train A Emergency Core Cooling Systems (HPI and LPI) to be inoperable. Train B is unaffected.
CSF 3, RCS Integrity: Maintain the capability to control heatup and cooldown rates and control RCS pressure to prevent reactor vessel brittle fracture or LTOP events. Maintain RCP seal cooling to prevent excessive loss of RCS inventory through RCP seals. A loss of the 1D 4Kv bus will cause RC-V-1, Pressurizer Spray Valve, and the Emergency power to Pressurizer Group 8 Heaters to be unavailable.
CSF 4, Core Heat Removal: Provide the capability to remove core heat production at all times. A loss of the 1D 4Kv bus will cause EF-P-2A and DHR Train A to be inoperable. EF-P-2B, EF-P-1, and DHR Train B are available.
CSF 5, Containment Integrity: Provide means to prevent or minimize fission product release to the environment. (1) Maintain containment pressure below design and (2) Provide capability to isolate the containment when required. A loss of the 1D 4Kv bus causes Reactor Building Emergency Cooling Train A and Building Spray Train A to be unavailable. Train B is available as well as Normal Reactor Building Cooling.
CSF 7, Electrical Power: Provide electrical power as required to accomplish the other Critical Safety Functions. Provide AC and DC power for emergency equipment operation and instrumentation systems. A loss of the 1D 4Kv bus due to a bus fault will prevent use of the Emergency Diesel Generators. Prolonged loss of AC increases the risk of a Loss of DC and Vital busses.
CSF 8, Auxiliary Emergency Systems: Provide equipment cooling (closed cooling & ventilation), and other support requirements to accomplish the other Critical Safety Functions. Provide Instrument Air for operation of EFW, ADVs, RCP Support Systems and some containment isolation valves. A loss of the 1D 4Kv bus will cause Train A of all emergency cooling support systems (Control Building cooling, Nuclear Services Closed Cooling Water cooling, Intermediate Closed Cooling water cooling, River Water Systems, Equipment ventilation, etc.) to be unavailable. Train B is unaffected.
CSF 9, Fire Protection & Remote Shutdown Capability: Maintain means to prevent, detect, and suppress fires, as well as the capability to perform a plant shutdown without access to the Control Room. A loss of the 1D 4Kv bus will cause the Relay Room Cardox and Screen House detection systems to be inoperable. Fire watches will be initiated.

The operators will diagnose a Loss of the 1D 4Kv Bus based on a loss of running equipment powered from the 1D 4Kv bus (Secondary Closed Cooling Water Pumps, Secondary River Water Pumps, Intermediate Closed Cooling Water Pumps, etc.), LO-P-6 starting, and half of the Control Room lighting out.

The CRS will enter OP-TM-AOP-013, LOSS OF 1D 4160V BUS. The ARO will place the Control Switch for LO-P-6 to Pull-To-Lock to minimize loading on the Battery. The CRS will evaluate and declare Tech Spec 3.7.2.f:

- T.S. 3.7.2 – The reactor shall not remain critical unless all of the following requirements are satisfied:
  - 3.7.2.f – The Engineered Safeguards bus, switchgear, load shedding, and automatic diesel start systems shall be operable except as provided in Specification 3.7.2.c above and as required for testing.

Once LO-P-6 is in Pull-To-Lock and the Tech Spec has been declared, then the scenario can continue.

Event #7: The lead examiner will cue the Loss of Stator Coolant Pumps, causing a lack of Main Generator Stator Cooling.

The crew will diagnose a Loss of Stator Cooling Pumps by Main Annunciators L-2-7, GEN STATOR STBY CLG PUMP RUN, and Main Annunciator L-1-7, GEN STATOR CLG LOSS RUNBACK, in alarm, and no operating Stator Coolant Pump indications on PCL.

The crew will enter OP-TM-MAP-L0107 and OP-TM-MAP-L0207. The ARO will attempt to start the standby pump from PCL and will identify that it will not start. The crew will identify that the Turbine Control Valves are not closing as expected and, if the condition continues for more than 3.5 minutes, that the Main Turbine did not automatically trip.

- IAW OP-TM-MAP-L0107:
  - Automatic Actions:
    - Turbine Control Valves close at 23.4% vmo/min.
    - Turbine trips after 2 minute time delay should turbine not runback to approximately 95% power.
    - Turbine trips after 3.5 minute time delay should turbine not runback to approximately 29.6% power.

The CRS will enter OP-TM-EOP-001, Reactor Trip. The URO will trip the reactor (CT-18). The Main Turbine will fail to automatically trip and the URO will trip it manually. The ARO will adjust Main Feedwater flow to avoid an overcooling event.

Once the Reactor and Main Turbine have been tripped and a symptom check has been performed, then the scenario can continue.

Event #8: The lead examiner will cue the Loss of Offsite Power.

The crew will diagnose a Loss of Offsite Power by the Control Room lighting going dark for 10 seconds, followed by half of the lights returning once the B Emergency Diesel Generator powers the 1E 4Kv bus, as well as several alarms indicating a loss of the 4 bus and 8 bus, and a loss of seal injection and letdown. The CRS will direct entry into OP-TM-AOP-020, Loss of Station Power.

OP-TM-AOP-020 addresses two types of Loss of Station Power events. (1) Loss of Offsite Power with one or both diesels supplying the ES 4160V busses and (2) a loss of offsite power with no diesels supplying the ES 4160V busses.

- If one or both diesels are supplying the ES busses, the procedure walks the operators through ensuring RCS cooling and RCP seal cooling is established. As long as one diesel is available, the reactor can be stabilized and the operators can methodically walk through powering additional busses and starting additional equipment.

The ARO will establish Natural Circulation by feeding the OTSG's IAW Rule 4 (CT-12) and verify that Natural Circulation exists IAW Guide 10, Natural Circulation.

- IAW Guide 10:
  - Natural Circulation exists when the following conditions exist:
    - RCS differential temperature develops and stabilizes <50F.
    - Thot < 600F.
    - Incore temperatures stabilize and track Thot indications.
    - Tcold reflects OTSG saturation temperature for the existing OTSG pressure.
    - Primary to Secondary heat transfer is demonstrated by steaming or feeding OTSGs.
    - Adequate SCM exists.

Once Natural Circulation has been established, then the scenario can continue.

Event #9: The lead examiner will cue the Loss of EG-Y-1B.

The crew will diagnose a Loss of Offsite Power by the Control Room lighting going dark as well as several alarms indicating a loss of EG-Y-1B. The CRS will direct entry into OP-TM-AOP-020, Loss of Station Power, Section 4.0, Station Blackout.

OP-TM-AOP-020 addresses two types of Loss of Station Power events. (1) Loss of Offsite Power with one or both diesels supplying the ES 4160V busses and (2) a loss of offsite power with no diesels supplying the ES 4160V busses.

