Exelon Generation

10 CFR 55.40

LG-21-005

January 28, 2021

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission, Region I
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: Submittal of Initial License Operator Training Program Exam Outlines

Reference: NRC Letter "Limerick Generating Station Units 1 and 2 - Senior Reactor and Reactor Operator Initial License Examinations," dated September 24, 2020

As requested in the reference, the following NUREG-1021 required documents are being provided to Mr. Joseph DeMarshall regarding the Initial Operator Licensing Exams to be administered beginning on July 12, 2021 at Limerick Generating Station.

ES-201-2	Exam Outline Quality Checklist and Supporting Information
ES-201-3	Examination Security Agreement and Supporting Information
ES-301-1	Administrative Topics Outline and Supporting Information
ES-301-2	Control Room/In-plant Outline and Supporting Information
ES-301-5	Transient and Event Checklists and Supporting Information
ES-D-1	Scenario Outlines and Supporting Information

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,

Digitally signed by
Sturniolo, Frank
Date: 2021.01.25 14:03:03
-05'00'

Frank Sturniolo
Vice President – Limerick Generating Station
Exelon Generation Co., LLC

Enclosure 1:	ES-201-2 Exam Outline Quality Checklist and Supporting Information
Enclosure 2:	ES-201-3 Examination Security Agreement and Supporting Information
Enclosure 3:	ES-301-1 Administrative Topics Outline and Supporting Information
Enclosure 4:	ES-301-2 Control Room/In-plant Outline and Supporting Information
Enclosure 5:	ES-301-5 Transient and Event Checklists and Supporting Information
Enclosure 6	ES-D-1 Scenario Outlines and Supporting Information

cc:	J. DeMarshall, USNRC, Region I	(w/encl)
	D. Jackson, USNRC, Region I	(w/o encl)
	USNRC Senior Resident Inspector, LGS	(w/o encl)

Facility: Limerick 20-1 Date of Examination: 07-12-21

Item	Task Description	Initials a	Initials b*	Initials c**
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model in accordance with ES-401 or ES-401N.	B	JRW	JD
	b. Assess whether the outline was systematically and randomly prepared in accordance with Section D.1 of ES-401 or ES-401N and whether all K/A categories are appropriately sampled.	B	JRW	JD
	c. Assess whether the outline overemphasizes any systems, evolutions, or generic topics.	B	JRW	JD
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	B	JRW	JD
2. SIMULATOR	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, technical specifications, and major transients.	B	JRW	JD
	b. Assess whether there are enough scenario sets (and spares) to test the projected number and mix of applicants in accordance with the expected crew composition and rotation schedule without compromising exam integrity, and ensure that each applicant can be tested using at least one new or significantly modified scenario, that no scenarios are duplicated from the applicants' audit test(s), and that scenarios will not be repeated on subsequent days.	B	JRW	JD
	c. To the extent possible, assess whether the outline(s) conforms with the qualitative and quantitative criteria specified on Form ES-301-4 and described in Appendix D and in Section D.5, "Specific Instructions for the 'Simulator Operating Test,'" of ES-301 (including overlap).	B	JRW	JD
3. WALKTHROUGH	a. Verify that the systems walkthrough outline meets the criteria specified on Form ES-301-2: (1) The outline(s) contains the required number of control room and in-plant tasks distributed among the safety functions as specified on the form. ✓ (2) Task repetition from the last two NRC examinations is within the limits specified on the form. ✓ (3) No tasks are duplicated from the applicant's audit test(s). ✓ (4) The number of new or modified tasks meets or exceeds the minimums specified on the form. ✓ (5) The number of alternate-path, low-power, emergency, and radiologically controlled area tasks meets the criteria on the form. ✓	B	JRW	JD
	b. Verify that the administrative outline meets the criteria specified on Form ES-301-1: (1) The tasks are distributed among the topics as specified on the form. ✓ (2) At least one task is new or significantly modified. ✓ (3) No more than one task is repeated from the last two NRC licensing examinations. ✓	B	JRW	JD
	c. Determine whether there are enough different outlines to test the projected number and mix of applicants and ensure that no items are duplicated on subsequent days.	B	JRW	JD
4. GENERAL	a. Assess whether plant-specific priorities (including probabilistic risk assessment and individual plant examination insights) are covered in the appropriate exam sections.	B	JRW	JD
	b. Assess whether the 10 CFR 55.41, 55.43, and 55.45 sampling is appropriate.	B	JRW	JD
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	B	JRW	JD
	d. Check for duplication and overlap among exam sections and the last two NRC exams.	B	JRW	JD
	e. Check the entire exam for balance of coverage.	B	JRW	JD
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	B	JRW	JD

		Printed Name/Signature	Date
a.	Author	Christopher J. Bihl / C.J. Bihl	1/25/21
b.	Facility Reviewer (*)	Jeffrey R. Weaver / [signature]	1/25/21
c.	NRC Chief Examiner (#)	Joseph DeMarshall / Joseph DeMarshall	2-5-21
d.	NRC Supervisor	Donald E Jackson / [signature]	2/8/21

* Not applicable for NRC-prepared examination outlines.

\# The independent NRC reviewer initials items in column "c"; the chief examiner's concurrence is required.

Facility: Limerick 20-1 Date of Examination: 07/12/21

Examination Level: RO ☒ SRO ☐ Operating Test Number: 1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, N	Temperature Effects on Reactor Level Instrumentation (G2.1.20 – Ability to interpret and execute procedure steps, IMP RO 4.6 SRO 4.6) (LOJPM6756)
Conduct of Operations	R, N	Determine Drywell Venting Parameters (G2.1.25 – Ability to interpret reference materials, such as graphs, curves, tables, etc. IMP RO 3.9 / SRO 4.2) (LOJPM6755)
Equipment Control	R, D	Review Drywell Floor Drain Sump/Equipment Drain Tank Logs and Determine Compliance With TS 3.4.3.2 (2.2.12 – Knowledge of Surveillance Procedures, IMP RO 3.7 SRO 4.1) (LOJPM6708)
Radiation Control	R, D	Area Rad Monitor(s) Fail Downscale (G2.3.15 – Knowledge of Rad. Monitoring Systems such as fixed rad monitors and alarms, portable survey inst., personnel monitoring equip. etc, IMP RO 2.9 SRO 3.1)(LOJPM6718)
Emergency Plan		N/A

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

Facility: Limerick 20-1	Date of Examination: 07/12/21
Examination Level: RO ☐ SRO ☒	Operating Test Number: 1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R, N	Determination of Adequate Shift Staffing (G2.1.5 – Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations etc, IMP RO 2.9 SRO 3.9)(LOJPM6757)
Conduct of Operations	R, D	Determine Acceptability Of Installing Fuel Pool Gates (G2.1.40 – Knowledge of Refueling administrative requirements, IMP RO 2.8 SRO 3.9)(LOJPM6763)
Equipment Control	R, D	Review Drywell Floor Drain Sump/Equipment Drain Tank Logs and Determine Compliance With TS 3.4.3.2 (2.2.12 – Knowledge of Surveillance Procedures, IMP RO 3.7 SRO 4.1)(LOJPM6708)
Radiation Control	R, D	Area Rad Monitor(s) Fail Downscale (G2.3.15 – Knowledge of Rad. Monitoring Systems such as fixed rad monitors and alarms, portable survey inst., personnel monitoring equip. etc, IMP RO 2.9 SRO 3.1)(LOJPM6718)
Emergency Plan	R, N	ERP Classification and Reporting (Time Critical) (G2.4.41 – Knowledge of the emergency action level thresholds and classifications. IMP RO 2.9 SRO 4.6 (LOJPM3132)

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

Facility: Limerick NRC 20-1	Date of Examination: 07/12/21
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐	Operating Test Number: 1

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

System/JPM Title	Type Code*	Safety Function
a. Start a Recirc Pump (System 202001- Recirculation System, A3.02, Imp 3.1)(LOJPM3121)	A,D,S	1
b. RCIC Manual Slow Start (System 217000 – Reactor Core Isolation Cooling System, A4.01, Imp 3.7) (LOJPM3015)	D, S	2
c. Roll the Main Turbine (System 241000 – Reactor/Turbine Pressure Regulating System, A4.19, Imp 3.5)(LOJPM3029)	A,N,S	3
d. Shutdown Cooling Flow Adjustment (System 205000 – Shutdown Cooling System, K1.15, Imp 3.5)(LOJPM3515)	A, D, L, S	4
e. Vent Containment Using HCVS (System 223001 –Primary Containment System and Aux, A4.07, Imp 4.2) (LOJPM3070)	N, S, EN, L	5
f. Scram Channel A1 and A2 Functional Test (System 212000 – Reactor Protection System, K4.05, Imp 3.4) (LOJPM3031)	D,EN,S	7
g. Supply RECW To The Drywell Coolers (System 40000 – Component Cooling Water, A2.01, Imp 3.3)(LOJPM3028)	D, L, S	8
h. Standby Gas Treatment Manual Startup (System 261000 – Standby Gas Treatment, A4.02, Imp 3.1)(LOJPM3528)	A,D, EN, S	9

