The Attachments to this letter contain NRC Examination Material which should be withheld from public disclosure for 2 years following the Examination per 10 CFR 2.390, 55.49, and NUREG-1021. Upon separation, this cover letter is decontrolled.

Exelon Generation.

Paul M. Swift
Site Vice President

R.E. Ginna Nuclear Power Plant
1503 Lake Rd.
Ontario, NY 14519

315 791 5200 Office
www.exeloncorp.com
paul.swift@exeloncorp.com

May 7, 2020

David Lew
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
2100 Renaissance Boulevard
Renaissance Park
King of Prussia, PA 19406-2713

R.E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

Subject: Submittal of Initial Operator Licensing Examination Outline

Dear Mr. Lew:

Below is a list of the examination outlines supporting the initial license exam scheduled for October 05, 2020, through October 16, 2020, at R. E. Ginna Station which have been sent to the respective Chief Examiner.

This submittal includes all appropriate examination standard forms and outlines in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 11.

In accordance with NUREG-1021, Revision 11, Section ES-201, "Initial Operator Licensing Examination Process," and Section ES-501, "Initial Post-Examination Activities," please ensure that these materials are withheld from public disclosure for two (2) years after the examinations are complete.

Should you have any questions concerning this letter, please contact the Manager, Site Regulatory Assurance, George Wrobel, at 585-315-0552. For questions concerning examination materials, please contact the Regulatory Exam Author, David Eckert, at 315-791-5523.

Respectfully,

Paul Swift
Site Vice President

SRRS: 3D.100; Retain this document until all regulatory actions associated with the exam are complete, at which time it may be discarded.

The Attachments to this letter contain NRC Examination Material which should be withheld from public disclosure for 2 years following the Examination per 10 CFR 2.390, 55.49, and NUREG-1021. Upon separation, this cover letter is decontrolled.

Enclosures: (Hard copy via FedEx to Joseph DeMarshall, Chief Examiner, NRC Region I)

Examination Security Agreements (Form ES-201-3)
Administrative Topics Outlines (Form ES-301-1)
- Reactor Operator (RO) ES-301-1
- Senior Reactor Operator (SRO) ES-301-1

Control Room/In-Plant Systems Outlines (Form ES-301-2)
- Reactor Operator (RO) ES-301-2
- Senior Reactor Operator - Instant (SRO-I) ES-301-2
- Senior Reactor Operator - Upgrade (SRO-U) ES-301-2

Scenario Outlines (Form ES-D-1)
- Scenario #1 ES-D-1
- Scenario #2 ES-D-1
- Scenario #3 ES-D-1
- Scenario #4 ES-D-1
- Scenario #5 ES-D-1

NRC Developed Written Examination Outlines
- PWR Examination Outline (Form ES-401-2)
- Generic Knowledge and Abilities Outline (Tier 3) (Form ES-401-3)
- Record of Rejected K/As (Form ES-401-4)

Completed Checklists
- Examination Outline Quality Checklist (Form ES-201-2)
- Transient and Event Checklist (Form ES-301-5)

cc: (without attachments)
Chief, NRC Operator Licensing Branch – Mr. Donald Jackson
Ginna Station Project Manager, NRR – Mr. V. Sreenivas
NRC Senior Resident Inspector - Ginna Station – Mr. Jason Schussler (E-mail)

The Attachments to this letter contain NRC Examination Material which should be withheld from public disclosure for 2 years following the Examination per 10 CFR 2.390, 55.49, and NUREG-1021. Upon separation, this cover letter is decontrolled.