- In a Station Blackout (no power to either 4160V ES bus), immediate attempts are made to energize the ES busses. Actions are taken to prevent inventory loss, including protecting the RCP seals. The operators take action to minimize DC loads to maximize station battery life.

Scenario Set-up
NRC Scenario 1

The URO will place Intermediate Closed Cooling Water Pumps in Pull-To-Lock and ensure that RCP Seal Return and Seal Injection Valves are closed (CT-3).

The scenario can be terminated when the Main Turbine has been tripped, Natural Circulation has been established, Intermediate Closed Cooling Water Pumps have been placed in Pull-To-Lock, and RCP Seal Return and Seal Injection lines have been isolated.

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-3 - Isolate Possible RCS Leak Paths – All isolable leaks should be isolated, if possible. There are several possible leaks in the RCS which may be isolated by closing certain valves. These may include:

- PORV and PORV Block Valve.
- Pressurizer Spray Valve and Spray Block Valve
- Pressurizer Vent Valves.
- Pressurizer Sample Valves.
- Hot Leg High Point Vent Valves.
- Reactor Vessel Head Vent Valves.

Station Blackout procedures include provisions for minimizing primary inventory losses, including:

- Isolating all Letdown flows.
- Closing Seal Return Valves.
- Isolating other known leak paths.

Protecting against RCP Seal LOCA outside of the following limits should be considered **grounds for failure of the critical task**:

- Isolating Seal Return Lines while any Reactor Coolant Pump #1 seal inlet temperature > 235°F.
- Isolating Intermediate Closed Cooling Water System while any Reactor Coolant Pump #1 seal inlet temperature > 235°F.

Safety Significance: Elevated RCP seal temperatures will result in increased seal leakage of approximately 21 gpm / pump. To avoid seal damage and excessive seal leakage, do **not** restore RCP seal injection.
To avoid water hammer, thermal barrier cooler damage, and RCS leakage to the ICCW system, do **not** restore RCP thermal barrier cooling.
Either Seal Injection restoration or Intermediate Closed Cooling Water restoration would lead to a reduction of RCS inventory.

Cues:
1. Computer indication for RCP #1 Seal Inlet Temperature

Performance Indicators:
1. Operation of associated RCP Seal Valve controls.
2. Operation of associated RCP Seal Pump controls.

Feedback:
1. Indications of Valve status indications associated with RCP Seals.
2. Indications of Pump status indications associated with RCP Seals.

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-12 – Establish Natural Circulation RC Flow – Whenever forced RC flow is not available, NC flow should be established. Maintaining primary to secondary heat transfer via NC eliminates the need to add RC to the RB as would occur with the back up feed and bleed HPI core cooling mode.

- o If primary to secondary heat transfer has been lost, then establish and maintain appropriate SG levels in accordance with Rule 4.0.
- o Reduce SG pressure using the TBVs/ADVs to establish a positive primary to secondary side ΔT of - 50°F.
- o RCS pressure should be maintained constant or slightly increasing using MU or HPI. RCS pressure should not be increased if PTS guidance is invoked.
- o Trying to establish Natural Circulation RC flow outside of the following limits should be considered **grounds for failure of the critical task**:
    - ▪ Establish Emergency Feedwater flow IAW Rule 4 to each OTSG (less than 515 gpm total), with a target band of 50-85% in the Operating Range.
    - ▪ Establish Natural Circulation prior to transitioning into OP-TM-EOP-009, HPI Cooling.

Safety Significance: Enhances the transient mitigation capability of the plant by maintaining SGs operable and eliminates the need to add RC to the RB as with HPI Cooling.

Cues:
- Low RC flow alarm
- Verbal alert by plant staff that all RCPs have tripped
- SCM monitor and associated alarms
- P-T display and associated alarms

Performance Indicators:
- Operation of EFW/FW pump and valve controls
- Operation of TBV/ADV controls
- Operation of MU/HPI pump and valve controls

Feedback:
- Verbal verification that natural circulation has been established
- SG pressure
- RC temperature

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-18 – Turbine Trip - Whenever conditions exist such that a reactor trip is required, then the normally redundant actions of tripping the reactor and main turbine should be accomplished immediately. Tripping the main turbine provides assurance of a redundant trip signal to the main turbine electro-hydraulic control unit.

- Due to Excessive Heat Transfer concerns, tripping the Main Turbine outside of the following limit should be considered **grounds for failure of the critical task**:
    - Trip the Main Turbine to ensure that it does not cause a continuous excessive cooldown rate or Tcold to be less than 329°F.
    - Once the reactor is shutdown, prompt isolation of the turbine steam flow path is significant. Until the major steam flow path through the turbine to the condenser is isolated, RCS heat removal will be much larger than heat generation, and a rapid RCS cooldown will continue. (Source: OP-TM-EOP-0011)

Safety Significance: When the reactor is tripped (shutdown), steam flow to the main turbine must be stopped in order to maintain the appropriate primary to secondary heat balance. When the appropriate primary to secondary heat balance is established, the normal heat removal systems are available for plant control thus enhancing the transient mitigation capability of the plant. If the turbine steam flow path is not isolated after a rapid reduction in reactor heat generation (reactor trip), extremely rapid RCS cooldown is possible. Prompt operator action can minimize the extent of this overcooling and potential consequences to RCS pressure boundary. A prolonged rapid cooldown will complicate plant control and could challenge OTSG tube integrity or Reactor Vessel integrity.

Cues:

- Visual indications (closed generator output and exciter breakers, main turbine stop and control valves are not closed)
- P-T display and associated alarms
- Verbal alert by plant staff that all main turbine stop and control valves are not closed immediately following actuation of a reactor trip signal
- Verbal alert by plant staff that main alternator output/exciter breakers are not open immediately following actuation of a reactor trip signal

Performance Indicators:

- Operation of control room manual main turbine trip pushbutton
- Main turbine trip alarm
- Main turbine-generator exciter alarms
- Main turbine-generator breaker status alarms

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-18 – Turbine Trip (cont)

Feedback:

- RC temperature and pressure
- SG level and pressure
- Mega-Watt electric indication
- Main turbine-generator breaker status indications
- Verbal notification by plant staff of main turbine trip status

Scenario Set-up
NRC Scenario 1

Industry Experience:

1. OE28735 – Main Turbine High Vibration Trips (Palo Verde Unit 1 and 3) (5/5/09)

2. Fort St. Vrain Loss of all AC Power (Blackout) (10/27/83)

3. SOER 99-1 Loss of Grid (12/99)

4. TMI Inadvertent ESAS Actuation Due to Operator error (7/2/90)

5. OE25328 – Engineered Safeguards Actuation System (ESAS) Relay Failure Contributes to Inadvertant Safety System Actuation (Three Mile Island Unit 1) (6/27/07)

PRA

- Diesel Generator 1A loss (Risk Increase Factor)

- Loss of Offsite Power (Initiating Event)

