Exelon Generation

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June 30, 2015
TMI-15-075

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 September 28, 2015, 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 for Power Reactors," Revision 10.

In accordance with NUREG 1021, Revision 10, 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) 948-2023.

Respectfully,

Thomas Haaf
Site Vice President (Acting), Three Mile Island Unit 1
Exelon Generation Co., LLC

TPH/mdf

Enclosures: (Mailed to Peter Presby, Chief Examiner, NRC Region I)

Examination Security Agreement (Form ES-201-3)
Administrative Topics Outline (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 Outline (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)

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

Statement Detailing Method of Written Outline Generation:

- All original and replacement K/A's for Three Mile Island ILT Class 14-01 NRC Written Examination have been randomly selected utilizing Westinghouse NRC K/A Exam Generator (NKEG) software.

<table>
<tr><td colspan="19">Facility: Three Mile Island Date of Exam: 09/28/2015</td></tr>
<tr><td rowspan="2">Tier</td><td rowspan="2">Group</td><td colspan="12">RO K/A Category Points</td><td colspan="5">SRO-Only Points</td></tr>
<tr><td>K 1</td><td>K 2</td><td>K 3</td><td>K 4</td><td>K 5</td><td>K 6</td><td>A 1</td><td>A 2</td><td>A 3</td><td>A 4</td><td>G *</td><td>Total</td><td colspan="2">A2</td><td colspan="2">G*</td><td>Total</td></tr>
<tr><td rowspan="3">1. Emergency & Abnormal Plant Evolutions</td><td>1</td><td>3</td><td>3</td><td>3</td><td colspan="3" rowspan="3">N/A</td><td>3</td><td>3</td><td colspan="2" rowspan="3">N/A</td><td>3</td><td>18</td><td colspan="2">3</td><td colspan="2">3</td><td>6</td></tr>
<tr><td>2</td><td>1</td><td>2</td><td>2</td><td>2</td><td>1</td><td>1</td><td>9</td><td colspan="2">2</td><td colspan="2">2</td><td>4</td></tr>
<tr><td>Tier Totals</td><td>4</td><td>5</td><td>5</td><td>5</td><td>4</td><td>4</td><td>27</td><td colspan="2">5</td><td colspan="2">5</td><td>10</td></tr>
<tr><td rowspan="3">2. Plant Systems</td><td>1</td><td>3</td><td>2</td><td>3</td><td>3</td><td>3</td><td>2</td><td>2</td><td>3</td><td>2</td><td>3</td><td>2</td><td>28</td><td colspan="2">3</td><td colspan="2">2</td><td>5</td></tr>
<tr><td>2</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>0</td><td>10</td><td>0</td><td>2</td><td colspan="2">1</td><td>3</td></tr>
<tr><td>Tier Totals</td><td>4</td><td>3</td><td>4</td><td>4</td><td>4</td><td>3</td><td>3</td><td>4</td><td>3</td><td>4</td><td>2</td><td>38</td><td colspan="2">5</td><td colspan="2">3</td><td>8</td></tr>
<tr><td colspan="5" rowspan="2">3. Generic Knowledge and Abilities Categories</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">4</td><td rowspan="2">10</td><td>1</td><td>2</td><td>3</td><td>4</td><td rowspan="2">7</td></tr>
<tr><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">2</td><td colspan="2">3</td><td>2</td><td>2</td><td>1</td><td>2</td></tr>
</table>

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO 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). (One Tier 3 Radiation Control K/A is allowed if the K/A is replaced by a K/A from another Tier 3 Category.)

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 $\pm$ 1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

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

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

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

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

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

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

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

G* Generic K/As

ES-401							PWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 1 (RO / SRO)		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 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1	X						EK1.2 - Knowledge of the operational implications of the following concepts as they apply to the (Post-Trip Stabilization): Normal, abnormal and emergency operating procedures associated with (Post-Trip Stabilization)	3.5	3
000008 Pressurizer Vapor Space Accident / 3		X					AK2.03 - Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Controllers and positioners	2.5	4
000009 Small Break LOCA / 3					X		EA2.04 - Ability to determine or interpret the following as they apply to a small break LOCA: PZR level	3.8	13
000011 Large Break LOCA / 3		X					EK2.02 - Knowledge of the interrelations between the and the following Large Break LOCA: Pumps	2.6	5
000015/17 RCP Malfunctions / 4				X			AA1.02 - Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCP oil reservoir level and alarm indicators	2.8	10
000022 Loss of Rx Coolant Makeup / 2						X	2.2.39 - Knowledge of less than or equal to one hour Technical Specification action statements for systems.	3.9	16
000025 Loss of RHR System / 4			X				AK3.02 - Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System: Isolation of RHR low-pressure piping prior to pressure increase above specified level	3.3	7
000026 Loss of Component Cooling Water / 8			X				AK3.02 - Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS	3.6	8
000027 Pressurizer Pressure Control System Malfunction / 3						X	2.4.18 - Knowledge of the specific bases for EOPs.	3.3	17
000038 Steam Gen. Tube Rupture / 3	X						EK1.01 - Knowledge of the operational implications of the following concepts as they apply to the SGTR: Use of steam tables	3.1	1
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4	X						AK1.06 - Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: High energy steam line break considerations	3.7	2
000054 (CE/E06) Loss of Main Feedwater / 4				X			AA1.01 - Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): AFW controls, including the use of alternate AFW sources	4.5	11
000055 Station Blackout / 6						X	2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	3.9	18
000057 Loss of Vital AC Inst. Bus / 6					X		AA2.07 - Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: Valve indicator of charging pump suction valve from RWST	3.3	14
000058 Loss of DC Power / 6			X				AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by D/Gs	3.4	9

