Exelon Generation

10 CFR 55.40

LG-16-119

October 18, 2016

Mr. Daniel Dorman, Regional Administrator
U.S. Nuclear Regulatory Commission, Region 1
2100 Renaissance Blvd., Suite 100
King of Prussia, PA 19406-2713

Limerick Generating Station (LGS), Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NFP-85
Docket Nos. 50-352 & 50-353

Subject: Initial License Operator Training Program Exam Outlines

Reference: NRC Letter “Senior Reactor and Reactor Operator Initial License Examinations – Limerick Generating Station Units 1 and 2,” dated July 14, 2016

As requested in the reference, the following NUREG–1021 required documents are being provided to Mr. P. Presby regarding the Initial Operator Licensing Exams to be administered beginning on January 16, 2017 at Limerick Generating Station.

ES-201-2	Exam Outline Quality Checklist
ES-201-3	Examination Security Agreement
ES-301-1	Administrative Topics Outlines
ES-301-2	Control Room/In-plant Systems Outlines
ES-301-5	Transient and Event Checklists
ES-401-1	BWR Exam Outline
ES-401-3	Generic Knowledge and Abilities Outline
ES-401-4	Record of Rejected K/As
ES-D-1	Scenario Outlines

Additionally the following Items are being provided:

- A copy of the suppressed K/A’s with the bases for each suppression
- A draft exam schedule
- Reference materials required to support exam material and outline review

There are no regulatory commitments contained in this letter.

If you have any questions or need additional information, please contact Mr. Daniel Semeter at (610) 718-4040.

Respectfully,

Rick Libra

Richard W. Libra
Vice President – Limerick Generating Station
Exelon Generation Co., LLC

Enclosures:	Form ES-201-2	Exam Outline Quality Checklist
	Form ES-201-3	Examination Security Agreement
	Form ES-301-1	Administrative Topics Outlines
	Form ES-301-2	Control Room/In-plant Systems Outlines
	Form ES-301-5	Transient and Event Checklists
	Form ES-401-1	BWR Exam Outline
	Form ES-401-3	Generic Knowledge and Abilities Outline
	Form ES-401-4	Record of Rejected K/As
	Form ES-D-1	Scenario Outlines

cc:	P. Presby USNRC, Region I	(w/encl)
	D. Jackson, USNRC, Region I	(w/o encl)
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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). (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 ±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 with justification; 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/A's
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 a category 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.

G* Generic K/As

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
295018 Partial or Total Loss of CCW / 8					X		AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Component temperatures	3.4	76
295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1					X		EA2.05 - Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Control rod position	4.3	77
600000 Plant Fire On-site / 8					X		AA2.17 - Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Systems that may be affected by the fire	3.6	78
295019 Partial or Total Loss of Inst. Air / 8						X	2.2.36 - Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	4.2	79
295005 Main Turbine Generator Trip / 3						X	2.4.6 - Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.	4.7	80
295026 Suppression Pool High Water Temp. / 5						X	2.4.47 - Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	4.2	81
700000 Generator Voltage and Electric Grid Disturbances					X		AA2.10 - Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator overheating and the required actions.	3.8	82
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	X						AK1.01 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Natural circulation	3.5	39
295026 Suppression Pool High Water Temp. / 5	X						EK1.01 - Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE : Pump NPSH	3.0	40
295038 High Off-site Release Rate / 9		X					EK2.04 - Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Stack-gas monitoring system: Plant-Specific	4.2	41
295031 Reactor Low Water Level / 2		X					EK2.03 - Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Low pressure core spray	4.2	42
600000 Plant Fire On-site / 8		X					AK2.04 - Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Breakers, relays, and disconnects	2.5	43
295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1		X					EK2.11 - Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: RMCS: Plant-Specific	3.8	44
295024 High Drywell Pressure / 5			X				EK3.04 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : Emergency depressurization	3.7	45

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
295021 Loss of Shutdown Cooling / 4			X				AK3.05 - Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING : Establishing alternate heat removal flow paths	3.6	46
295016 Control Room Abandonment / 7			X				AK3.02 - Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT : Turbine trip	3.7	47
295003 Partial or Complete Loss of AC / 6				X			AA1.04 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : D.C. electrical distribution system	3.6	48
295018 Partial or Total Loss of CCW / 8				X			AA1.02 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : System loads	3.3	49
700000 Generator Voltage and Electric Grid Disturbances				X			AA1.05 - Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Engineered safety features.	3.9	50
295023 Refueling Acc Cooling Mode / 8					X		AA2.04 - Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : Occurrence of fuel handling accident	3.4	51
295028 High Drywell Temperature / 5					X		EA2.01 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Drywell temperature	4.0	52
295005 Main Turbine Generator Trip / 3					X		AA2.03 - Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Turbine valve position	3.1	53
295025 High Reactor Pressure / 3						X	2.4.18 - Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.	3.3	54
295030 Low Suppression Pool Water Level / 5						X	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	55
295006 SCRAM / 1						X	2.4.46 - Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions.	4.2	56
295004 Partial or Total Loss of DC Pwr / 6						X	2.4.49 - Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.6	57
295019 Partial or Total Loss of Inst. Air / 8			X				AK3.01 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Backup air system supply: Plant-Specific	3.3	58
K/A Category Totals:	2	4	4	3	3/4	4/3	Group Point Total:		20/7

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
295010 High Drywell Pressure / 5					X		AA2.03 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE : Drywell radiation levels	3.6	83
295002 Loss of Main Condenser Vac / 3						X	2.4.49 - Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.4	84
295017 High Off-site Release Rate / 9					X		AA2.04 - Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : Source of off-site release	4.3	85
295012 High Drywell Temperature / 5	X						AK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE : Pressure/temperature relationship	3. 3	59
295014 Inadvertent Reactivity Addition / 1		X					AK2.04 - Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: Void concentration	3.2	60
295017 High Off-site Release Rate / 9			X				AK3.02 - Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE : Plant ventilation	3. 3	61
295008 High Reactor Water Level / 2				X			AA1.08 - Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL : Feedwater system	3. 5	62
295022 Loss of CRD Pumps / 1					X		AA2.02 - Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : CRD system status	3. 3	63
295002 Loss of Main Condenser Vac / 3						X	2.1.31 - Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.	4. 6	64
295010 High Drywell Pressure / 5					X		AA2.02 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE : Drywell pressure	3. 8	65
K/A Category Totals:	1	1	1	1	2/2	1/1	Group Point Total:		7/3

