10 CFR 55.40

NMP1L3120
December 6, 2016

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 Examinations

Reference (1) Letter from D. E. Jackson (NRC) to B. Hanson (Exelon Nuclear), dated August 11, 2016, 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 February 6, 2017. The examinations are being prepared based on guidelines in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 10.

Enclosed are the draft examinations, operating tests, and the supporting reference materials for the Unit 1 Senior Reactor and Reactor Operator initial license examinations, as requested in Reference (1).

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,

for Jim Tsardakas
Greg Elkins (Acting Director Site Training)

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

JNT/RSP

Enclosure: Draft examinations, operating tests, and the supporting reference materials for the Unit 1 Senior Reactor and Reactor Operator initial license examinations

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

Enclosure

Draft examinations, operating tests, and the supporting reference materials for the Unit 1

Senior Reactor and Reactor Operator initial license examinations

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

Facility: Nine Mile Point Unit 1 Scenario No.: NRC-1 Op-Test No.: LC1 15-1

Examiners: ____________ Operators: ____________

Initial Conditions: The plant is operating at approximately 1% in 5 loop operation during a startup. ESW Pump 12 is out of service for maintenance.

Turnover: The reactor is critical at approximately 1% power. Perform N1-ST-W15, Manual and Automatic Scram Instrument Channel Test. Continue the startup IAW N1-OP-43A, starting with control rod withdrawals.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N –BOP, SRO	Perform N1-ST-W15, Sections 6.2 and 6.3 (partial), Channel 11 Auto and Manual Scram Tests N1-ST-W15
2	N/A	R - ATC, SRO	Withdraw control rods N1-OP-43A, N1-OP-5
3	RD04	C –ATC, SRO SRO-TS	Stuck control rod N1-OP-5, Technical Specifications
4	NM01C	I – ATC, SRO SRO-TS	SRM Fails Upscale N1-OP-38A, Technical Specifications
5	CW05A CW05B	C –All	Loss of all TBCLC requires scram and MSIV isolation N1-SOP-24.1, N1-SOP-1
6	RP05A RP05B	I –BOP, SRO	RPS fails to scram the reactor, ARI is successful N1-EOP-2, N1-SOP-1
7	EC02	M –All	EC 11 Steam Leak in the Reactor Building ARP K1-1-1, K1-3-4, N1-SOP-1, N1-EOP-2, N1-EOP-5
8	EC07A EC08A EC08B	C –All	EC 11 will not isolate, requiring a RPV Blowdown N1-EOP-8

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

Facility: **Nine Mile Point Unit 1** Scenario No.: **NRC-1**		Op-Test No.: **LC1 15-1**
1. Total malfunctions (5-8) **Events 3, 4, 5, 6, 7, 8**	6	
2. Malfunctions after EOP entry (1-2) **Event 6, 7, 8**	3	
3. Abnormal events (2-4) **Events 3, 4, 5, 6**	4	
4. Major transients (1-2) **Event 7**	1	
5. EOPs entered/requiring substantive actions (1-2) **N1-EOP-2, N1-EOP-5**	2	
6. EOP contingencies requiring substantive actions (0-2) **N1-EOP-8**	1	
7. EOP Based Critical tasks (2-3)	2	

CRITICAL TASK DESCRIPTIONS:	CRITICAL TASK JUSTIFICATION:
CT-1.0: Given a complete loss of TBCLC, the crew will insert a manual reactor scram in accordance with N1-SOP-24.1.	TBCLC provides cooling to major heat loads in the plant, including Recirc Pump MG sets and Instrument Air Compressors. Either of which, when lost, greatly reduces the level of safety of the reactor.
CT- 2.0: Given conditions requiring a scram and failure of an RPS channel to trip, the crew will manually initiate Alternate Rod Insertion (ARI) to shutdown the reactor, in accordance with N1-SOP-1 and/or N1-EOP-3.	Inserting control rods during a transient lowers reactor power, which reduces challenges to the plant during the transient. With RPS failing to trip, the crew must rely on backup shutdown methods to ensure control rods are inserted to provide long term, stable, core shutdown.
CT- 3.0: Given an un-isolable Emergency Condenser leak outside Primary Containment and two general area temperatures above the maximum safe limit, 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 1% power with a startup in progress. ESW pump 12 is out of service for maintenance. IRM 11 is bypassed due to spiking. Control rod withdrawal has been halted for the turnover and to permit the crew to perform N1-ST-W15, Manual Scram Instrument Channel Test, sections 6.1 to 6.3 for Channel 11 only. The surveillance test results will be satisfactory.

After completion of the surveillance test the crew will resume withdrawing control rods in approach to criticality. The second control rod to be moved will be stuck. The crew will enter N1-OP-5, Section H.13 and raise drive water pressure to move the control rod. The control rod will withdraw with increased drive water pressure.
Next, an SRM upscale trip will occur. The crew will respond per ARP F3-4-1 and N1-OP-38A to bypass the SRM.