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U

System/JPM Title	Type Code*	Safety Function
i. T-242 Defeat of HPCI/RCIC Test Return and Injection Valve Isolation Logic (System 217000 – Reactor Core Isolation Cooling, A2.01, Imp 3.8)(LOJPM2275)	N, E, L, R	4
j. Manual Isolation and Vent of the Scram Air Header (System 212000 – Reactor Protection System, A4.17, Imp 4.1)(LOJPM2210)	D, E, R	7
k. T-244 Diesel Driven Fire Pump Manual Start (System 286000 – Fire Protection System, A2.08, Imp 3.2)(LOJPM2232)	D, E	8

* 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, and in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for **R** /SRO-I/SRO-U	
(A)lternate path	**4–6**/4–6 /2–3	4
(C)ontrol room		
(D)irect from bank	**≤ 9**/≤ 8/≤ 4	8
(E)mergency or abnormal in-plant	**≥ 1**/≥ 1/≥ 1	3
(EN)gineered safety feature	**≥ 1**/≥ 1/≥ 1 (control room system)	3
(L)ow-Power/Shutdown	**≥ 1**/≥ 1/≥ 1	4
(N)ew or (M)odified from bank including 1(A)	**≥ 2**/≥ 2/≥ 1	3/1
(P)revious 2 exams	**≤ 3**/≤ 3/≤ 2 (randomly selected)	0
(R)CA	**≥ 1**/≥ 1/≥ 1	2
(S)imulator		8

Facility: Limerick NRC 20-1	Date of Examination: 07/12/21
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐	Operating Test Number: 1

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

System/JPM Title	Type Code*	Safety Function
a. Start a Recirc Pump (System 202001- Recirculation System, A3.02, Imp 3.1)(LOJPM3121)	A,D,S	1
b. RCIC Manual Slow Start (System 217000 – Reactor Core Isolation Cooling System, A4.01, Imp 3.7) (LOJPM3015)	D, S	2
c. Roll the Main Turbine (System 241000 – Reactor/Turbine Pressure Regulating System, A4.19, Imp 3.5)(LOJPM3029)	A,N,S	3
d. Shutdown Cooling Flow Adjustment (System 205000 – Shutdown Cooling System, K1.15, Imp 3.5)(LOJPM3515)	A, D, L, S	4
e. Vent Containment Using HCVS (System 223001 –Primary Containment System and Aux, A4.07, Imp 4.2) (LOJPM3070)	N, S, EN, L	5
f. Scram Channel A1 and A2 Functional Test (System 221000 – Reactor Protection System, K4.05, Imp 3.4) (LOJPM3031)	D,EN,S	7
h. Standby Gas Treatment Manual Startup (System 261000 – Standby Gas Treatment, A4.02, Imp 3.1)(LOJPM3528)	A, D, EN, S	9
In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U		
i. T-242 Defeat of HPCI/RCIC Test Return and Injection Valve Isolation Logic (System 217000 – Reactor Core Isolation Cooling, A2.01, Imp 3.8)(LOJPM2275)	N, E, L, R	4
j. Manual Isolation and Vent of the Scram Air Header (System 212000 – Reactor Protection System, A4.17, Imp 4.1)(LOJPM2210)	D, E, R	7
k. T-244 Diesel Driven Fire Pump Manual Start (System 286000 – Fire Protection System, A2.08, Imp 3.2)(LOJPM2232)	D, E	8

* 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, and in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for R /**SRO-I**/SRO-U	
(A)lternate path	4–6/**4–6** /2–3	4
(C)ontrol room		
(D)irect from bank	≤ 9/**≤ 8**/≤ 4	7
(E)mergency or abnormal in-plant	≥ 1/**≥ 1**/≥ 1	3
(EN)gineered safety feature	≥ 1/**≥ 1**/≥ 1 (control room system)	3
(L)ow-Power/Shutdown	≥ 1/**≥ 1**/≥ 1	3
(N)ew or (M)odified from bank including 1(A)	≥ 2/**≥ 2**/≥ 1	3/1
(P)revious 2 exams	≤ 3/**≤ 3**/≤ 2 (randomly selected)	0
(R)CA	≥ 1/**≥ 1**/≥ 1	3
(S)imulator		7

Facility: Limerick NRC 20-1 Date of Examination: 07/12/21

Exam Level: RO ☐ SRO-I ☐ SRO-U ☒ Operating Test Number: 1

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

System/JPM Title	Type Code*	Safety Function
c. Roll the Main Turbine (System 241000 – Reactor/Turbine Pressure Regulating System, A4.19, Imp 3.5)(LOJPM3029)	A,N,S	3
e. Vent Containment Using HCVS (System 223001 –Primary Containment System and Aux, A4.07, Imp 4.2) (LOJPM3070)	N, S, EN, L	5
h. Standby Gas Treatment Manual Startup (System 261000 – Standby Gas Treatment, A4.02, Imp 3.1)(LOJPM3528)	A, D, EN, S	9
In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U		
i. T-242 Defeat of HPCI/RCIC Test Return and Injection Valve Isolation Logic (System 217000 – Reactor Core Isolation Cooling, A2.01, Imp 3.8)(LOJPM2275)	N, E, L, R	4
j. Manual Isolation and Vent of the Scram Air Header (System 212000 – Reactor Protection System, A4.17, Imp 4.1)(LOJPM2210)	D, E, R	7

* 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, and in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for R /SRO-I/**SRO-U**	
(A)lternate path	4–6/4–6 /**2–3**	2
(C)ontrol room		
(D)irect from bank	≤ 9/≤ 8/**≤ 4**	2
(E)mergency or abnormal in-plant	≥ 1/≥ 1/**≥ 1**	2
(EN)gineered safety feature	≥ 1/≥ 1/**≥ 1** (control room system)	2
(L)ow-Power/Shutdown	≥ 1/≥ 1/**≥ 1**	2
(N)ew or (M)odified from bank including 1(A)	≥ 2/≥ 2/**≥ 1**	3/1
(P)revious 2 exams	≤ 3/≤ 3/**≤ 2** (randomly selected)	0
(R)CA	≥ 1/≥ 1/**≥ 1**	2
(S)imulator		3

Facility: Limerick 1 & 2 Scenario No.: SEG-2158E Rev 2 Op-Test No.: 1 .

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions:

Unit 1 is at 5.0 % power with a GP-2, Normal Plant Startup in progress.
Unit 2 is at 100% power.

Turnover:

GP-2 is complete up to step 3.4.31 ready to raise Rx press to 960 psig using DEHC, and withdraw control rods to maintain Main Turbine BPVs controlling reactor pressure.

Critical Tasks:

OT-200.1 Insert Control Rods with Alternate Rod Insertion (ARI)
T-102.9 Conduct an Emergency Blowdown due to High Drywell Press

Per NUREG-1021, App. D, If an operator or crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis for a CT identified in the post scenario review.

Event No.	Malfunction Number	Event Type*	Event Description
1	N/A	R-ATC N-BOP N-SRO	Continue raising reactor power and RPV pressure
2	MRD016D	C-ATC C-SRO	Control Rod (42-43) fails stuck **(Abnormal)**
3	MPC257	C-BOP C-SRO TS	‘1D’ RHR Pump suction leak **(Abnormal)**
4	MED282A	C-BOP C-SRO TS	Loss of Div 1 DC **(Abnormal)**
5	MRR441	C-BOP C-SRO	Small coolant leak in Drywell **(Abnormal) (CT-1)**
6	MRP029C	C-ATC C-SRO	RPS ‘A’ fails to scram (ARI successful) **(Malfunction)**
7	MMS067	M-ALL	Steam leak in the Drywell
8	MPC476	C-BOP C-SRO	Downcomer break results in Suppression Pool pressure equalizing with Drywell pressure requiring blowdown on Pressure Suppression Curve **(Malfunction) (CT-2)**
9	MRH600B	C-BOP C-SRO	'1B' RHR Pump fails to start **(Malfunction)**

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

QUANTITATIVE ATTRIBUTES

A. ILT

Target Quantitative Attributes (Per Scenario; See ES-301 Section D.5.d)	
1. Malfunctions after EOP entry (1-2): Steam Leak in Drywell; Downcomer break; 1B RHR PP fails to start.	3
2. Abnormal events (2-4): Stuck Control Rod, 1D RHR PP Suction Leak, Loss of Div 1 DC, Small Coolant Leak in DW	4
3. Major transients (1–2): Steam Leak in DW	1
4. EOPs entered/requiring substantive actions (1–2): T-101, T-102	2
5. EOP contingencies requiring substantive actions (0-2): T-112	1
6. Critical tasks (2–3): Manually Scram Reactor; Perform Emergency Blowdown.	2

Simulator Scenario Summary

SEG-2158E

Initial Conditions:
Unit 1 is at 5.0% power with a GP-2, Normal Plant Startup in progress.
Unit 2 is at 100% power.