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
400000 Component Cooling Water								X				A2.04 - Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Radiation monitoring system alarm	3.0	86
215005 APRM / LPRM								X				A2.01 - Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions Power supply degraded	3.1	87
263000 DC Electrical Distribution											X	2.1.27 - Conduct of Operations: Knowledge of system purpose and / or function.	4.0	88
215003 IRM											X	2.2.40 - Equipment Control: Ability to apply technical specifications for a system.	4.7	89
239002 SRVs								X				A2.06 - Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Reactor high pressure	4.3	90
263000 DC Electrical Distribution	X											K1.02 - Knowledge of the physical connections and/or cause- effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Battery charger and battery	3.2	1
217000 RCIC	X											K1.07 - Knowledge of the physical connections and/or cause- effect relationships between REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and the following: Leak detection	3.1	2
205000 Shutdown Cooling		X										K2.02 - Knowledge of electrical power supplies to the following: Motor operated valves	2.5	3
206000 HPCI		X										K2.04 - Knowledge of electrical power supplies to the following: Turbine control circuits: BWR-2,3,4	2.5	4
300000 Instrument Air			X									K3.02 - Knowledge of the effect that a loss or malfunction of the (INSTRUMENT AIR SYSTEM) will have on the following: Systems having pneumatic valves and controls	3.3	5

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
259002 Reactor Water Level Control			X									K3.02 - Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following: Reactor feedwater system	3.7	6
262001 AC Electrical Distribution				X								K4.04 - Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Protective relaying	2.8	7
218000 ADS				X								K4.01 - Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: Prevent inadvertent initiatior of ADS logic	3.7	8
209001 LPCS					X							K5.04 - Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : Heat removal (transfer) mechanisms	2.8	9
215003 IRM					X							K5.01 - Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : Detector operation	2.6	10
264000 EDGs						X						K6.07 - Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET) : Cooling water system	3.8	11
400000 Component Cooling Water						X						K6.06 - Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Heat exchangers and condensers	2.9	12
203000 RHR/LPCI: Injection Mode							X					A1.09 - Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Component cooling water systems	2.9	13
223002 PCIS/Nuclear Steam Supply Shutoff							X					A1.03 - Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: SPDS/ERIS/CRIDS/GDS: Plant-Specific	2.5	14

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G		Imp.	Q#
239002 SRVs								X				A2.02 - Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Leaky SRV	3.1	15
215004 Source Range Monitor								X				A2.05 - Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/system	3.3	16
262002 UPS (AC/DC)									X			A3.01 - Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source	2.8	17
212000 RPS									X			A3.07 - Ability to monitor automatic operations of the REACTOR PROTECTION SYSTEM including: SCRAM air header pressure	3.6	18
215005 APRM / LPRM										X		A4.01 - Ability to manually operate and/or monitor in the control room: IRM/APRM recorder	3.2	19
211000 SLC										X		A4.06 - Ability to manually operate and/or monitor in the control room: RWCU system isolation	3.9	20
261000 SGTS											X	2.1.27 - Conduct of Operations: Knowledge of system purpose and / or function.	3.9	21
211000 SLC											X	2.2.36 - Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	3.1	22
261000 SGTS										X		A4.05 - Ability to manually operate and/or monitor in the control room: Drywell to suppression chamber/torus differential pressure: Mark-I,II	2.9	23
263000 DC Electrical Distribution					X							K5.01 - Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION : Hydrogen generation during battery charging.	2.6	24

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
206000 HPCI							X					A1.06 - Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including: System flow: BWR-2,3,4	3.8	25
203000 RHR/LPCI: Injection Mode									X			A3.02 - Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including: Pump start	4.0	26
K/A Category Totals:	2	2	2	2	3	2	3	2/3	3	3	2/2	Group Point Total:	26/5	

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q #
202002 Recirculation Flow Control								X				A2.04 - Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation pump speed mismatch between loops: Plant-Specific	3.2	91
226001 RHR/LPCI: CTMT Spray Mode											X	2.4.35 - Emergency Procedures / Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects.	4.0	92
290001 Secondary CTMT								X				A2.03 - Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High area radiation	3.6	93
245000 Main Turbine Gen. / Aux.	X											K1.09 - Knowledge of the physical connections and/or cause- effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: D. C . electrical distribution	2.7	27
202001 Recirculation		X										K2.01 - Knowledge of electrical power supplies to the following: Recirculation pumps: Plant-Specific	3.2	28
290001 Secondary CTMT			X									K3.01 - Knowledge of the effect that a loss or malfunction of the SECONDARY CONTAINMENT will have on following: Off-site radioactive release rates	4.0	29
271000 Off-gas				X								K4.06 - Knowledge of OFFGAS SYSTEM design feature(s) and/or interlocks which provide for the following: Decay of fission product gases to particulate daughters	2.7	30
256000 Reactor Condensate					X							K5.08 - Knowledge of the operational implications of the following concepts as they apply to REACTOR CONDENSATE SYSTEM: Heat removal (transfer) mechanisms	2.6	31
286000 Fire Protection						X						K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the FIRE PROTECTION SYSTEM A.C. electrical distribution: Plant-Specific	3.1	32

Limerick January 2017 ILT NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G		Imp.	Q #
219000 RHR/LPCI: Torus/Pool Cooling Mode							X					A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE controls including: Suppression pool temperature	4.0	33
214000 RPIS								X				A2.03 - Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Overtravel/in-out	3.6	34
288000 Plant Ventilation									X			A3.01 - Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: Isolation/initiation signals	3.8	35
230000 RHR/LPCI: Torus/Pool Spray Mode										X		A4.13 - Ability to manually operate and/or monitor in the control room: Suppression chamber pressure	4.0	36
201001 CRD Hydraulic											X	2.2.44 - Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives effect plant and system conditions.	4.2	37
223001 Primary CTMT and Aux.				X								K4.04 - Knowledge of PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES design feature(s) and/or interlocks which provide for the following: Prevents hydrogen from reaching an explosive mixture	3.5	38
K/A Category Totals:	1	1	1	2	1	1	1	1/2	1	1	1/1	Group Point Total:		12/3

Facility:	Limerick January 2017 ILT NRC Exam	Date:				
Category	K/A #	Topic	RO		SRO-Only	
			IR	Q#	IR	Q#
1. Conduct of Operations	2.1.13	Knowledge of facility requirements for controlling vital / controlled access.			3.2	94
	2.1.36	Knowledge of procedures and limitations involved in core alterations.			4.1	99
	2.1.8	Ability to coordinate personnel activities outside the control room.	3.4	66		
	2.1.32	Ability to explain and apply all system limits and precautions.	3.8	67		
	2.1.25	Ability to interpret reference materials, such as graphs, curves, tables, etc.	3.9	75		
	Subtotal			3		2
2. Equipment Control	2.2.20	Knowledge of the process for managing troubleshooting activities.			3.8	95
	2.2.43	Knowledge of the process used to track inoperable alarms.			3.3	98
	2.2.21	Knowledge of pre- and post-maintenance operability requirements.	2.9	68		
	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	69		
	2.2.39	Knowledge of less than one hour technical specification action statements for systems.	3.9	74		
	Subtotal			3		2
3. Radiation Control	2.3.6	Ability to approve release permits.			3.8	96
	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personell monitoring equipment, etc.			2.9	100
	2.3.13	Knowledge of Radialogical 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	70		
	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	71		