000062 Loss of Nuclear Svc Water / 4					X		AA2.02 - Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The cause of possible SWS loss	2.9	15
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4		X					EK2.1 - Knowledge of the interrelations between the (Inadequate Heat Transfer) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.8	6
000077 Generator Voltage and Electric Grid Disturbances / 6				X			AA1.04 - Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Reactor controls	4.1	12
000009 Small Break LOCA / 3						X	2.4.41 - Knowledge of the emergency action level thresholds and classifications.	4.6	79
000011 Large Break LOCA / 3					X		EA2.01 - Ability to determine or interpret the following as they apply to a Large Break LOCA: Actions to be taken, based on RCS temperature and pressure - saturated and superheated	4.7	76
000015/17 RCP Malfunctions / 4					X		AA2.09 - Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on high stator temperatures	3.5	77
000057 Loss of Vital AC Inst. Bus / 6						X	2.4.30 - 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.	4.1	80
000058 Loss of DC Power / 6						X	2.4.11 - Knowledge of abnormal condition procedures.	4.2	81
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1					X		EA2.1 - 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.	4.0	78
K/A Category Totals:	3	3	3	3	3/ 3	3/ 3	Group Point Total:		18/6

ES-401	PWR Examination Outline Emergency and Abnormal Plant Evolutions – Tier 1/Group 2 (RO / SRO)						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	#
000005 Inoperable/Stuck Control Rod / 1		X					AK2.01 - Knowledge of the interrelations between the Inoperable / Stuck Control Rod and the following: Controllers and positioners	2.5	20
000024 Emergency Boration / 1				X			AA1.19 - Ability to operate and / or monitor the following as they apply to Emergency Boration: Makeup control system selector switch for CVCS isolation valve	3.2	23
BW/A02&A03 Loss of NNI-X/Y / 7			X				AK3.3 – Knowledge of the reasons for the following responses as they apply to the (Loss of NNI-X): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	3.7	25
000036 (BW/A08) Fuel Handling Accident / 8	X						AK1.1 - Knowledge of the operational implications of the following concepts as they apply to the (Refueling Canal Level Decrease): Components, capacity, and function of emergency systems.	3.7	19
000051 Loss of Condenser Vacuum / 4			X				AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum	2.8	22
000069 (W/E14) Loss of CTMT Integrity / 5		X					AK2.03 - Knowledge of the interrelations between the Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch	2.8	21
BW/A01 Plant Runback / 1						X	2.1.25 - Ability to interpret reference materials, such as graphs, curves, tables, etc.	3.9	27
BW/A05 Emergency Diesel Actuation / 6					X		AA2.2 - Ability to determine and interpret the following as they apply to the (Emergency Diesel Actuation): Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.	3.5	26
BW/E13&E14 EOP Rules and Enclosures				X			EA1.2 - Ability to operate and / or monitor the following as they apply to the (EOP Enclosures): Operating behavior characteristics of the facility.	2.8	24
000024 Emergency Boration / 1						X	2.4.6 - Knowledge of EOP mitigation strategies.	4.7	84
000036 (BW/A08) Fuel Handling Accident / 8					X		AA2.02 - Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurrence of a fuel handling incident	4.1	82
BW/A02&A03 Loss of NNI-X/Y / 7					X		AA2.1 - Ability to determine and interpret the following as they apply to the (NNI-X): Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.0	83
BW/E08; W/E03 LOCA Cooldown - Depress. / 4						X	2.4.18 - Knowledge of the specific bases for EOPs.	4.0	85
K/A Category Point Totals:	1	2	2	2	1/ 2	1/ 2	Group Point Total:		9/4

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

Plant Systems – Tier 2/Group 1 (RO / SRO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
003 Reactor Coolant Pump					x							K5.02 - Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP coastdown on RCS parameters	2.8	39
003 Reactor Coolant Pump						x						K6.14 - Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Starting requirements	2.6	42
004 Chemical and Volume Control				x								K4.10 - Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Minimum temperature requirements on borated systems (prevent crystallization)	3.2	36
005 Residual Heat Removal								x				A2.04 - Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction	2.9	46
005 Residual Heat Removal										x		A4.04 - Ability to manually operate and/or monitor in the control room: Controls and indication for closed cooling water pumps	3.1	51
006 Emergency Core Cooling	x											K1.13 - Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: CSS	3.3	28
007 Pressurizer Relief/Quench Tank					x							K5.02 - Knowledge of the operational implications of the following concepts as they apply to PRTS: Method of forming a steam bubble in the PZR	3.1	40
008 Component Cooling Water	x											K1.04 - Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: RCS, in order to determine source(s) of RCS leakage into the CCWS	3.3	29
008 Component Cooling Water										x		A4.10 - Ability to manually operate and/or monitor in the control room: Conditions that require the operation of two CCW coolers	3.1	52
010 Pressurizer Pressure Control			x									K3.02 - Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RPS	4.0	33
012 Reactor Protection						x						K6.11 - Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Trip setpoint calculators	2.9	43
013 Engineered Safety Features Actuation		x										K2.01 - Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control	3.6	31
022 Containment Cooling								x				A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Fan motor over-current	2.5	47
022 Containment Cooling										x		A4.05 - Ability to manually operate and/or monitor in the control room: Containment readings of temperature, pressure, and humidity system	3.8	53

