Exelon Generation.

10 CFR 55.40

NMP1L3230
July 2, 2018

U.S. Nuclear Regulatory Commission
Attn: Regional Administrator, Region I
2100 Renaissance Blvd, Suite 100
King of Prussia, PA 19406-2713

Nine Mile Point Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-63
NRC Docket No. 50-220

Subject: Nine Mile Point Unit 1 Initial License Examination Outlines

Reference: (1) Letter from D. E. Jackson (NRC) to B. Hanson (Exelon Nuclear), dated April 26, 2018, Senior Reactor and Reactor Operator Initial License Examinations - (Nine Mile Point, Unit 1)

As discussed in Reference (1), arrangements have been made for the administration of license examinations at Nine Mile Point Unit 1 during the week of December 3, 2018. The examinations are being prepared based on guidelines in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 11.

Enclosed are the approved examination outlines for the Unit 1 Senior Reactor and Reactor Operator initial license examinations. The written examination outline was originally developed by the Chief Examiner, Mr. Peter Presby.

In accordance with NUREG-1021, ES-201, "Initial Operator Licensing Examination Process," Attachment 1, Nine Mile Point Nuclear Station, LLC (NMPNS) requests that the examination materials be withheld from public disclosure until two years after the examinations have been completed. The enclosed materials are appropriately marked in accordance with NUREG-1021.

Should you have any questions regarding the information in this submittal, please contact Greg Elkins, Manager Operations Training, at (315) 349-1261.

Sincerely,

James N. Tsardakas
Director Site Training, Nine Mile Point Nuclear Station
Exelon Generation Company, LLC

JNT/RSP

Enclosure: Examination Outlines for the Unit 1 Senior Reactor and Reactor Operator Initial License Examinations

cc: P. Presby, NRC Chief Examiner (with enclosure)
D. Jackson, NRC (without enclosure)
NRC Resident Inspector (without enclosure)

Enclosure

Examination Outlines for the Unit 1
Senior Reactor and Reactor Operator Initial License Examinations

Facility: Nine Mile Point Unit 1 Date of Examination: 12/03/2018

Developed by: Written: Facility ☒ NRC ☐ // Operating: Facility ☒ NRC ☐

Target Date*	Task Description (Reference)	Chief Examiner's Initials
-240	1. Examination administration date confirmed (C.1.a; C.2.a–b). For NRC-prepared exams, arrangements are made for the facility to submit reference materials (C.1.e; C.3.c; Attachment 3).	04/06/2018
-210	2. NRC examiners and facility contact assigned (C.1.d; C.2.f).	04/26/2018
-210	3. Facility contact briefed on security and other requirements (C.2.c). As applicable, the facility contact submits to the NRC any prescreened K/As for elimination from the written examination outline, with a description of the facility's prescreening process (ES-401, D.1.b).	04/26/2018
-210	4. Reference material due for NRC-prepared exams (C.1.e; C.3.c; Attachment 3). 0	N/A
-210	5. Corporate notification letter sent (C.2.e).	04/30/2018
-195	6. NRC-developed written examination outline (ES-401-1/2 or ES-401N-1/2 and ES-401-3 or ES-401N-3) sent to facility contact (must be on the exam security agreement) (C.1.e–f; C.2.h; C.3.d–e).	01/02/2018
-150	7. Operating test outline(s) and other checklists due, including Forms ES-201-2, ES-201-3, ES-301-1, ES-301-2, ES-301-5, and ES-D-1, as applicable (C.1.e–f; C.3.d–e).	07/03/2018
-136	8. Operating test outline(s) reviewed by the NRC and feedback provided to facility licensee (C.2.h; C.3.d–e).	07/19/2018
-75	9. Proposed examinations (written, JPMs, and scenarios, as applicable) and outlines (Forms ES-301-1, ES-301-2, ES-D-1, ES-401-1/2 or ES-401N-1/2, and ES-401-3 or ES-401N-3); supporting documentation (including Forms ES-301-3, ES-301-4, ES-301-5, ES-301-6, ES-401-6, ES-401N-6, and any Form ES-201-2 and ES-201-3 updates); and reference materials due (C.1.e–h; C.3.d).	09/18/2018
-75	10. Examinations prepared by the NRC are approved by the NRC supervisor and forwarded for facility licensee review (C.1.i; C.2.h; C.3.f–g).	N/A
-60	11. Preliminary waiver/excusal requests due (C.1.m; C.2.c; ES-202).	10/03/2018
-50	12. Written exam and operating test reviews completed (C.3.f).	10/13/2018
-35	13. Examination review results discussed between the NRC and facility licensee (C.1.i; C.1.k–l; C.2.h; C.3.g). The NRC and the facility licensee conduct exam preparatory week.	10/28/2018 xx/xx/2018
-30	14. Preliminary license applications and waiver/excusal requests, as applicable (NRC Form 398) due (C.1.m; C.2.i; ES-202).	11/02/2018
-14	15. Final license applications and waiver/excusal requests, as applicable (NRC Form 398), due and Form ES-201-4 prepared (C.1.m; C.2.k; ES-202).	11/18/2018
-7	16. Written examinations and operating tests approved by the NRC supervisor (C.2.j–k; C.3.h).	11/25/2018
-7	17. Request facility licensee management feedback on the examination (C.2.l).	11/25/2018
-7	18. Final applications reviewed; one or two (if more than 10) applications audited to confirm qualifications/eligibility; and examination approval and waiver/excusal letters sent (C.2.k; Attachment 5; ES-202, C.3.j; ES-204).	11/25/2018
-7	19. Proctoring/written exam administration guidelines reviewed with facility licensee (C.3.k).	11/25/2018
-7	20. Approved scenarios and job performance measures distributed to NRC examiners (C.3.i).	11/25/2018

* Target dates are based on facility-prepared examinations and the examination date identified in the corporate notification letter. These dates are for planning purposes and may be adjusted on a case-by-case basis in coordination with the facility licensee.