	Subtotal			2		2
4. Emergency Procedures / Plan	2.4.38	Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.			4.4	97
	2.4.11	Knowledge of abnormal condition procedures.	4.0	72		
	2.4.35	Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects.	3.8	73		
	Subtotal			2		1
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/1 Q#1	263000 K1.03	K1.03 - Knowledge of the physical connections and/or cause-effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Battery ventilation There are no DC components, all fans are MCC level 480 volt power (ac control power) all solenoids dampers are ac powered. This leaves the cause and effect which would lead to duplication/overlap with Q 24 263000 K5.01 Hydrogen generation during battery charging. K1.03 removed random number generator used to select from remaining KA's K1.02 selected Battery charger and battery
1/1 Q# 46	295021 AK3.03	AK3.03 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Increasing drywell cooling There is no procedural direction at LGS to increase Drywell cooling on a loss of Shutdown cooling, also the difficulty of any question for this would be low and 4 plausible distractors would not be possible AK3.03 removed from 295021 AK3's random number generator used to reselect AK3.05 selected
1/1 Q51	295023 Refueling Acc Cooling Mode / 8	AA2.05 - Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : Entry conditions of emergency plan Unable to write an RO level question to AA2.05 AA2.05 removed random number generator used to select from the remaining 4 AA2.01-2.04 AA2.04 Occurrence of fuel handling accident selected
1/1 Q55	295030 Low Suppression Pool Water Level / 5	2.4.41 - Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications. Unable to write an RO question to this K/A Reselected with random number generator from generic 2.4 not already used in RO exam 2.4.4 selected
1/1 Q41	295038 High Off-site Release Rate / 9	EK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE : Protection of the general public KA deselected due to inability to write question with required LOD due to overlap with other questions in this exam or cert exam. Unable to write questions with sufficient LOD for the remaining 2 EK1 KAs Used random number generator to select EK2.04

Facility: Limerick	Date of Examination: 01/17/17
Examination Level: RO X SRO	Operating Test Number: 1

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, D	2.1.19 (Ability to use plant computers to evaluate system or component status. (CFR: 41.10 / 45.12) IMPORTANCE RO 3.9 SRO 3.8 Evaluate jet pump operability (LOJPM6717)
Conduct of Operations	R, N	G2.1.25 (Ability to interpret reference materials, such as graphs, curves, tables, etc. (CFR: 41.10 / 43.5 / 45.12) IMPORTANCE RO 3.9 SRO 4.2) Determine Maximum Generator VARS (LOJPM6719)
Equipment Control	R, D	G2.2.42 (Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3) IMPORTANCE RO 3.9 SRO 4.6) Administrative Actions on a Thermal Limit Violation (LOJPM6714)
Radiation Control	R, N	G2.3.15 (Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9) IMPORTANCE RO 2.9 SRO 3.1) Administrative Requirements for Bypassing a Spiking Radiation Monitor (LOJPM6762)
Emergency Procedures/Plan		

NOTE: All items (five total) are required for SROs. RO applicants require only four 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)

Facility: Limerick	Date of Examination: 01/17/17
Examination Level: RO SRO X	Operating Test Number: 1

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, D	G2.1.5 (Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc. (CFR: 41.10 / 43.5 / 45.12) IMPORTANCE RO 2.9* SRO 3.9) Determination of Adequate Shift Staffing (LOJPM6712)
Conduct of Operations	R, N	G2.1.40 (Knowledge of refueling administrative requirements. (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 2.8 SRO 3.9) Refueling: Determine Acceptability of Installing Fuel Pool Gates (LOJPM6763)
Equipment Control	R, D	G2.2.40 (Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3) IMPORTANCE RO 3.4 SRO 4.7) Evaluate Jet Pump Operability (LOJPM6717)
Radiation Control	R, M	G2.3.11 (Ability to control radiation releases. (CFR: 41.11 / 43.4 / 45.10) IMPORTANCE RO 3.8 SRO 4.3 DETERMINE OFFGAS EFFLUENT ACTIVITY RELEASE RATE (LOJPM6760)
Emergency Procedures/Plan	S, D	G2.4.41 (Knowledge of the emergency action level thresholds and classifications. (CFR: 41.10 / 43.5 / 45.11) IMPORTANCE RO 2.9 SRO 4.6) EAL declaration (LOJPM3097)

NOTE: All items (five total) are required for SROs. RO applicants require only four 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)

Facility: LIMERICK Date of Examination: 1/17/2017
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐ Operating Test No.: 1

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

System / JPM Title	Type Code*	Safety Function
a. MANUAL SLC INITIATION WITH B SLC PUMP FAILURE (System 211000 – Standby Liquid Control; A4.08; 4.2/4.2) (LOJPM3094)	A, D, EN, S	1 – Reactivity Control
b. RCIC MANUAL SLOW START USING FIC-49-1R600 (System 217000 – RCIC; A4.04; 3.6/3.6) (LOJPM3015)	D, EN, S	2 – Rx Water Inventory Ctrl.
c. BPV EXERCISE TEST (System 241000 – Rx/Turb Pressure Regulating System; A4.06; 3.9/3.9) (LOJPM3083)	N, S	3 – Reactor Pressure Control
d. SHUTDOWN COOLING FLOW ADJUSTMENTS - RHRSW HI RAD ADJUSTMENTS – RHRSW HI RAD (System 205000 – Shutdown Cooling; A4.04; 3.4/3.3) (LOJPM3515)	A, D, EN, L, S	4 – Heat removal from core
e. PLACE FEEDWATER FILL SYSTEM IN SERVICE (System 223001- Primary Containment System and Auxiliaries; A2.01; 4.3/4.4) (LOJPM3118)	EN, L, N, S	5 – Containment Integrity
f. DIESEL GENERATOR FAST START FROM THE MCR (System - 264000 – Emergency Diesel Generator; A4.04; 3.7/3.7) (LOJPM3130)	A, D, EN, S	6 - Electrical
g. SUPPLY RECW TO THE DRYWELL COOLERS (System 400000 – Component Cooling Water (CCWS); A2.01; 3.3/3.4) (LOJPM3028)	D, L, S	8 – Plant service systems
h. MANUALLY INITIATE A CONTROL ROOM CHLORINE/TOXIC CHEMICAL ISOLATION (System 290003 – Control Room HVAC; A3.01; 3.3/3.5) (LOJPM3023)	D, S	9 – Radioactivity Release
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. Reset RDCS leading to the need for bypass a control rod (System 201002 – Reactor Manual Control; A2.04; 3.2/3.1) (LOJPM2119)	A, N, R	1 – Reactivity Control
j. T-239, Defeating RFPT, HPCI, RCIC High Level interlock (EPE295031 - Reactor Low Water Level; EA1.02; 4.5/4.5) (LOJPM2273)	E, N, R	2 – Reactor Water Inventory
k. START ESW PUMP PER SE-1 (APE295016 – Control Room Abandonment; AA1.04; 3.1/3.2) (LOJPM2258) **(Unit 2)**	A, D, E, R	7 - Instrumentation