026 Containment Spray				X								K4.06 - Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Iodine scavenging via the CSS	2.8	37
039 Main and Reheat Steam				X								K4.07 - Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Reactor building isolation	3.4	38
039 Main and Reheat Steam									X			A3.02 - Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS	3.1	49
059 Main Feedwater											X	2.2.44 - Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	54
061 Auxiliary/Emergency Feedwater					X							K5.01 - Knowledge of the operational implications of the following concepts as the apply to the AFW: Relationship between AFW flow and RCS heat transfer	3.6	41
061 Auxiliary/Emergency Feedwater								X				A2.08 - Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Flow rates expected from various combinations of AFW pump discharge valves	2.7	48
062 AC Electrical Distribution	X											K1.02 - Knowledge of the physical connections and/or cause effect relationships between the ac distribution system and the following systems: ED/G	4.1	30
062 AC Electrical Distribution			X									K3.01 - Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: Major system loads	3.5	34
063 DC Electrical Distribution							X					A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including: Battery capacity as it is affected by discharge rate	2.5	44
064 Emergency Diesel Generator							X					A1.01 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ED/G system controls including: ED/G lube oil temperature and pressure	3.0	45
073 Process Radiation Monitoring			X									K3.01 - Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases	3.6	35
076 Service Water		X										K2.08 - Knowledge of bus power supplies to the following: ESF-actuated MOVs	3.1	32
078 Instrument Air									X			A3.01 - Ability to monitor automatic operation of the IAS, including: Air pressure	3.1	50
103 Containment											X	2.2.37 - Ability to determine operability and/or availability of safety related equipment.	3.6	55

010 Pressurizer Pressure Control								x				A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PORV failures	4.2	86
013 Engineered Safety Features Actuation								x				A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: LOCA	4.8	87
063 DC Electrical Distribution											x	2.4.20 - Knowledge of the operational implications of EOP warnings, cautions, and notes.	4.3	89
073 Process Radiation Monitoring								x				A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Calibration drift	2.9	88
076 Service Water											x	2.4.1 - Knowledge of EOP entry conditions and immediate action steps.	4.8	90
K/A Category Point Totals:	3	2	3	3	3	2	2	3/3	2	3	2/2	Group Point Total:		28/5

ES-401												PWR Examination Outline Plant Systems – Tier 2/Group 2 (RO / SRO)		Form ES-401-2
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
001 Control Rod Drive			x									K3.01 - Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: CVCS	2.9	58
011 Pressurizer Level Control		x										K2.02 - Knowledge of bus power supplies to the following: PZR heaters	3.1	57
029 Containment Purge										x		A4.04 - Ability to manually operate and/or monitor in the control room: Containment evacuation signal	3.5	65
034 Fuel Handling Equipment						x						K6.02 - Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System: Radiation monitoring systems	2.6	61
035 Steam Generator					x							K5.01 - Knowledge of operational implications of the following concepts as the apply to the S/GS: Effect of secondary parameters, pressure, and temperature on reactivity	3.4	60
041 Steam Dump/Turbine Bypass Control							x					A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure	3.1	62
068 Liquid Radwaste				x								K4.01 - Knowledge of design feature(s) and/or interlock(s) which provide for the following: Safety and environmental precautions for handling hot, acidic, and radioactive liquids	3.4	59
071 Waste Gas Disposal									x			A3.03 - Ability to monitor automatic operation of the Waste Gas Disposal System including: Radiation monitoring system alarm and actuating signals	3.6	64
075 Circulating Water	x											K1.01 - Knowledge of the physical connections and/or cause effect relationships between the circulating water system and the following systems: SWS	2.5	56
086 Fire Protection								x				A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low FPS header pressure	3.0	63
001 Control Rod Drive								x				A2.15 - Ability to (a) predict the impacts of the following malfunction or operations on the CRDS- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Quadrant power tilt	4.2	91

016 Non-nuclear Instrumentation								X				A2.01 - 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	92
035 Steam Generator											X	2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm.	4.3	93
K/A Category Point Totals:	1	1	1	1	1	1	1	½	1	1	0/1	Group Point Total:		10/3

Facility: Three Mile Island Date of Exam: 09/28/2015

Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.19	Ability to use plant computers to evaluate system or component status.	3.9	66		
	2.1.25	Ability to interpret reference materials, such as graphs, curves, tables, etc.	3.9	67		
	2.1.5	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.			3.9	94
	2.1.35	Knowledge of the fuel-handling responsibilities of SROs.			3.9	95
	2.1.					
	2.1.					
	Subtotal			2		2
2. Equipment Control	2.2.13	Knowledge of tagging and clearance procedures.	4.1	68		
	2.2.38	Knowledge of conditions and limitations in the facility license.	3.6	69		
	2.2.42	Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	3.9	70		
	2.2.12	Knowledge of surveillance procedures.			4.1	96
	2.2.25	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.			4.2	97
	2.2.			3		2
	Subtotal					
3. Radiation Control	2.3.13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	3.4	71		
	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	72		
	2.3.6	Ability to approve release permits.			3.8	98
	2.3.					
	2.3.					
	2.3.					
	Subtotal			2		1
4. Emergency Procedures /	2.4.4	Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	4.5	73		

Plan	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.	3.7	74		
	2.4.47	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	4.2	75		
	2.4.18	Knowledge of the specific bases for EOPs.			4.0	99
	2.4.25	Knowledge of fire protection procedures.			3.7	10
	2.4.					
	Subtotal			3		2
Tier 3 Point Total				10		7