Then, a complete loss of TBCLC occurs requiring a manual reactor Scram (**Critical Task**). The Mode Switch will fail to scram the Reactor, however manual ARI actuation will result in successful control rod insertion (**Critical Task**).

Next, a steam leak will develop from Emergency Condenser 11. The crew will attempt to isolate the leak, however the Emergency Condenser will fail to isolate both automatically and manually. Two General Areas of the Reactor Building will exceed the maximum safe temperatures. The crew will blowdown the Reactor per N1-EOP-8 (**Critical Task**).

Facility: Nine Mile Point Unit 1 Scenario No.: NRC-2 Op-Test No.: LC1 15-1

Examiners: ______________ Operators: ______________

Initial Conditions: The plant is operating at approximately 100% power. ESW Pump 12 is out of service for maintenance. IRM 11 is bypassed due to spiking. PB 11 is aligned to reserve power in preparation for cross-tying PB 16.

Turnover: Lower reactor power to 98% using recirc flow. Cross-tie PB 16A to PB 16B with PB 16B supplying. Then, return PB-11 to normal.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R-ATC, SRO	Lower reactor power with recirc. N1-OP-1
2	N/A	N-BOP, SRO	Cross-tie PB 16A to PB 16B N1-OP-30
3	RD02	C –ATC, SRO	Control Rod 26-35 Drifts Out N1-SOP-5.2
4	FW02A Override	C – BOP, SRO	Feedwater Booster Pump 11 Trips with Failure of Feedwater Booster Pump 13 to Auto-start N1-SOP-16.1, Technical Specifications
5	RP25	C-All	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 N1-EOP-2, N1-EOP-5
7	CU14	C -All	Failure of the RWCU Isolation Valves to isolate N1-EOP-5, N1-EOP-8
8	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 15-1**
1. Total malfunctions (5-8) **Events 3, 4, 5, 6, 7, 8**	6	
2. Malfunctions after EOP entry (1-2) **Events 7, 8**	2	
3. Abnormal events (2-4) **Events 3, 4, 5**	3	
4. Major transients (1-2) **Event 6**	1	
5. EOPs entered/requiring substantive actions (1-2) **N1-EOP-2, N1-EOP-5**	2	
6. EOP contingencies requiring substantive actions (0-2) **N1-EOP-8**	1	
7. EOP Based 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 100% power. ESW pump 12 is out of service for maintenance. IRM 11 is bypassed due to spiking. The crew will start by lowering Reactor power to approximately 98% with Recirculation flow. Then the crew will cross-tie PB 16A to PB 16B.

Following the power board transfer, 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 15-1

Examiners: ______________ Operators: ______________

Initial Conditions: The plant is operating at approximately 100% power. ESW Pump 12 is out of service for maintenance. IRM 11 is bypassed due to spiking.

Turnover: Remove Line 4 from service per N1-OP-33A to support National Grid maintenance.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Remove Line 4 from Service N1-OP-33A, Technical Specifications
2	AD05	C – BOP, SRO R – ATC	ERV Inadvertently Opens N1-SOP-1.4, N1-SOP-1.1, Technical Specifications
3	ED04	C – All	Powerboard 11 Electrical Fault N1-SOP-30.1, N1-SOP-1.3, N1-SOP-1.1, Technical Specifications
4	EC01	M – All	Steam Leak in Primary Containment N1-SOP-1, N1-EOP-2, N1-EOP-4, N1-EOP-8
5	PC10A PC10C	C – All	Torus to Drywell Vacuum Breaker Inadvertently Opens N1-EOP-4
6	FW28A FW28B FW06 CS07	C – BOP, SRO	HPCI Fails to Auto-Initiate, Feedwater Pump 13 Disengages, and Core Spray Valves Fail to Auto-Open N1-EOP-2
7	Overrides	C – ATC, SRO	Partial Primary Containment Isolation Failure N1-SOP-40.2
8	CT01A	C – All	Containment Spray Pump 111 Trips N1-EOP-4

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

Facility: **Nine Mile Point Unit 1**	Scenario No.: **NRC-3**	Op-Test No.: **LC1 15-1**
1. Total malfunctions (5-8) **Events 2, 3, 4, 5, 6, 7, 8**	7	
2. Malfunctions after EOP entry (1-2) **Events 5, 6, 7, 8**	4	
3. Abnormal events (2-4) **Events 2, 3, 7**	3	
4. Major transients (1-2) **Event 4**	1	
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-8**	1	
7. EOP Based Critical tasks (2-3)	2	