Bcc: Site Vice President - Ginna Station – Mr. P. Swift
Manager, Site Regulatory Assurance - Ginna Station – Mr. G. Wrobel
Vice President - Licensing and Regulatory Affairs – Mr. G. Gellrich
Director, Licensing – Mrs. S. Rafferty-Czincila
Exelon Document Control Desk Licensing (Hard Copy)
Exelon Document Control Desk Licensing (Electronic Copy)
Human Resources - Ginna Station – Mr. S. Moore
Training Manager - Ginna Station – Mr. S. Minya

Facility:	Ginna Date of Examination:	10/2020		
Item	Task Description	Initials		
		a	b*	c#
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model per ES-401 or ES-401N.	DE	JWK	JD
	b. Assess whether the outline was systematically and randomly prepared in accordance with Section D.1 of ES-401 or ES-401N and whether all K/A categories are appropriately sampled.	DE	JWK	JD
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	DE	JWK	JD
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	N/A DE	N/A	JD
2. SIMULATOR	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, technical specifications, and major transients.	DE	JWK	JD
	b. Assess whether there are enough scenario sets (and spares) to test the projected number and mix of applicants in accordance with the expected crew composition and rotation schedule without compromising exam integrity; and ensure that each applicant can be tested using at least one new or significantly modified scenario, that no scenarios are duplicated from the applicants' audit test(s), and scenarios will not be repeated on subsequent days.	DE	JWK	JD
	c. To the extent possible, assess whether the outline(s) conform(s) with the qualitative and quantitative criteria specified on Form ES-301-4, described in Appendix D and described in Section D.5 Specific Instructions for the "Simulator Operating Test" of ES-301 (including overlap).	DE	JWK	JD
3. WALKTHROUGH	a. Verify that systems walk-through outline meets the criteria specified on Form ES-301-2: (1) the outline(s) contain(s) 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 applicants' 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 RCA tasks meet the criteria on the form.	DE	JWK	JD
	b. Verify that the administrative outline meets the criteria specified on Form ES-301-1: (1) the tasks are distributed among the topics as specified on the form (2) at least one task is new or significantly modified (3) no more than one task is repeated from the last two NRC licensing examinations	DE	JWK	JD
	c. Determine if there are enough different outlines to test the projected number and mix of applicants and ensure that no items are duplicated on subsequent days.	DE	JWK	JD
4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam section.	DE	JWK	JD
	b. Assess whether the 10CFR 55.41/43 and 55.45 sampling is appropriate.	DE	JWK	JD
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	DE	JWK	JD
	d. Check for duplication and overlap among exam sections and the last two NRC exams.	DE	JWK	JD
	e. Check the entire exam for balance of coverage.	DE	JWK	JD
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	DE	JWK	JD

	Printed Name / Signature	Date
a. Author	DAVID ECKERT / David L Eckert	5/1/2020
b. Facility Reviewer (*)	JUSTIN KNOWLES / Justin W Knowles	5/4/2020
c. NRC Chief Examiner (#)	Joseph DeMarshall / Joseph DeMarshall	5/20/2020
d. NRC Supervisor	Donald E. Jackson / DEJ	6/15/2020

* Not applicable for NRC-prepared examination outlines.
\# The independent NRC reviewer initials items in column "c"; the chief examiner's concurrence is required.

Facility: **Ginna**		Date of Examination:	**10/2020**
Examination Level:	**RO**	Operating Test Number:	**N20-1**

Administrative Topic (see Note)	Type Code*	Describe activity to be performed	
Conduct of Operations	D, P, R	2.1.5 (2.9)	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.
		JPM:	Determine the Allowable Hours an Operator Can Work
Conduct of Operations	N, R	2.1.25 (3.9)	Ability to interpret reference materials, such as graphs, curves, tables, etc.
		JPM:	Complete Plant System Checks Inside Control Room (Verify SDM)
Equipment Control	N, R	2.2.41 (3.5)	Ability to obtain and interpret station electrical and mechanical drawings.
		JPM:	Determine Leak Isolation Boundaries
Radiation Control	D, R	2.3.5 (2.9)	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
		JPM:	Evaluate Steam Generator Tube Leakage from R-47 Reading

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

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

RO Admin JPM Summary

A1a This is a Bank JPM. The operator will be given a previous work history, a report to work time and a time that they assumed the watch. The operator will be directed to determine the latest time that they can be relieved without violating Work Hour Rules or requiring a waiver or exception; and assuming that they get relieved and leave the site at the time determined, determine the earliest time that they can next report to work without violating any Work Hour Rules or requiring a waiver or exception. The operator will be expected to determine these dates and times in accordance with LS-AA-119 per the attached KEY. This JPM appeared on the 2018 Initial License Exam and was randomly selected for the 2020 Exam.