Tier / Group	Original K/A	Reason for Rejection
1 / 1	054 / AA1.03	The subject K/A is not relevant at the subject facility. Replaced with 054 / AA1.01
2 / 1	008 / A4.11	The subject K/A is not relevant at the subject facility. Replaced with 008 / A4.10
2 / 1	059 / K3.03	Topic overlaps with the Audit Written Exam. Replaced with 062 / K3.01
2 / 1	103 / A1.01	Topic overlaps with the Audit Written Exam. Replaced with 003 / K5.02
2 / 1	078 / 2.2.44	Generic K/A oversampled. 078 / 2.2.44 overlaps with 059 / 2.2.44. Replaced with 022 / A4.05
2 / 1	013 / 2.1.27	Original K/A is LOD 1. Replaced with 103 / 2.2.37
2 / 2	014 / 2.1.27	Original K/A is LOD 1. Replaced with 086 / A2.02
2 / 2	015 / K1.04	The subject K/A is not relevant at the subject facility. Replaced with 075 / K1.01
3 / 1	2.1.30	The subject K/A is not generic enough for Tier level at the subject facility. Replaced with 2.1.25
3 / 3	2.3.11	Topic overlaps with the Audit Written Exam. Replaced with 2.3.13
1 / 1 SRO	009 / 2.4.45	Generic K/A oversampled. 009 / 2.4.45 overlaps with 035 / 2.4.45. Replaced with 009 / 2.4.41
1 / 1 SRO	E05 / 2.2.36	The subject K/A is not relevant to the topic at the subject facility. Replaced with 057 / 2.4.30
1 / 2 SRO	051 / 2.4.1	Original K/A is RO LOK. Replaced with E08 / 2.4.18
1 / 2 SRO	060 / 2.4.47	Topic oversampled. 060 / 2.4.47 overlaps with RO Generic K/A 2.4.47. Replaced with 024 / 2.4.6
2 / 1 SRO	006 / A2.04	Original K/A is RO LOK. Replaced with 010 / A2.03
2 / 2 SRO	075 / A2.03	Topic overlaps with the Audit Written Exam / NRC Operational. Replaced with 001 / A2.15
3 / 2 SRO	2.2.1	Original K/A is RO LOK. Replaced with 2.2.21
3 / 3 SRO	2.3.15	Topic oversampled. 2.3.15 overlaps with RO Generic K/A 2.3.15. Replaced with 2.3.6

Facility: Three Mile Island Date of Examination: 09/28/2015
Examination Level: RO ☒ SRO ☐ Operating Test Number: 2015-301

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M/R	Verify Watchstanding Requirements – Work-Hour Rules 2.1.5 (2.9): Ability to use procedures related to shift staffing, such as minimum crew compliment, overtime limitations, etc.
Conduct of Operations	M/R	Given a Dropped Rod at Power, Calculate SDM 2.1.25 (3.9): Ability to interpret station reference materials such as graphs, curves, tables, etc.
Equipment Control	N/R	Given Intermediate Closed Cooling Water System Electrical and Mechanical Print Drawings, Identify the Status of Associated Containment Isolation Valves 2.2.41 (3.5): Ability to obtain and interpret station electrical and mechanical drawings.
Radiation Control		Category Not Selected for RO Applicants.
Emergency Plan	M/S	Perform State and Local Event Notification 2.4.43 (3.2): Knowledge of Emergency Communications Systems and Techniques.

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

* 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)

JPM A1-1 - Conduct of Operations: Verify Watchstanding Requirements – Work-Hour Rules

Given plant conditions and references OP-TM-1010-111-1001, Shift Manning Requirements, and LS-AA-119, Overtime Controls, identify which requested days of overtime the candidate may work while staying within the requirements of Work-Hour rules.

> Safety Significance: Exelon procedures associated with work-hour rules implement requirements for managing fatigue and controlling work hours in accordance with 10 CFR 26, Subpart I, "Managing Fatigue." The requirements are intended to provide reasonable assurance that worker fatigue will be avoided and that all individuals will be able to safely perform their duties and maintain the health and safety of the public.

This JPM has been modified to ensure that the allowable days are completely different than the previous JPM.

JPM A1-2 - Conduct of Operations: Given a Dropped Rod at Power, Calculate SDM

Given a dropped rod at power, calculate Shutdown Margin IAW OP-TM-300-205, Shutdown Margin for Hot Shutdown Conditions.

> Safety Significance: Tech Specs require that a Shutdown Margin of > 1% $\Delta k/k$ must be maintained at all times.

This JPM has been modified to ensure that the data given and resultant calculations are completely different than the previous JPM.

JPM A2 - Equipment Control: Given Intermediate Closed Cooling Water System Electrical and Mechanical Print Drawings, Identify the Status of Associated Containment Isolation Valves

Given a set of conditions, a timeline of events, and Intermediate Closed Cooling Water System electrical and mechanical prints, the candidates will determine the status of multiple Containment Isolation Valves.

> Safety Significance: Tech Specs require that containment integrity shall be maintained whenever all three of the following conditions exist:
>
> a. Reactor coolant pressure is 300 psig or greater.
> b. Reactor coolant temperature is 200 degrees F or greater.
> c. Nuclear fuel is in the core.
>
> Containment Integrity exists when the following conditions are satisfied:
>
> c. All active CIVs, including power-operated valves, check valves, and relief valves, are OPERABLE or locked closed. Normally closed active CIVs (other than the purge valves) may be unisolated intermittently or manual control of power-operated valves may be substituted for automatic control under administrative control.

This is a new JPM created for the ILT 14-01 NRC examination.

JPM A4 – Emergency Plan: Perform State and Local Event Notification

Given a faulted State and Local Notification form, the candidate will identify the faulted errors on the form and then will simulate performance of making State and Local Event notifications.

> Safety Significance: Provides prompt and accurate notification of nuclear station emergencies to local, state and federal agencies.

This JPM has been modified to ensure that the combination of faults given is completely different than the previous JPM.

Facility: Three Mile Island | Date of Examination: 09/28/2015

Examination Level: RO ☐ SRO ☒ | Operating Test Number: 2015-301

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M/R	Maintain Minimum Shift Staffing – Control Overtime 2.1.5 (3.9): Ability to use procedures related to shift staffing, such as minimum crew compliment, overtime limitations, etc.
Conduct of Operations	M/R	Given a Dropped Rod at Power, Review Submitted SDM for Approval 2.1.25 (4.2): Ability to interpret station reference materials such as graphs, curves, tables, etc.
Equipment Control	N/R	Given Intermediate Closed Cooling Water System Electrical and Mechanical Print Drawings, Identify the Status of Associated Containment Isolation Valves with Tech Spec LCO 2.2.41 (3.9): Ability to obtain and interpret station electrical and mechanical drawings.
Radiation Control	M/R	Review RB Entry Survey Log 2.3.13 (3.8): Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
Emergency Plan	M/S	Determine the Emergency Action Level (EAL) and Make a Protective Action Recommendation (PAR) IAW the Facility Emergency Plan 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 (which would require all five items).