CRITICAL TASK DESCRIPTIONS:	CRITICAL TASK JUSTIFICATION:
CT-1.0: Given an inadvertently open ERV at power, close the ERV or insert a manual scram prior to Torus temperature exceeding 110°F, in accordance with N1-SOP-1.4	A manual Reactor scram is required before Torus temperature exceeds 110°F. This reduces the rate of energy production and thus heat input to the Torus. Additionally, this allows evaluating the success of the Reactor scram before boron injection would be required due to Torus temperature in the event of a failure to scram. Closing the ERV prior to the need for the scram avoids the need for these more substantial actions, prevents challenging the plant with a scram, and stops heat input to the Torus.
CT-2.0: Given a LOCA in the Drywell and a failure of HPCI to initiate, inject with preferred and alternate injection systems to restore and maintain RPV water level above -84 inches, in accordance with N1-EOP-2.	Maintaining Reactor water level above -84 inches ensures adequate core cooling through the preferred method of core submergence. This protects the integrity of the fuel cladding.
CT- 3.0: Given a LOCA in the Drywell and degraded Containment Spray capability, execute N1-EOP-8, RPV Blowdown, when it is determined Torus pressure cannot be maintained below the Pressure Suppression Pressure limit, in accordance with N1-EOP-4.	A Blowdown is required to limit further release of energy into the Primary Containment and to ensure that the RPV is depressurized while pressure suppression capability is still available. This protects the integrity of the Primary Containment.

SCENARIO SUMMARY

The scenario begins at approximately 100% power. ESW pump 12 is out of service for maintenance. IRM 11 is bypassed due to spiking. The crew will remove Line 4 from service for maintenance. The SRO will determine the Tech Spec impact.

Next, ERV 111 will inadvertently open. The crew will enter N1-SOP-1.4, Stuck Open ERV. The crew will perform an emergency power reduction to approximately 85% power, then take actions to close ERV 111 **(Critical Task)**. These actions will close the ERV, but leave it inoperable. The SRO will determine the Tech Spec impact.

Next, Powerboard 11 will de-energize due to an electrical fault. This will cause loss of multiple major loads, including a second Recirculation pump, a Service Water pump, and a Circulating Water pump. The crew will respond per N1-SOP-30.1. This will include lowering Reactor power to restore the plant within single Circulating Water pump operating limitations. The SRO will determine the Tech Spec impact of this power loss.

Next, a steam leak will develop inside Primary Containment. The crew will scram the Reactor and execute N1-EOP-2, RPV Control, and N1-EOP-4, Primary Containment Control. After the scram, Feedwater pump 13 will dis-engage early and Feedwater will fail to automatically shift to the HPCI flow-control mode on low Reactor water level. At lower Reactor pressure, Core Spray Isolation Valves will also fail to automatically open. The crew will be able to restore and maintain Reactor water level by manually injecting with preferred and alternate systems (**Critical Task**). Multiple primary containment isolation valves will fail to close on either manual or automatic containment isolation. The crew will be able to manually close these valves. Two Torus-to-Drywell vacuum breakers will fail open, resulting in some steam escaping from the Drywell directly into the Torus airspace. When the crew initiates Containment Spray, Containment Spray pump 111 will trip. These failures will further degrade Primary Containment pressure control. The Pressure Suppression Pressure (PSP) will be exceeded. The crew will blowdown the Reactor per N1-EOP-8 (**Critical Task**).

Facility: Nine Mile Point Unit 1   Scenario No.: NRC-4   Op-Test No.: LC1 15-1

Examiners: ______________________   Operators: ______________________

______________________   ______________________

______________________   ______________________

Initial Conditions: The plant is operating at approximately 95% power. ESW Pump 12 is out of service for maintenance. IRM 11 is bypassed due to spiking.

Turnover: Start TBCLC Pump 12 and secure TBCLC Pump 11. Then, raise Reactor power to 98% with Recirculation flow.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Swap Running TBCLC Pumps N1-OP-24
2	N/A	R – ATC, SRO	Raise Reactor Power with Recirculation Flow N1-OP-1, N1-OP-43B
3	NM19A	I-ATC, SRO	APRM #13 fails upscale, half scram, bypass. ARP Tech Spec
4	RD36A	C –BOP, SRO	Control Rod Drive Flow Control Valve 44-151 fails closed, requiring shifting to the alternate FCV N1-SOP-5.1, Tech Spec
5	CW16A	C-BOP, SRO	Service Water Adams Strainer 11 High D/P N1-OP-18, ARP H1-3-2
6	MC01	C –ATC, SRO	Loss of main condenser vacuum N1-SOP-25.1, N1-SOP-1.1, N1-SOP-1
7	RD33	M -ALL	ATWS N1-EOP-2, N1-EOP-3
8	TC12	C -ALL	All Turbine Bypass Valves Fail Closed N1-EOP-8