A1b This is a new JPM. The operator will be told that they have just relieved the watch on day shift, given a set of plant parameters that places the plant in Mode 3 with RCS Temperature at 530ºF at a specific time in life and told that they are performing Attachment 1, Modes 1, 2 and 3 of O-6.13, DAILY SURVEILLANCE LOG. The operator will be directed to (1) Complete the Attachment 1 check to Verify SDM on Page 6 of 10, (2) Identify all required Technical Specification ACTION, if any, based on the check and (3) Identify the latest time that this check can be complete. The operator will be expected to determine that adequate SDM does not exist, that the required ACTION is to Initiate boration to restore SDM to within limit within 15 minutes and identify that Attachment 1 must be complete by 0800 per the attached KEY.

A2 This is a new JPM. The operator will be told that a leak has occurred in the Service Water System and that the crew has entered AP-SW.1, SERVICE WATER LEAK. The operator will be directed to (1) identify the closest leak isolation boundary valves for this leak, (2) identify which, if any, of these valves need to be re-positioned from their current position, and (3) identify the Breaker location for any electrically operated leak isolation boundary valve that may need to be operated. The operator will review the Piping & Instrumentation Diagram of the Service Water System as well as system procedures and determine the closest leak isolation boundary valves for this leak, the boundary valves that need to be re-positioned and identify the Breaker location for the two MOVs in accordance with the Attached KEY.

A3 This is a Bank JPM. The operator will be told that the plant is operating at 100% power, that the crew has been notified that RMS-R47, R-47 AIR EJECTOR NOBLE GAS MONITOR, has alarmed in the TSC, provided with current readings, and told that the crew is evaluating the need to enter AP-SG.1, Steam Generator Tube Leak. The operator will be directed to determine (1) if the R-47 alarm setpoint is set properly, (2) the estimated Steam Generator Tube Leakrate based on the local R-47 reading and (3) if AP-SG.1, Steam Generator Tube Leak, should be entered based on the known information. The operator will be expected to determine that the R-47 alarm setpoints are set lower than identified by P-9, that the leakrate is 4.5 gpd based on the R-47 reading and that this leakrate is below the entry conditions into AP-SG.1.

Facility: **Ginna**		Date of Examination:	**10/2020**
Examination Level:	**SRO**	Operating Test Number:	**N20-1**

Administrative Topic (see Note)	Type Code*	Describe activity to be performed	
Conduct of Operations	M, R	2.1.25 (4.2) JPM:	Ability to interpret reference materials, such as graphs, curves, tables, etc. Determine Time to Boil for a Loss of Shutdown Cooling
Conduct of Operations	N, R	2.1.18 (3.8) JPM:	Ability to make accurate, clear, concise logs, records, status boards, and reports. Determine Reportability Requirements
Equipment Control	M, R	2.2.42 (4.6) JPM:	Ability to recognize system parameters that are entry-level conditions for Technical Specifications A-52.12, Inoperability of a Fire Damper
Radiation Control	D, P, R	2.3.6 (3.8) JPM:	Ability to approve release permits. Review and Approve Gas Decay Tank Release Permit
Emergency Procedures/Plan	N, R	2.4.37 (4.1) JPM:	Knowledge of the lines of authority during implementation of the emergency plan. Determine if Emergency Dose Limits and Thyroid Blocking Agents (Potassium Iodide [KI]) Should be Approved for Use

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

*Type Codes & Criteria:

(C)ontrol room, **(0)** (S)imulator, **(0)** or Class(R)oom **(5)**
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes) **(1)**
(N)ew or (M)odified from bank (≥ 1) **(4)**
(P)revious 2 exams (≤ 1; randomly selected) **(1)**

SRO Admin JPM Summary

A1a This is a modified Bank JPM. The operator will be provided with two sets of shutdown conditions; one current, and one projected to exist within ten days. The operator will be directed to determine the Time to Boil given a Loss of RHR for each of the two sets of conditions. The operator will be expected to use IP-OUT-2 and determine that the Time to Boil for the present plant conditions is 3.6 hours, and for the projected conditions ten days from now is 1.65 hours.