Turnover:
GP-2 is complete up to the point to raise RPV pressure to 960 psig using DEHC, and withdraw control rods to maintain Main Turbine BPVs controlling reactor pressure.

Event 1: When the crew takes the shift, the BOP will raise RPV pressure using DEHC, and continue raising power by withdrawing control rods.

Evaluation: The BOP will use GP-2 to raise RPV Pressure to 960 psig by raising DEHC Target Pressure with “SLOW RATE” (0.5 psig/sec) or GP-2 Attachment 15. The SRO should direct further rod withdrawal to continue power ascension.

Event 2: As the ATC withdraws control rods, one of the control rods will fail stuck.

Evaluation: During the rod withdrawal the ATC will determine that control rod 42-43 is not moving on demand. The crew will use normal operating procedures to attempt to move the stuck control rod. The crew should adjust drive water pressure to free the struck rod and put it in its proper position.

Event 3: Following the stuck control rod event, a leak will develop in the suction line for the ‘1D’ RHR Pump.

Evaluation: The crew should react to the indications and dispatch a floor operator the the location. Upon confirmation of the active leak into the RHR pump room, the crew should enter AOP SE-4, Flooding, and take action to mitigate/isolate the condition and the degradation of secondary containment.

Event 4: After the RHR leak has been isolated and Tech Specs referenced, a loss of Division I DC will occur

Evaluation: The crew is expected to diagnose a loss of Division 1 DC from the indications in the MCR. Once diagnosed the crew should enter AOP E-1FA and take actions a prescribed in the procedure, including placing the MCR HVAC into a Chlorine Isolation, Verify Unit Secondary Containment integrity, hold a crew brief on plant condition and contingencies.

Event 5: After the crew responds to the loss of DC, a small reactor coolant leak will occur in the Drywell.

Evaluation: The crew will recognize rising Drywell pressure and enter AOP OT-101. The crew will then determine that the Drywell pressure increase is the result of an unisolable coolant leak the crew will perform a rapid plant shutdown using GP-4.

Event 6: After the BOP isolates RWCU in response to the increase in Drywell pressure, the ATC will perform a manual scram but 'A' RPS will fail to de-energize.

Evaluation: The ATC will take the reactor mode switch to the shutdown position but 'A' RPS will fail to de-energize. The ATC will use OT-200 Appendix 1, Reactor Scram Hard Card, and manually activate RRCS (Alternate Rod Insertion), which will succeed in inserting the control rods. The crew will continue with post scram actions to stabilize the plant using EOP T-101.

Event 7: When the Reactor Mode Switch has been placed in Shutdown, the steam leak in the Drywell will increase requiring actions directed by T-102.

Evaluation: As Drywell pressure begins to rise rapidly the SRO will enter EOP T-102 at 1.68 psig Drywell pressure and direct actions to initiate suppression pool spray. HPCI will initiate and the ATC and BOP will have to coordinate what to do with this injection source. The SRO will evaluate/direct Drywell sprays when Drywell pressure exceeds 7.5 psig.

Event 8: As the Operator attempt to place Drywell Spray in service, a downcomer break will occur resulting in Suppression Pool pressure rising faster and exceeding the Pressure Suppression Curve.

Evaluation: The SRO will monitor T-102, Pressure Suppression Curve and once it is evident that the unsafe side of the curve cannot be avoided, enter EOP T-112, Emergency Depressurization, and direct an emergency blowdown of the RPV.

Event 9: When Suppression Pool pressure exceeds 7.5 psig with the permissive to spray the Drywell, the '1B' RHR Pump will trip.

Evaluation: With '1A' RHR not available without DIV 1 DC power the crew will utilize RHRSW to spray containment using the 'B' loop of RHR.

Termination: The scenario may be terminated when the reactor level stabilized within required band, emergency RPV depressurization has been completed and Containment Spray is in service per T-225.

Simulator Scenario Critical Tasks

<table>
<tr><th colspan="3">OT-200.1</th><th colspan="6">Insert Control Rods with Alternate Rod Insertion (ARI)</th></tr>
<tr><td colspan="9">K/A information obtained from NUREG-1123 Revision 3</td></tr>
<tr><td colspan="3"><u>Identifier</u></td><td colspan="2"><u>K/A</u></td><td colspan="2"><u>Importance</u></td><td rowspan="2"><u>Safety Function (for Systems)</u></td></tr>
<tr><td>System/EPE/APE</td><td>Number</td><td>Title</td><td>Number</td><td>Ability</td><td>RO</td><td>SRO</td></tr>
<tr><td>EPE</td><td>295037</td><td>Redundant Reactivity Control System</td><td>EA1.02</td><td>Operate the Redundant Reactivity Control System</td><td>3.8</td><td>4.0</td><td>7</td></tr>
<tr><td>Safety Significance</td><td colspan="7">Correct performance of this action prevents a significant reduction of safety margin beyond that irreparably introduced by the scenario. Failure to perform this task correctly also represents a degradation of a fission product barrier.</td></tr>
<tr><td>Initiating Cue</td><td colspan="7">The Reactor Protection System is activated either automatically by exceeding an RPS setpoint, or manually, and the reactor control rods fail to insert to ensure reactor shutdown under all conditions.</td></tr>
<tr><td rowspan="2">Measurable Performance Standard</td><td colspan="2">Expected action</td><td colspan="5">Initiate manual RRCS activation.</td></tr>
<tr><td colspan="2">Safety-significant boundary condition</td><td colspan="5">Prior to requirement for emergency blowdown of RPV.</td></tr>
<tr><td>Performance Feedback</td><td colspan="7">The scram air header is depressurized, and control rods insert to ensure the reactor is shutdown under all conditions.</td></tr>
<tr><td>Evaluation</td><td colspan="7">SAT / UNSAT</td></tr>
</table>

<table>
<tr><td>T-102.9</td><td colspan="7">Conduct an Emergency Blowdown due to High Drywell Press</td></tr>
<tr><td colspan="8">K/A information obtained from NUREG-1123 Revision 3</td></tr>
<tr><td colspan="3"><u>Identifier</u></td><td colspan="2"><u>K/A</u></td><td colspan="2"><u>Importance</u></td><td rowspan="2"><u>Safety Function (for Systems)</u></td></tr>
<tr><td>System/EPE/APE</td><td>Number</td><td>Title</td><td>Number</td><td>Ability</td><td>RO</td><td>SRO</td></tr>
<tr><td>System</td><td>223001</td><td>PCS Primary Containment System and Auxiliaries</td><td>A2.07</td><td>High drywell pressure</td><td>4.4</td><td>4.3</td><td>5. Containment Integrity</td></tr>
<tr><td>EPE</td><td>295024</td><td>High Drywell Pressure</td><td>EA2.01</td><td>Drywell Pressure</td><td>4.4</td><td>4.4</td><td>N/A</td></tr>
<tr><td>Safety Significance</td><td colspan="7">Correct performance of this action prevents a significant reduction of safety margin beyond that irreparably introduced by the scenario. Failure to perform this task correctly also represents a degradation of a fission product barrier.</td></tr>
<tr><td>Initiating Cue</td><td colspan="7">Pri Cont pressure and Supp Pool level cannot be maintained on the safe side of the Pressure Suppression Pressure (PSP) curve limit</td></tr>
<tr><td rowspan="2">Measurable Performance Standard</td><td colspan="2">Expected action</td><td colspan="5">Commence an emergency blowdown in accordance with the RPV Control contingency in effect. If no RPV Control contingency is in effect, commence an Emergency Blowdown in accordance with T-112</td></tr>
<tr><td colspan="2">Safety-significant boundary condition</td><td colspan="5">Prior to exceeding Pressure Suppression Pressure (PSP) Curve.</td></tr>
<tr><td>Performance Feedback</td><td colspan="7">Either:
1. T-112 has been commenced, or
2. Emergency Blowdown commenced per contingency TRIP in effect</td></tr>
<tr><td>Evaluation</td><td colspan="7">SAT / UNSAT</td></tr>
</table>

Facility: Limerick 1 & 2 Scenario No.: SEG-5006E Rev 2 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 Control Enclosure HVAC in a B Subsystem Radiation Isolation per S78.8.A section 4.3 for CMO evaluation.