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

Facility: **Nine Mile Point Unit 1** Scenario No.: **NRC-4**		Op-Test No.: **LC1 15-1**
1. Total malfunctions (5-8) **Events 3, 4, 5, 6, 7, 8**	6	
2. Malfunctions after EOP entry (1-2) **Event 8**	2	
3. Abnormal events (2-4) **Events 3, 4, 5**	3	
4. Major transients (1-2) **Event 6**	1	
5. EOPs entered/requiring substantive actions (1-2) **N1-EOP-2**	1	
6. EOP contingencies requiring substantive actions (0-2) **N1-EOP-3, N1-EOP-8**	2	
7. EOP Based Critical tasks (2-3)	3	

CRITICAL TASK DESCRIPTIONS:	CRITICAL TASK JUSTIFICATION:
CT-1.0: Given a failure of the reactor to scram with power above 6% and RPV water level above -41 inches, the crew will terminate and prevent all injection except boron and CRD, in accordance with N1-EOP-3.	High Reactor power after a scram is attempted indicates a challenge to nuclear fuel and to plant heat sinks. In the event of a loss of the normal heat sink, this may result in adding heat to the Torus and challenging the Primary Containment. Lowering Reactor power reduces these challenges.
CT- 2.0: Given a failure of the reactor to scram with power above 6%, the crew will lower reactor power by inserting control rods or injecting boron, in accordance with N1-EOP-3.	Inserting control rods lowers Reactor power, which reduces challenges to the plant during a failure to scram. Additionally, inserting control rods ultimately provides a long-term, stable core shutdown. Boron injection will lower power, however, alone may not provide a stable shutdown condition.
CT- 3.0: Given a failure to scram and the inability to restore and maintain RPV water level above -109 inches with the preferred ATWS injection systems, the crew will execute N1-EOP-8, RPV Blowdown and restore and maintain RPV water level above -109 inches with preferred and alternate ATWS injection systems, in accordance with N1-EOP-3.	Reactor water level must be maintained above limits to ensure adequate core cooling. With only low pressure systems available to inject and Reactor pressure above the pressure limits of these systems, Reactor pressure must be quickly lowered to allow injection. This protects the integrity of the fuel cladding.

SCENARIO SUMMARY

The scenario begins at approximately 95% power. ESW pump 12 is out of service for maintenance. IRM 11 is bypassed due to spiking. The crew is to start TBCLC pump 12 and secure TBCLC pump 11 per N1-OP-24 section F.1. After the TBCLC pumps are successfully shifted, the crew will raise reactor power with recirc.

Then, APRM 13 will fail upscale causing a half scram. The crew will bypass the APRM and reset the half scram. The CRS will make a Tech Spec call. When the half scram is reset, the Control Rod Drive Flow Control Valve fails closed, requiring shifting to the alternate FCV. After CRD flow is returned to normal, Service Water Adams Strainer 11 will clog, resulting in a high D/P annunciator in the control room. The crew will dispatch an Operator to manually backwash the strainer. When this is unsuccessful at clearing the alarm, the crew will place Service Water pump 12 in service.

Main condenser vacuum will begin to degrade as a result of air in-leakage. The crew will enter N1-SOP-25.1. Power will be lowered per N1-SOP-1.1 and the reactor will be scrammed before vacuum lowers to 22.1" Hg. When the scram is inserted, control rods will only partially insert. The crew will enter N1-EOP-3 to respond to the failure to scram. The crew will terminate and prevent all injection except boron and CRD **(Critical Task)**. The ATWS will be complicated with the failure of TBVs to open. The crew will initiate liquid poison injection and/or take actions in accordance with N1-EOP-3.1 to insert control rods **(Critical Task)**.

Facility: Nine Mile Point Unit 1 Date of Examination: February 2017
Examination Level: RO Operating Test Number: LC1 15-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, S	Perform Control Rod Position Verification and Determine Reactivity Event Severity N1-OP-42, OP-AA-300, N1-OP-5, K/A 2.1.37 (4.3)
Conduct of Operations		
Equipment Control	P, D, R 2015 NRC	Perform Daily Thermal Limit Surveillance N1-RESP-1A, K/A 2.2.12 (3.7)
Radiation Control	N, R	Application of Radiation Exposure Limits IAW RP-AA-203 – SDC Room RP-AA-203, K/A 2.3.4 (3.2)
Emergency Procedures/Plan	D, R	Actions For External Security Threats OP-NM-106-104, 2.4.28 (3.2)