A1b This is a new JPM. The operator will be provided with a set of plant conditions that ultimately led to an automatic reactor trip from 100% power. The operator will be directed to determine reportability requirements, including completion of any necessary paperwork. The operator will be expected to identify that this issue requires a 4-hour notification to the NRC in accordance with LS-AA-1020 (Reportability Tables and Decision Trees), and complete NRC Form 361 (Reactor Plant Event Notification Worksheet) in accordance with the attached Key.

A2 This is a modified Bank JPM. The operator will be told that the plant is operating at 100% power and that a report from the field has indicated that Fire Damper BA-28, located in the wall separating the Turbine Building from Battery Room A, has been determined to be Non-Functional. The operator will be directed to document the failure of the Fire Damper as required by A-52.12, Nonfunctional Equipment Important to Safety; and identify all, if any, required ACTION. The operator will be expected to complete an A-52.12-F-03 in accordance with the KEY provided and identify that ACTION A.1.1 or A.1.2.1 and A.1.2.2; and A.2 of TRM 3.7.5 must be taken.

A3 This is a Bank JPM. The operator will be given a set of initial data on the D Gas Decay Tank, the current time and date and told that the on-shift Chemistry Technician has initiated a release for the D Gas Decay Tank and brought the Gaseous Waste Release Form (Attachment 1) from CH-703, RELEASE OF GDT'S AND OTHER GASEOUS BATCH RELEASES, to the Control Room. The operator will be directed to review and approve the release permit for the D Gas Decay Tank and as the Shift Manager approve the release OR identify all entries that must be changed to permit approval. The operator will be expected to review the release permit for Gas Decay Tank 'D' and identify the following deficiencies: (1) Chemistry Supervision signature should be signed, (2) Incorrect date entered for "Date Tank Held", (3) R-14 Alarm listed setpoint is incorrect, (4) R-14 Alarm actual setpoint is incorrect, and (5) the release has to be started within 12 hours vice 24 hours. The operator will NOT sign to approve the release. This JPM appeared on the 2019 Initial License Exam and was randomly selected for the 2020 Exam.

A4 This is a new JPM. The operator will be given a set of emergency conditions at Ginna, told that a General Emergency has been declared, that they are the Shift Emergency Director and that they are implementing EP-AA-112-100-F-01, SHIFT EMERGENCY DIRECTOR CHECKLIST. The operator will be directed to determine if Emergency Exposure Limits need to be approved for the rescue individuals, determine if the use of Thyroid Blocking Agents must be recommended for the rescue individuals, and if so, identify recommended dosage; and to prepare any paperwork that needs to be completed. The operator will be expected to determine that the rescuers need to be approved to receive Emergency Exposure Limits > 5 REM TEDE and the Thyroid Blocking Agent must be recommended such that one (1) 130 mg KI tablet (or (2) 65 mg KI tablets) is issued to each rescuer per day for 10 consecutive days or until directed that the risk no longer exists; and prepare an EP-AA-113-F-02 for each rescuer and EP-AA-113-F-03 per the attached KEY.