Facility:	Three Mile Island	Scenario No.: 2	Op Test No.: 289-2014-301
Examiners:		Operators:	
Initial Conditions:	• (Temporary IC-165)		
	• 100% Power, MOL		
	• SBO OOS For Maintenance, expected to return to service in 10 hours		
	• Crane work is occurring on the West side of the Plant to stage new piping.		
Turnover:	Maintain 100% Reactor Power		
Critical Tasks:	• Minimize SCM (CT-7)		
	• Limit Uncontrolled Radiation Release (CT-21)		
	• Reduce Steaming/Isolate Affected SGs (CT-22)		

Event No.	Malf. No.	Event Type*	Event Description
1	MSR01	C CRS C URO C ARO	MSIV Inadvertent Closure, entry into OP-TM-PPC-L2204. (URO: Lowers power in ICS Auto, ARO: Opens MS-V-1A)
2	RW02C	TS CRS C ARO	NR-P-1C Trips, NR-P-1B Fails to Auto-Start, entry into OP-TM-MAP-B0105, and OP-TM-MAP-B0205 (ARO: Starts NR-P-1B from CR)
3	MU19D R	N CRS C URO C ARO	Reactor Coolant Pump Seal Leakage, entry into OP-TM-AOP-040. (URO: Lower Reactor Power, ARO: Manually Control Feedwater Pumps)
4	ED09D	TS CRS C ARO	Loss of Vital Bus D, entry into OP-TM-AOP-018 (ARO: Restore Control Building Ventilation)
5	IC20	I CRS I URO I ARO	Total FW Demand Fails to Zero Volts, ICS Transient, entry into OP-TM-AOP-070. (URO/ARO: Coordinate to stabilize plant in ICS HAND control)
6	TH17B	TS CRS R URO N ARO	~30 gpm "B" OTSG Tube Leak (TS), entry into OP-TM-EOP-005 (URO: Guide 9)
7	TH16B	M CRS M URO M ARO	~800 gpm "B" OTSG Tube Rupture with an elevated offsite dose, entry into OP-TM-EOP-005
8	Override	C CRS C URO	MU-P-1A Overcurrent. (URO: Secure MU-P-1A)

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

Three Mile Island NRC Scenario #2

Event #1: When the crew has accepted the watch, the Lead Examiner will cue the Closure of MS-V-1A. The crew will identify this by the green closed light lit, white test light lit (during travel) and the red open light not lit (after travel is complete) (CC). The crew will enter OP-TM-PPC-L2204, which will direct lowering power less than 90% and reopening MS-V1A.

When MS-V-1A has been opened, the scenario can continue.

Event #2: When the crew has accepted the watch, the Lead Examiner will cue trip of the "C" Nuclear River Pump. "B" Nuclear River Pump fails to auto-start in standby, leaving only one (1) Nuclear River Pump running. One Nuclear River Pump may not be sufficient to cool both the Nuclear Service Closed Cooling System (NSCCW) and the Intermediate Closed Cooling System (ICCW).

It is considered a loss of NSCCW if NSCCW temperatures reach 100°F, and the following Critical Safety Functions are affected:

> **CSF 4**, Core Heat Removal: Provide the capability to remove core heat production at all times: Loss of Nuclear Services cooling function: RC pumps must shutdown. Natural Circulation will be used RCS heat removal.
> **CSF 8**, Auxiliary Emergency Systems: Provide equipment cooling (closed cooling and ventilation), and other support requirements to accomplish the other Critical Safety Functions. Provide Instrument Air for operation of EFW, ADVs, RCP Support Systems, and some containment isolation valves: Loss of Nuclear Services cooling function: Other CSFs are affected as follows: (1) the reliability of safety related power sources and instrumentation system is degraded by the loss of the control building chillers and (2) the reliability of the decay closed pump motors and emergency feed pump motors is degraded by the loss of cooling to the area ventilation coolers.
> **CSF 10**, Chemistry Control: Provide the means to monitor and control primary and secondary water chemistry in order to ensure the long term reliability of plant systems and limit the potential release of radioactive materials: Loss of Nuclear Services cooling function would result in the loss of the capability to obtain an RCS or OTSG sample.

It is considered a loss of ICCW if ICCW temperatures reach 120°F, and the following Critical Safety Functions are affected:

> **CSF 1**, Reactivity and Reactor Power Control: Maintain control of the fission process, maintain the capability to shutdown the reactor and the capability to maintain the reactor in a shutdown condition. Control energy production and reactor power distribution based on design limits and current core heat removal capability. Loss of Intermediate Component Cooling: The reactor is tripped in the event of loss of cooling to the CRD stators in order to prevent stator damage. Loss of CRD stator cooling would not prevent CRD insertion on RPS actuation. Maintaining reactor shutdown is not affected by loss of IC component cooling.

CSF 3, RCS Integrity: Maintain the capability to control heatup and cooldown rates and control RCS pressure prevent reactor vessel brittle fracture or LTOP events. Maintain RCP seal cooling to prevent excessive loss of RCS inventory through RCP seals. Loss of Intermediate Component Cooling: One of two RCP seal cooling methods is lost. Loss of seal injection would require RCP shutdown. If SI is lost, overheating of RCP seals is likely. If seal injection maintained, solid operation may be required due to the loss of letdown.

The crew will diagnose the trip of NR-P-1C by an amber disagreement light on the NR-P-1C control switch and Annunciator alarms A-1-5 and A-2-5. The ARO will manually start "B" Nuclear River Pump to provide sufficient cooling for NSCCW and ICCW. The CRS will identify and declare the following Tech Spec: 3.3.2.

When NR-P-1B is running and the Tech Spec has been declared, the scenario can continue.

Event #3: The Lead Examiner will cue the Reactor Coolant Pump Seal Leak on RC-P-1D. IAW OP-TM-AOP-040, RCP #1 SEAL FAILURE, Basis Document:

The objective is to shutdown the RCP before any further significant RCP damage occurs. The goal is to attempt to reduce power to allow stopping pump without having to trip the reactor. If tripping the RCP would challenge RPS, then the reactor will be tripped, the IMAs and symptom check performed and then the affected RCP tripped.

These actions should be performed promptly. If seal leakoff flow exceeds 8 gpm or temperature exceeds limits at seal water to radial bearing or #1 seal inlet, the RCP must be shutdown immediately. Once an RCP is tripped, #1 seal return is isolated (close MU-V-33) and the #2 seal becomes the primary seal.

The crew will diagnose the #1 Seal Leak on RC-P-1D by the PPC alarm, the digital recorder (PC), and Lab Seal D/P indicated abnormal (CC). The CRS will announce entry into OP-TM-AOP-040 and will determine that, based on Seal Leakage being greater than 6 gpm but less than 8 gpm, a power reduction will be made in order to secure the RCP.

The URO will have to reduce power with ICS in AUTO to secure the RCP. The ARO will place Main Feedwater Pumps to HAND control per 1102-4, Power Operation.