Facility: **Nine Mile Point Unit 1** Date of Examination: **December 2018**

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.	PFI	PMN	[illegible]
	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.	PFI	PMN	[illegible]
	c. Assess whether the outline overemphasizes any systems, evolutions, or generic topics.	PFI	PMN	[illegible]
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	PFI	PMN	[illegible]
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.	PFI	PMN	[illegible]
	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.	PFI	PMN	[illegible]
	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).	PFI	PMN	[illegible]
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.	PFI	PMN	[illegible]
	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.	PFI	PMN	[illegible]
	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.	PFI	PMN	[illegible]
4. GENERAL	a. Assess whether plant-specific priorities (including probabilistic risk assessment and individual plant examination insights) are covered in the appropriate exam sections.	PFI	PMN	[illegible]
	b. Assess whether the 10 CFR 55.41, 55.43, and 55.45 sampling is appropriate.	PFI	PMN	[illegible]
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	PFI	PMN	[illegible]
	d. Check for duplication and overlap among exam sections and the last two NRC exams.	PFI	PMN	[illegible]
	e. Check the entire exam for balance of coverage.	PFI	PMN	[illegible]
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	PFI	PMN	[illegible]

		Printed Name/Signature	Date
a.	Author	Paul Isham / [signature]	9/7/18
b.	Facility Reviewer (*)	Phil Nichols / [signature]	9.7.18
c.	NRC Chief Examiner (#)	P. Presby / [signature]	11/9/18
d.	NRC Supervisor	Donald Jackson / [signature]	~~11/29/18~~ 12/3/18

* Not applicable for NRC-prepared examination outlines.

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

Facility: Nine Mile Point Unit 1 Date of Examination: December 2018
Examination Level: RO Operating Test Number: LC1 17-1 NRC

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M, R	Verification Of Active License Status OP-AA-105-101, OP-AA-105-102, K/A 2.1.4 (3.3)
Conduct of Operations	D, R	DWFDT / DWEDT Leak Rate Determination and Evaluation N1-OP-8, K/A 2.1.18 (3.6)
Equipment Control	D, R	Develop a clearance boundary for the Liquid Poison Test Tank OP-CE-109-101 KA 2.2.13 (4.1)
Radiation Control	P, D, R (2017 NRC)	Application of Radiation Exposure Limits IAW RP-AA-203 – SDC Room RP-AA-203, K/A 2.3.4 (3.2)
Emergency Procedures/Plan		

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

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

Facility: Nine Mile Point Unit 1 Date of Examination: December 2018
Examination Level: SRO Operating Test Number: LC1 17-1 NRC

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, R	Reactivate SRO Licenses OP-AA-105-102, KA 2.1.4 (3.8)
Conduct of Operations	D, R	Perform Time to Boil Calculation for Reactor Coolant System OP-NM-108-117-1002, K/A 2.1.40 (3.9)
Equipment Control	N, R	Review and Approval of Completed Surveillance Test, N1-ST-Q13, Emergency Service Water Pump and Check Valve Operability Test N1-ST-Q13, K/A 2.2.12 (4.1)
Radiation Control	P, D, R (2017 NRC)	Determine Actions for Inoperable Service Water Radiation Monitor N1-ARP-H1, ODCM, K/A 2.3.15 (3.1)
Emergency Procedures/Plan	D, R	Emergency Event Reclassification and Notification EP-CE-111, EPIP-EPP-01 EAL Flowchart, K/A 2.4.41 (4.6)

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

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

Facility: Nine Mile Point Unit 1	Date of Examination: December 2018
Exam Level: RO/SRO-I/**SRO-U**	Operating Test No.: LC1 17-1 NRC

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

System / JPM Title	Type Code*	Safety Function
a. Swap CRD Pumps **K/A 201001 A4.01 (3.1/3.1), N1-OP-5**	**M, A, S**	**1**
b. Perform N1-ST-M8, Reactor Building Emergency Ventilation System Operability Test **K/A 288000 A4.01 (3.1/2.9), N1-ST-M8**	**N, S, EN**	**9**
c. Vent the Drywell Prior to Personnel Entry >212 **K/A 223001 A4.03 (3.4/3.4), N1-OP-9**	**M, S, L, A**	**5**
d. Rapid RWCU System Restoration for Level Control (RO Only) K/A 204000 A4.06 (3.0/2.9), N1-EOP-HC	D, S, L	2
e. Restore Emergency Condenser To Service K/A 207000 A4.05 (3.5/3.7), N1-OP-13	D, A, EN, S	4
f. Swap PB 101 from 1014 to R1011 K/A 262001 A4.01 (3.4/3.7), N1-OP-30	D, S	6
g. Control Rod Exercising Operability Test K/A 214000 A4.02 (3.8/3.8), N1-ST-W1, N1-OP-5	P, D, A, S (2015 NRC)	7
h. MSIV Stroke test and Limit Switch Test K/A 239001 A4.01 (4.2/4.0), N1-ST-Q26	P, S, D (2015 NRC)	3
In-Plant Systems* (3 for RO); (3 for SRO-I) ; **(3 or 2 for SRO-U)**		
i. Swap CRD Stabilizing Valves K/A 201001 A2.08 (2.8/2.8), N1-OP-5	D, R	1
j. Lineup Lake Water to Supply the Emergency Condenser Makeup Tanks using the Electric Fire Pump **K/A 207000 A1.01 (3.7/3.8), N1-SOP-21.2**	**D, E, A, R**	**4**
k. Supply Emergency Cooling Water to EDG from the Diesel Fire Pump **K/A 400000 K1.02 (3.2/3.4), N1-OP-45**	**D, E, R**	**8**

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

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

Pairings:
A then B
E then F

Facility: Nine Mile Point Unit 1 Scenario No.: NRC-2 Op-Test No.: LC1 17-1 NRC

Examiners: ______________________ Operators: ______________________

Initial Conditions: The plant is operating at approximately 90% power. Containment Spray Pump 112 is out of service for maintenance. Steam Packing Exhauster 12 is out of service due to high vibrations. PB 11 is aligned to reserve power in preparation for cross-tying PB 16.