* 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	5
(C)ontrol room		
(D)irect from bank	**≤ 9** / ≤ 8 / ≤ 4	7
(E)mergency or abnormal in-plant	**≥ 1** / ≥ 1 / ≥ 1	2
(EN)gineered safety feature	**≥ 1** / ≥ 1 / ≥ 1 (control room system)	5
(L)ow-Power / Shutdown	**≥ 1** / ≥ 1 / ≥ 1	3
(N)ew or (M)odified from bank including 1(A)	**≥ 2** / ≥ 2 / ≥ 1	4
(P)revious 2 exams	**≤ 3** / ≤ 3 / ≤ 2 (randomly selected)	0
(R)CA	**≥ 1** / ≥ 1 / ≥ 1	3
(S)imulator		

Facility: LIMERICK	Date of Examination: 1/17/2017	
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐	Operating Test No.: 1	

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

System / JPM Title	Type Code*	Safety Function
a. MANUAL SLC INITIATION WITH B SLC PUMP FAILURE (System 211000 – Standby Liquid Control; A4.08; 4.2/4.2) (LOJPM3094)	A, D, EN, S	1 – Reactivity Control
b. RCIC MANUAL SLOW START USING FIC-49-1R600 (System 217000 – RCIC; A4.04; 3.6/3.6) (LOJPM3015)	D, EN, S	2 – Rx Water Inventory Ctrl.
d. SHUTDOWN COOLING FLOW ADJUSTMENTS - RHRSW HI RAD ADJUSTMENTS – RHRSW HI RAD (System 205000 – Shutdown Cooling; A4.04; 3.4/3.3) (LOJPM3515)	A, D, EN, L, S	4 – Heat removal from core
e. PLACE FEEDWATER FILL SYSTEM IN SERVICE (System 209001 – Low Pressure Core Spray; A4.07; 2.7/2.8) (LOJPM3118)	EN, L, N, S	5 – Containment Integrity
f. DIESEL GENERATOR FAST START FROM THE MCR (System - 264000 – Emergency Diesel Generator; A4.04; 3.7/3.7) (LOJPM3130)	A, D, EN, S	6 - Electrical
g. SUPPLY RECW TO THE DRYWELL COOLERS (System 400000 – Component Cooling Water (CCWS); A2.01; 3.3/3.4) (LOJPM3028)	D, L, S	8 – Plant service systems
h. MANUALLY INITIATE A CONTROL ROOM CHLORINE/TOXIC CHEMICAL ISOLATION (System 290003 – Control Room HVAC; A3.01; 3.3/3.5) (LOJPM3023)	D, S	9 – Radioactivity Release
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. Reset RDCS leading to the need for bypass a control rod (System 201002 – Reactor Manual Control; A2.04; 3.2/3.1) (LOJPM2226)	A, N, R	1 – Reactivity Control
j. T-239, Defeating RFPT, HPCI, RCIC High Level interlock (EPE295031 - Reactor Low Water Level; EA1.02; 4.5/4.5) (LOJPM2273)	E, N, R	2 – Reactor Water Inventory
k. START ESW PUMP PER SE-1 (APE295016 – Control Room Abandonment; AA1.04; 3.1/3.2) (LOJPM2258) **(Unit 2)**	A, D, E, R	7 - Instrumentation

* 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	5
(C)ontrol room		
(D)irect from bank	≤ 9 / **≤ 8** / ≤ 4	7
(E)mergency or abnormal in-plant	≥ 1 / **≥ 1** / ≥ 1	2
(EN)gineered safety feature	≥ 1 / **≥ 1** / ≥ 1 (control room system)	5
(L)ow-Power / Shutdown	≥ 1 / **≥ 1** / ≥ 1	3
(N)ew or (M)odified from bank including 1(A)	≥ 2 / **≥ 2** / ≥ 1	3
(P)revious 2 exams	≤ 3 / **≤ 3** / ≤ 2 (randomly selected)	0
(R)CA	≥ 1 / **≥ 1** / ≥ 1	3
(S)imulator		

Facility: LIMERICK Date of Examination: 1/17/2017
Exam Level: RO ☐ SRO-I ☐ SRO-U ☒ Operating Test No.: 1

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

System / JPM Title	Type Code*	Safety Function
f. DIESEL GENERATOR FAST START FROM THE MCR (System - 264000 – Emergency Diesel Generator; A4.04; 3.7/3.7) (LOJPM3130)	A, D, EN, S	6 - Electrical
g. SUPPLY RECW TO THE DRYWELL COOLERS (System 400000 – Component Cooling Water (CCWS); A2.01; 3.3/3.4) (LOJPM3028)	D, L, S	8 – Plant service systems

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

i. Reset RDCS leading to the need for bypass a control rod (System 201002 – Reactor Manual Control; A2.04; 3.2/3.1) (LOJPM2226)	A, N, R	1 – Reactivity Control
j. T-239, Defeating RFPT, HPCI, RCIC High Level interlock (EPE295031 - Reactor Low Water Level; EA1.02; 4.5/4.5) (LOJPM2273)	E, N, R	2 – Reactor Water Inventory
k. START ESW PUMP PER SE-1 (APE295016 – Control Room Abandonment; AA1.04; 3.1/3.2) (LOJPM2258) **(Unit 2)**	A, D, E, R	7 - Instrumentation

* 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**	3
(C)ontrol room		
(D)irect from bank	≤ 9 / ≤ 8 / **≤ 4**	3
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / **≥ 1**	2
(EN)gineered safety feature	≥ 1 / ≥ 1 / **≥ 1** (control room system)	1
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / **≥ 1**	1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / **≥ 1**	2
(P)revious 2 exams	≤ 3 / ≤ 3 / **≤ 2** (randomly selected)	0
(R)CA	≥ 1 / ≥ 1 / **≥ 1**	3
(S)imulator		

Limerick Generating Station ILT NRC Exam (1/17/17) JPM Summaries

Simulator JPMs

(LOJPM3094) MANUAL SLC INITIATION WITH B SLC PUMP FAILURE (Alternate Path)

The conditions of the simulator are that an ATWS is in progress. The candidate is directed to manually initiate the Unit 1 SLC System using S48.1.B. The candidate will start the 1A and 1B SLC pumps and verify system operation. They will verify the RWCU system isolated. During this process the candidate will identify that the 1B SLC Pump trips. Upon trip the candidate dispatches an Equipment Operator, based on reports they will determine that the 1B SLC pump is not available. The candidate will now determine that 2 SLC pumps are still required and obtain the key for the 1C SLC Pump and place the 1C SLC pump in operation and verify correct system response.

(LOJPM3015) RCIC MANUAL SLOW START USING FIC-49-1R600

The conditions of the simulator are normal full power operation. The candidate is directed to place U1 RCIC in full flow test for a 15 minute PMT using the manual slow start method. The slow start method requires the candidate to place the flow controller in manual and minimized. Once turbine speed is observed a flow path from the Condensate Storage Tank is established and turbine speed is raised establishing the required flow and pressure values.