Once the "D" Reactor Coolant Pump is tripped, then the scenario can continue.

Event #4: The Lead Examiner will cue the Loss of Vital Bus "D".

The effects of a loss of VBD which are significant to plant safety or operation are numerous. For each effect the required compensatory action is described in OP-TM-AOP-018.

This procedure stabilizes the plant and performs compensatory actions for equipment failures. It is considered a loss of Vital Bus "D" if the OTSGs are being used for RCS heat removal and an unplanned deenergization of VBD has occurred, and the following Critical Safety Functions are affected:

CSF 1, Reactivity and Reactor Power Control: Maintain control of the fission process, maintain the capability to shutdown the reactor and the capability to maintain the reactor in a shutdown condition. Control energy production and reactor power distribution based on design limits and current core heat removal capability. **Loss of VBD:** NI-4 and NI-8 are lost, but the remaining channels of nuclear instrumentation and incore detectors provide sufficient information to control power level and reactor power distribution.
CSF 6, Radiation Control and Control Room Habitability: Monitor and control the release of radiation to the environment. Maintain access to critical plant equipment and use of the Control Room. **Loss of VBD**: RM-L-6, RM-A-7, RM-A-4, and RM-A-6 are deenergized. Compensatory actions will be taken IAW ODCM requirements. Access to plant equipment and Control Room is not affected.
CSF 7, Electrical Power: Provide electrical power as required to accomplish the other Critical Safety Functions. Provide AC and DC power for emergency equipment operation and instrumentation systems. **Loss of VBD:** VBD is deenergized. Ability to accomplish other Critical Safety Functions is not compromised.
CSF 8, Auxiliary Emergency Systems: Provide equipment cooling (closed cooling and ventilation), and other support requirements to accomplish the other Critical Safety Functions. Provide Instrument Air for operation of EFW, ADVs, RCP Support Systems and some containment isolation valves. **Loss of VBD:** Ventilation will be lost to CB 322' Battery Rooms, Inverter Rooms, ES 480V Switchgear Rooms, and Remote Shutdown Area. Compensatory actions will be performed IAW OP-TM-AOP-034, "Loss of Control Building Cooling." Instrument air is not affected.
CSF 9, Fire Protection and Remote Shutdown Capability: Maintain means to prevent, detect and suppress fires, as well as the capability to perform a plant shutdown without access to the Control Room. **Loss of VBD:** Loss of PRF annunciators disables alarms PRF-5-1, Relay Room Fire, PRF-7-1, IWFS/TS Bldg Fire, PRF-7-6, UPS Fire, and PRF-7-7, Process Center Fire. HVB-4-10 remains available to annunciate a fire in the Relay Room, and Relay Room CO2 fire suppression remains operable. Sprinkler systems remain operable in IWFS/TS Bldg, UPS room, and Processing Center.

The crew will diagnose the loss of Vital Bus "D" by the "D" Reactor Protection System Cabinet being deenergized, NI-8 indication deenergized (CC), Multiple annunciator alarms, including one for a failed inverter, "D" powered HSPS lights lit, and a loss of the right monitor of the Position Monitor Panel. The ARO will place Radiation Monitor Interlock switches to Defeat, and restore Control Building, Auxiliary Building, and Fuel Handling Building Ventilation. The CRS will identify and declare the following Tech Spec: 3.5.5.2.

When the radiation monitor interlock switches are in defeat, ventilation is running and the Tech Spec has been declared, the scenario can continue.

Event #5: The Lead Examiner will cue Total FW Demand Fails to Zero Volts. This will cause an ICS transient, which if not responded to swiftly, will cause a Reactor Trip. The crew will diagnose the ICS failure by a rapid change in RCS pressure, multiple annunciator alarms, and/or changes in indications at multiple ICS stations. Entry into OP-TM-AOP-070, PRIMARY TO SECONDARY HEAT TRANSFER UPSET will be required based on RCS pressure not being controlled in ICS AUTO. "RCS pressure is not being controlled" requires the operator to make a subjective determination, based on their skills, training, and experience. A determination that RCS pressure is being controlled should include the following elements: 1) The reason for the transient is understood 2) RCS pressure response is consistent with the expected response for the event 3) Automatic or manual control in accordance with normal operating procedures is effectively controlling RCS pressure. A conservative assessment (i.e. concluding that RCS pressure is not being controlled) is appropriate when the three conditions above cannot be satisfied. ICS failures are one class of events that can lead to an upset in primary to secondary heat transfer. Most ICS failures can be mitigated by use of the appropriate manual control normal operating procedures.

This entry into OP-TM-AOP-070 is different from the others because FeedWater will rise significantly and the crew will need to establish ICS controls in manual at the FeedPump and Feedwater Valve ICS controllers.

Once the plant is stabilized in ICS HAND control, the scenario can continue.

Event #6: The Lead Examiner will cue the "B" OTSG Tube Leak. Any OTSG tube leak causes an abnormal increase in the release of radioactive materials to the environment. The most fundamental objective is to minimize this release. The prioritized objectives of this procedure are:

- Maintain core cooling.
- Minimize the activity release to the atmosphere (minimize release duration, rate and concentration of radioisotopes, particularly iodine)
- Minimize the integrated tube leakage

30 seconds of MSSV actuation can release 75% of the iodine for the entire event. The main condenser provides for iodine removal due to the ability of water to absorb the iodine (at least temporarily) through the condensers 104 "partitioning factor". This phenomenon reduces off site does consequences to the point that the radiation limits listed in the EOP can not be reached regardless of the tube leak size and the number of fuel pin leaks. MSSV actuation provides an opportunity for failure resulting in an unisolable path for reactor coolant directly to the environment.

The crew will diagnose an OTSG tube leak based on RM-G-27, RM-A-5, and RM-A-15 indications (PR), Annunciator C-1-1 in alarm, and/or pressurizer level lowering (CC). The CRS will announce entry into OP-TM-EOP-005, OTSG TUBE LEAKAGE. This is a reactivity manipulation event. The URO will perform reactor shutdown with ICS in HAND. The ARO will place control Main Feedwater in HAND and may lineup to feed to the RCS from the "B" RBCT for inventory control. The CRS will evaluate and declare Tech Spec 3.1.6.3

When sufficient reactivity manipulation has been observed, the scenario can continue.

Event #7/8: The Lead Examiner will cue the "B" OTSG Tube Rupture. The CRS will continue in OP-TM-EOP-005, OTSG Tube Leak.

The URO will minimize SCM to lower pressure and therefore lower the OTSG tube leak rate. Guide 8 and OS-24 give direction to maintain SCM 30-70F, but as close to 30F as possible. The ARO will preferentially steam the "B" OTSG:

Upon receiving the elevated offsite dose projections, the CRS should invoke 10CFR 50.54x and raise the cooldown rate IAW OP-TM-EOP-005, Step 3.31.