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: February 2017
Examination Level: SRO	Operating Test Number: LC1 15-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, S	Perform Control Rod Position Verification and Determine Reactivity Event Severity and Notification Requirements OP-AA-300, N1-OP-5, N1-OP-42, K/A 2.1.37 (4.6)
Conduct of Operations	N, R	Determine Operator Qualifications Using LMS K/A 2.1.8 (4.1)
Equipment Control	P, D, R 2015 NRC	Perform Daily Thermal Limit Surveillance and Determine Corrective Actions N1-RESP-1A, K/A 2.2.12 (4.1)
Radiation Control	D, R	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 Classification and PARs Based on Dose Assessment EP-CE-111, EPIP-EPP-01 EAL Flowchart, K/A 2.4.29 (4.4)

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: February 2017
Exam Level: RO/SRO-I	Operating Test No.: LC1 15-1 NRC

Control Room Systems* (8 for RO); (7 for SRO-I)

System / JPM Title	Type Code*	Safety Function
a. Synchronize Main Generator to the Grid, Main Generator Locks Out K/A 245000 A4.02 (3.1/2.9), N1-OP-32, N1-SOP-31.1	D, A, S	4
b. Startup Control Room Ventilation (RO Only) K/A 290003 A4.01 (3.2/3.2), N1-OP-49	D, S	9
c. Channel 11 Non-Coincident Scram Test K/A 215004 A4.05 (3.1/3.2), N1-ST-R4, N1-OP-5	N, S, L, EN	7
d. Vent the Drywell Prior to Personnel Entry <212 K/A 223001 A4.03 (3.4/3.4), N1-OP-9	D, P, S, L (2015 NRC)	5
e. Sequential Loss of Service Water K/A 295018 AA1.01 (3.3/3.4) N1-SOP-18.1	D, A, S	8
f. Secure a Reactor Recirculation Pump, Pump Trips K/A 202001 A4.01 (3.7/3.7), N1-OP-1, N1-SOP-1.3	M, A, S	1
g. Core Spray Pump Quarterly ST, Suction Strainer Clogging K/A 209001 A4.01 (3.8/3.6), N1-ST-Q1C	D, A, S, EN	2
h. EDG 103 Control Room Start Following Station Blackout K/A 295003 AA1.02 (4.2/4.3), N1-OP-45	D, S, EN	6
In-Plant Systems* (3 for RO); (3 for SRO-I)		
i. Remove ERV Fuses in the Plant K/A 239002 A2.03 (4.1/4.2), N1-SOP-1.4	D, E, R	3
j. Inject Boron Into the Reactor Using the Hydro Pump (PRA) K/A 295037 EA1.10 (3.7/3.9), N1-EOP-3.2	D, E, R	1
k. Transfer Reactor Trip Bus 141 to I&C Bus 130A K/A 295003 AA1.01 (3.7/3.8), N1-SOP-30.2 Att 3	D, R	6

* 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
C and D

Facility: Nine Mile Point Unit 1 Date of Exam: February 2017

Tier	Group	RO K/A Category Points												SRO-Only Points			
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2		G*	Total
1. Emergency & Plant Evolutions	1	4	3	3				4	3			3	20	3		4	7
	2	2	1	1				1	1			1	7	2		1	3
	Tier Totals	6	4	4				5	4			4	27	5		5	10
2. Plant Systems	1	3	2	2	3	2	3	2	2	2	2	3	26	3		2	5
	2	1	1	1	1	1	1	2	1	1	1	1	12	0	2	1	3
	Tier Totals	4	3	3	4	3	4	4	3	3	3	4	38	5		3	8

3. Generic Knowledge & Abilities Categories	1	2	3	4	Total	1	2	3	4	Total
	3	3	2	2	10	2	2	2	1	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 outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 Radiation Control K/A is allowed if the K/A is replaced by a K/A from another Tier 3 Category.)
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted with justification; operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in 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

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
295028 High Drywell Temperature / 5					X		EA2.03 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level	3.9	76
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4					X		AA2.06 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Nuclear boiler instrumentation	3.3	77
295030 Low Suppression Pool Water Level / 5					X		EA2.01 - Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool level	4.2	78
295038 High Off-site Release Rate / 9						X	2.1.23 - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.4	79
295025 High Reactor Pressure / 3						X	2.4.8 - Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	4.5	80
295026 Suppression Pool High Water Temperature / 5						X	2.2.25 - Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	4.2	81
295004 Partial or Complete Loss of DC Power / 6						X	2.2.36 - Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	4.2	82
600000 Plant Fire On-site / 8	X						AK1.02 - Knowledge of the operational implications of the following concepts as they apply to Plant Fire On Site: Fire Fighting	2.9	39
295038 High Off-site Release Rate / 9	X						EK1.03 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: Meteorological effects on off-site release	2.8	40
295028 High Drywell Temperature / 5	X						EK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification	2.9	41
295031 Reactor Low Water Level / 2		X					EK2.09 - Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Recirculation system: Plant-Specific	3.3	42
295019 Partial or Complete Loss of Instrument Air / 8		X					AK2.11 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Radwaste	2.5	43
295016 Control Room Abandonment / 7		X					AK2.01 - Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Remote shutdown panel: Plant-Specific	4.4	44
295004 Partial or Complete Loss of DC Power / 6			X				AK3.01 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Load shedding: Plant-	2.6	45