Turnover: Cross-tie PB 16A to PB 16B with PB 16B supplying. Power board 11 will remain aligned to reserve power. Then, raise reactor power to 95% using recirc flow.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N-BOP, SRO	Cross-tie PB 16A to PB 16B N1-OP-30
2	N/A	R-ATC, SRO	Raise reactor power with recirc. N1-OP-1
3	RD02	C –ATC, SRO	Control Rod 26-35 Drifts Out N1-SOP-5.2
4	FW02A Override	C – BOP, SRO TS-SRO	Feedwater Booster Pump 11 Trips with Failure of standby Feedwater Booster Pump to Auto-start N1-SOP-16.1, Technical Specifications
5	RP25	C-All TS-SRO	Respond to trip of Reactor Protection System (RPS) UPS 172 Technical Specification N1-SOP-40.1
6	CU11	M -All	RWCU break in the Secondary Containment requiring scram; RWCU Isolation Valves to isolate N1-EOP-2, N1-EOP-5, N1-EOP-8
7	Overrides	C – ATC, SRO	Mode Switch Fails to Scram N1-SOP-1

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

Facility: **Nine Mile Point Unit 1** Scenario No.: **NRC-2** Op-Test No.: **LC1 17-1 NRC**		
1. Malfunctions after EOP entry (1-2) **Events 7**	1	
2. Abnormal events (2-4) **Events 3, 4, 5**	3	
3. Major transients (1-2) **Event 6**	1	
4. EOPs entered/requiring substantive actions (1-2) **N1-EOP-2, N1-EOP-5**	2	
5. EOP contingencies requiring substantive actions (0-2) **N1-EOP-8**	1	
6. Preidentified Critical tasks (2-3)	2	

CRITICAL TASK DESCRIPTIONS:	CRITICAL TASK JUSTIFICATION:
CT-1.0: Given an un-isolable RWCU leak outside primary containment and one general area temperature above the maximum safe limit, the crew will insert a manual reactor scram, in accordance with N1-EOP-5.	With an un-isolable primary system discharging outside of Primary Containment resulting in general area temperature above the maximum safe limit, the Reactor must be scrammed. This reduces the rate of energy production and thus the heat input, radioactivity release, and break flow into the Secondary Containment. This also ensures the Reactor is shutdown prior to need for a blowdown.
CT- 2.0: Given an un-isolable RWCU leak outside primary containment and two general area temperatures above the maximum safe limit, the crew will execute N1-EOP-8, RPV Blowdown, in accordance with N1-EOP-5.	An un-isolable primary system discharging outside of Primary Containment resulting in two general area temperatures above the maximum safe limit indicates a wide-spread problem posing a direct and immediate threat to Secondary Containment. A blowdown minimizes flow through the break, rejects heat to the suppression pool in preference to outside the containment, and places the primary system in the lowest possible energy state.

SCENARIO SUMMARY

The scenario begins at approximately 90% power. Containment Spray Pump 112 is out of service for maintenance. Steam Packing Exhauster 12 is out of service due to high vibrations. The crew will start by cross-tying PB 16A to PB 16B. Then, the crew will raise Reactor power to approximately 95% with Recirculation flow.

During the power ascension, a control rod will begin to drift out. The crew will select the drifting control rod and drive it full in. The crew will dispatch an operator to valve out the affected Hydraulic Control Unit to prevent the control rod from continuing to drift.
Then, Feedwater Booster pump 11 will trip. The standby Feedwater Booster pump will fail to auto-start. The crew will manually start the standby Feedwater Booster pump to restore normal system pressures. The SRO will determine the Tech Spec impact for loss of a redundant HPCI component.

RPS UPS 172 will develop an internal fault and drop out the #12 RPS system and RPS Bus 12. The crew will respond to the trip of UPS per N1-SOP-40.1. The SRO will direct the bus be repowered from I&C Bus 130A and will determine the most limiting Tech Spec condition. The BOP and the RO will reset ½ scram and ½ isolations and perform recovery actions after the bus is repowered. The SRO will determine Tech Spec 3.1.2, 3.6.11 and 3.4.4 are the limiting 7 day LCO's applicable with the RPS 12 Bus tripped.

A Reactor Water Cleanup system line break will occur in the Secondary Containment downstream of the Supply Isolation Valves. Reactor Water Cleanup will fail to isolate on high area temperature. The crew will attempt to isolate the system, but the valves will fail to fully close. This break will require a scram **(Critical Task)** and RPV blowdown **(Critical Task)** due to exceeding the Maximum Safe Value for general area temperatures. The Mode Switch will fail to scram the Reactor, however either RPS pushbuttons or manual ARI actuation will result in successful control rod insertion.

Facility: Nine Mile Point Unit 1 Scenario No.: NRC-3 Op-Test No.: LC1 17-1 NRC

Examiners: ______________ Operators: ______________

Initial Conditions: The plant is operating at approximately 100% power. Containment Spray Pump 112 is out of service for maintenance. Steam Packing Exhauster 12 is out of service due to high vibrations.