The URO will secure MU-P-1A due to an overcurrent condition. This will leave the "B" Train of HPI operational, which should be sufficient to maintain Pressurizer level once RCS pressure has lowered.

The scenario can be terminated the "B" OTSG has been isolated, HPI flow is terminated, minimum Makeup flow is established, and SCM has been minimized.

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-7 – Minimize SCM - HPI must be throttled to minimize SCM while maintaining margin> 30°F this minimizes primary to secondary leakage and reduces dose on the secondary side of the plant as well as minimizing release to the public. If HPI is allowed to raise OTSG pressure above 1000 psig after OTSG is full, a liquid RCS release to atmosphere would occur. Task failure would be to not throttle and challenge this.

- Minimizing SCM outside of the following limits should be considered grounds for failure of the critical task:
  - Do not allow SCM to fall below 25F.
  - Do not allow SCM to rise above 70F sustained.

Safety Significance: Except when RCP NPSH limits are applicable and are more restrictive, RCS pressure should be maintained close to, but above, the minimum SCM to minimize RCS-SG ΔP. The reason for minimizing RCS-SG ΔP is to reduce the leak flowrate from primary to secondary to as low as possible. Therefore, this procedure (minimizing SCM) is desirable whenever possible during SGTR mitigation.
Reducing the leak flowrate from the RCS to the secondary side of a SG reduces RCS losses and when accomplished with an impaired steam system (e.g., weeping MSSV and MSL leak) should reduce integrated radiation releases from the impaired system. If the level of the leaking SG can be maintained within normal operating limits, then the SG will remain available for continued use during the cooldown, thus enhancing the transient mitigation capability of the plant.

Cues:
1. SCM monitor
2. SPDS displays and associated alarms
3. P-T display and associated alarms

Performance Indicators:
1. Operation of MU/HPI pump and valve controls
2. Operation of normal or auxiliary spray valve controls

Feedback:
1. SCM meter and/or plant SPDS and/or P-T display
2. RCS pressure and temperature
3. MU/HPI pump and valve status indications
4. Normal and auxiliary spray valve status indications

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-21 – Limit Uncontrolled Radiation Release – During SGTR mitigation:

- Attempt to control PZR level such that it does not continually decrease by increasing MU or HPI flow and reducing letdown flow
- If the reactor has not tripped, then perform controlled shutdown to prevent lifting MSSVs.
- Isolate non-essential steam loads from affected SG(s).
- If required to prevent exceeding [SG overfill setpoint] or [radiation limit], use emergency cooldown rate limit to 500'F.

The typical plant design allows for 40 cycles of an emergency cooldown to 500'F Thot at 240°F/hr. This rate is allowed for any SGTR event. However, it is recommended that the use of this emergency cooldown rate be limited to situations where:

a) the affected SG level(s) will reach the SG level limit before the SG can be isolated using the normal cooldown rate, including the use of SG drains if available, or

b) activity release rates are projected to reach the integrated limit before 500°F Thot at the normal cooldown rate.

The emergency cooldown rate is recommended for the two cases noted because several large SGTRs and/or a relatively high percentage of failed fuel already exist. In these cases, it is most important to prevent liquid discharge through the MSSVs and limit the duration of high activity release rates.

- Limiting Uncontrolled Radiation Release outside of the following limits should be considered grounds for failure of the critical task:
  - Do not allow affected OTSG MSSV's to lift.
  - Do not allow affected OTSG to go dry.

Safety Significance: If pressurizer level can be controlled, then the operator's ability to perform a controlled shutdown is greatly enhanced. Whenever possible power should be reduced as quickly as possible, but in a controlled manner, to well within the turbine bypass system capacity before tripping the reactor to prevent lifting of the MSSVs. This includes cases where maximum MU or HPI flow and letdown isolation are required to keep up with the tube leak and maintain pressurizer level. Power reduction is intended to minimize atmospheric radiation releases due to SG safety valve operation. Also, if a reactor trip can be averted through controlled operations, then ability to mitigate the transient is expected to be enhanced as normal transition from power operations to a controlled cooldown occurs.

Cues:

1. Main steam line radiation alarm
2. SG high level alarm
3. Pressurizer low level alarm
4. Verbal alert by plant staff that a SGTR is occurring and the reactor has not tripped

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-21 – Limit Uncontrolled Radiation Release (cont)

Performance Indicators:

1. Operation of MU/HPI pump controls
2. Operation of MU/HPI valve controls
3. Operation of letdown valve controls
4. Operation of TBV/ADV controls

Feedback:

1. Pressurizer level
2. Letdown flow
3. MU/HPI flow
4. RCS temperature
5. Verbal alert by plant staff of pressurizer level status
6. Verbal alert by plant staff of RCS cooldown rate

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-22 – Reduce Steaming/Isolate Affected SGs (includes use of SG drains) – Steam affected SGs to maintain level < [overfill setpoint]. If steaming alone cannot prevent SG fill, then use SG drains (if available) to maintain SG level below [overfill setpoint]. Isolate SG(s) if steaming and draining cannot prevent overfill and maintain RCS and isolated SG pressures < 1000 PSIG by use of [primary and secondary relief paths].

- Isolating Affected SGs outside of the following limits should be considered grounds for failure of the critical task:
  - Do not allow isolation to occur with RCS pressure > 1000 psig.

Safety Significance: The more probable tube rupture scenario is a tube leak in one SG with both SGs available. The preferred mitigation strategy is therefore isolation of the affected SG following the initial cooldown and depressurization to <1000 PSIG. This limits the radiological consequences of the event, but does require cooldown to DHRS operation using one SG.

Both SGs are always used in the initial cooldown and depressurization to < 1000 PSIG. Prevention of MSSV lift on the affected SG(s) is integral to the goal of minimizing off-site release, and assurance requires RCS temperatures at or below 500°F in order to maintain SCM when RCS pressure is < 1000 PSIG. Once this initial cooldown and RCS depressurization to <1000 PSIG is completed, then SG isolation can be considered.

There are limitations on continued steaming of a SG with a SGTR. These limitations consider the overriding concerns of SGTR transients that dictate the isolation of the SG(s) and initiation of HPI cooling, if necessary. These limits are based on integrated radiation dose reaching predetermined values and SG filling due to tube leakage despite steaming to achieve maximum allowable cooldown rate.

SGs isolated due to SG fill criteria pose concerns related to liquid passing through MSSVs. MSSVs should be prevented from passing liquid, since their failure to reseat becomes more probable. For this reason, RCS and SG pressures are maintained <1000 PSIG by use of [primary and secondary relief paths]. These relief paths may include such things as letdown, PZR vents, HPVs, the PORV, TBVs and ADVs.