(LOJPM3083) BPV EXERCISE TEST

The conditions of the simulator are power is reduced to 99% or less. The candidate is then directed to perform the Bypass Valve Exercise Test which involves multiple manipulations at the Digital EHC computer in the Simulator.

(LOJPM3515) SHUTDOWN COOLING FLOW ADJUSTMENTS - RHRSW HI RAD ADJUSTMENTS – RHRSW HI RAD (Alternate Path)

The conditions of the simulator are U1 is shutdown with Reactor temperature of 125 degrees F with 1A RHR is in service in the Shutdown Cooling Mode. The candidates are told additional cooling is required and are directed to make adjustments to provide additional cooling. Following cooling adjustments, a high radiation condition in the

RHRSW system will occur. The candidate responds to this condition by tripping the RHR pump and isolating the RHR Heat Exchanger.

(LOJPM3118) PLACE FEEDWATER FILL SYSTEM IN SERVICE

The conditions of the simulator are U1 experienced a LOCA with a LOOP. The candidates are directed to place the Safeguard Piping Fill system in service using S52.1.C. This requires the start of the A and B Safeguard Piping Fill Pumps and opening the 4 associated injection valves.

(LOJPM3130) DIESEL GENERATOR FAST START FROM THE MCR (Alternate Path)

The conditions of the simulator are U1 is at power with D14 EDG running. The candidates are asked to continue with the D14 EDG fast start testing. Once the candidates synchronize the EDG to the Bus, D14 EDG experiences elevated jacket water temperature conditions and the candidates are expected to separate the EDG from the Bus and secure the engine.

(LOJPM3028) SUPPLY RECW TO THE DRYWELL COOLERS

The conditions of the simulator are U1 has been shutdown due to rising Drywell Pressure, both Drywell chillers are unavailable. The candidates are directed to align Reactor Enclosure Cooling Water (RECW) to the A Drywell Chilled Water Loop. This involves isolating RECW to loads that are not required and valve alignments to establish the appropriate flow path.

(LOJPM3023) MANUALLY INITIATE A CONTROL ROOM CHLORINE/TOXIC CHEMICAL ISOLATION

The conditions of the simulator are U1 is at normal full power operation. The candiates are directed to manually initiate a Main Control Room HVAC Chlorine/Toxic Chemical Isolation for maintenance using the "B" subsystem only. This involves placing the system in reset, aligning selectors to chlorine, restoring reset switches, depressing isolation pushbuttons and monitoring system performance.

In Plant JPMs

(LOJPM2119) Reset RDCS leading to the need for bypass a control rod (Alternate Path)

The conditions for this JPM are the unit has experienced a Rod Drive Control System (RDCS) INOP condition. The candidates are directed to reset RDCS in the Aux Equipment Room (AER). The reset is unsuccessful and the candidate is expected to

determine that the cause is a particular control rod and proceed to bypassing the control rod in RDCS and perform another RDCS reset.

(LOJPM2273) T-239, Defeating RFPT, HPCI, RCIC High Level interlock

The conditions for this JPM are the unit has experienced an RPV Level Unknown condition with RPV flooding required. The candidates are directed to defeat the feed pump high level interlocks which involves lifting and taping leads in the Aux Equipment Room for all 3 feed pumps.

(LOJPM2258) START ESW PUMP PER SE-1 **(Unit 2)**

The conditions for this JPM are the Main Control Room has been evacuated and the Emergency Diesel Generators running but one is reported to have no cooling water flow. The candidates are directed to start the A Emergency Service Water (ESW) Pump using SE-1, which involves using a transfer switch and control switch to close the 4 kV Breaker for the ESW Pump.

RO Admin

(LOJPM6717) Evaluate jet pump operability

The candidates are directed to Perform ST-6-043-320-1, Daily Jet Pump Operability Verification for Two Recirculation Loop Operation and report the results. As the candidate progresses through the performance of the test, they determine that the test is Unsatisfactory due to failure of Speed – Loop flow characteristics curve and Jet pump diffuser to lower plenum deferential pressure.

(LOJPM6719) Determine Maximum Generator VARS

The candidates is directed to respond to the TSOs request, and determine the Maximum VAR output that remains within the capability of the U2 Generator for the current plant conditions, and provide a "what if" for a reduced generator hydrogen pressure condition. This is accomplished by interpreting a graph containing three variables: real power, reactive power, and generator hydrogen pressure.

(LOJPM6714) Administrative Actions on a Thermal Limit Violation

The candidates are directed to review the official 3D Monicore Periodic Log (P1) following power ascension. The candidates are expected to determine that FMLCPR is exceeding Tech Spec Limits and that GP-14, Resolution of Thermal Limit Violations is required to be implemented. Candidate determines a Reactor power reduction is required with Control Rods only using RMSI **AND** GP-5 Appendix 2 concurrently until core Thermal Limits are less than 1.000.

(LOJPM6762) Administrative Requirements for Bypassing a Spiking Radiation Monitor

The candidates are presented with a specific radiation monitor that is spiking and they are directed to take the required actions to ensure it is appropriately bypassed.

SRO Admin

(LOJPM6712) Determination of Adequate Shift Staffing

Candidate is presented with a staffing issue where several operators become ill and they are tasked with determining if staffing requirements for current operating modes are met and to include any immediate and long term (greater than 2 hours) corrective actions that are required to ensure adequate shift staffing is met.

(LOJPM6763) Refueling: Determine Acceptability of Installing Fuel Pool Gates

Candidate is provided with Fuel Pool Cooling Heat Exchange cooling capability, Decay Heat Load estimates and available systems and are tasked with determining if it is acceptable to separate the Spent Fuel Pool from the Reactor Cavity.

(LOJPM6717) Evaluate Jet Pump Operability

The candidates are directed to review the official 3D Monicore Periodic Log (P1) follow power ascension. The candidates are expected to determine that FMLCPR is exceeding Tech Spec Limits and that GP-14, Resolution of Thermal Limit Violations is required to be implemented. Candidate determines a Reactor power reduction required with Control Rods only using RMSI and GP-5 Appendix 2 concurrently until core Thermal Limits are less than 1.000. Additionally the SRO determines compensatory actions (LCO action).

(LOJPM6760) Determine Offgas Effluent Activity Release Rate

Per GP-5, Steady State Operations, calculate the average offgas pre-treatment radioactivity release rate. Activity rates will be determined to exceed thresholds and subsequent results will be such that Tech Specs will require a coolant sample. The coolant sample will exceed the Dose Equivalent Iodine Value requiring a LCO entry.

(LOJPM3097) ERP Classification and Reporting (Time Critical)

Degraded plant conditions will require the candidates to determine that the thresholds have been met for the declaration of a General Emergency. They will then complete the notification form including the Protective Action Recommendation (PAR).

Facility: Limerick 1 & 2 Scenario No.: SEG-2007E Rev 2 Op-Test No.: 1

Examiners: ____________ Operators: ____________

Initial Conditions:

Unit 1 is at 90% power. Unit 2 is at 100% power.