* 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)

JPM A1-1 - Conduct of Operations: Maintain Minimum Shift Staffing – Control Overtime

Given plant conditions and references OP-TM-1010-111-1001, Shift Manning Requirements, and LS-AA-119, Overtime Controls, a prepared Shift Staffing Report, LMS Qual Matrix Report, and a prepared overtime List, identify the required actions to restore minimum staffing and select personnel IAW the requirements to control overtime.

> Safety Significance: Exelon procedures associated with work-hour rules implement requirements for managing fatigue and controlling work hours in accordance with 10 CFR 26, Subpart I, "Managing Fatigue." The requirements are intended to provide reasonable assurance that worker fatigue will be avoided and that all individuals will be able to safely perform their duties and maintain the health and safety of the public.

This JPM has been modified to ensure that the allowable personnel and circumstances are different than the previous JPM.

JPM A1-2 - Conduct of Operations: Given a Dropped Rod at Power, Review Submitted SDM for Approval

Given a dropped rod at power, review the submitted Shutdown Margin calculation for approval (by calculating Shutdown Margin IAW OP-TM-300-205, Shutdown Margin for Hot Shutdown Conditions, to verify accuracy) and identify the faults. Additionally determines Tech Spec action, and does not approve the submitted SDM.

> Safety Significance: Tech Specs require that a Shutdown Margin of > 1% $\Delta k/k$ must be maintained at all times.

This JPM has been modified to ensure that the data given and resultant calculations are completely different than the previous JPM.

JPM A2 - Equipment Control: Given Intermediate Closed Cooling Water System Electrical and Mechanical Print Drawings, Identify the Status of Associated Containment Isolation Valves with Tech Spec LCO

Given a set of conditions, a timeline of events, and Intermediate Closed Cooling Water System electrical and mechanical prints, the candidates will determine the status of multiple Containment Isolation Valves. Additionally determines Tech Spec action.

> Safety Significance: Tech Specs require that containment integrity shall be maintained whenever all three of the following conditions exist:
>
> a. Reactor coolant pressure is 300 psig or greater.
> b. Reactor coolant temperature is 200 degrees F or greater.
> c. Nuclear fuel is in the core.
>
> Containment Integrity exists when the following conditions are satisfied:
>
> c. All active CIVs, including power-operated valves, check valves, and relief valves, are OPERABLE or locked closed. Normally closed active CIVs (other than the purge valves) may be unisolated intermittently or manual control of power-operated valves may be substituted for automatic control under administrative control.

This is a new JPM created for the ILT 14-01 NRC examination.

JPM A3 – Radiation Control: Review RB Entry Survey Log

Given a faulted Reactor Building Entry Survey Log and while referencing RP-TM-460-1007, Access to TMI-1 Reactor Building, identify the faults. Additionally, does not approve the RB entry.

> Safety Significance: The Material describes sampling, equipment and conditional requirements needed prior to entry into the TMI-1 Reactor Building. Possible hazards which may exist in Reactor Building include gamma and neutron radiation (reactor critical), airborne radioactive contamination, and explosive or oxygen-deficient atmosphere.

This JPM has been modified to ensure that the combination of faults given is completely different than the previous JPM.

JPM A4 – Emergency Plan: Determine the Emergency Action Level (EAL) and Make a Protective Action Recommendation (PAR) IAW the Facility Emergency Plan

Given a set of conditions, declare the appropriate Emergency Classification (a Time Critical component). Additionally, declare the associated Protective Action Recommendation (also a Time Critical component).

> Safety Significance: As required in the conditions set forth by the Nuclear Regulatory Commission (NRC) for the operating licenses for the Exelon Nuclear Stations, the management of Exelon recognizes its responsibility and authority to operate and maintain the nuclear power stations in such a manner as to provide for the safety of the general public. The Exelon Emergency Preparedness Program consists of the Exelon Nuclear Standardized Radiological Emergency Plan ("Standard Plan"), Station Annexes, emergency plan implementing procedures, and associated program administrative procedures. The Standard Plan outlines the basis for response actions that would be implemented in an emergency. Planning efforts common to all Exelon Nuclear stations are encompassed within the Standard Plan.

This JPM has been modified to ensure that the conditions given and the method for deciding the PAR are completely different than the previous JPM.

Facility: Three Mile Island Date of Examination: 09/28/2015

Exam Level: RO ☒ SRO-I ☐ SRO-U ☐ Operating Test No.: 2015-301

Control Room Systems: *8 for RO; 7 for SRO-I; 2 or 3 for SRO-U

System / JPM Title	Type Code*	Safety Function
a. Respond to an Inoperable/Stuck Control Rod (005) AA1.01	M/S	1
b. Respond to a Loss of Pressurizer Level Control with Failures (011) A2.03	D/A/S	2
c. Restore Seal Injection with a Loss of ICCW (003) K6.02	P/S/A	4P
d. Respond to an OTSG Overfeed (035) A2.04	N/A/S	4S
e. Initiate RB Spray (026) A2.03	D/L/S/A/EN	5
f. Lower CFT Level and Pressure from the Control Room (006) A4.02	N/S	3
g. Startup Reactor Protection System Channel (012) A4.02	D/S	7
h. Respond to Loss of SCCW (026) AA1.05	D/A/S	8

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

i. Initiate Emergency Boration IAW EOP-020 (004) G2.1.30	D/E/L/R	1
j. Respond to a Loss of Instrument Air (078) A3.01	D/E	8
k. EFW from Fire Service using FS-P-15 (061) A2.04	D/E/L	4S