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
							Specific		
295021 Loss of Shutdown Cooling / 4			X				AK3.04 - Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Maximizing reactor water cleanup flow	3.3	46
295005 Main Turbine Generator Trip / 3			X				AK3.06 - Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: Realignment of electrical distribution	3.3	47
700000 Generator Voltage and Electric Grid Disturbances				X			AA1.01 - Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Grid frequency and voltage	3.6	48
295006 SCRAM / 1				X			AA1.02 - Ability to operate and/or monitor the following as they apply to SCRAM: Reactor water level control system	3.9	49
295030 Low Suppression Pool Water Level / 5				X			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	3.6	50
295025 High Reactor Pressure / 3					X		EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor power	4.2	51
295023 Refueling Accidents / 8					X		AA2.04 - Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Occurrence of fuel handling accident	3.4	52
295026 Suppression Pool High Water Temperature / 5					X		EA2.01 - Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature	4.1	53
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4						X	2.2.40 - Equipment Control: Ability to apply technical specifications for a system.	3.4	54
295018 Partial or Complete Loss of CCW / 8						X	2.2.44 - Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	55
295003 Partial or Complete Loss of AC / 6						X	2.2.36 - Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	3.1	56
295024 High Drywell Pressure / 5				X			EA1.11 - Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Drywell spray: Mark-I&II	4.2	57
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1	X						EK1.07 - Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Shutdown margin	3.4	58

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#

K/A Category Totals:	4	3	3	4	**3/3**	**3/4**	Group Point Total:	20/7

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
295035 Secondary Containment High Differential Pressure / 5					X		EA2.02 - Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Off-site release rate: Plant-Specific	4.1	83
295015 Incomplete SCRAM / 1						X	2.4.35 - Emergency Procedures / Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects.	4.0	84
295009 Low Reactor Water Level					X		AA2.01 - Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water level	4.2	85
295036 Secondary Containment High Sump/Area Water Level / 5	X						EK1.02 - Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Electrical ground/ circuit malfunction	2.6	59
295002 Loss of Main Condenser Vacuum / 3		X					AK2.07 - Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: Offgas system	3.1	60
500000 High Containment Hydrogen Concentration / 5			X				EK3.06 - Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Operation of wet well vent	3.1	61
295008 High Reactor Water Level / 2				X			AA1.08 - Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL: Feedwater system	3.5	62
295007 High Reactor Pressure / 3					X		AA2.01 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor pressure	4.1	63
295022 Loss of CRD Pumps / 1						X	2.4.45 - Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm.	4.1	64
295035 Secondary Containment High Differential Pressure / 5	X						EK1.01 - Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Secondary containment integrity	3.9	65
K/A Category Totals:	2	1	1	1	**1/2**	**1/1**	Group Point Total:		7/3

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
300000 Instrument Air								X				A2.01 - Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Air dryer and filter malfunctions	2.8	86
259002 Reactor Water Level Control								X				A2.03 - 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 reactor water level input	3.7	87
400000 Component Cooling Water											X	2.1.28 - Conduct of Operations: Knowledge of the purpose and function of major system components and controls.	4.1	88
239002 SRVs											X	2.4.41 - Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications.	4.6	89
218000 ADS								X				A2.04 - 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: ADS failure to initiate	4.2	90
212000 RPS	X											K1.12 - Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following: Reactor/turbine pressure control system: Plant-Specific	3.4	1
264000 EDGs	X											K1.01 - Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: A.C. electrical distribution	3.8	2
206000 HPCI		X										K2.03 - Knowledge of electrical power supplies to the following: Initiation logic: BWR-2,3,4	2.8	3

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
400000 Component Cooling Water		X										K2.02 - Knowledge of electrical power supplies to the following: CCW valves	2.9	4
263000 DC Electrical Distribution			X									K3.02 - Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Components using D.C. control power (i.e. breakers)	3.5	5
300000 Instrument Air			X									K3.02 - Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following: Systems having pneumatic valves and controls	3.3	6
239002 SRVs				X								K4.04 - Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Ensures even distribution of heat load to suppression pool, and adequate steam condensing	3.4	7
215005 APRM / LPRM				X								K4.06 - Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Effects of detector aging on LPRM/APRM readings	2.6	8
215003 IRM					X							K5.01 - Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: Detector operation	2.6	9
205000 Shutdown Cooling					X							K5.02 - Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE): Valve operation	2.8	10
218000 ADS						X						K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: Nuclear boiler instrument system (level indication)	3.8	11
259002 Reactor Water Level Control						X						K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: Plant air systems	3.2	12