Critical Tasks:

T-101.2 Inhibit Auto ADS due to Low RPV Level
T-101.5 Conduct an Emergency Blowdown due to Low RPV Level

Per NUREG-1021, App. D, If an operator or crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis for a CT identified in the post scenario review

Event No.	Malfunction Number	Event Type*	Event Description
1.	N/A	N-BOP N-SRO	Place CE HVAC in Radiation Isolation
2.	HS26-190A/C	TS	Containment Leak Detector Inadvertent Isolation
3.	MPR020C	I-ATC I-SRO	#3 APRM fails upscale **(Malfunction)**
4.	MFH563C	R-ATC C-ALL	Low Pressure FWH Level Transient **(Abnormal)**
5.	MRR209A1 MRR209A2	C-BOP C-SRO TS	‘1A’ Loss of ASD Cooling **(Abnormal)**
6.	MPR003A	C-ATC C-SRO	Core Power Oscillations **(Malfunction)**
7.	MFW252A MRR440A	M-ALL	Loss of High Pressure Injection / LOCA Inside Containment **(CT-1, CT-2)**
8.	MRC457B	C-BOP C-SRO	RCIC controller in AUTO failure **(Malfunction)**
9.	MAD151M	C-ATC C-SRO	‘1M’ Tailpipe break at 50% **(CT-2 continued) (Malfunction)**

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

QUANTITATIVE ATTRIBUTES

A. ILT

Target Quantitative Attributes (Per Scenario; See ES-301 Section D.5.d)	
1. Malfunctions after EOP entry (1-2): RCIC controller in AUTO failure; ‘1M’ Tailpipe break at 50%	2
2. Abnormal events (2-4): Low Pressure FWH Level Transient; ‘1A’ Loss of ASD Cooling	2
3. Major transients (1–2): Loss of High Pressure Injection / LOCA Inside Containment	1
4. EOPs entered/requiring substantive actions (1–2): T-101, T-102	2
5. EOP contingencies requiring substantive actions (0-2): T-112	1
6. Critical tasks (2–3)	2

Simulator Scenario Summaries

<u>SEG-5006E</u>

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

Turnover:
- Maintain 100% Reactor power
- Place Control Enclosure HVAC in a B Subsystem Radiation Isolation per S78.8.A section 4.3 for CMO evaluation run.

Event 1: Shortly after the crew has assumed responsibility, they will be required to place Control Enclosure HVAC in a B Subsystem Radiation Isolation per S78.8.A section 4.3

Evaluation: The crew executes the procedure.

Event 2: Shortly after the evolution of inserting a CE HVAC Radiation Isolation, an inadvertent isolation of the Unit 1 Containment Leak Detector occurs.

Evaluation: The crew determines that the Unit 1 Containment Leak Detector is isolated and after referencing the Alarm Response Card (ARC), will dispatch floor operators to investigate. The SRO will evaluate Tech Specs and enter LCO 3.4.3.1 and direct Chemistry to sample.

Event 3: Following the isolation of the Containment Leak Detector, APRM #3 fails upscale.

Evaluation: The ATC determines a failure of the APRM #3. After referencing the Alarm Response Card (ARC), and Tech Specs the Crew places APRM #3 in bypass.

Event 4: After the crew takes action to bypass the #3 APRM an isolation of one LP Feedwater Heater String occurs forcing the crew to reduce Rx Power.

Evaluation: The crew identifies the isolation of the ‘1C’ LP Feedwater Heater String and enters AOP OT-104, Reactivity Addition. The ATC reduces Reactor power to maintain less than 100%. The crew further reduces power to meet feedwater inlet temperature constraints.

SEG-5006E (continued)

Event 5: As the crew is recovering from the low pressure FWH level transient a sequential loss of the '1A' ASD cooling pumps occurs requiring entry into single loop operations.

Evaluation: The crew identifies the loss of cooling to the '1A' ASD. After a period of time, cell temperature alarms will occur on the '1A' ASD, and the crew will trip the ASD and enter AOP OT-112, Recirc Flow. The SRO enters GP-15 and directs actions for single loop operations. The SRO then evaluates and enters Tech Spec 3.4.1.1.a.1 for single loop operation.

Event 6: As the crew responds to the Reactor Recirc Pump Trip transient, Thermal Hydraulic Instabilities (THI) occur.

Evaluation: The ATC observes THI indications and scrams the unit. The SRO enters EOP T-101 and the ATC and BOP perform their scram actions using OT-200 appendices 1 and 10 respectively.

Event 7: Following the reactor shutdown, a Feedwater line break inside primary containment occurs resulting in a loss of all feedwater to the RPV followed by a progressively worsening RPV coolant leak into the drywell.

Evaluation: The crew identifies rising DW pressure and enters EOP T-102. The ATC recognizes the Feedwater line break and actions are taken to isolate the feedwater. The crew then takes action to identify HP injection sources and primary containment controls (Drywell and Suppression pool sprays).

Event 8: When RCIC is initiated its flow controller fails in automatic.

Evaluation: The crew determines that RCIC is not functioning with its flow controller in automatic, and takes manual control to allow injection with RCIC

Event 9: As RPV level decreases below -161" the crew performs the T-112, Emergency Blowdown allowing low pressure ECCS systems to maintain RPV level.

Evaluation: During the emergency depressurization the '1M' SRV is identified as having a broken tailpipe. The operator closes the '1M' SRV and opens an additional SRV to satisfy the RPV T-112 depressurization requirements.

Termination: 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.

Simulator Scenario Critical Tasks

T-101.2	Inhibit Auto ADS due to Low RPV Level						
K/A information obtained from NUREG-1123 Revision 3							
Identifier			**K/A**		**Importance**		**Safety Function (for Systems)**
System/EPE/APE	**Number**	**Title**	**Number**	**Ability**	**RO**	**SRO**	
System	218000	ADS Automatic Depressurizatio n System	A2.06	ADS initiation signals present	4.5	4.3	3. Reactor Pressure Control
EPE	295031	Reactor Low Water Level	EA2.01	Reactor water level	4.7	4.6	N/A

Safety Significance	Correct performance of this action prevents a significant reduction of safety margin beyond that irreparably introduced by the scenario.	
Initiating Cue	Mode switch in shutdown and either: 1. Reactor power above 4% or unknown 2. RPV level cannot be maintained above -129"	
Measurable Performance Standard	**Expected action**	Inhibit auto ADS by placing ADS Auto Inhibit switches B21C-S15A and B21C-S15C in the inhibit position
	Safety-significant boundary condition	Prior to automatic ADS initiation (5 open SRVs)
Performance Feedback	ADS Auto Inhibit switches B21C-S15A and B21C-S15C in the inhibit position	
Evaluation	**SAT / UNSAT**	

T-101.5	Conduct an Emergency Blowdown due to Low RPV Level

K/A information obtained from NUREG-1123 Revision 3

Identifier			K/A		Importance		Safety Function (for Systems)
System/EPE/APE	**Number**	**Title**	**Number**	**Ability**	**RO**	**SRO**	
System	218000	ADS Automatic Depressurization System	A2.02	Loss of coolant accident	4.2	4.1	3. Reactor Pressure Control
EPE	295031	Reactor Low Water Level	EA2.03	Reactor pressure	4.2	4.0	N/A

Safety Significance	Correct performance of this action prevents a significant reduction of safety margin beyond that irreparably introduced by the scenario. Failure to perform this task correctly also represents a degradation of a fission product barrier.	
Initiating Cue	Mode switch in shutdown with RPV level unable to be restored and maintained above -161"	
Measurable Performance Standard	**Expected action**	Commence an emergency blowdown per T-112 to restore core cooling
	Safety-significant boundary condition	Before RPV level reaches Minimum Steam Cooling RPV Water Level (MSCRWL) (-186")
Performance Feedback	Lowering RPV pressure.	
Evaluation	**SAT / UNSAT**	

Facility: Limerick 1 & 2 Scenario No.: SEG-6215E Rev 2 Op-Test No.: 1 .

Examiners: ______________________ Operators: ______________________

______________________ ______________________

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Initial Conditions:

Unit 2 is at 100% power
Unit 1 is at 96% power for rod recovery following control rod maintenance.
Reactor Services personnel are loading spent fuel into shipping casks.
Refuel Floor Secondary Containment is established.

Turnover:

The crew is expected to withdraw the 2 maintenance rods per the ReMA and restore power to 100% with Recirc flow.

Critical Tasks:

T-117.1 Inhibit Auto ADS due to Low RPV Level.
T-117.3 Control Reactor Power by Controlling RPV Injection.
T-117.6 Insert Control Rods.