Turnover: Reduce reactor power to 98% with recirc flow. Then, start TBCLC Pump 12 and secure TBCLC Pump 11.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R-ATC, SRO	Lower reactor power to 98% with recirc flow N1-OP-1
2	N/A	N – BOP, SRO	Swap Running TBCLC Pumps **(2017 Scenario 4),** N1-OP-24
3	ED06	C –BOP, SRO TS-SRO	Powerboard 101 fault N1-SOP-1.3, Technical Specifications
4	RP17B	I-SRO TS-SRO	Reactor Pressure Instrument 36-07C Fails Low Technical Specifications
5	TC06	I – ATC, SRO TS-SRO	EPR Oscillation N1-SOP-31.1, Technical Specifications
6	CW04A CW04B CW04C	C – All	All RBCLC Pumps Trip **(2015 Scenario 5)**, N1-SOP-11.1, N1-SOP-1, N1-EOP-2
7	FW03A FW03B FW06	C – BOP, SRO	Motor Driven Feedwater Pumps Fail to Operate and 13 FW Pump clutch disengages **(2015 Scenario 5)**, N1-EOP-2
8	CU01 EC01	M – All	Coolant Leak Inside Primary Containment **(2015 Scenario 5)**, N1-EOP-2, N1-EOP-4
9	VICP201 68/69	M – All	Fuel Zone Level Instrument Sporadic Indication **(2015 Scenario 5)**, N1-EOP-2, N1-EOP-7

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

Facility: **Nine Mile Point Unit 1** Scenario No.: **NRC-3** Op-Test No.: **LC1 17-1 NRC**		
2. Malfunctions after EOP entry (1-2) **Events 7, 8, 9**	3	
3. Abnormal events (2-4) **Events 3, 5, 6**	3	
4. Major transients (1-2) **Events 8 & 9**	2	
5. EOPs entered/requiring substantive actions (1-2) **N1-EOP-2, N1-EOP-4**	2	
6. EOP contingencies requiring substantive actions (0-2) **N1-EOP-7**	1	
7. Preidentified Critical tasks (2-3)	2	

CRITICAL TASK DESCRIPTIONS:	**CRITICAL TASK JUSTIFICATION:**
CT-1.0: Given a LOCA in the Drywell with the inability to maintain containment parameters within the Pressure Suppression Pressure limit, initiate Containment Sprays, in accordance with N1-EOP-4.	Initiating Containment Sprays reduces Primary Containment pressure. This reduces stresses on the Drywell and Torus, assists in avoiding "chugging" that may cause fatigue failure of the LOCA downcomers, and avoids the need for a blowdown. These benefits reduce challenges to the fuel cladding, the RPV, and the Primary Containment.
CT- 2.0: Given the plant with RPV water level unknown, execute N1-EOP-7, RPV Flooding, in accordance with N1-EOP-2.	With Reactor water level unknown, the status of core cooling is unknown. RPV flooding is required to establish conditions to cool the core. This protects the fuel cladding integrity.

SCENARIO SUMMARY

The scenario begins at approximately 100% power. Containment Spray Pump 112 is out of service for maintenance. Steam Packing Exhauster 12 is out of service due to high vibrations. The crew is directed to lower power to 98% with recirc flow, and then swap TBCLC pump 11 per N1-OP-24 section F.1.

After shifting the TBCLC pumps, a fault will occur causing a loss of Powerboard 101. The crew will respond to a loss of RRP 13 per N1-SOP-1.3. Additional lost loads include Condensate Pump 12, Feedwater Booster Pump 12 and the Electric Fire Pump.

After the loss of PB 101, Reactor pressure instrument 36-07C fails low. The SRO will review Technical Specifications for the loss of automatic scram instrumentation.

Next, EPR oscillations begin. The crew will implement N1-SOP-31.2 place the MPR in service and secure the EPR. SRO will address Technical Specifications.

Next, the running RBCLC pumps will trip. The standby RBCLC pump will trip upon being started. The crew will enter N1-SOP-11.1, RBCLC Failure. The crew will scram the Reactor, trip Recirculation pumps, initiate Emergency Condensers, and shut the MSIVs. The high pressure Feedwater pumps will fail to operate on the scram, complicating Reactor water level control.

Once the crew stabilizes the plant after the scram, a coolant leak will develop inside the Primary Containment. The crew will re-enter N1-EOP-2, RPV Control, and N1-EOP-4, Primary Containment Control. Containment parameters will degrade and the crew will initiate Containment Sprays **(Critical Task)**. The elevated Containment temperature will cause the Fuel Zone level indications to become erratic. With all other Reactor water level indicators downscale, the crew will execute N1-EOP-7, RPV Flooding, to lower Reactor pressure and flood the Reactor to the Main Steam lines **(Critical Task)**.

Facility: Nine Mile Point Unit 1 Scenario No.: NRC-4 Op-Test No.: LC1 17-1 NRC

Examiners: ______________________ Operators: ______________________

Initial Conditions: A plant startup is in progress with reactor power approximately 2-3%. Containment Spray Pump 112 is out of service for maintenance. Steam Packing Exhauster 12 is out of service due to high vibrations.

Turnover: Continue power ascension by withdrawing control rods.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R-ATC, SRO	Raise power with control rods N1-OP-43A, N1-OP-5
2	RD42	C –ATC, SRO	Control Rod Double Notches N1-OP-5
3	RR06A RR07A	I –ATC, SRO	IRM Downscale Failure N1-SOP-1.2,
4	ED12A	C –BOP, SRO	Powerboard 16A Electrical Fault **(2015 Scenario 5)**, ARP L4-3-6, N1-EOP-4
5	RR06A RR07A	C –BOP, SRO TS-SRO	Recirc Pump 11 seal failure requiring isolation of the pump N1-SOP-1.2, Technical Specification 3.2.5, 3.1.7.e
6	PC05 CT04A	C –BOP, SRO TS-SRO	Seismic Event; Isolable Leak on Containment Spray Suction Line N1-SOP-28, N1-EOP-5, Technical Specifications
7	PC05 PC04	M – All	Second Seismic Event; Torus Break; Multiple Control Rods Fail to Insert N1-EOP-5, N1-EOP-4, N1-SOP-1, N1-EOP-2, N1-EOP-8, N1-EOP-3
8	CT02B CT02C	C – All	Containment Spray Raw Water Pumps 112 and 121 Trips N1-EOP-4