* All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all five 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 / ≥ 1 / ≥ 1 (control room system)
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: Three Mile Island Date of Examination: 09/28/2015
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐ Operating Test No.: 2015-301

Control Room Systems: *8 for RO; 7 for SRO-I; 2 or 3 for SRO-U

System / JPM Title	Type Code*	Safety Function
a. Respond to an Inoperable/Stuck Control Rod (005) AA1.01	M/S	1
b. Respond to a Loss of Pressurizer Level Control with Failures (011) A2.03	D/A/S	2
c. Restore Seal Injection with a Loss of ICCW (003) K6.02	P/S/A	4P
d. Respond to an OTSG Overfeed (035) A2.04	N/A/S	4S
e. Initiate RB Spray (026) A2.03	D/L/S/A/EN	5
f. Lower CFT Level and Pressure from the Control Room (006) A4.02	N/S	3
g.		
h. Respond to Loss of SCCW (026) AA1.05	D/A/S	8

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

i. Initiate Emergency Boration IAW EOP-020 (004) G2.1.30	D/E/L/R	1
j. Respond to a Loss of Instrument Air (078) A3.01	D/E	8
k. EFW from Fire Service using FS-P-15 (061) A2.04	D/E/L	4S

* All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all five 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 / ≥ 1 / ≥ 1 (control room system)
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: Three Mile Island	Date of Examination: 09/28/2015
Exam Level: RO ☐ SRO-I ☐ SRO-U ☒	Operating Test No.: 2015-301

Control Room Systems: *8 for RO; 7 for SRO-I; 2 or 3 for SRO-U

System / JPM Title	Type Code*	Safety Function
a.		
b.		
c.		
d. Respond to an OTSG Overfeed (035) A2.04	N/A/S	4S
e. Initiate RB Spray (026) A2.03	D/L/S/A/EN	5
f. Lower CFT Level and Pressure from the Control Room (006) A4.02	N/S	3
g.		
h.		

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

i. Initiate Emergency Boration IAW EOP-020 (004) G2.1.30	D/E/L/R	1
j. Respond to a Loss of Instrument Air (078) A3.01	D/E	8
k.		

* All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all five 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 / ≥ 1 / ≥ 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	

JPM A: Respond to an Inoperable/Stuck Control Rod (Modified JPM): The candidate will take control of an individual Control Rod which is greater than 7 inches off from the rest of the rod group and return it to within the acceptable band IAW OP-TM-622-414, Exercising One or More Control Rods.

> Safety Significance: Nine inches is a Tech Spec limit that requires the Control Rod to be declared inoperable. The alarm comes in at 7 inches in order to take action prior to reaching the Tech Spec limit. The Tech Spec axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate. These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K.

This JPM is modified from the previous JPM in that this is designed for the Digital Control Rod Drive System, whereas the original JPM was designed for the older Analog Digital Control Rod System.

JPM B: Respond to a Loss of Pressurizer Level Control with Failures (Bank JPM): The candidate will take manual control of the Pressurizer makeup valve to avoid improper Pressurizer level while at power. Once the alternate instrument is selected and the Pressurizer makeup valve is placed back in automatic control, the upstream Pressurizer makeup valve will fail closed, forcing the candidate to control Pressurizer level with an HPI valve IAW OP-TM-EOP-010, Guide 9, RCS Inventory Control.

> Safety Significance: Tech Specs require that the reactor shall be maintained subcritical by at least one percent delta k/k until a steam bubble is formed and an indicated water level between 80 and 385 inches is established in the pressurizer. If level is too low, there will not be enough inventory to keep the core covered on an event, even a reactor trip, which would cause a transfer of the bubble from the Pressurizer to the Reactor Vessel. If level is too high, there would not be sufficient room in the Pressurizer to prevent severe overpressurization in the event of any single failure.

This JPM is alternate path because the candidate must identify that the upstream valve has gone closed, leave the alarm response, and enter Guide 9.

JPM C: Restore Seal Injection with a Loss of ICCW (Previous two JPM's): The candidate will restore seal injection IAW OP-TM-AOP-041, Loss of Seal Injection. The first Makeup Pump (MU-P-1A) will not start and the candidate will continue in the procedure to start MU-P-1C. As soon as MU-P-1C starts, a loss of Intermediate Closed Cooling Water occurs. The candidate will identify no seal cooling to the Reactor Cooling Pumps and will trip the reactor and secure the reactor coolant pumps.

> Safety Significance: To avoid seal damage, seal injection water flow is required to all RCPs when reactor coolant temperature is above 190°F and pressure is above 100 psig, except while operating in the loss of injection mode. Operating the RCPs in the loss of seal injection mode without intermediate cooling water operating may result in damage to the pump bearing and/or seals from particles in the reactor coolant.

This JPM is alternate path because the candidate must identify that the Reactor Coolant Pumps have no seal cooling and return from Section 6 to Section 3 of OP-TM-AOP-041. The candidate will then enter OP-TM-EOP-001, Reactor Trip.

JPM was randomly chosen via selecting from playing cards representing JPM's from the last two NRC examinations.

JPM D: Respond to an OTSG Overfeed (New Alternate JPM):
The candidate will take manual control of Feedwater regulating valves. While manipulating the "B" set of valves, Feedwater to the "B" OTSG will become excessive, causing the candidate to trip the reactor based on OTSG isolation. Additionally, a Main Feedwater Valve does not automatically isolate on high level and the operator must manually close the valve.

> Safety Significance: If FW is not being controlled and level exceeds 97.5% operating range, actions are taken to promptly stop the overfeed and minimize possible water carryover or main steam line flooding. At 97.5%, HSPS should have stopped FW flow by closing the main FW valves.