Per NUREG-1021, App. D, If an operator or crew significantly deviates from or fails to follow procedures that affect the maintenance of basic safety functions, those actions may form the basis for a CT identified in the post scenario review

Event No.	Malfunction Number	Event Type*	Event Description
1	N/A	R-ATC	Withdraw control rods and restore power to 100%
2	MPR011B	C-ATC C-SRO TS	'1B' RBM fails INOP **(Malfunction)**
3	MED279B	C-BOP C-SRO TS	Loss of 10-Y202 instrument bus **(Abnormal)**
4	MCR547	C-ATC C-SRO	CRD pump trip due to clogged suction strainer **(Abnormal)**
5	MMS063D MRD556 MSL559	M-ALL	MSIV closure with Hydraulic ATWS and SLC Line rupture (**CT-1, CT-2, CT-3)**
6	MRD024	C-ATC C-SRO	RDCS fails **(Malfuction)**
7	MMT100 MEH108	C-BOP C-SRO	Turbine high vibration requiring manual turbine trip / Bypass Valves fail closed **(Malfuction)**

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

QUANTITATIVE ATTRIBUTES

A. ILT

Target Quantitative Attributes (Per Scenario; See ES-301 Section D.5.d)	
1. Malfunctions after EOP entry (1-2): Turbine high vibration requiring manual turbine trip / Bypass Valves fail closed; RDCS Fails	2
2. Abnormal events (2-4): Loss of 10-Y202 instrument bus; CRD pump trip due to clogged suction strainer	2
3. Major transients (1–2): MSIV closure with Hydraulic ATWS and SLC Line rupture	1
4. EOPs entered/requiring substantive actions (1–2): T-101, T-102	2
5. EOP contingencies requiring substantive actions (0-2): T-117	1
6. Critical tasks (2–3)	3

SEG-6215E

Initial Conditions:
Unit 1 is at 96% power for rod recovery following on-line HCU maintenance.
Unit 2 is at 100% power.
Reactor Services personnel are loading spent fuel into shipping casks.
Refuel Floor Secondary Containment is established.

Turnover:
The crew is expected to withdraw the 2 maintenance rods per the ReMA and restore power to 100% with Recirc flow.

Event 1: When the crew takes responsibility, the ATC is directed to withdraw control rods per the provided ReMA and restore power to 100% with Recirc flow.

Evaluation: The crew takes action to withdraw the control rods using the ReMA and normal operating procedure S73.1.A while monitoring control rod drive parameters and Reactor Power. A coupling check is performed for each control rod withdrawn.

Event 2: When the 2nd control rod is selected, the '1B' RBM fails INOP and generates a control rod block requiring a Tech Spec evaluation.

Evaluation: The ATC will determine the rod block condition and the crew will determine that the '1B' Rod Block Monitor has as failed. After referencing Tech Specs, and thermal limits the crew determines that it is permissible to bypass the RBM and continue with recovering the control rod.

Event 3: Once both maintenance rods have been withdrawn and Reactor power has been restored to 100% with Recirc, the safeguard instrument bus 10-Y202 trips.

Evaluation: Using the indication in the MCR the crew determines the trip of 10-Y202 safeguard instrument bus and enters AOP E-10Y202. From the procedure the crew makes various callouts to the field. The crew recognizes a loss of Drywell cooling and enters AOP OT-101, High Drywell Pressure. The crew addresses the loss of Drywell cooling by starting the standby Drywell Chiller to restore Drywell cooling.

Event 4: After the 10-Y202 trip has been addressed the running control rod drive pump trips due to a clogged common suction strainer.

Evaluation: The crew enters AOP OT-107. During execution of the procedure the crew bypasses the strainer, and starts a CRD pump.

SEG-6215E (continued)

Event 5: After the crew has restarted a CRD pump , the ‘D’ Inboard MSIV fails closed resulting in an automatic scram signal. A hydraulic ATWS occurs with 185 control rods (all) failing to scram. Complicating the event, the SLC injection line ruptures in the Drywell.

Evaluation: The ATC recognizes the scram condition and place Reactor Mode Switch in Shutdown. The SRO enters EOP T-101. The crew then recognizes the ATWS, the ATC and BOP execute T-200 appendices 1 and 10 respectively. The ATC performs the first RPV lowering below -50 inches. The failure of SLC is identified and procedure T-209 called out to be performed. Two RHRSW pumps are then started in preparation for placing two loops of suppression pool cooling in service. The SRO directs performance of T-221 to keep the MSIVs open, and T-217 to insert control rods.

Event 6: Complicating the event, the rod drive control system fails at the time of the scram.

Evaluation: The ATC recognizes the inability to insert control rods manually and dispatches an operator to reset the rod drive control system.

Event 7: After Reactor level has been lowered to less than -50”, the Main Turbine experiences high vibrations which requires a turbine trip. Complicating the event will be a failure of the Turbine Bypass Valves to control pressure, forcing pressure control with SRVs and heat addition to containment.

Evaluation: The crew identifies that the Main Turbine vibrations are rising and briefs a pressure control contingency to trip it. Once the Main Turbine is tripped the crew recognizes the DEHC Bypass Valve malfunction and establishes pressure control with SRVs. As Suppression Pool temperature reaches 110 °F, the crew performs a second lowering to below top of active fuel (-161”). At this point the rods will be inserted and the crew slowly recovers level to the normal band (+12.5” to +54”) to mitigate RPV cooldown.

Termination: The scenario may be terminated when all control rods have been inserted and RPV level has been stabilized above top of active fuel.

Simulator Scenario Critical Tasks

<table>
<tr><td>T-117.1</td><td colspan="8">Inhibit Auto ADS due to Low RPV Level</td></tr>
<tr><td colspan="9">K/A information obtained from NUREG-1123 Revision 3</td></tr>
<tr><td colspan="3">Identifier</td><td colspan="2">K/A</td><td colspan="2">Importance</td><td rowspan="2">Safety Function (for Systems)</td></tr>
<tr><td>System/EPE/APE</td><td>Number</td><td>Title</td><td>Number</td><td>Ability</td><td>RO</td><td>SRO</td></tr>
<tr><td>System</td><td>218000</td><td>ADS Automatic Depressurization System</td><td>A2.06</td><td>ADS initiation signals present</td><td>4.5</td><td>4.3</td><td>3. Reactor Pressure Control</td></tr>
<tr><td>EPE</td><td>295031</td><td>Reactor Low Water Level</td><td>EA2.01</td><td>Reactor water level</td><td>4.7</td><td>4.6</td><td>N/A</td></tr>
<tr><td>Safety Significance</td><td colspan="7">Correct performance of this action prevents a significant reduction of safety margin beyond that irreparably introduced by the scenario.</td></tr>
<tr><td>Initiating Cue</td><td colspan="7">ATWS condition</td></tr>
<tr><td rowspan="2">Measurable Performance Standard</td><td colspan="2">Expected action</td><td colspan="5">Inhibit auto ADS by placing ADS Auto Inhibit switches B21C-S15A and B21C-S15C in the inhibit position</td></tr>
<tr><td colspan="2">Safety-significant boundary condition</td><td colspan="5">Prior to automatic ADS initiation (5 open SRVs)</td></tr>
<tr><td>Performance Feedback</td><td colspan="7">ADS Auto Inhibit switches B21C-S15A and B21C-S15C in the inhibit position</td></tr>
<tr><td>Evaluation</td><td colspan="7">SAT / UNSAT</td></tr>
</table>

T-117.3	Control Reactor Power by Controlling RPV Injection						
K/A information obtained from NUREG-1123 Revision 3							
Identifier			K/A		Importance		Safety Function (for Systems)
System/EPE/APE	Number	Title	Number	Ability	RO	SRO	
EPE	295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown	EA2.01	Reactor power	4.3	4.7	N/A
EPE	295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown	EA2.02	Reactor Water Level	4.3	4.4	N/A

Safety Significance	Correct performance of this action prevents a significant reduction of safety margin beyond that irreparably introduced by the scenario. Failure to perform this task correctly also represents a degradation of a fission product barrier and incorrect reactivity control.	
Initiating Cue	With mode switch in shutdown, Reactor power above 4%, RPV level between TAF (-161”) and -50”, Supp Pool temperature above 110 ºF, and either an SRV open or drywell pressure above 1.68 psig	
Measurable Performance Standard	Expected action	Control RPV injection to lower RPV level until reactor power is <4%, RPV level reaches TAF (-161”), or all SRVs are closed with drywell pressure less than 1.68 psig.
	Safety-significant boundary condition	Prior to exceeding Heat Capacity Temperature Limit (HCTL)
Performance Feedback	RPV level restored and maintained in a band between -186” and the level to which it was lowered.	
Evaluation	SAT / UNSAT	

T-117.6	Insert Control Rods						
K/A information obtained from NUREG-1123 Revision 3							
Identifier			**K/A**		**Importance**		**Safety Function (for Systems)**
System/EPE/APE	**Number**	**Title**	**Number**	**Ability**	**RO**	**SRO**	
System	201001	CRDH Control Rod Drive Hydraulic System	A2.04	SCRAM conditions	4.7	4.2	1. Reactivity Control
EPE	295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown	EA2.05	Control rod position	4.2	4.5	N/A