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

Facility: **Nine Mile Point Unit 1**	Scenario No.: **NRC-4**	Op-Test No.: **LC1 17-1 NRC**
1. Malfunctions after EOP entry (1-2) **Events 8**	1	
2. Abnormal events (2-4) **Events 3, 4, 5, 6**	4	
3. Major transients (1-2) **Event 7**	1	
4. EOPs entered/requiring substantive actions (1-2) **N1-EOP-2, N1-EOP-4, N1-EOP-5**	3	
5. EOP contingencies requiring substantive actions (0-2) **N1-EOP-3, N1-EOP-8**	2	
6. Preidentified Critical tasks (2-3)	2	

CRITICAL TASK DESCRIPTIONS:	**CRITICAL TASK JUSTIFICATION:**
CT-1.0: Given an un-isolable Torus leak exceeding makeup capacity, scram the Reactor, in accordance with N1-EOP-4.	Lowering Torus water level challenges the pressure suppression function of the Primary Containment. Continued Reactor operation is not allowed with an inoperable Primary Containment. A Reactor scram also allows subsequent mitigating actions, such as Reactor cooldown and/or blowdown.
CT- 2.0: Given an un-isolable Torus leak exceeding makeup capacity, perform an RPV Blowdown, in accordance with N1-EOP-4.	If torus water level lowers below the elevation of the ERV discharge holes, opening ERVs would discharge steam directly into the torus airspace. The resulting pressure increase could exceed the maximum pressure capability of the Primary Containment. Since the RPV may not be kept at pressure under these conditions, a blowdown is required.

SCENARIO SUMMARY

The scenario begins with the plant in a startup at approximately 2-3% power and raising power by control rod withdrawal. While withdrawing control rods, one control rod will double notch to one position past its intended position. The crew will respond per N1-OP-5 section H.9.0 and re-insert the control rod to the intended position.

Next, IRM 11 will fail downscale. The SRO will determine the impact of the failure on Tech Specs and the crew will bypass the IRM.

Next, Powerboard 16A will develop an electrical fault. This will cause a loss of power to three Drywell cooling fans. The crew will start an additional Drywell cooling fan to stabilize Drywell temperature and pressure. The electrical loss will also affect EDG 103 auxiliary equipment. The US will evaluate for Tech Spec impacts.

Then, the inner seal will fail on Reactor Recirculation Pump 11. A few minutes later, the outer seal will fail, affecting drywell leakage. The crew will remove the pump from service and isolate it. The SRO will review Technical Specifications for drywell leakage and partial loop operation.

Then, a seismic event occurs and results in an isolable leak on Containment Spray pump 121 suction line. The crew will execute N1-SOP-28, Seismic Event, and N1-EOP-5, Secondary Containment Control. The crew will close Containment Spray suction valve 121 to isolate the leak. The SRO will address Tech Specs.

Next, a second seismic event occurs and results in an un-isolable Torus break. The crew will re-enter N1-EOP-5 and enter N1-EOP-4, Primary Containment Control. The crew will attempt to add water to the Torus, however makeup efforts will be complicated by trip of Containment Spray Raw Water pumps. The crew will be unable to raise Torus water level. The crew will insert a manual Reactor scram **(Critical Task)**. Multiple control rods will fail to fully insert. The crew will enter N1-EOP-2, RPV Control, and transition to N1-EOP-3, Failure to Scram. The crew will be unable to drive control rods in with RMCS. As Torus water level lower further, the crew will perform an RPV Blowdown per N1-EOP-8 **(Critical Task)**. During the RPV Blowdown, the crew will terminate and prevent all injection except boron and CRD and later re-inject to restore/maintain Reactor water level above the top of active fuel.

Facility: **Nine Mile Point Unit 1** Date of Exam: **December 2018**

<table>
<tr><th rowspan="2">Tier</th><th rowspan="2">Group</th><th colspan="12">RO K/A Category Points</th><th colspan="4">SRO-Only Points</th></tr>
<tr><th>K1</th><th>K2</th><th>K3</th><th>K4</th><th>K5</th><th>K6</th><th>A1</th><th>A2</th><th>A3</th><th>A4</th><th>G*</th><th>Total</th><th colspan="2">A2</th><th>G*</th><th>Total</th></tr>
<tr><td rowspan="3">1. Emergency and Abnormal Plant Evolutions</td><td>1</td><td>3</td><td>4</td><td>2</td><td colspan="3" rowspan="3">N/A</td><td>4</td><td>3</td><td colspan="2" rowspan="3">N/A</td><td>4</td><td>20</td><td colspan="2">3</td><td>4</td><td>7</td></tr>
<tr><td>2</td><td>1</td><td>1</td><td>1</td><td>2</td><td>1</td><td>1</td><td>7</td><td colspan="2">2</td><td>1</td><td>3</td></tr>
<tr><td>Tier Totals</td><td>4</td><td>5</td><td>3</td><td>6</td><td>4</td><td>5</td><td>27</td><td colspan="2">5</td><td>5</td><td>10</td></tr>
<tr><td rowspan="3">2. Plant Systems</td><td>1</td><td>3</td><td>2</td><td>3</td><td>3</td><td>2</td><td>2</td><td>2</td><td>3</td><td>2</td><td>2</td><td>2</td><td>26</td><td colspan="2">3</td><td>2</td><td>5</td></tr>
<tr><td>2</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>2</td><td>0</td><td>1</td><td>2</td><td>1</td><td>1</td><td>12</td><td>0</td><td>1</td><td>2</td><td>3</td></tr>
<tr><td>Tier Totals</td><td>4</td><td>3</td><td>4</td><td>4</td><td>3</td><td>4</td><td>2</td><td>4</td><td>4</td><td>3</td><td>3</td><td>38</td><td colspan="2">4</td><td>4</td><td>8</td></tr>
<tr><td colspan="5" rowspan="2">3. Generic Knowledge and Abilities Categories</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">4</td><td rowspan="2">10</td><td>1</td><td>2</td><td>3</td><td>4</td><td rowspan="2">7</td></tr>
<tr><td colspan="2">3</td><td colspan="2">2</td><td colspan="2">2</td><td colspan="2">3</td><td>2</td><td>1</td><td>2</td><td>2</td></tr>
</table>