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
207000 Isolation (Emergency) Condenser							X					A1.10 - Ability to predict and/or monitor changes in parameters associated with operating the ISOLATION (EMERGENCY) CONDENSER controls including: Primary side temperature: BWR-2,3	3.2	13
223002 PCIS/Nuclear Steam Supply Shutoff							X					A1.04 - Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: Individual system relay status	2.6	14
209001 LPCS								X				A2.03 - 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: A.C. failures	3.4	15
215004 Source Range Monitor								X				A2.01 - 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: Power supply degraded	2.7	16
262002 UPS (AC/DC)									X			A3.01 - Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source	2.8	17
261000 SGTS									X			A3.04 - Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including: System temperature	3.0	18
262001 AC Electrical Distribution										X		A4.05 - Ability to manually operate and/or monitor in the control room: Voltage, current, power, and frequency on A.C. buses	3.3	19
211000 SLC										X		A4.07 - Ability to manually operate and/or monitor in the control room: Lights and alarms	3.6	20
212000 RPS											X	2.4.34 - Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	4.2	21

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
218000 ADS											X	2.4.11 - Emergency Procedures / Plan: Knowledge of abnormal condition procedures.	4.0	22
400000 Component Cooling Water						X						K6.07 - Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Breakers, relays, and disconnects	2.7	23
205000 Shutdown Cooling				X								K4.03 - Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: Low reactor water level: Plant-Specific	3.8	24
215004 Source Range Monitor											X	2.1.20 - Conduct of Operations: Ability to interpret and execute procedure steps.	4.6	25
206000 HPCI	X											K1.04 - Knowledge of the physical connections and/or cause-effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following: Reactor feedwater system: BWR-2,3,4	3.6	26
K/A Category Totals:	3	2	2	3	2	3	2	**2/3**	2	2	**3/2**	Group Point Total:	26/5	

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q #
201006 RWM								X				A2.02 - 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: Loss of steam flow input: P-Spec(Not-BWR6)	2.9	91
271000 Off-gas											X	2.2.37 - Equipment Control: Ability to determine operability and / or availability of safety related equipment.	4.6	92
286000 Fire Protection								X				A2.11 - Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Pump trips: Plant-Specific	3.2	93
215001 Traversing In-core Probe	X											K1.01 - Knowledge of the physical connections and/or cause-effect relationships between TRAVERSING IN-CORE PROBE and the following: Local power range monitors (Not-BWR1)	2.5	27
202001 Recirculation		X										K2.02 - Knowledge of electrical power supplies to the following: MG sets: Plant-Specific	3.2	28
233000 Fuel Pool Cooling/Cleanup			X									K3.08 - Knowledge of the effect that a loss or malfunction of the FUEL POOL COOLING AND CLEAN-UP will have on following: Refueling operations	2.9	29
241000 Reactor/Turbine Pressure Regulator				X								K4.19 - Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature(s) and/or interlocks which provide for the following: Steam bypass valve control	3.6	30
226001 RHR/LPCI: Containment Spray Mode					X							K5.06 – Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE: Vacuum breaker operation	2.6	31
290001 Secondary Containment						X						K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the SECONDARY CONTAINMENT: Reactor building ventilation: Plant-Specific	3.5	32

Nine Mile Point Unit 1 2017 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q #
272000 Radiation Monitoring							X					A1.02 - Ability to predict and/or monitor changes in parameters associated with operating the RADIATION MONITORING SYSTEM controls including: Lights, alarms, and indications associated with surveillance testing	2.9	33
290003 Control Room HVAC								**X**				A2.01 - Ability to (a) predict the impacts of the following on the CONTROL ROOM HVAC; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Initiation/reconfiguration	3.1	34
201003 Control Rod and Drive Mechanism									X			A3.01 - Ability to monitor automatic operations of the CONTROL ROD AND DRIVE MECHANISM including: Control rod position	3.7	35
201002 RMCS										X		A4.06 - Ability to manually operate and/or monitor in the control room: Rod select matrix power switch	2.8	36
201001 CRD Hydraulic											**X**	2.1.7 - Conduct of Operations: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.	4.4	37
245000 Main Turbine Generator and Auxiliaries							X					A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS controls including: Generator megawatts	2.7	38
K/A Category Totals:	1	1	1	1	1	1	2	**1/2**	1	1	**1/1**	Group Point Total:		12/3

Facility:	Nine Mile Point Unit 1	Date:	February 2017			
Category	K/A #	Topic	RO		SRO-Only	
			IR	Q#	IR	Q#
1. Conduct of Operations	2.1.32	Ability to explain and apply all system limits and precautions.			4.0	94
	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.			3.8	99
	2.1.26	Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).	3.4	66		
	2.1.1	Knowledge of conduct of operations requirements.	3.8	67		
	2.1.8	Ability to coordinate personnel activities outside the control room.	3.4	74		
	Subtotal			3		2
2. Equipment Control	2.2.40	Ability to apply technical specifications for a system.			4.5	95
	2.2.23	Ability to track Technical Specification limiting conditions for operations.			4.6	100
	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	4.5	68		
	2.2.42	Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	3.9	69		
	2.2.18	Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.	2.6	75		
	Subtotal			3		2
3. Radiation Control	2.3.13	Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.			3.8	96