Facility: **Ginna**		Date of Examination: **10/2020**	
Exam Level (circle one): **RO**		Operating Test No.: **N20-1**	
Control Room Systems@ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U)			
	System / JPM Title	Type Code*	Safety Function
A.	004 Chemical and Volume Control System [004 A4.07 (3.9/3.7)] Borate for one rod not fully inserted following trip per ES-0.1	S, P, D, A, L	1
B.	008 Component Cooling Water [008 A2.01 (3.3\3.6)] Initiating a Plant Heatup on RHR w/Subsequent Failure of CCW Pump	S, N, A, L	8
C.	006 Emergency Core Cooling System [006 A1.13 (3.5/3.7)] Makeup to the B Accumulator from the RWST	S, D, EN	2
D.	EPE 074 Inadequate Core Cooling (074 EA1.05 (3.9 / 4.1)] Vent RCS for Accumulator/RHR Injection	S, P, D, A, L, EN	4P
E.	045 Main Turbine Generator System [045 A2.17 (2.7*/2.9*)] Synchronize Generator On-Line with Improper Load Pickup	S, D, A	4S
F.	EPE E14 High Containment Pressure [EPE E14 EA1.1 (3.7/3.7)] Verify Containment Isolation and Heat Removal	S, M, A, L, EN	5
G.	062 A. C. Electrical Distribution [062 A4.01 (3.3/3.1)] Establish 0/100 Electric Lineup on Circuit 7T	S, N	6
H.	010 Pressurizer Pressure Control System [010 A4.01 (3.7/3.5)] Depressurize the RCS During a SGTR	S, N	3
In-Plant Systems:* 3 for RO; 3 for SRO-I; 3 or 2 for SRO-U			
I.	071 Waste Gas Disposal System [071 A3.03 (3.6/3.8)] Release D Gas Decay Tank	P, D, R, A	9
J.	E04 LOCA Outside Containment [E04 EA2.2 (3.6/4.2)] Perform SFP-RWST Attachment	D, R, E	3
K.	061 Auxiliary Feedwater System [061 K4.13 (2.7/2.9)] Align Self-Cooling to the TDAFW Pump	D, E	4S
*	All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions, all five SRO-U systems must serve different safety functions, and 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 (6)
(C)ontrol room	
(D)irect from bank	≤ 9 (7)
(E)mergency or abnormal in-plant	≥ 1 (2)
(EN)gineered Safety Feature	≥ 1 (3) (Control Room System)
(L)ow-Power / Shutdown	≥ 1 (4)
(N)ew or (M)odified from bank including 1(A)	≥ 2 (4)
(P)revious 2 exams	≤ 3 (3) (Randomly Selected)
(R)CA	≥ 1 (2)
(S)imulator	

JPM Summary

JPM A This is a Bank JPM. The operator will be told that the plant has experienced a reactor trip and that the crew has completed ES-0.1, REACTOR TRIP RESPONSE, through step 3. After being told that another operator will handle all other MCB operations and alarms not associated with the assigned task, the operator will be directed to perform Step 4 of ES-0.1 (Verify MRPI Indicates – ALL CONTROL AND SHUTDOWN RODS ON BOTTOM). The operator will be expected to determine that one control rod has not fully inserted in accordance with Step 4 of ES-0.1 and attempt to initiate boration via the normal method. When it is discovered that boration cannot be started in accordance with step 4 RNO of ES-0.1 **(Alternate Path)**, the operator will initiate RCS boration in accordance with ER-CVCS.1, Section 6.2.1. This JPM appeared on the 2018 Initial License Exam and was randomly selected for the 2020 Exam.

JPM B This is a new JPM. The operator will be told that the plant is in Mode 5, that the crew has completed Section 6.5 of O-1.1, PLANT HEATUP FROM COLD SHUTDOWN TO HOT SHUTDOWN, the current RCS heatup rate is 11ºF/hour that Chemistry has completed all required RCS sampling and has added Hydrazine and that another operator is standing by to monitor and plot heatup rate. The operator will be directed to initiate an RCS heatup at maximum rate not to exceed 50°F/Hour with a maximum RHR System Total Flow of 1500 gpm using Section 6.6 of O-1.1. The operator will be expected to raise the RCS heatup rate to greater than 35ºF/hour in accordance with section 6.6. of O-1.1, and then respond to a loss of CCW using AP-CCW.3, restarting the A CCW Pump **(Alternate Path)** and stopping the RCPs if necessary.

JPM C This is a Bank JPM. The operator will be told that AR-C-4, ACCUMULATOR 1B (LOOP A) LEVEL 60 % 75, has alarmed. The operator will be directed to refill the B Accumulator to 69% using the C SI Pump. The operator will be expected to start the C SI Pump and raise the level of the B Accumulator to ≈69% in accordance with Section 6.2 of S-16.13, RWST WATER MAKEUP TO THE ACCUMULATORS, without the B Accumulator Level rising to greater than 75%.