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-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 Form ES-401-1
Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/**SRO**)

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4					X		AA2.02, Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Neutron monitoring	3.1	27
295003 (APE 3) Partial or Complete Loss of AC Power / 6				X			AA1.01, Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: A.C. electrical distribution system	3.7	29
295004 (APE 4) Partial or Complete Loss of DC Power / 6	X						AK1.04, Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Effect of battery discharge rate on capacity	2.8	30
295005 (APE 5) Main Turbine Generator Trip / 3						X	G.2.4.31, Knowledge of annunciator alarms, indications, or response procedures.	4.2	31
295006 (APE 6) Scram / 1		X					AK2.06, Knowledge of the interrelations between SCRAM and the following: Reactor power	4.2	32
						X	**G2.1.19, Ability to use plant computers to evaluate system or component status.**	**3.8**	**76**
295016 (APE 16) Control Room Abandonment / 7						X	G2.4.35, Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	3.8	33
295018 (APE 18) Partial or Complete Loss of CCW / 8	X						AK1.01, Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations	3.5	34
295019 (APE 19) Partial or Complete Loss of Instrument Air / 8			X				AK3.03, Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Service air isolations: Plant-Specific	3.2	35
295021 (APE 21) Loss of Shutdown Cooling / 4						X	G2.2.37, Ability to determine operability and/or availability of safety related equipment.	3.6	36
					X		**AA2.06, Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor pressure**	**3.3**	**77**
295023 (APE 23) Refueling Accidents / 8	X						AK1.01, Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: Radiation exposure hazards	3.6	37
						X	**G2.2.37, Ability to determine operability and/or availability of safety related equipment.**	**4.6**	**78**
295024 High Drywell Pressure / 5		X					EK2.15, Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Containment spray logic: Plant-Specific	3.8	38

295025 (EPE 2) High Reactor Pressure / 3						X	G2.4.8, Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	3.8	39
295026 (EPE 3) Suppression Pool High Water Temperature / 5				X			EA1.03, Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature monitoring	3.9	28
					X		**EA2.01, Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature**	**4.2**	**79**
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5			X				EK3.06, Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: ADS	3.4	40
					X		**EA2.01, Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell temperature**	**4.1**	**80**
295030 (EPE 7) Low Suppression Pool Water Level / 5					X		EA2.04, Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Drywell/ suppression chamber differential pressure: Mark I & II	3.5	41
295031 (EPE 8) Reactor Low Water Level / 2				X			EA1.13, Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: Reactor water level control	4.3	42
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1				X			EA1.01, Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor Protection System	4.6	43
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9		X					EK2.08, Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: SPDS/ERIS/CRIDS/GDS: Plant-Specific.	2.6	44
						X	**G2.4.41, Knowledge of the emergency action level thresholds and classifications.**	**4.6**	**81**
600000 (APE 24) Plant Fire On Site / 8					X		AA2.05, Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Ventilation alignment necessary to secure affected area	2.9	45
						X	**2.1.7, Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.**	**4.7**	**82**
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6		X					AK2.07, Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Turbine/generator control	3.6	46
K/A Category Totals:	3	4	2	4	3/**3**	4/**4**	RO/**SRO** Group Point Total:		20/**7**

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO/**SRO**)		Form ES-401-1
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	Q#
295002 (APE 2) Loss of Main Condenser Vacuum / 3	X						AK1.03, Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM: Loss of heat sink	3.6	47
295007 (APE 7) High Reactor Pressure / 2						**X**	**G2.2.42, Ability to recognize system parameters that are entry-level conditions for Technical Specifications.**	**4.6**	**83**
295009 (APE 9) Low Reactor Water Level / 2				X			AA1.04, Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor water cleanup	2.7	48
295012 (APE 12) High Drywell Temperature / 5			X				AK3.01, Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: Increased drywell cooling	3.5	49
295013 (APE 13) High Suppression Pool Temperature. / 5						X	G2.4.20, Knowledge of the operational implications of EOP warnings, cautions, and notes.	3.8	50
295015 (APE 15) Incomplete Scram / 1				X			AA1.05, Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: Rod worth minimizer: Plant-Specific	2.5	51
295022 (APE 22) Loss of Control Rod Drive Pumps / 1		X					AK2.07, Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: Reactor pressure (SCRAM assist): Plant-Specific	3.4	52
295029 (EPE 6) High Suppression Pool Water Level / 5					X		EA2.03, Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level	3.4	53
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9					**X**		**EA2.03, Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Cause of high area radiation**	**4.2**	**84**
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5					**X**		**EA2.03, Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Cause of the high water level**	**3.8**	**85**
K/A Category Point Totals:	1	1	1	2	1/**2**	1/**1**	RO/**SRO** Group Point Total:		7/**3**

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

System # / Name	K 1	K 2	K 3	K4	K 5	K 6	A 1	A 2	A 3	A 4	G *	K/A Topic(s)	IR	Q#
205000 (SF4 SCS) Shutdown Cooling			X									K3.02, Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor water level: Plant-Specific	3.2	6
								X				**A2.02, Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low shutdown cooling suction pressure: Plant-Specific**	**2.7**	**86**
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection		X										K2.01, Knowledge of electrical power supplies to the following: System valves: BWR-2,3,4	3.2	3
				X								K4.07, Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Automatic system initiation: BWR-2,3,4	4.3	24
207000 (SF4 IC) Isolation (Emergency) Condenser						X						K6.04, Knowledge of the effect that a loss or malfunction of the following will have on the ISOLATION (EMERGENCY) CONDENSER: Plant air systems: BWR-2,3	3.2	11
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray	X											K1.09, Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Nuclear boiler instrumentation	3.2	1
								X				**A2.01, 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: Pump trips**	**3.4**	**87**
211000 (SF1 SLCS) Standby Liquid Control							X					A1.03, Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: Pump discharge pressure	3.6	13