This JPM is alternate path because the candidate must identify that the "B" OTSG hi level has occurred and that the reactor may not remain critical. Therefore, the candidate will exit the alarm response and trip the reactor IAW OP-TM-EOP-001, Reactor Trip.

JPM E: Initiate RB Spray (Bank Alternate JPM): The candidate will initiate Reactor Building Spray IAW OP-TM-214-901, RB Spray Operation.

> Safety Significance: The Reactor Building Spray System is a Safety Related system that provides for protection of the integrity of the Reactor Building and limits the release of radioactivity to less than 10CFR100 limits following a Loss of Coolant Accident. The Building Spray system accomplishes this by:
>
> a. Maintaining Reactor Building pressure less than 55 psig
> b. Absorbing Iodine
> c. Providing a means for measurement of Reactor Building pressure
> d. Providing a means to establish post LOCA liquid inventory in an acceptable long term pH range

This JPM is alternate path because the candidate must identify that the "B" Building Spray Train has not properly actuated and then route from section 4.1 to 4.2 of OP-TM-214-901 to take the compensating actions.

JPM F: Lower CFT Level and Pressure from the Control Room (New JPM): The candidate will restore Core Flood Tank "A" level and pressure IAW OP-TM-213 series procedures.

> Safety Significance: Unlike any of the other ECCS components, which require that only train be operational, Tech Specs state that both CFTs are required because a single CFT has insufficient inventory to reflood the core for hot and cold line breaks.

This JPM is a new JPM, created for ILT 14-01 NRC examination.

JPM G: Startup Reactor Protection System Channel (Bank JPM): The candidate will startup Reactor Protection System Channel "D" IAW OP-TM-641-404, De-energizing RPS Channel D.

> Safety Significance: IAW Tech Specs, There are four reactor protection channels. Normal trip logic is two out of four. Minimum required trip logic is one out of two. Every reasonable effort will be made to maintain all safety instrumentation in operation.

This JPM is a bank JPM.

JPM H: Respond to Loss of SCCW (Bank Alternate JPM): The candidate will identify that a Secondary Closed Cooling Water Pump has tripped with no automatic start of the standby pump. The candidate will manually start the standby pump and then recognize that SCCW surge tank level has dropped and will secure SCCW cooled components IAW OP-TM-AOP-033, Loss of Secondary Component Cooling.

Safety Significance: OP-TM-AOP-033, Loss of Secondary Component Cooling, is designed to mitigate the effects of loss of cooling to the components cooled by the secondary closed cooling system. This procedure provides the mitigation strategy for events that challenge the system function. If secondary closed flow is lost (or pumps must be shutdown), then each component cooled by secondary closed is shutdown or otherwise protected from loss of cooling. CSF 4, Core Heat Removal, is affected by the following means:

- Main Feedwater capability is lost (Condensate, Condensate Booster & Main FW Pumps are not available). Condenser Vacuum may be lost. EFW and ADVs are used for RCS heat removal via OTSGs.

This JPM is alternate path because the candidate must identify that the the SCCW Surge Tank has lowered to the point where it is no longer providing net positive suction head to the SCCW Pumps and then must exit the alarm response and route to OP-TM-AOP-033, Loss of Secondary Component Cooling.

JPM I: Initiate Emergency Boration IAW EOP-020 (Bank JPM): The candidate will perform the in-plant steps required to initiate Emergency Boration. This task includes signing onto an RWP.

Safety Significance: Emergency boration is desired to insert negative reactivity and ensure the reactor remains shutdown during a cooldown.

This JPM is a bank JPM.

JPM J: Respond to a Loss of Instrument Air (Bank JPM): The candidate will perform the in-plant steps required to start and maintain backup Instrument Air Compressors.

Above 60 psig, the actions are focused on restoring IA system pressure and identifying the problem, Below 60 psig, the actions are focused on safety and equipment protection. The most challenging post trip threat is the potential to lose all means of RCP seal cooling. Actions must be quickly performed to maintain seal injection or thermal barrier cooling. Primary inventory control (letdown and bleed capabilities), RCS heat removal (OTSG feeding and steaming capabilities) are also affected. CSF-5, Containment Integrity, is affected by the following means:

- All containment isolation valves fail closed on loss of IA.

This JPM is a bank JPM.

JPM K: EFW from Fire Service using FS-P-15 (Bank JPM): The candidate will perform the in-plant steps required to align a portion of the Emergency Feedwater System to be supplied with Fire Service Water.

Safety Significance: The purpose is to provide a means to remove decay heat following a complete loss of the control tower, AC/DC power and licensed operators. When OTSG pressures are approximately 250 psig, FS-P-15 will be capable of providing sufficient head/flow for adequate decay heat removal. Once FS-P-15 is aligned and capable of providing adequate cooling flow (i.e., OTSG pressure < 250 psig and flow requirements less than 200 gpm), OTSG feed will be swapped to the portable pump with flow from the pump matching decay heat.

This JPM is a bank JPM.

Facility:	Three Mile Island	Scenario No.:	1	Op Test No.:	14-01 NRC
Examiners:		Operators:			
Initial Conditions:	• (Temporary IC-175)				
	• 85% Power, MOL				
	• BS-P-1A is OOS for maintenance, expected to return to service in 6 hours.				
	• Crane work is occurring on the West side of the Plant to stage new piping				
Turnover:	Maintain 85% Power Operations				
Critical Tasks:	• PORV Control for Heat Transfer (CT-13) (If conditions are met)				
	• Shutdown Reactor - ATWS (CT-24)				
	• Restore Feed to a Dry OTSG (CT-26)				