JPM D This is a Bank JPM. The operator will be told that the plant has experienced a LOCA, that due to other malfunctions with the ARVs and the Safety Injection system the ability to adequately cool the core has been lost, that the crew has initiated FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, that initial attempts to restore core cooling per the procedure have been ineffective and that Bus 15 tripped during the initial transient. The operator will be directed to continue the actions of FR-C.1 beginning at Step 23 (Check if RCPs should be started). The operator will be expected to attempt to establish Instrument Air to the Containment and when this fails **(Alternate Path)**, use ATT-12.0, ATTACHMENT N2 PORVS to vent the RCS using both PORVs and the Reactor Vessel Head Vent Valves. This JPM appeared on the 2019 Initial License Exam and was randomly selected for the 2020 Exam.

JPM E This is a Bank JPM. The operator will be told that the generator is being started following a refueling outage, that the generator is at 1800 rpm and the turbine is fully warmed up, and that the generator output voltage is 19 KV. The operator will be directed to synchronize the generator on-line per O-1.2, PLANT STARTUP FROM HOT SHUTDOWN TO FULL LOAD, steps 6.13.1 through 6.13.11.7. During this action, the operator will recognize that Automatic Load Pickup feature of the Turbine Control System has failed **(Alternate Path)**. The operator will be expected to synchronize the Main Generator to the Electrical Grid, and when it is recognized that the Automatic Load Pickup has failed to function, the operator will manually load the Turbine to 40 to 60 MW, without reverse powering the Main Generator.

JPM F This is a modified Bank JPM. The operator will be told that the plant tripped from 100% power and safety injection has actuated, that the crew entered E-0, Reactor Trip or Safety Injection, and then transitioned to E-2, Faulted Steam Generator Isolation, and that due to a degrading transient, an Orange Path now exists on the Containment Critical Safety Function Status Tree. The operator will be directed to verify Containment Isolation and Heat Removal systems are operating as expected by performing FR-Z.1, Response to High Containment Pressure, starting from Step 1. The operator will be expected to establish Containment Spray flow including flow from the Sodium Hydroxide Tank from at least one train of Containment Spray and manually start the C CNMT Recirculation Fan Cooler. When the operator discovers that one Containment Isolation Valve has failed to close as expected and is unable to take specified response actions **(Alternate Path)** the operator will take alternative action in accordance with ATT-3.0, ATTACHMENT CI/CVI.

JPM G This is a new JPM. The operator will be told that the plant is operating at 100% power, that the Electric Plant is currently in a 50/50 NORMAL lineup, and that RG&E ECC has requested that the plant be placed in a 0/100 lineup on Circuit 7T for scheduled maintenance on offsite Circuit 767 later today. The operator will be directed to establish a 0/100 Electric Plant alignment and de-energize the 12B Station Aux Transformer per O-6.9.2, ESTABLISHING AND/OR TRANSFERRING OFFSITE POWER TO BUS 12A / BUS 12B, Section 6.4. The operator will be expected to transfer 4160V buses from a 50/50 NORMAL Lineup to 0/100 Lineup on Circuit 7T and de-energize the 12B Station Aux Transformer per Section 6.4.2 and 6.4.3 of O-6.9.2.

JPM H This is a new JPM. The operator will be told that the plant has experienced a Steam Generator Tube Rupture, that the crew has completed E-3, Steam Generator Tube Rupture, through Step 17, and that the crew is ready to commence RCS cooldown. The operator will be directed to depressurize the RCS to minimize break flow and refill the Pressurizer starting with Step 18. The operator will be expected to depressurize the RCS using maximum Pressurizer Spray and terminate the depressurization once one of the four criteria listed in E-3, Step 18.b, are met.