212000 (SF7 RPS) Reactor Protection					X							K5.02, Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements	3.3	9
											X	**2.4.46, Ability to verify that the alarms are consistent with the plant conditions.**	**4.2**	**88**
215003 (SF7 IRM) Intermediate-Range Monitor					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
215004 (SF7 SRMS) Source-Range Monitor				X								K4.02, Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Reactor SCRAM signals	3.4	7
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor							X					A1.07, Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: APRM (gain adjustment factor)	3.0	14
											X	G2.1.32, Conduct of Operations: Ability to explain and apply all system limits and precautions.	3.8	22
218000 (SF3 ADS) Automatic Depressurization								X				A2.01, Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Small steam line break LOCA	4.1	16
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff									X			A3.01, Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: System indicating lights and alarms	3.4	17
								X				**A2.04, Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Process radiation monitoring system failures**	**3.2**	**89**

239002 (SF3 SRV) Safety Relief Valves								X				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	3.0	15
	X											K1.07, Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: Suppression Pool	3.6	23
259002 (SF2 RWLCS) Reactor Water Level Control										X		A4.01, Ability to manually operate and/or monitor in the control room: All individual component controllers in the manual mode	3.8	20
											X	**G2.4.9, Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.**	**4.2**	**90**
261000 (SF9 SGTS) Standby Gas Treatment						X						K6.03, Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : Emergency diesel generator system	3.0	12
								X				A2.07, 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: A.C. electrical failure	2.7	25
262001 (SF6 AC) AC Electrical Distribution				X								K4.03, Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Interlocks between automatic bus transfer and breakers	3.1	8
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)			X									K3.10 - Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: Containment isolation: Plant-Specific	2.7	5
263000 (SF6 DC) DC Electrical Distribution									X			A3.01, Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights	3.2	18
			X									K3.03, Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Systems with D.C. components (i.e. valves, motors, solenoids, etc.)	3.4	26

264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG										X		A4.04, Ability to manually operate and/or monitor in the control room: Manual start, loading, and stopping of emergency generator: Plant-Specific	3.7	19
300000 (SF8 IA) Instrument Air		X										K2.02, Knowledge of electrical power supplies to the following: Emergency air compressor	3.0	4
											X	2.1.30, Conduct of Operations: Ability to locate and operate components, including local controls.	4.4	21
400000 (SF8 CCS) Component Cooling Water	X											K1.02, Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following: Loads cooled by CCWS	3.2	2
K/A Category Point Totals:	3	2	3	3	2	2	2	3/ **3**	2	2	2/ **2**	RO/**SRO** Group Point Total:		26/5

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	Q#
201002 (SF1 RMCS) Reactor Manual Control								**X**				**A2.01, Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod movement sequence timer malfunctions**	**2.8**	**91**
201003 (SF1 CRDM) Control Rod and Drive Mechanism											**X**	**G2.2.38, Knowledge of conditions and limitations in the facility license.**	**4.5**	**92**
201006 (SF7 RWMS) Rod Worth Minimizer								X				A2.01, Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply loss: P-Spec(Not-BWR6)	2.5	54
202001 (SF1, SF4 RS) Recirculation											**X**	**G2.2.36, Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.**	**4.2**	**93**
202002 (SF1 RSCTL) Recirculation Flow Control									X			A3.03, Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Scoop tube operation: BWR-2,3,4	3.1	55
215001 (SF7 TIP) Traversing In-Core Probe				X								K4.01, Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation: Mark I & II (Not-BWR1)	3.4	56
223001 (SF5 PCS) Primary Containment and Auxiliaries											X	G2.4.3, Ability to identify post-accident instrumentation.	3.7	57
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode						X						K6.11, Knowledge of the effect that a loss or malfunction of the following will have on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE: Component cooling water systems	2.8	58
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary										X		A4.09, Ability to manually operate and/or monitor in the control room: Hydrogen seal oil pressure	2.6	59

259001 (SF2 FWS) Feedwater						X						K6.06, Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM: Plant service water	2.7	60
268000 (SF9 RW) Radwaste	X											K1.05, Knowledge of the physical connections and/or cause-effect relationships between RADWASTE and the following: Drywell equipment drains	2.9	61
272000 (SF7, SF9 RMS) Radiation Monitoring		X										K2.05, Knowledge of electrical power supplies to the following: Reactor building ventilation monitors: Plant-Specific	2.6	62
288000 (SF9 PVS) Plant Ventilation					X							K5.02, Knowledge of the operational implications of the following concepts as they apply to PLANT VENTILATION SYSTEMS: Differential pressure control	3.2	63
290001 (SF5 SC) Secondary Containment									X			A3.01, Ability to monitor automatic operations of the SECONDARY CONTAINMENT including: Secondary containment isolation	3.9	64
290002 (SF4 RVI) Reactor Vessel Internals			X									K3.01, Knowledge of the effect that a loss or malfunction of the REACTOR VESSEL INTERNALS will have on following: Reactor water level	3.2	65
K/A Category Point Totals:	1	1	1	1	1	2	0	1/**1**	2	1	1/**2**	RO/**SRO** Group Point Total:		12/3