Turnover:

Withdraw Control Rods per ReMA and raise Recirc flow to return to 100% Reactor power

'1B' EHC Pump is blocked for pump replacement. Repairs are scheduled to be complete in two (2) hours. Maintain 100% power and support PMT of '1B' EHC pump when it is returned to operations.

Event No.	Malfunction Number	Event Type*	Event Description
1.	MRD016E (46-15)	C-RO TS-SRO	Control Rod (46-15) uncoupled **(Abnormal)**
2.	MAD149E	R-RO C-PRO	'1N' SRV fails open / closes when Rx power lowered to <90 % **(Abnormal)**
3.	MCS183C	C-RO C-PRO TS-SRO	Inadvertent Div 4 LOCA Signal **(Abnormal)**
4.	MRH172D	C-PRO TS-SRO	'1D' RHR Pump fails to auto start **(Malfunction)**
5.	MRR441	C-PRO	Small leak in Drywell **(Abnormal)**
6.	MMS067	M-ALL	Large Steam Leak in Drywell
7.	MRH600B	C-PRO	'1B' RHR Pump Trips **(Malfunction)**
8.	MRH573A	C-PRO	HV-51-1F024A Thermal Overload condition **(Malfunction)**

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

SCENARIO EVENT AND EVALUATION SUMMARY:

Event One: As the crew assumes responsibility for the shift, they are directed to withdraw Control Rods and adjust Recirc flow per the ReMA and return the reactor to 100% power. During the power ascension and as Control Rod 46-15 is being withdrawn, a "Rod Overtravel" alarm will annunciate, and Control Rod 46-15 will become uncoupled.

Evaluation: To evaluate the crews ability to raise Recirc flow and withdraw Control Rods per a ReMA, and when the 'Rod Overtravel' alarms, identify an uncouple Control Rod, implement ON-104, Control Rod Problems, and OT-104, Unexpected/ Unexplained Positive or Negative Reactivity Insertion, isolate the Control Rod (46-15) and evaluate Tech Specs for the INOP Control Rod.

Event Two: When the Control Rod has been fully inserted as directed by the ON procedure, and Tech Specs are evaluated for the Control Rod being INOP, the '1N' SRV will inadvertently open. The SRV will close when reactor power is lowered to ~87%.

Evaluation: To evaluate the crews' response to an open SRV. The crew will enter and execute OT-114, Inadvertent Opening Of A Relief Valve. The crew will identify that when reducing reactor power to ~87%, per the OT procedure, the SRV closes. The crew will then evaluate the plants response to the now closed SRV.

Event Three: After the SRV is closed, an inadvertent Division 4 LOCA signal will occur.

Evaluation: To evaluate the crews response to use MCR instrumentation and determine the Division 4 LOCA signal is inadvertent. The crew will identify the cause of the Division 4 LOCA signal to be an excess flow check valve actuation. The crew will recognize a reactor power increase and that HPCI is running and injecting into the RPV, and isolate the HPCI system. The crew will also identify the D14 EDG running unloaded, and a Core Spray Pump running. The CRS will determine the required Tech Spec Actions for systems that are unavailable due to the LOCA signal.

Event Four: During the inadvertent Division 4 LOCA signal, the '1D' RHR Pump will trip.

Evaluation: To evaluate the crews response to the tripped '1D' RHR Pump, that should be running due to the DIV 4 LOCA signal. Identify the effect due to the loss of the '1D' RHR Pump, and Tech Spec requirements for an RHR system being OOS.

SCENARIO EVENT AND EVALUATION SUMMARY cont'd:

Event Five: and Six: When actions for the inadvertent LOCA signal are complete, a gradually increasing coolant leak in the drywell will develop requiring the plant to shutdown, which will eventually require a T-112, Emergency Blowdown be performed.

Evaluation: To evaluate the crews response to increasing drywell pressure and temperature, including the initial execution of OT-101, Drywell High Pressure, and eventually a plant shutdown and execution of T-101, RPV Control, and T-102, Primary Containment Control.

Event Seven: The '1B' RHR Pump, which was placed in service when the SRV opened, trips on a time delay after the plant is shutdown, and when the crew transitions to the 'A' Loop of RHR for containment cooling, the HV-51-1F024A, 'A' Loop RHR Test Return Valve, will trip on a Thermal Overload condition.

Evaluation: To evaluate the crew's response to the tripped '1B' RHR Pump and to transition the '1A' RHR Pump when directed to re-direct from Suppression Pool Spray to Drywell Spray. As the crew re-directs to 'A' RHR for Drywell Spray they will manually open the HV-51-1F024A valve when a thermal overload trip is identified.

Termination Point: The scenario may be terminated when Drywell Spray is in service, control rods inserted, reactor level is restored and a(n) ALERT (FA1) has been declared and the EP plan has been implemented.

Facility: Limerick 1 & 2 Scenario No.: SEG-5006E Rev 0 Op-Test No.: 1 .

Examiners: ____________ Operators: ____________

Initial Conditions:
Unit 1 is at 100 % power. Unit 2 is at 100 % power.

Turnover:
Maintain 100% Reactor power
Place '1C' SBLC Pump in Automatic Injection Mode per S48.1.A, Standby Liquid Control System Set-Up For Normal Operation, step 4.7 in preparation of a 1B SLC System Outage Window.

Event No.	Malfunction Number	Event Type*	Event Description
1.	MSL198B C41-S1C	N-PRO TS-SRO	Align '1C' SLC Pump for automatic operation
2.	MPR020C	C-RO	#3 APRM fails upscale **(Malfunction)**
3.	MRR430A MRD024	R-RO C-PRO TS-SRO	'1A' Reactor Recirc Pump shaft seizure resulting in Recirc Pump trip **(Abnormal)**
4.	MFH563C	C-PRO	Low Pressure FWH Level Transient **(Abnormal)**
5.	MFH116C MFH016C MPR003A	C-RO	'6C' FWH Isolation **(Abnormal)** Core Power Oscillations
6.	MFW252A MRR440A	M-ALL	LOCA Inside Containment **(T-111)**
7.	MRC457B	C-PRO	RCIC controller in AUTO failure **(Malfunction)**
8.	MAD148D	C-PRO	'1M' SRV fails to open **(Malfunction)**
*	**(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor**		

SCENARIO EVENT AND EVALUATION SUMMARY:

Event One: Shortly after the crew has assumed responsibility, they will be required to align the '1C' SLC Pump for automatic operation, using S48.1.A, Standby Liquid Control System Set-Up For Normal Operation, and remove the '1B' SLC Pump from service.

Evaluation: Evaluate the crew's ability to perform the procedure and place equipment in service and to evaluate the CRSs ability to apply Tech Spec 3.1.5. for the SLC Pump being removed from service.

Event Two: Shortly after the evolution of placing the '1C' SLC Pump in standby service, APRM #3 will fail upscale.