Safety Significance	Correct performance of this action prevents a significant reduction of safety margin beyond that irreparably introduced by the scenario. Failure to perform this task correctly is also representative of incorrect reactivity control.	
Initiating Cue	ATWS condition	
Measurable Performance Standard	**Expected action**	Insert control rods with a strategy appropriate to conditions, using T-210, T-214, T-215, T-216, T-213, T-219, T-217, or T-218 based on ATWS conditions.
	Safety-significant boundary condition	Prior to entry into SAMP-1 and SAMP-2
Performance Feedback	All rods inserted to or beyond Maximum Subcritical Bank Withdrawal Position (MSBWP), or all rods fully inserted except one.	
Evaluation	**SAT / UNSAT**	

Facility: **Limerick** Date of Exam: **07/12/2021**

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1	4	3	3	N/A			4	3	N/A		3	20	4	3	7
	2	1	1	1				2	1			1	7	2	1	3
	Tier Totals	5	4	4				6	4			4	27	6	4	10
2. Plant Systems	1	3	1	3	3	2	2	3	3	2	2	2	26	3	2	5
	2	1	1	1	1	1	2	1	1	1	1	1	12	0	1	2	3
	Tier Totals	4	2	4	4	3	4	4	4	3	3	3	38	4	4	8

3. Generic Knowledge and Abilities Categories	RO 1	RO 2	RO 3	RO 4	RO Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	2	3	2	3	10	2	2	1	2	7

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it 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 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. 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' 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.

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

* These systems/evolutions must be included as part of the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan. They are not required to be included when using earlier revisions of the K/A catalog.

** These systems/evolutions may be eliminated from the sample (as applicable to the facility) when Revision 3 of the K/A catalog is used to develop the sample plan.

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO)	Form ES-401-1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4			05				AK3.05 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Reduced loop operating requirements: Plant-Specific. (CFR: 41.5 / 45.6)	3.2	1
295003 (APE 3) Partial or Complete Loss of AC Power / 6	06						AK1.06 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Station blackout: Plant-Specific. (CFR: 41.8 to 41.10)	3.8	2
295004 (APE 4) Partial or Total Loss of DC Power / 6		03					AK2.03 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: D.C. bus loads. (CFR: 41.7 / 45.8)	3.3	3
295005 (APE 5) Main Turbine Generator Trip / 3						04.31	Generic 2.4.31 - Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)	4.2	4
295006 (APE 6) Scram / 1	02						AK1.02 - Knowledge of the operational implications of the following concepts as they apply to SCRAM: Shutdown margin. (CFR: 41.8 to 41.10)	3.4	5
295016 (APE 16) Control Room Abandonment / 7						01.30	Generic 2.1.30 – Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)	4.4	6
295018 (APE 18) Partial or Complete Loss of CCW / 8					01		AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Component temperatures. (CFR: 41.10 / 43.5 / 45.13)	3.3	7
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8						02.44	Generic 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. (CFR: 41.5 / 43.5 / 45.12)	4.2	8
295021 (APE 21) Loss of Shutdown Cooling / 4				04			AA1.04 - Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: Alternate heat removal methods. (CFR: 41.7 / 45.6)	3.7	9
295023 (APE 23) Refueling Accidents / 8			02				AK3.02 - Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS: Interlocks associated with fuel handling equipment. (CFR: 41.5 / 45.6)	3.4	10
295024 High Drywell Pressure / 5	01						EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Drywell integrity: Plant-Specific. (CFR: 41.8 to 41.10)	4.1	11
295025 (EPE 2) High Reactor Pressure / 3			03				EK3.03 - Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: HPCI operation: Plant-Specific. (CFR: 41.5 / 45.6)	3.8	12

295026 (EPE 3) Suppression Pool High Water Temperature / 5				03			EA1.03 - Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature monitoring. (CFR: 41.7 / 45.6)	3.9	13
295027 (EPE 4) High Containment Temperature (Mark III Containment Only) / 5									
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5		03					EK2.03 - Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Reactor water level indication. (CFR: 41.7 / 45.8)	3.6	14
295030 (EPE 7) Low Suppression Pool Water Level / 5				01			EA1.01 - Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: ECCS systems (NPSH considerations): Plant-Specific. (CFR: 41.7 / 45.6)	3.6	15
295031 (EPE 8) Reactor Low Water Level / 2					04		EA2.04 - Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Adequate core cooling. (CFR: 41.10 / 43.5 / 45.13)	4.6	16
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1		02					EK2.02 - Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: RRCS: Plant-Specific. (CFR: 41.7 / 45.8)	4.0	17
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9				06			EA1.06 - Ability to operate and/or monitor the following as they apply to HIGH OFFSITE RELEASE RATE: Plant ventilation. (CFR: 41.7 / 45.6)	3.5	18
600000 (APE 24) Plant Fire On Site / 8					13		AA2.13 - Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE: Need for emergency plant shutdown.	3.2	19
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6	03						AK1.03 - Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Under-excitation. (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)	3.3	20
K/A Category Totals:	4	3	3	4	3	3	Group Point Total:		20

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E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295002 (APE 2) Loss of Main Condenser Vacuum / 3					02		AA2.02 - Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Reactor power: Plant-Specific. (CFR: 41.10 / 43.5 / 45.13)	3.2	21
295007 (APE 7) High Reactor Pressure / 3									
295008 (APE 8) High Reactor Water Level / 2	03						AK1.03 - Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL: Feed flow/steam flow mismatch. (CFR: 41.8 to 41.10)	3.2	22
295009 (APE 9) Low Reactor Water Level / 2									
295010 (APE 10) High Drywell Pressure / 5									
295011 (APE 11) High Containment Temperature (Mark III Containment only) / 5									
295012 (APE 12) High Drywell Temperature / 5									
295013 (APE 13) High Suppression Pool Temperature. / 5									
295014 (APE 14) Inadvertent Reactivity Addition / 1									
295015 (APE 15) Incomplete Scram / 1				01			AA1.01 - Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: CRD hydraulics. (CFR: 41.7 / 45.6)	3.8	23
295017 (APE 17) Abnormal Offsite Release Rate / 9									
295020 (APE 20) Inadvertent Containment Isolation / 5 & 7				02			AA1.02 - Ability to operate and/or monitor the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Drywell ventilation/cooling system. (CFR: 41.7 / 45.6)	3.2	24
295022 (APE 22) Loss of Control Rod Drive Pumps / 1		07					AK2.07 - Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: Reactor Pressure (SCRAM assist): Plant-Specific. (CFR: 41.7 / 45.8)	3.4	25
295029 (EPE 6) High Suppression Pool Water Level / 5									
295032 (EPE 9) High Secondary Containment Area Temperature / 5						04.01	Generic 2.4.1 – Knowledge of EOP entry conditions and immediate action steps. (CFR: 41.10 / 43.5 / 45.13)	4.6	26
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9									
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									
295035 (EPE 12) Secondary Containment High Differential Pressure / 5			01				EK3.01 - Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Blow-out panel operation: Plant-Specific. (CFR: 41.5 / 45.6)	2.8	27

295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5									
500000 (EPE 16) High Containment Hydrogen Concentration / 5									
K/A Category Point Totals:	1	1	1	2	1	1	Group Point Total:		7

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Plant Systems—Tier 2/Group 1 (RO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode									09			A3.09 - Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including: Emergency generator load sequencing. (CFR: 41.7 / 45.7)	3.6	28
205000 (SF4 SCS) Shutdown Cooling			03									K3.03 - Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor temperatures (moderator, vessel, flange). (CFR: 41.7 / 45.4)	3.8	29
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection	07											K1.07 - Knowledge of the physical connections and/or cause-effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following: D.C. power: BWR-2,3,4. (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.7	30
				14								K4.14 - Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature(s) and or interlocks which provide for the following: Control oil to turbine speed controls: BWR-2,3,4. (CFR: 41.7)	3.4	31
207000 (SF4 IC) Isolation (Emergency) Condenser														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray										03		A4.03 - Ability to manually operate and/or monitor in the control room: Injection valves. (CFR: 41.7 / 45.5 to 45.8)	3.7	32
209002 (SF2, SF4 HPCS) High-Pressure Core Spray														
211000 (SF1 SLCS) Standby Liquid Control					04							K5.04 - Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: Explosive valve operation. (CFR: 41.5 / 45.3)	3.1	33
212000 (SF7 RPS) Reactor Protection						04						K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM: D.C. electrical distribution. (CFR: 41.7 / 45.7)	2.8	34
					02							K5.02 - Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements. (CFR: 41.5 / 45.3)	3.3	35
215003 (SF7 IRM) Intermediate-Range Monitor							05					A1.05 - Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: SCRAM and rod block trip setpoints. (CFR: 41.5 / 45.5)	3.9	36
215004 (SF7 SRMS) Source-Range Monitor									02			A3.02 - Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: Annunciator and alarm signals. (CFR: 41.7 / 45.7)	3.4	37