Facility: Nine Mile Point Unit 1		Date of Exam: January 2019				
Category	K/A #	Topic	RO		SRO-only	
			IR	Q#	IR	Q#
1. Conduct of Operations	2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.	3.3	66		
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management.	4.3	67		
	2.1.31	**Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.**			**4.3**	**94**
	2.1.40	**Knowledge of refueling administrative requirements.**			**3.9**	**95**
	2.1.43	Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.	4.1	68		
	Subtotal			3		**2**
2. Equipment Control	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	4.6	69		
	2.2.21	**Knowledge of pre- and post-maintenance operability requirements.**			**4.1**	**96**
	2.2.22	Knowledge of limiting conditions for operations and safety limits.	4.0	70		
	Subtotal			2		**1**
3. Radiation Control	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	71		
	2.3.7	**Ability to comply with radiation work permit requirements during normal or abnormal conditions.**			**3.6**	**97**
	2.3.13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	3.4	72		
	2.3.14	**Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.**			**3.8**	**98**
	Subtotal			2		**2**
4. Emergency Procedures/Plan	2.4.14	Knowledge of general guidelines for EOP usage.	3.8	73		
	2.4.17	**Knowledge of EOP terms and definitions.**			**4.3**	**99**

	2.4.21	**Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.**			**4.6**	**100**
	2.4.32	Knowledge of operator response to loss of all annunciators.	3.6	74		
	2.4.22	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.	3.6	75		
	Subtotal			3		**2**
Tier 3 Point Total				10		**7**

Tier / Group	Randomly Selected K/A	Reason for Rejection
The systematic and random sampling process utilized the pre-approved Nine Mile Point Unit 1 K/A suppression list.		
The following K/As were rejected following the systematic and random sampling process:		
2 / 1	Question 9 212000 RPS K5.01 - Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Fuel thermal time constant	A discriminating question could not be developed without testing generic fundamentals knowledge. Randomly reselected K/A 212000 RPS K5.02 - Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements.
2 / 1	Question 14 215005 APRM / LPRM A1.06 - Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: Recirculation flow control valve position: Plant-Specific	This facility does not have recirculation flow control valves. Randomly reselected K/A 215005 A1.07 - Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: APRM (gain adjustment factor).
2 / 1	Question 23 239002 SRVs K1.05 - Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: Plant air systems: Plant-Specific	An acceptable question could not be written for the randomly selected K/A due to limited interrelations between SRVs and plant air systems. Randomly reselected K/A 239002 K1.07 - Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: Suppression Pool.

2 / 1	Question 25 261000 SGTS A2.14 - 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 system pressure: Plant-Specific	There are no interlocks or initiations for RBEVS (SGTS) related to high system pressure at this facility. An acceptable question could not be developed without testing minutia. Randomly reselected K/A 261000 A2.07 - 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: A.C. electrical failure.
1 / 1	Question 28 295001 Partial or Complete Loss of Forced Core Flow Circulation AA1.03 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: RMCS: Plant-Specific	295001 was inadvertently sampled twice on the RO exam prior to sampling 295026. Reselected 295026 Suppression Pool High Water Temperature and randomly reselected EA1.03 - Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature monitoring.
1 / 1	Question 36 295021 Loss of Shutdown Cooling 2.2.25 - Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	An acceptable question could not be developed for the randomly sampled K/A due to lack of Technical Specification bases related to loss of Shutdown Cooling that are RO level. Randomly reselected K/A 295021 Loss of Shutdown Cooling 2.2.37 - Ability to determine operability and/or availability of safety related equipment.
1 / 1	Question 40 295028 High Drywell Temperature EK3.04 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: Increased drywell cooling	An acceptable question could not be developed for the randomly sampled K/A without overlapping Question 49. Randomly reselected K/A 295028 High Drywell Temperature EK3.06 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: ADS.

2 / 2	Question 63 288000 Plant Ventilation K5.03 - Knowledge of the operational implications of the following concepts as they apply to PLANT VENTILATION SYSTEMS: Temperature control	An acceptable question could not be developed for the randomly sampled K/A without testing minutia due to a lack of operationally relevant references related to Plant Ventilation temperature control. Randomly reselected K/A 288000 Plant Ventilation K5.02 - Knowledge of the operational implications of the following concepts as they apply to PLANT VENTILATION SYSTEMS: Differential pressure control.
2 / 2	Question 64 290001 Secondary Containment A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the SECONDARY CONTAINMENT controls including: System lineups	An acceptable question could not be developed for the randomly sampled K/A due to lack of Secondary Containment controls and associated system lineups. Randomly reselected K/A 290001 Secondary Containment A3.01 - Ability to monitor automatic operations of the SECONDARY CONTAINMENT including: Secondary containment isolation.
3	Question 67 2.1.19 - Ability to use plant computers to evaluate system or component status.	An acceptable question could not be developed for the randomly sampled K/A without oversampling plant computer topics (see Questions 44 & 76). Use of plant computers is also tested extensively on the operating exam. Randomly reselected K/A 2.1.37 - Knowledge of procedures, guidelines, or limitations associated with reactivity management.
3	Question 75 2.4.35 - Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	The randomly sampled generic K/A is also tested on Question 33. Reselecting for better balance of coverage. Randomly reselected K/A 2.4.22 - Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.
2 / 1	Question 88 212000 Reactor Protection 2.4.34 - Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	An acceptable question could not be developed for the randomly sampled K/A due to lack of RO tasks performed outside the control room related to the Reactor Protection System. Randomly reselected K/A 212000 Reactor Protection 2.4.46 - Ability to verify that the alarms are consistent with the plant conditions.

2 / 2	Question 91 201002 Reactor Manual Control A2.02 - Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod drift alarm	An acceptable question could not be developed for the randomly sampled K/A without overlapping the operating exam. Additionally, the K/A did not readily support testing at the SRO level. Randomly reselected K/A 201002 Reactor Manual Control A2.01 - Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod movement sequence timer malfunctions.
2 / 2	Question 93 202001 Recirculation 2.2.12 - Knowledge of surveillance procedures.	An acceptable question could not be developed for the randomly sampled K/A due to lack of surveillance procedures for the Recirculation system. Randomly reselected K/A 202001 Recirculation 2.2.36 - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.
1 / 2	Question 83 295008 High Reactor Pressure	Editorial error - APE 295008 does not coincide with High Reactor Pressure. Conferred with Chief Examiner to change 295008 to 295007 to coincide with High Reactor Pressure