Event No.	Malf. No.	Event Type*	Event Description
1	RM0323	TS CRS	Reactor Building Hi Range Radiation Monitor, RM-G-23, Failure
2	ZAIRC1LIC	C CRS C URO	MU-V-17 Fails Closed in Auto, entry into OP-TM-211-472 (URO: Controls Pressurizer Level with MU-V-17 in Manual)
3	ED09D	TS CRS C ARO	Loss of D Inverter, Loss of VBD, entry into OP-TM-AOP-018 (ARO: Place Rad Monitors Interlock switches to Defeat, Restore Control Building Ventilation)
4	02A5S81	C CRS C URO	Low Makeup Tank Pressure, entry into OP-TM-MAP-D0303 (URO: Raise Makeup Tank pressure)
5	IC23	I CRS I URO I ARO	SG/RX Demand Station fails to 0 Volts, Entry into OP-TM-AOP-070 (URO/ARO: ICS station to Manual, Stabilize Power)
6	MU29	C CRS R URO C ARO	RCS leak through the Letdown Line, entry into OP-TM-AOP-050 (URO: Lowers power in Manual ARO: Isolate the Letdown Line)
7	FW15B RD28 RD32	M CRS M URO M ARO	"B" Main Feed Pump trips, "A" Main Feed Pump Runs to 0 rpm, ATWS, Lack of Primary to Secondary Heat Transfer.
8	FW19	C CRS C ARO	EFW Control Valves fail to operate, EF-V-52A-D Closed (ARO: Establish PSHT via Condensate Booster Pump flow)
9 (if required)	MU35B	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

Facility:	Three Mile Island	Scenario No.: 2	Op Test No.: 14-01 NRC
Examiners:		Operators:	
Initial Conditions:	• (Temporary IC-176)		
	• 100% Power, MOL		
	• BS-P-1A is OOS for maintenance, expected to return to service in 6 hours.		
	• Crane work is occurring on the West side of the Plant to stage new piping		
Turnover:	Maintain 100% Power Operations		
Critical Tasks:	• Electrical Power Alignment (CT-8)		
	• Turbine Trip (CT-18)		
	• Protect against RCP Seal LOCA (CT-*)		

Event No.	Malf. No.	Event Type*	Event Description
1	NI27A	I CRS I URO I ARO	Pressurizer Pressure Instrument Fails High, entry into OP-TM-MAP-G0106, OP-TM-MAP-G0107 (URO: Blocks PORV, closes Spray Valve, Pressurizer Heater Control in Manual, ARO: "A" RPS to Manual Bypass)
2	IC12	C CRS C URO C ARO	Total RCS Flow IN Fails to Zero Volts, entry into OP-TM-AOP-070 and 1102-4. (URO/ARO: ICS station to Manual, Stabilize Power)
3	03A3S09 - ZDI1SAE 2(1)	TS CRS C URO C ARO	Loss of 1E 4KV Bus, Entry into OP-TM-AOP-014 (URO: Manual control of Makeup valves. ARO: Restore Seal Injection)
4	TU01D	C CRS R URO N ARO	High Vibrations on Main Turbine, entry into OP-TM-MAP-K0201 and 1102-4 (URO/ARO: Power reduction with ICS in Manual)
5	EG04A EG04B	I CRS I URO	Loss of Stator Coolant Pumps, Main Turbine fails to automatically runback and trip (URO: Trip Reactor)
6	HVB-1-1 HVB-2-1 A-1-4	TS CRS C ARO	Fire in EG-Y-1B Room, entry into OP-TM-AOP-001 (ARO: Secure EG-Y-1B)
7	ED01	M CRS M URO M ARO	Loss of Offsite Power, entry into OP-TM-AOP-020.
8	EG01A	C CRS C URO	"A" EDG fails to start, SBO start required. (URO)

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

Facility:	Three Mile Island	Scenario No.: 3	Op Test No.: 14-01 NRC
Examiners:		Operators:	
Initial Conditions:	• (Temporary IC-177)		
	• 85% Power, MOL		
	• BS-P-1A is OOS for maintenance, expected to return to service in 6 hours.		
	• Crane work is occurring on the West side of the Plant to stage new piping		
Turnover:	Maintain 85% Reactor Power		
Critical Tasks:	• Control HPI (CT-5)		
	• Establish FW Flow and Feed SG(s) (CT-10)		
	• Natural Circulation RCS Flow (CT-12)		

Event No.	Malf. No.	Event Type*	Event Description
1	DHR32	TS CRS	BWST level lowers, entry into OP-TM-MAP-E0204
2	03A4S01 - ZDIPB1R CB ON	TS CRS I URO I ARO	Inadvertent ES Actuation, "B" Train (TS), entry into OP-TM-AOP-046 (URO: Defeats signal, ARO: Opens MU-V-2A/B)
3	RC08B IC51	I CRS I URO I ARO	Tc Instrument Fails High, SASS Fails to Actuate, entry into OP-TM-AOP-070 (URO: Manual control of Control Rods, ARO: Manual control of Feedwater)
4	MU19	C CRS R URO N ARO	RC-P-1A #1 Seal Leak, leak at 6.5 gpm, Entry into OP-TM-AOP-040 (URO/ARO: Power reduction in manual)
5	MU19	C CRS C URO	RC-P-1A #1 Seal Failure, leak at 10 gpm, Entry into OP-TM-AOP-040 (URO: Secure RC-P-1A)
6	MS19A	C CRS C ARO	Isolable Steam Leak in Turbine Bldg, entry into OP-TM-AOP-051. (ARO: Isolate Steam Leak)
7	TH06	M CRS M URO M ARO	RCS LOCA, entry into OP-TM-EOP-001.
8	CC06A	C CRS C URO	NSCCW Rupture in RC-P-1A Motor Air Cooler, Loss of NSCCW, Reactor trip, entry into OP-TM-AOP-031, and OP-TM-EOP-001 (URO: Reactor Trip IMA's)
9	ICR02 ICR04	C CRS C ARO	HSPS fails to feed OTSG's to 50% (ARO: Feed OTSG's to >50% in manual)

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