215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor	04											K1.04 - Knowledge of the physical connections and/or cause-effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: LPRM channels. (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.6	38
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling											01.32	Generic 2.1.32 - Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)	3.8	39
218000 (SF3 ADS) Automatic Depressurization						02						K6.02 - Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: Low pressure core spray system pressure: Plant-Specific. (CFR: 41.7 / 45.7)	4.1	40
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff							02					A1.02 - Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF SYSTEM controls including: Valve closures. (CFR: 41.5 / 45.5)	3.7	41
239002 (SF3 SRV) Safety Relief Valves								01				A2.01 - 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: Stuck open vacuum breakers. (CFR: 41.5 / 45.6)	3.0	42
259002 (SF2 RWLCS) Reactor Water Level Control				11				06				K4.11 - Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and or interlocks which provide for the following: DP control: Plant-Specific. (CFR: 41.7) A2.06 - Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of controller output signal. (CFR: 41.5 / 45.6)	3.3 3.3	43 44
261000 (SF9 SGTS) Standby Gas Treatment			01									K3.01 - Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on the following: Secondary containment and environment differential pressure. (CFR: 41.7 / 45.6)	3.3	45
262001 (SF6 AC) AC Electrical Distribution				01			05					A1.05 - Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: Breaker lineups. (CFR: 41.5 / 45.5) K4.01 - Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and or interlocks which provide for the following: Bus lockouts. (CFR: 41.7)	3.2 3.0	46 47
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)			02									K3.02 - Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on the following: Recirculation pump speed: Plant-Specific. (CFR: 41.7 / 45.4)	2.9	48

263000 (SF6 DC) DC Electrical Distribution											04.45	Generic 2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.12)	4.1	49
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG	06											K1.06 - Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Starting system. (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.2	50
								07				A2.07 - Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of off-site power during full-load testing. (CFR: 41.5 / 45.6)	3.5	51
300000 (SF8 IA) Instrument Air		02										K2.02 - Knowledge of electrical power supplies to the following: Emergency air compressor. (CFR: 41.7)	3.0	52
400000 (SF8 CCS) Component Cooling Water										01		A4.01 - Ability to manually operate and/or monitor in the control room: CCW indications and control. (CFR: 41.7 / 45.5 to 45.8)	3.1	53
510000 (SF4 SWS*) Service Water (Normal and Emergency)														
K/A Category Point Totals:	3	1	3	3	2	2	3	3	2	2	2	Group Point Total:		26

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Plant Systems—Tier 2/Group 2 (RO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 (SF1 CRDH) CRD Hydraulic														
201002 (SF1 RMCS) Reactor Manual Control											02.12	Generic 2.2.12 – Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)	3.7	54
201003 (SF1 CRDM) Control Rod and Drive Mechanism				04								K4.04 - Knowledge of CONTROL ROD AND DRIVE MECHANISM design feature(s) and/or interlocks which provide for the following: The use of either accumulator or reactor water to SCRAM the control rod. (CFR: 41.7)	3.6	55
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer						05						K6.05 - Knowledge of the effect that a loss or malfunction of the following will have on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC): Steam flow input: P-Spec (Not-BWR6). (CFR: 41.7 / 45.7)	2.7	56
202001 (SF1, SF4 RS) Recirculation														
202002 (SF1 RSCTL) Recirculation Flow Control								01				A2.01 - 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 trip. (CFR: 41.5 / 45.6)	3.4	57
204000 (SF2 RWCU) Reactor Water Cleanup														
214000 (SF7 RPIS) Rod Position Information														
215001 (SF7 TIP) Traversing In-Core Probe							03					A1.03 - Ability to predict and/or monitor changes in parameters associated with operating the TRAVERSING IN-CORE PROBE controls including: Valve status: Mark-I&II (Not-BWR1). (CFR: 41.5 / 45.5)	2.6	58
215002 (SF7 RBMS) Rod Block Monitor														
216000 (SF7 NBI) Nuclear Boiler Instrumentation			02									K3.02 - Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER INSTRUMENTATION will have on following: PCIS/NSSSS. (CFR: 41.7 / 45.4)	4.0	59
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode														
223001 (SF5 PCS) Primary Containment and Auxiliaries		09										K2.09 - Knowledge of electrical power supplies to the following: Drywell cooling fans: Plant-Specific. (CFR: 41.7)	2.7	60
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode														

233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup														
234000 (SF8 FH) Fuel-Handling Equipment														
239001 (SF3, SF4 MRSS) Main and Reheat Steam														
239003 (SF9 MSVLCS) Main Steam Isolation Valve Leakage Control														
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating					04							K5.04 - Knowledge of the operational implications of the following concepts as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM: Turbine inlet pressure vs. reactor pressure. (CFR: 41.5 / 45.3)	3.3	61
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary						05						K6.05 - Knowledge of the effect that a loss or malfunction of the following will have on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Stator water cooling. (CFR: 41.7 / 45.7)	2.9	62
256000 (SF2 CDS) Condensate										03		A4.03 - Ability to manually operate and/or monitor in the control room: Hotwell level controls. (CFR: 41.7 / 45.5 to 45.8)	3.2	63
259001 (SF2 FWS) Feedwater														
268000 (SF9 RW) Radwaste														
271000 (SF9 OG) Offgas	02											K1.02 - Knowledge of the physical connections and/or cause-effect relationships between OFFGAS SYSTEM and the following: Process radiation monitoring system. (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.1	64
272000 (SF7, SF9 RMS) Radiation Monitoring														
286000 (SF8 FPS) Fire Protection														
288000 (SF9 PVS) Plant Ventilation														
290001 (SF5 SC) Secondary Containment														
290003 (SF9 CRV) Control Room Ventilation									01			A3.01 - Ability to monitor automatic operations of the CONTROL ROOM HVAC including: Initiation/reconfiguration. (CFR: 41.7 / 45.7)	3.3	65
290002 (SF9 RVI) Reactor Vessel Internals														
51001 (SF8 CWS*) Circulating Water														
K/A Category Point Totals:	1	1	1	1	1	2	1	1	1	1	1	Group Point Total:		12

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4						02. 25	Generic 2.2.25 – Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2)	4.2	76
295003 (APE 3) Partial or Complete Loss of AC Power / 6									
295004 (APE 4) Partial or Total Loss of DC Power / 6									
295005 (APE 5) Main Turbine Generator Trip / 3									
295006 (APE 6) Scram / 1									
295016 (APE 16) Control Room Abandonment / 7									
295018 (APE 18) Partial or Complete Loss of CCW / 8									
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8					02		AA2.02 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of safety-related instrument air system loads (see AK2.1 – AK2.19). (CFR: 41.10 / 43.5 / 45.13)	3.7	77
295021 (APE 21) Loss of Shutdown Cooling / 4									
295023 (APE 23) Refueling Accidents / 8					05		AA2.05 -Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Entry conditions of emergency plan. (CFR: 41.10 / 43.5 / 45.13)	4.6	78
295024 High Drywell Pressure / 5									
295025 (EPE 2) High Reactor Pressure / 3									
295026 (EPE 3) Suppression Pool High Water Temperature / 5					03		EA2.03 -Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor pressure. (CFR: 41.10 / 43.5 / 45.13)	4.0	79
295027 (EPE 4) High Containment Temperature (Mark III Containment Only) / 5									
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5									
295030 (EPE 7) Low Suppression Pool Water Level / 5									
295031 (EPE 8) Reactor Low Water Level / 2									
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1						04. 18	Generic 2.4.18 – Knowledge of the specific bases for EOPs. (CFR: 41.10 / 43.1 / 45.13)	4.0	80
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9									
600000 (APE 24) Plant Fire On Site / 8						04. 08	Generic 2.4.8 – Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)	4.5	81

700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6					05		AA2.05 - Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Operational status of offsite circuit. (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)	3.8	82
K/A Category Totals:	0	0	0	0	4	3	Group Point Total:		7

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Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (SRO)