Evaluation: Evaluate the crew's response to the plant for the failed APRM, reference Tech Spec 3.3.1 and place the inoperable APRM in the BYPASS position.

Event Three: After the failed APRM issue is resolved, the '1A' Recirc Pump shaft will seize resulting in a Recirc Pump trip and reduction in core flow and reactor power.

Evaluation: To evaluate the crew's ability to address the sudden change in reactor power by entering and executing OT-104, Unexpected/ Unexplained Positive or Negative Reactivity Insertion, and OT-112, Unexpected/ Unexplained Change In Core Flow, for the tripped Recirc Pump. The crew will isolate the failed Recirc Pump and insert Control Rods to exit the Restricted Region of the Power/Flow Map. The crew will also execute GP-5 to stabilize the plant.

Event Four: and Five: As the crew is recovering from the tripped Recirc Pump and attempting to exit the Restricted Region of the Power/flow Map a low pressure FWH level transient will occur requiring re-entry into OT-104. As the crew addresses the transient another FWH level control problem will result in a 6^{th} FWH isolation causing a larger positive reactivity addition and subsequent Thermal Hydraulic Instabilities (THI) .

Evaluation: To evaluate the crew's ability to diagnose the positive reactivity addition from the loss of FWH while in the Restricted Region of the Power/Flow Map and to detect/suppress core THI, by monitoring LPRM, APRM and period meters to detect indications for signs of core THI. The crew will shutdown the plant due to the Thermal Hydraulic Instabilities.

SCENARIO EVENT AND EVALUATION SUMMARY cont'd

Event Six: After the reactor is shutdown, a Feedwater line break will occur resulting in a loss of all feedwater to the RPV.

Evaluation: To evaluate the crew's ability to take appropriate actions to control RPV level using T-101, RPV Control, and eventually T-111, Level Restoration/Steam Cooling. The crew will start ECCS Pumps in preparation to maintain RPV level following an RPV Emergency Blowdown. Also to maintain Containment parameters, enter T-102, Primary Containment Control and, as RPV level drops due to the loss of high pressure feed, enter T-112, Emergency Blowdown to assure adequate core cooling with the low pressure injection systems.

Event Seven: During reactor RPV level restoration the RCIC flow controller will fail in automatic.

Evaluation: To evaluate the PROs response to the RCIC system failure and diagnose that RCIC is available only when manual control is taken.

Event Eight: As RPV level decreases to -161" the crew performs the T-112, Emergency Blowdown allowing low pressure ECCS systems to maintain RPV level. To accomplish the blowdown, five ADS valves are selected to be opened, however the '1M' SRV will fail to open from the handswitch, and require an alternate SRV be opened to ensure 5 SRVs are opened.

Evaluation: Evaluate the crew's ability to identify failure of the '1M' SRV to open, and open a non-ADS valve to ensure 5 SRV's are open.

Termination Point: The scenario may be terminated when the Emergency Blowdown is complete, RPV level is restored to normal band with ECCS systems and Containment Spray is in service and a(n) Alert (FA1) has been declared with the Emergency Plan implemented.

Appendix D **Scenario Outline** **Form ES-D-1**

Facility: Limerick 1 & 2 Scenario No.: SEG-7016E Rev 002 Op-Test No.: 1

Examiners: ______ Operators: ______

Initial Conditions:

Unit 1 is at 100% power. Unit 2 is at 100% power.

Turnover:

None

Event No.	Malfunction Number	Event Type*	Event Description
1.	MFW001	R-RO C-PRO TS-SRO	HWC System Failure **(Abnormal)**
2.	VIC104C8 MFW250A MFW246A	C-RO C-PRO	‘1C’ RFP Min Flow Valve fails open **(Abnormal)**
3.	MCW481A MCW486B	C-PRO	‘1A’ TECW Pump trip ‘1B’ TECW fails to auto start **(Abnormal)**
4.	MVI234C	C-RO TS- SRO	Reactor High Pressure Transmitter Failure **(Abnormal)**
5.	MRP029D MSL559 MRP407C	M-ALL	ATWS The ATWS is mitigated by the insertion of control rods via T-215. **(Major)**
6.	MRD024	C-RO	RDCS Inoperative after reactor shutdown **(Malfunction)**
7.	MMT002 MEH108	C-PRO C-RO	Turbine Trip / Bypass Valves Fail Closed **(Malfunction)**
8.	MRSW600A MRSW600C	C-PRO	‘0A’ or ‘0C’ RHRSW Pump Trips **(Malfunction)**

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

SCENARIO EVENT AND EVALUATION SUMMARY:

Event One: Shortly after the crew has taken responsibility of the shift, the HWC system will fail such that the Hydrogen Flow Controller output will rise and the PLC will lock up. The MSL Rad Monitor alarms will be received due to excess Hydrogen injection.

Evaluation: Evaluate the crews' response to the HWC malfunction including actions taken per ON-102, Air Ejector Discharge Or Main Steam Line High Radiation and T-103, Secondary Containment Control. The crew will reduce power and will trip the HWC System as well as reference Tech Specs due to excessive H2 injection.

Event Two: After the crew has taken Tech Spec actions for HWC, the '1C' RFP Min Flow Valve will fail open causing '1C' RFP injection rate to lower and the '1A' and '1B' RFPs to increase injection to the vessel.

Evaluation: Evaluate the crew's response to the '1C' RFP Min Flow Valve failing open and to secure the '1C' RFP, and isolate the minimum flow valve. The crew will execute OT-104, Unexpected/ Unexplained Positive or Negative Reactivity Insertion and reduce total Feedwater less than 13 Mlb/hr per GP-5.

Event Three: After the '1C' RFP is secured and reactor power lowered, the in-service '1A' TECW Pump will inadvertently trip, with the '1B' TEWC Pump failing to auto start.

Evaluation: Evaluate the crew's response to identify the TECW Pump failure, enter and execute ON-117, Loss of TECW, manually start the '1B' TECW Pump, and monitor Condensate Pump and Air Compressor temperatures.

Event Four: and Five: After the crew has started the '1B' TECW Pump a RPS reactor pressure transmitter fails upscale. The failed RPS instrument will result in a failed RPS channel. As a result of the RPS channel failure the plant will scram. On the scram the Control Rods will fail to insert, resulting in an ATWS.

Evaluation: Evaluate the crews' ability to address the failed pressure transmitter and implement OT-117, RPS Failures, and respond to a failure of the reactor to scram, and subsequent ATWS. The crew will be evaluated on their ability to implement T-101, RPV Control and T-117, Level/Power Control, and the initial level lowering per T-270, Terminate And Prevent Injection Into The RPV. Complicating the event, the SLC discharge line will rupture in the DW.

SCENARIO EVENT AND EVALUATION SUMMARY: cont'd

Event Six: After the Mode Switch is taken to shutdown, RDCS will become inoperative and prevent manual insertion of control rods.

Evaluation: Evaluate the crew's ability to recognize the failure of RDCS and call out for floor assistance in resetting RDCS in the AER. Once RDCS is reset, the crew is expected to manually insert control rods to help mitigate the ATWS condition.