Cues:

1. Rising OTSG level
2. Rad Monitor Alarms
3. Lowering Pressurizer level
4. Lowering RCS Pressure
5. Automatic initiation of HPI

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-22 – Reduce Steaming/Isolate Affected SGs (includes use of SG drains) (cont)

Performance Indicators:
1. Operation of TBV/ADV controls

Feedback:
1. SG(s) level and pressure
2. RCS pressure
3. MFW/EFW flow
4. MFW/EFW pump and valve status indication
5. TBV/ADV status indication

Industry Experience:

- TMI Reactor Trip (11/2/06) Main Steam Safety Valves remained open longer than expected. (IR 552591)
- Indian Point 2 (2/15/00) - Steam Generator Tube Failure (380 litres per minute)
- Palo Verde 2 (3/14/93) - Steam Generator Tube Leak ranged between 11 and 39 litres per day, suddenly turned to 900 litres per minute tube rupture.

PRA

- Steam Generator Tube Rupture (Initiating Event)

No Reactivity Transient?

Facility:	Three Mile Island	Scenario No.:	4	Op Test No.:	289-2014-301
Examiners:		Operators:			

Initial Conditions:
- (Temporary IC-238)
- 85% Power, MOL
- I&C Maintenance is occurring on HSPS, Train B
- SBO OOS for Maintenance, expected to return to service in 10 hours.

Turnover: Maintain 85% Reactor Power

Critical Tasks:
- PORV Control for Heat Transfer (CT-13)
- Shutdown Reactor - ATWS (CT-24)
- Restore Feed to a Dry OTSG (CT-26)

Event No.	Malf. No.	Event Type*	Event Description
1	IA08	TS CRS C ARO	Instrument Air Leak Requiring Isolation of "A" Side 2-Hour Air and "A" EFW Valves, entry into OP-TM-AOP-028 (ARO:Start IA-P-1A/B)
2	IC38B	C CRS C ARO	Invalid "B" OTSG Low Level, "B" EFW inadvertent actuation. (ARO: defeats invalid signal, secures EF-P-2B)
3	RD10B	I CRS I URO I ARO	Uncontrolled Inward Rod Motion. (URO: Assumes Manual Control of Control Rods)
4		TS CRS C URO	MU-V-18 Fails Closed, entry into Guide 9. (URO: Controls Pressurizer Level with HPI valve)
5	FW15A	C CRS R URO N ARO	"A" Main Feed Pump Trips, Manual runback required. (URO: Runback in Manual, ARO: Runback in Manual)
6	TH18B	C CRS C URO C ARO	Sheared Shaft on RC-P-1B (URO: Secures RC-P-1B, ARO: re-ratios Main Feedwater)
7	RD28 RD32	M CRS M URO M ARO	"B" Main Feed Pump Runs to 0 rpm, ATWS, Lack of Primary to Secondary Heat Transfer. What's success path on ATWS?!
8		C CRS C ARO	EFW Control Valves fail to operate. (ARO: Establish PSHT via Condensate Booster Pump flow)
9		C CRS C URO	MU-P-1A/C will not start, MU-P-1B trips. (URO: Establish PORV control for Heat Transfer)

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

Three Mile Island NRC Scenario #4

Event #1: When the crew has accepted the watch, the Lead Examiner will cue the Loss of Instrument Air.

The operators will diagnose the Loss of Instrument Air based on a report from the field and slightly lower IA pressure.

Upon a Loss of Instrument Air, the following Critical Safety Functions are affected:

> **CSF-1, Reactivity & Reactor Power Control:** Maintain control of the fission process, maintain the capability to shutdown the reactor and the capability to maintain the reactor in a shutdown condition. Control energy production and reactor power distribution based on design limits and current core heat removal capability. Loss of IA: If IA pressure < 60 psig, then the reactor will be tripped. At above 60 psig, IA will be sufficient for plant control and proper reactivity management. Reactor trip capability is not impacted by loss of IA. Emergency boration can be performed via BWST or by local operation of MU-V-51 when emergency borating from BAMT. Emergency boration via the RBAT is inoperable (several air operated valves w/o handwheel). DC-V-20A & B are closed to prevent flooding new fuel storage area with non-borated water.

> **CSF-3, RCS Integrity:** Maintain the capability to control heatup and cooldown rates and control RCS pressure prevent reactor vessel brittle fracture or LTOP events. Maintain RCP seal cooling to prevent excessive loss of RCS inventory through RCP seals. Loss of IA: Normal RCS pressure control (spray & heaters) is not affected by loss of IA. Loss of IA will result in a loss of letdown. This will eventually cause problems with RCS pressure control (i.e., with seal injection and no letdown, pressurizer level will increase until solid ops is required). This is mitigated by providing a method to restore letdown without IA IAW OP-TM-211-950. Emergency pressure control functions (PORV & ES powered heaters) are not affected by loss of IA. Both methods (ICCW to thermal barrier & seal injection) of RCP seal cooling are adversely affected by loss of IA. IC-V-3, IC-V-4, and MU-V-20 fail closed on loss of air (after depletion of local air reservoir). Local operator action to block OPEN IC-V-3, 4, and MU-V-20 is needed to maintain redundant means of seal cooling. Loss of ICCW more than 10 minutes after loss of seal injection, or Loss of Seal Injection for more than 10 minutes without ICCW will result in unacceptable RCP seal & thermal barrier temperatures which will preclude restoration of RCP seal cooling. (Reference 6.1) Therefore, the Loss of IA procedure is designed to ensure that both SI and ICCW are not lost.

start if air to IB is lost. MS-V-13A & B fail open. The pump will remain operating until IA pressure is restored. EF-P-1 remains operable during a loss of IA event. CO-V-13 is closed at less than 60 psig to prevent loss of EFW inventory caused by dumping the CSTs to the hotwell. CO-V-8 and hotwell makeup level control is supplied by BUIA.

CSF-5, Containment Integrity: Provide means to prevent or minimize fission product release to the environment. (1) Maintain containment pressure below design and (2) Provide capability to isolate the containment when required. Loss of IA: Containment cooling. Industrial cooler operation will be degraded. RBEC will be initiated IAW MAP N-1-6 if RB air temperature is high.

The CRS will enter OP-TM-AOP-028, LOSS OF INSTRUMENT AIR. The ARO will start IA-P-1A and/or IA-P-1B from the Control Room prior to Instrument Air reaching 60psig (setpoint below which a manual Reactor Trip must occur). The CRS will determine isolation points and order the appropriate valves to be closed. The CRS will evaluate and declare T.S. 3.4.1.1.a.(2) based on the following Limit/Precaution in 1104-25, Instrument and Control Air System:

> Both "A" and "B" trains of two-hour BUIA must be pressurized and inservice prior to exceeding 250°F in RCS. (Tech. Spec. Definition of OPERABLE (1.3) applied to EFW.)

When IA-P-1A and/or IA-P-1B are/is running, the air leak has been isolated and the Tech Spec has been declared, the scenario can continue.

Event #2: The Lead Examiner can cue the Inadvertent Low Level Signal on the "B" OTSG with EFW actuation.