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295002 (APE 2) Loss of Main Condenser Vacuum / 3									
295007 (APE 7) High Reactor Pressure / 3									
295008 (APE 8) High Reactor Water Level / 2									
295009 (APE 9) Low Reactor Water Level / 2									
295010 (APE 10) High Drywell Pressure / 5									
295011 (APE 11) High Containment Temperature (Mark III Containment only) / 5									
295012 (APE 12) High Drywell Temperature / 5									
295013 (APE 13) High Suppression Pool Temperature. / 5									
295014 (APE 14) Inadvertent Reactivity Addition / 1						04.30	Generic 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. (CFR: 41.10 / 43.5 / 45.11)	4.1	83
295015 (APE 15) Incomplete Scram / 1									
295017 (APE 17) Abnormal Offsite Release Rate / 9									
295020 (APE 20) Inadvertent Containment Isolation / 5 & 7									
295022 (APE 22) Loss of Control Rod Drive Pumps / 1									
295029 (EPE 6) High Suppression Pool Water Level / 5					03		EA2.03 - Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level. (CFR: 41.10 / 43.5 / 45.13)	3.5	84
295032 (EPE 9) High Secondary Containment Area Temperature / 5									
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9									
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									
295035 (EPE 12) Secondary Containment High Differential Pressure / 5									
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5									
500000 (EPE 16) High Containment Hydrogen Concentration / 5					03		EA2.03 - Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible limits for drywell. (CFR: 41.10 / 43.5 / 45.13)	3.8	85

K/A Category Point Totals:	0	0	0	0	2	1	Group Point Total:	3

ES-401 BWR Examination Outline Form ES-401-1
Plant Systems—Tier 2/Group 1 (SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode														
205000 (SF4 SCS) Shutdown Cooling											02.37	Generic 2.2.37 - Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)	4.6	86
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection														
207000 (SF4 IC) Isolation (Emergency) Condenser														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray								05				A2.05 - Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Core spray line break. (CFR: 41.5 / 45.6)	3.6	87
209002 (SF2, SF4 HPCS) High-Pressure Core Spray														
211000 (SF1 SLCS) Standby Liquid Control														
212000 (SF7 RPS) Reactor Protection														
215003 (SF7 IRM) Intermediate-Range Monitor														
215004 (SF7 SRMS) Source-Range Monitor								02				A2.02 - 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: SRM inop condition. (CFR: 41.5 / 45.6)	3.7	88
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor														
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling											01.23	Generic 2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)	4.4	89
218000 (SF3 ADS) Automatic Depressurization														
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff														
239002 (SF3 SRV) Safety Relief Valves														
259002 (SF2 RWLCS) Reactor Water Level Control														

261000 (SF9 SGTS) Standby Gas Treatment								03				A2.03 - Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High train temperature. (CFR: 41.5 / 45.6)	3.2	90
262001 (SF6 AC) AC Electrical Distribution														
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)														
263000 (SF6 DC) DC Electrical Distribution														
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG														
300000 (SF8 IA) Instrument Air														
400000 (SF8 CCS) Component Cooling Water														
510000 (SF4 SWS*) Service Water (Normal and Emergency)														
K/A Category Point Totals:	0	0	0	0	0	0	0	3	0	0	2	Group Point Total:		5

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems—Tier 2/Group 2 (SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 (SF1 CRDH) CRD Hydraulic														
201002 (SF1 RMCS) Reactor Manual Control														
201003 (SF1 CRDM) Control Rod and Drive Mechanism														
201004 (SF7 RSCS) Rod Sequence Control														
201005 (SF1, SF7 RCIS) Rod Control and Information														
201006 (SF7 RWMS) Rod Worth Minimizer														
202001 (SF1, SF4 RS) Recirculation								01				A2.01 - Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Jet pump failure: Plant-Specific. (CFR: 41.5 / 45.6)	3.9	91
202002 (SF1 RSCTL) Recirculation Flow Control														
204000 (SF2 RWCU) Reactor Water Cleanup														
214000 (SF7 RPIS) Rod Position Information											01.07	Generic 2.1.7 - Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation. (CFR: 41.5 / 43.5 / 45.12 / 45.13)	4.7	92
215001 (SF7 TIP) Traversing In-Core Probe														
215002 (SF7 RBMS) Rod Block Monitor														
216000 (SF7 NBI) Nuclear Boiler Instrumentation														
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode														
223001 (SF5 PCS) Primary Containment and Auxiliaries														
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Spray Mode														
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup														
234000 (SF8 FH) Fuel-Handling Equipment														
239001 (SF3, SF4 MRSS) Main and Reheat Steam														
239003 (SF9 MSVLCS) Main Steam Isolation Valve Leakage Control														
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating														
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary														
256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater														

268000 (SF9 RW) Radwaste														
271000 (SF9 OG) Offgas														
272000 (SF7, SF9 RMS) Radiation Monitoring														
286000 (SF8 FPS) Fire Protection											02.38	Generic 2.2.38 - Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)	4.5	93
288000 (SF9 PVS) Plant Ventilation														
290001 (SF5 SC) Secondary Containment														
290003 (SF9 CRV) Control Room Ventilation														
290002 (SF9 RVI) Reactor Vessel Internals														
51001 (SF8 CWS*) Circulating Water														
K/A Category Point Totals:	0	0	0	0	0	0	0	1	0	0	2	Group Point Total:		3

Facility: **Limerick** Date of Exam: **07/12/2021**

Category	K/A #	Topic	RO IR	RO #	SRO-only IR	SRO-only #
1. Conduct of Operations	G2.1.1	Knowledge of conduct of operations requirements. (CFR: 41.10 / 45.13)	3.8	66		
	G2.1.26	Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen). (CFR: 41.10 / 45.12)	3.4	67		
	G2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management. (CFR: 41.1 / 43.6 / 45.6)			4.6	94
	G2.1.40	Knowledge of refueling administrative requirements (CFR: 41.10 / 43.5 / 45.13)			3.9	95
	Subtotal			2		2
2. Equipment Control	G2.2.6	Knowledge of the process for making changes to procedures. (CFR: 41.10 / 43.3 / 45.13)	3.0	68		
	G2.2.13	Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)	4.1	69		
	G2.2.35	Ability to determine Technical Specification Mode of Operation. (CFR: 41.7 / 41.10 / 43.2 / 45.13)	3.6	70		
	G2.2.11	Knowledge of the process for controlling temporary design changes. (CFR: 41.10 / 43.3 / 45.13)			3.3	96
	G2.2.21	Knowledge of pre- and post-maintenance operability requirements. (CFR: 41.10 / 43.2)			4.1	97
	Subtotal			3		2
3. Radiation Control	G2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions. (CFR: 41.12 / 43.4 / 45.10)	3.2	71		
	G2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)	3.4	72		
	G2.3.13	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. (CFR: 41.12 / 43.4 / 45.9 / 45.10)			3.8	98
	Subtotal			2		1
4. Emergency Procedures/Plan	G2.4.17	Knowledge of EOP terms and definitions. (CFR: 41.10 / 45.13)	3.9	73		
	G2.4.26	Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage. (CFR: 41.10 / 43.5 / 45.12)	3.1	74		
	G2.4.43	Knowledge of emergency communication systems and techniques. (CFR: 41.10 / 45.13)	3.2	75		
	G2.4.38	Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required. (CFR: 41.10 / 43.5 / 45.11)			4.4	99
	G2.4.44	Knowledge of emergency plan protective action recommendations. (CFR: 41.10 / 41.12 / 43.5 / 45.11)			4.4	100
	Subtotal			3		2
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
Tier 2 / Group 1	300000 **K2.02** RO Q52	Facility was unable to develop an operationally valid RO level question to test the selected K/A; i.e., *"Knowledge of electrical power supplies to the following: Emergency air compressor" (IR 3.0)*. Limerick does not have an Emergency air compressor. **2/23/21 – CHIEF EXAMINER re-selected K/A K2.01 for Q52 to maintain K/A category balance within the outline *(Note: There are only two K/As within the K2 Category for BWR System 300000, "Instrument Air System"):*** K2.01 - "Knowledge of electrical power supplies to the following: Instrument air compressor." (IR 2.8)
Tier 1 / Group 1	295019 **G2.2.44** RO Q8	Facility was unable to develop an operationally valid question to test the selected Generic K/A; i.e., *"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."* Procedure changes to both ON-119 and ON-104 resulted in two correct answers in the originally submitted version of the question. **5/13/21 – CHIEF EXAMINER re-selected Generic K/A 2.4.4 for Q8 to maintain K/A category balance within the outline.** G2.4.4 - "Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures." (IR 4.5)
Tier 2 / Group 1	263000 **G2.4.45** RO Q49	Facility was unable to develop an operationally valid and discriminating RO level question to test the selected Generic K/A; i.e., *"Ability to prioritize and interpret the significance of each annunciator or alarm,"* in combination with SYSTEM 263000, D.C. ELECTRICAL DISTRIBUTION. **7/1/21 – CHIEF EXAMINER randomly re-selected K/A A3.01 from amongst those categories with Tier 2 Group 1 totals ≤ 2 (i.e., K2, K5, K6, A3, and A4), to maintain outline category balance.** A3.01 - "Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights." (IR 3.2)