JPM I This is a Bank JPM. The operator will be told that the D Gas Decay Tank (GDT) is full and approved for release, that the D GDT does NOT have elevated activity and the contents are not from VCT gas and that Procedure S-4.2.5, Release of Gas Decay Tank, is in progress with the procedure signed off through Section 5.0. After being told that the Extra EO has removed the required tagout and completed the necessary paperwork, the operator will be directed to release the D GDT in accordance with S-4.2.5, Release of Gas Decay Tank. The operator will be expected to lineup and release the 'D' GDT in accordance with S-4.2.5, Section 6.4 and then stop the release in accordance with S-4.2.5, Step 4.2 upon an alarm condition on R-14 (**Alternate Path**). This JPM appeared on the 2018 Initial License Exam and was randomly selected for the 2020 Exam.

JPM J This is a Bank JPM. The operator will be told that the plant has experienced a LOCA outside of Containment, which cannot be isolated; and that the crew has entered ECA-1.1, Loss of Coolant Recirculation. The operator will be directed to transfer water from SFP to the RWST using Attachment ATT-18.0, SFP-RWST bypassing the SFP DI and Filter. The operator will be expected to initiate a transfer of water between the Spent Fuel Pool and the RWST bypassing the SFP DI and Filter in accordance with ATT-18.0.

JPM K This is a Bank JPM. The operator will be told that there has been a loss of the 480V safeguards buses, the crew is currently in ECA-0.0, Loss of All AC Power and the TDAFW has been running for one and a half hours with fire water not available. The operator will be directed to align self-cooling to the TDAFW pump per ATT-5.2, ALTERNATE COOLING TO TDAFW PUMP, Part B. The operator will be expected to align self-cooling to the TDAFW pump in accordance with ATT-5.2, ATTACHMENT ALTERNATE COOLING TO TDAFW PUMP within 30 minutes. This is a Time Critical JPM.

Facility: **Ginna**		Date of Examination: **10/2020**	
Exam Level (circle one): **SRO(I)**		Operating Test No.: **N20-1**	
Control Room Systems@ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U)			
	System / JPM Title	Type Code*	Safety Function
A.	004 Chemical and Volume Control System [004 A4.07 (3.9/3.7)] Borate for one rod not fully inserted following trip per ES-0.1	S, P, D, A, L	1
B.	008 Component Cooling Water [008 A2.01 (3.3\3.6)] Initiating a Plant Heatup on RHR w/Subsequent Failure of CCW Pump	S, N, A, L	8
C.	006 Emergency Core Cooling System [006 A1.13 (3.5/3.7)] Makeup to the B Accumulator from the RWST	S, D, EN	2
D.	EPE 074 Inadequate Core Cooling (074 EA1.05 (3.9 / 4.1)] Vent RCS for Accumulator/RHR Injection	S, P, D, A, L, EN	4P
E.	045 Main Turbine Generator System [045 A2.17 (2.7*/2.9*)] Synchronize Generator On-Line with Improper Load Pickup	S, D, A	4S
F.	EPE E14 High Containment Pressure [EPE E14 EA1.1 (3.7/3.7)] Verify Containment Isolation and Heat Removal	S, M, A, L, EN	5
G.	062 A. C. Electrical Distribution [062 A4.01 (3.3/3.1)] Establish 0/100 Electric Lineup on Circuit 7T	S, N	6
H.	NA		
In-Plant Systems:* 3 for RO; 3 for SRO-I; 3 or 2 for **SRO-U**			
I.	071 Waste Gas Disposal System [071 A3.03 (3.6/3.8)] Release D Gas Decay Tank	P, D, R, A	9
J.	E04 LOCA Outside Containment [E04 EA2.2 (3.6/4.2)] Perform SFP-RWST Attachment	D, R, E	3
K.	061 Auxiliary Feedwater System [061 K4.13 (2.7/2.9)] Align Self-Cooling to the TDAFW Pump	D, E	4S

* All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions, all five SRO-U systems must serve different safety functions, and 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 (6)
(C)ontrol room	
(D)irect from bank	≤ 8 (7)
(E)mergency or abnormal in-plant	≥ 1 (2)
(EN)gineered Safety Feature	≥ 1 (3) (Control Room System)
(L)ow-Power / Shutdown	≥ 1 (4)
(N)ew or (M)odified from bank including 1(A)	≥ 2 (3)
(P)revious 2 exams	≤ 3 (3) (Randomly Selected)
(R)CA	≥ 1 (2)
(S)imulator	