	2.3.14	Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.			3.8	98
	2.3.11	Ability to control radiation releases.	3.8	70		
	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		
	Subtotal			2		2
4. Emergency Procedures / Plan	2.4.30	Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.			4.1	97
	2.4.25	Knowledge of fire protection procedures.	3.3	72		
	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.	4.0	73		
	Subtotal			2		1
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:		
1 / 1	Question 80 295025 High Reactor Pressure 2.4.3 - Emergency Procedures / Plan: Ability to identify post-accident instrumentation.	An acceptable question could not be developed at the SRO level for with randomly sampled generic K/A with this evolution. Reactor pressure instrumentation is not required for post-accident monitoring per Technical Specifications. Randomly re-selected K/A 295025 High Reactor Pressure 2.4.8 - Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
1 / 2	Question 85 295013 High Suppression Pool Temperature AA2.01 - Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression pool temperature	An acceptable question could not be developed for the randomly sampled K/A without overlapping concepts with Questions 53 and 81. Randomly re-selected K/A 295009 Low Reactor Water Level AA2.01 - Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water level.
3	Question 95 2.2.15 - Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.	An acceptable question could not be developed at the SRO level for with randomly sampled generic K/A. Additionally, this generic K/A is better tested on the operating exam. Randomly re-selected K/A 2.2.40 - Ability to apply Technical Specifications for a system.

3	Question 97 2.4.47 - Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	The randomly sampled generic K/A is better suited for, and extensively tested on, the operating exam. Randomly re-selected K/A 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.
2 / 1	Question 12 259002 Reactor Water Level Control K6.05 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: Reactor water level input	An acceptable question could not be developed for the randomly sampled K/A without overlapping Question 87. Randomly re-selected K/A 259002 Reactor Water Level Control K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: Plant air systems.
2 / 1	Question 15 209001 LPCS A2.07 - 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: Loss of room cooling	An acceptable question could not be developed for the randomly sampled K/A because the facility's Core Spray pumps are not in separate rooms with dedicated room cooling, as in some other plants. Randomly re-selected K/A 209001 LPCS A2.03 - 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: A.C. failures.
2 / 1	Question 21 212000 RPS 2.1.25 - Ability to interpret reference materials, such as graphs, curves, tables, etc.	An acceptable question could not be developed for the randomly sampled generic K/A due to lack of applicable reference material (graphs, curves, tables, etc.) for RPS. Randomly re-selected K/A 212000 RPS 2.4.34 - Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

2 / 1	Question 25 215004 Source Range Monitor 2.4.6 - Knowledge of EOP mitigation strategies.	An acceptable question could not be developed for the randomly sampled generic K/A due to lack of use of SRMs in EOP mitigation strategies. Randomly re-selected K/A 215004 Source Range Monitor 2.1.20 – Ability to interpret and execute procedure steps.
1 / 1	Question 41 295028 High Drywell Temperature EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level measurement	An acceptable question could not be developed for the randomly sampled K/A without overlapping Question 76. Randomly re-selected K/A 295028 High Drywell Temperature EK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification.
1 / 1	Question 55 295018 Partial or Complete Loss of CCW 2.2.39 - Knowledge of less than or equal to one hour technical specification action statements for systems.	An acceptable question could not be developed for the randomly sampled generic K/A due to lack of a less than or equal to one hour Technical Specification related to Loss of CCW. Randomly re-selected K/A 295018 Partial or Complete Loss of CCW 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.
1 / 2	Question 65 295035 Secondary Containment High Differential Pressure EK1.02 - Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Radiation release	An acceptable question could not be developed for the randomly sampled K/A without overlapping Question 83. Randomly re-selected K/A 295035 Secondary Containment High Differential Pressure EK1.01 - Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Secondary containment integrity.

3	Question 67 2.1.6 - Ability to manage the control room crew during plant transients.	The randomly sampled generic K/A is better suited for, and extensively tested on, the operating exam. Randomly re-selected K/A 2.1.1 – Knowledge of conduct of operations requirements.
2 / 1	Question 6 300000 Instrument Air K3.03 - Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following: Cross-tied units	An acceptable question could not be developed for the randomly sampled K/A because there is no feature to cross-tie Instrument Air between Nine Mile Point Units 1 and 2, as there is at some other multiple unit sites. Randomly re-selected K/A 300000 Instrument Air K3.02 - Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following: Systems having pneumatic valves and controls.