Event Seven: After level is lowered and pressure has been stabilized, the Main Turbine will begin to experience vibration caused by bearing failure. The excess vibration will eventually cause a MT trip. Along with the Turbine trip, the Bypass Valves will fail to maintain RPV pressure, resulting in a rise in Suppression Pool temperature.

Evaluation: Evaluate the crews' ability to respond to a Main Turbine trip while maintaining reactor pressure control. The crew will perform a second lowering of reactor level per T-270, to less than –161" and maintain level between –186" to –161" as Suppression Pool temperature exceeds 110 deg F.

Event Eight: Also, after the Main Turbine Trip, the first started '0A' Loop RHRSW Pump will fail to start from the MCR.

Evaluation: To identify the trip of the lead RHRSW Pump and place the alternate pump in service in the '0A' RHRSW loop.

Termination Criteria: The scenario is terminated when RPV level has been maintained –186" to –161", Control Rods are inserted per T-215, De-energizing Of Scram Solenoids and a(n) Alert (MS3) has been declared, and the Emergency Plan has been implemented.

Facility: Limerick 1 & 2 Scenario No.: SEG-3005E Rev 0 Op-Test No.: 1 .

Examiners: ______________ Operators: ______________

SPARE

Initial Conditions:
Unit 1 is at 5.0 % power. Unit 2 is at 100% power.

Turnover:
D12 D/G is running parallel to 201 Safeguard Bus following 201-D12 Breaker compartment maintenance to repair a damaged cell switch linkage arm. D12 D/G has been running loaded for 1 hour 45 minutes of the required 2 hours per S92.2.N step 4.6.4.

Planned Evolutions :

When the 2 hour loaded run is complete, secure D12 D/G per S92.2.N beginning with step 4.6.5 .

Event No.	Malfunction Number	Event Type*	Event Description
1.	MESW600B	C-PRO TS-SRO	'0B' ESW Pump trip **(Malfunction)**
2.	MCU002A MCU002B MCU193	C-PRO TS-SRO	RWCU Isolation on differential flow **(Abnormal)**
3.	MRP029A MVI232B	I-RO TS-SRO	RPS level instrument fails downscale with failure to half-scram **(Abnormal)**
4.	MED280A	C-RO C-PRO TS-SRO	Trip of 1AY160 **(Abnormal)**
5.	MFW251B HS06-108A	C-RO	S/U Level Control Valve and HV-06-108A fail closed **(Malfunction)**
6.	MHP445	M-ALL	T-103 Steam Leak in HPCI
7.	MHP446A MHP446B	C-PRO	HPCI Steam Isolation Valves fail to close **(Malfunction)**
8.	VIC104B8 MFW245A	C-RO	'1B' Reactor Feedpump vibration / trip **(Abnormal)**
9	MRC460	C-PRO	RCIC injection valve fails to open in AUTO **(Malfunction)**

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

VI. SCENARIO EVENT AND EVALUATION SUMMARY:

Event One: At shift turnover, the crew will be directed to shutdown D12 D/G following an additional 15 minutes of running. The turnover directs the EDG be shutdown using S92.2.N, however before the diesel can be shutdown, the '0B' ESW Pump trips.

Evaluation: The crew is expected to identify the loss of cooling water to the operating D/G and start the alternate '0D' ESW Pump, or perform an emergency shutdown of the EDG. The SRO will identify the Tech Spec requirements for the loss of the '0B' ESW Pump.

Event Two: After the '0B' ESW Pump trips, and '0D' ESW Pump is started, and Tech Spec 3.7.1.2 referenced for the loss of an ESW Pump, a RWCU differential (delta) flow annunciator alarms. The RWCU delta flow is designed to isolate the RWCU system after a time delay.

Evaluation: Evaluate the crew's ability to diagnose that RWCU failed to isolate on delta flow. RWCU water is flowing into the RECW system causing a high rad condition in RECW. The crew will identify the delta flow and take action to manually isolate the RWCU system.

Event Three: After the RWCU system is isolated, Tech Specs referenced for the loss of conductivity monitoring, and chemistry notifications made, an 'A1' RPS RPV level instrument will fail downscale with a failure of an 'A' RPS half-scram signal being initiated.

Evaluation: Evaluate crew's ability to recognize the failure to half-scram on loss of the RPS reactor level instrument, enter OT-117 and insert a manual half-scram on the 'A' RPS channel. The SRO will reference Tech Spec 3.3.1 for the RPS channel.

Event Four: and Five: After the 'A' half-scram has been inserted, and Tech Spec 3.3.1 referenced, a loss of the 1A RPS UPS power supply, panel 1A-Y160 will occur. The power supply will result in a loss of RECW, DWCW, and PCIG, along with a Reactor HVAC isolation. A loss of 1A-Y160 at a power level when the Main Turbine is tripped will result in both Recirculation Pumps tripping. With no Recirc Pumps in operation the operators will scram the plant and enter T-100, Scram / Scram Recovery. During the scram the RO will identify a failure of the Reactor S/U Level Controller, and attempt to find alternate means to control RPV level.

Evaluation: Evaluate the crew's ability to diagnose the power loss and execute E-1AY160, identify both Recirc Pumps have tripped, scram the reactor, enter T-100, and bypass and restore the RECW, DWCW and PCIG systems. The RO will identify the failed S/U level controller while controlling reactor level, and maintain RPV level using the '1B' RFP.

SCENARIO EVENT AND EVALUATION SUMMARY: cont'd

Event Six and Seven: Once the reactor is shutdown form the tripped 1A-Y160 panel, and T-100 steps being executed, a steam leak will occur in the HPCI Room. Both the Inboard and Outboard HPCI steam lines will fail to isolate, and the steam leak will propagate into a second area.

Evaluation: Evaluate the crew's response to take actions per T-100 for the scram, and transition to T-101 as RVP level drops from the loss of the S/U level controller. Also, evaluate the crew's response during execution into T-103, Secondary Containment Control, due to the elevated HPCI room temperatures. Due to the unisolatable HPCI steam leak, and as the second area reaches MSO temperatures, a T-112, Emergency Blowdown will be performed.

Event Eight and Nine: After the reactor is scrammed, and with the S/U level controller failing, the RO will attempt to control RPV level with the standby '1B' RFP. The '1B' RFP will vibrate and eventually trip, and when the crew transitions to RCIC, the RCIC injection valve will fail to open in Auto.

Evaluation: To evaluate the crew's response to the high vibration condition on the '1B' RFPT, including a plan to remove the '1B' RFP from service. Also, evaluate the RO's ability to recognize the failures and control RPV level manually with a standby RFP or with the RCIC system.

Termination Criteria: This scenario may be terminated after two rooms have temperatures above MSO and when 5 SRVs opened to depressurize the reactor while reactor level is restored between 12.5" to 54" and a(n) SITE AREA EMERGENCY (FS1), has been declared and the Emergency Plan has been implemented.