The crew will diagnose the Inadvertent Low Level Signal by EF-P-2B and EF-P-1 running with no valid reason (OTSG level is greater than 10" in the Startup Range and OTSG pressure is greater than 600 psig).

The ARO will respond per OP-TM-424-901, and defeat the HSPS signal, secure the running EFW pump (EF-P-2B), and re-enable HSPS.

When HSPS is re-enabled, the scenario can continue.

Event #3: The Lead Examiner will cue the Uncontrolled Inward Rod Motion. This will cause an ICS transient.

The crew will diagnose the uncontrolled inward rod motion by an inward signal shown on the Diamond panel and the Position Indication Panel, Reactor Power lowering, RCS pressure and temperature lowering.

> *Note: OP-TM-AOP-064, UNCONTROLLED ROD MOTION, was deleted when the Digital Control Rod System was installed, and so the only way to address the situation is with OP-TM-AOP-070, PRIMARY TO SECONDARY HEAT TRANSFER UPSET.*

Entry into OP-TM-AOP-070, PRIMARY TO SECONDARY HEAT TRANSFER UPSET will be required based on RCS pressure lowering.

"RCS pressure is not being controlled" requires the operator to make a subjective determination, based on their skills, training, and experience. A determination that RCS pressure is being controlled should include the following elements:

1) The reason for the transient is understood
2) RCS pressure response is consistent with the expected response for the event
3) Automatic or manual control in accordance with normal operating procedures is effectively controlling RCS pressure.

A conservative assessment (i.e. concluding that RCS pressure is not being controlled) is appropriate when the three conditions above cannot be satisfied. ICS failures are one class of events that can lead to an upset in primary to secondary heat transfer. Most ICS failures can be mitigated by use of the appropriate manual control normal operating procedures.

This entry into AOP-070 is unique from the other scenarios because Reactor Power will lower due to a rod insertion signal, and not due to a failed ICS input.

Once the plant is stabilized in ICS HAND control, the scenario can continue.

Event #4: The Lead Examiner will cue the Closure of MU-V-18.

The crew will diagnose MU-V-18 closing by the green closed light lit and the red open light not lit (CC), lowering Pressurizer level indications (CC), and if left unattended long enough, MAP G-2-5, Pressurizer Level Hi/Lo in alarm.

The URO will establish Pressurizer Makeup manually via HPI Control Valve, MU-V-16B IAW OP-TM-EOP-010, Guide 9, RCS Inventory Control:

If normal makeup flow has not been established via MU-V-17 or MU-V-217 and through MU-V-18 (i.e. MU24-FI < 20 GPM), then MU-V-16B (or MU-V-16D) is used for normal MU flow. MU-V-16D is only used when the MU discharge cross connects are not in the normal lineup.

Once Pressurizer level is being restored and Makeup is being controlled manually, the scenario can continue.

Event #5: The Lead Examiner will cue the Loss of the "A" Main Feedwater Pump.

The crew will diagnose the Loss of FW-P-1A by an immediate drop in Feedwater flow, OTSG level decreasing rapidly, steam header pressure increasing, a neutron cross-limit alarm coming in, and the remaining feedwater pump speed increases causing feedwater flow to recover somewhat.

OP-TM-MAP-H0101, Plant Runback, will be entered and a manual runback will be performed to lower Reactor Power to approximately 68%. this will require coordination between the URO and ARO, ensuring that the Control Rods and Feedwater are run back in symphony. This is the reactivity manipulation for the scenario.

Once sufficient reactivity manipulation has been observed, the scenario can continue.

Event #6: The Lead Examiner will cue the Sheared Shaft on RC-P-1B.

The crew will diagnose the sheared shaft on RC-P-1B by MAP alarm F-3-1 in alarm and zero amps indicated on RC-P-1B.

OP-TM-MAP-F0301, RC LOOP FLOW LO, and OP-TM-226-152, Shutdown RC-P-1B, will be entered to secure RC-P-1B and to re-ratio Main Feedwater.

The URO will secure RC-P-1B and the ARO will re-ratio Main Feedwater due to the "A" and "B" Feed flow master ICS control stations being in HAND.

Once sufficient reactivity manipulation has been observed, the scenario can continue.

Event #7/8/9: The Lead Examiner will cue the "B" Main Feed Pump Lowering to 0 rpm. The crew will diagnose the loss of Main Feedwater by an immediate drop in Feedwater flow, OTSG level decreasing rapidly, and a rapid rise in RCS pressure and temperature. The URO will identify that an ATWS has occurred and will perform the Immediate Manual Actions of OP-TM-EOP-001, REACTOR TRIP.

> Memorized operator action is appropriate because operator response time can significantly alter the consequences of an ATWS or a failure of the turbine to trip when required.
>
> The reactor protection system is designed to prevent fuel clad or RCS pressure boundary failure. If RCS conditions are outside of the RPS envelope and RPS fails to de-energize the control rod drive mechanisms, prompt operator response can minimize the potential for fuel damage or an RCS pressure boundary failure
>
> EOP-001 is the primary entry point to the EOP network. This procedure is designed to address or direct procedures to address events from a reactor trip with no adverse plant conditions or equipment failures as well as events which challenge the fission product barriers.

The first priority (with the exception of entry into OP-TM-EOP-005 at power to mitigate an OTSG tube leak) to mitigate the consequences of a significant plant upset is to ensure the reactor is shutdown. Success of all subsequent EOP action is based on reducing core heat generation to reactor decay heat generation rates.

– Once the reactor is shutdown, prompt isolation of the turbine steam flow path is significant. Until the major steam flow path through the turbine to the condenser is isolated, RCS heat removal will be much larger than heat generation, and a rapid RCS cooldown will continue.

– When the turbine steam flow path is isolated, plant conditions are evaluated to determine if any symptoms of a core cooling upset are present.

Based on a symptom of low subcooling margin, excessive heat transfer, lack of primary to secondary heat transfer or primary to secondary leakage, rule based actions and entry into other sections of the EOP network is performed.

The crew will identify a Lack of Heat Transfer based on the following definition from OS-24, Conduct of Operations During Abnormal and Emergency Events:

LOHT is the inability of either OTSG to remove sensible heat from the RCS.

LOHT can be confirmed if one of the following sets of conditions exists:

- Incore temperatures or Thot rising above 580°F and at least one RC Pump operating
- Incore temperatures rising and NO FEEDWATER available
- Incore temperatures rising and RCS circulation can not be confirmed

The CRS will direct entry into OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER.

RCP philosophy IAW the GEOG and the OP-TM-EOP-004 Basis Document:

One or two RCPs (one in each loop) should be left running to reduce heat input to the RCS yet provide for heat transfer as soon as FW is restored to either OTSG.

The step intent is to reduce heat input to the RCS while maintaining forced flow in both RC loops.

Impact to pressurizer spray flow should be considered when selecting RCPs for shutdown.

Although one RCP is allowed, our procedure directs one RCP in each loop to allow for even flow between the loops (avoiding the reverse directional flow in the opposite loop), and to prevent a loss of forced RCS flow if the only running RCP were to trip.

The ARO will recognize that EFW control valves are inoperable and will establish primary to secondary heat transfer using Condensate Booster Pumps IAW OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER:

> If feedwater is NOT available, then efforts to establish EFW should continue. If this event occurs when the condensate booster pumps could provide a continuous feedwater supply, then the booster pumps may be used alone to feed the OTSGs.
>
> HPI COOLING will be initiated (EOP-009) when RCS pressure approaches the PORV setpoint. After initiating HPI COOLING, actions to restore feedwater (main or emergency) should continue.

Additionally, when the conditions are met during the scenario the CRS will direct entry into OP-TM-EOP-009, HPI COOLING.

HPI will not be adequate, which will force the URO to manually control the PORV to maintain RCS pressure while minimizing inventory losses until Primary to Secondary Heat Transfer exists.

Termination: The scenario can be terminated when the Reactor is shutdown, and Primary to Secondary Heat Transfer has been established via Condensate Booster Pump flow.

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-13 – PORV Control for Heat Transfer – During action to restore primary to secondary heat transfer, RCS pressure must be continually controlled. RCS pressure is controlled by manually opening and closing the PORV. This prevents excessive PORV cycling which could occur if the PORV were allowed to operate automatically.

During mitigation of LHT, manually cycle the PORV as necessary to maintain RCS pressure between the PORV setpoint or RV P-T limit and minimum SCM (if subcooled) or 1600 PSIG (if saturated).

- Performing PORV Control for Heat Transfer outside of the following limit should be considered **grounds for failure of the critical task:**
    - PORV automatically lifts three (3) times.

Safety Significance: During mitigation of LHT the PORV should be manually cycled to control RCS pressure between the PORV setpoint or RV P-T limit and 1600 PSIG if the RC is saturated or minimum allowable SCM if the RC is subcooled. The PORV opening values prevent challenges to the pressurizer safety valves and the RV P-T limit while the PORV closing values maintain a positive primary to secondary side AT (SGs remain heat sinks).

Cues:
1. High RCS pressure alarm
2. SCM monitor and associated alarms
3. SPDS displays and associated alarms
4. P-T display and associated alarms
5. Verbal alert by plant staff that RCS pressure has reached the PORV setpoint

Performance Indicators:
1. Operation of PORV and PORV block valve controls

Feedback:
1. PORV and PORV block valve status indication
2. RCS pressure and temperature
3. Verbal indication by plant staff that PORV flow has been initiated

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-24 – Shutdown Reactor - ATWS – Actuation of the manual reactor trip pushbutton, to backup the automatic trip and/or provide the necessary reactor trip, anytime the reactor trips or should have tripped. In the event the reactor fails to trip, in response to automatic and manual demands, then perform the following: Deenergize CRDMs

- Shutting down the reactor due to an ATWS outside of the following limit should be considered **grounds for failure of the critical task:**
    - Not deenergizing the CRDM power supplies:
        - 1G-02
        - 1L-02

Safety Significance: Without taking the proper actions, there exists a potential challenge to the Reactor Coolant System pressure boundary due to high RCS pressure.

An ATWS could occur due to a failure of the RPS to initiate a reactor trip signal upon one of the reactor trip parameters reaching its trip limit or the control and safety rods failing to insert once the RPS trip signal is given automatically or manually. A Diverse Scram System (DSS) is provided, independent of the RPS, to minimize the potential for an ATWS event. However, the operator must recognize and react to any of the reactor trip parameters that exceeds its limit but does not cause a reactor trip.

In this situation, the manual reactor trip button has been actuated but reactor power is not less than the plant specific reactor power level for verification of a reactor trip. Therefore, the reactor has not been shut down and there has been a failure of all or most of the control and safety rods to insert into the reactor core. Given that RPS, DSS and the manual reactor trip have failed to trip the reactor, then immediate actions to shut down the reactor by the alternate methods should be initiated. These methods include trip of CRDM breakers and maximum rate of boron addition to the RCS. Once the control and safety rods are successfully tripped into the core, or sufficient boric acid has been added to provide an adequate shutdown margin, the reactor will be shut down.

This should be achieved prior to taking additional mitigating actions because post-trip transient mitigation, from this point forward, is based on the assumption that the reactor is shutdown (subcritical).

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-24 – Shutdown Reactor - ATWS – Continued

Cues:
1. RPS channel alarms
2. RCS Power, Pressure and Temperature indications
3. P-T display and associated alarms
4. Verbal alert by plant staff that reactor shutdown requirements have not been met

Performance Indicators:
1. Operation of control rod drive feeder breakers

Feedback:
1. Nuclear Instruments
2. Control rod status indication
3. Control rod drive breaker status indication
4. Verbal indication from plant staff of reactor shutdown status

B&W Unit EOP Critical Task Description Document, 47-1229003-04:

CT-26 – Restore Feed to a Dry OTSG - If a RCP is running, establish FW to the SG(s) and control FW flow to maintain RCS cooldown rate within limits. EFW flow is established at less than 450 GPM total flow and MFW flow is established at less than 200,000 LBM/HR total flow.

- Restoring Feed to a Dry OTSG (sustained) outside of the following limits should be considered **grounds for failure of the critical task:**
  - To minimize OTSG stress, do not exceed MFW flow greater than than 200,000 LBM/HR total flow (sustained).
  - To ensure the main feedwater nozzles remain full and to prevent cavitation type damage to the Main Feedwater nozzles if they are not full of subcooled fluid, do not fall below less than 160,000 LBM/HR total flow (sustained).

Safety Significance:
If it is decided to perform the cooldown by using trickle feeding, it will be necessary to control the rate of FW addition to the SGs to maintain RCS cooldown limits. The FW flow rate should be adjusted to get the desired cooldown rate. If possible EFW should be used to limit SG thermal stresses. If MFW is used with the MFW nozzles, it will only be effective with forced flow.

Once heat transfer is restored in the SG, feed rates can be adjusted as necessary to control the cooldown and SG tube-to-shell ΔT.

Cues:
1. Low SG level alarms
2. Low SG pressure alarms
3. Verbal alert by plant staff that no SG is available for heat transfer

Performance Indicators:
1. Operation of EFW/MFW pump controls
2. Operation of EFW/MFW valve controls

Feedback:
1. EFW/MFW flow
2. SG level and pressure
3. RCS pressure and temperature
4. Verbal alert by plant staff of EFW/MFW flow status

Industry Experience:

- FW-P-1A Coupling Failure (TMI CR-00189457)
- Harris Nuclear Plant Manual Scram Due to Loss of Feedwater (12/14/99)
- Oconee 1 Loss of Feedwater (5/26/00)

PRA

- o Feedwater Transient (Initiating Event)