JPM Summary

JPM A This is a Bank JPM. The operator will be told that the plant has experienced a reactor trip and that the crew has completed ES-0.1, REACTOR TRIP RESPONSE, through step 3. After being told that another operator will handle all other MCB operations and alarms not associated with the assigned task, the operator will be directed to perform Step 4 of ES-0.1 (Verify MRPI Indicates – ALL CONTROL AND SHUTDOWN RODS ON BOTTOM). The operator will be expected to determine that one control rod has not fully inserted in accordance with Step 4 of ES-0.1 and attempt to initiate boration via the normal method. When it is discovered that boration cannot be started in accordance with step 4 RNO of ES-0.1 **(Alternate Path)**, the operator will initiate RCS boration in accordance with ER-CVCS.1, Section 6.2.1. This JPM appeared on the 2018 Initial License Exam and was randomly selected for the 2020 Exam.

JPM B This is a new JPM. The operator will be told that the plant is in Mode 5, that the crew has completed Section 6.5 of O-1.1, PLANT HEATUP FROM COLD SHUTDOWN TO HOT SHUTDOWN, the current RCS heatup rate is 11ºF/hour that Chemistry has completed all required RCS sampling and has added Hydrazine and that another operator is standing by to monitor and plot heatup rate. The operator will be directed to initiate an RCS heatup at maximum rate not to exceed 50°F/Hour with a maximum RHR System Total Flow of 1500 gpm using Section 6.6 of O-1.1. The operator will be expected to raise the RCS heatup rate to greater than 35ºF/hour in accordance with section 6.6. of O-1.1, and then respond to a loss of CCW using AP-CCW.3, restarting the A CCW Pump **(Alternate Path)** and stopping the RCPs if necessary.

JPM C This is a Bank JPM. The operator will be told that AR-C-4, ACCUMULATOR 1B (LOOP A) LEVEL 60 % 75, has alarmed. The operator will be directed to refill the B Accumulator to 69% using the C SI Pump. The operator will be expected to start the C SI Pump and raise the level of the B Accumulator to ≈69% in accordance with Section 6.2 of S-16.13, RWST WATER MAKEUP TO THE ACCUMULATORS, without the B Accumulator Level rising to greater than 75%.

JPM D This is a Bank JPM. The operator will be told that the plant has experienced a LOCA, that due to other malfunctions with the ARVs and the Safety Injection system the ability to adequately cool the core has been lost, that the crew has initiated FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, that initial attempts to restore core cooling per the procedure have been ineffective and that Bus 15 tripped during the initial transient. The operator will be directed to continue the actions of FR-C.1 beginning at Step 23 (Check if RCPs should be started). The operator will be expected to attempt to establish Instrument Air to the Containment and when this fails **(Alternate Path)**, use ATT-12.0, ATTACHMENT N2 PORVS to vent the RCS using both PORVs and the Reactor Vessel Head Vent Valves. This JPM appeared on the 2019 Initial License Exam and was randomly selected for the 2020 Exam.

JPM E This is a Bank JPM. The operator will be told that the generator is being started following a refueling outage, that the generator is at 1800 rpm and the turbine is fully warmed up, and that the generator output voltage is 19 KV. The operator will be directed to synchronize the generator on-line per O-1.2, PLANT STARTUP FROM HOT SHUTDOWN TO FULL LOAD, steps 6.13.1 through 6.13.11.7. During this action, the operator will recognize that Automatic Load Pickup feature of the Turbine Control System has failed **(Alternate Path)**. The operator will be expected to synchronize the Main Generator to the Electrical Grid, and when it is recognized that the Automatic Load Pickup has failed to function, the operator will manually load the Turbine to 40 to 60 MW, without reverse powering the Main Generator.

JPM F This is a modified Bank JPM. The operator will be told that the plant tripped from 100% power and safety injection has actuated, that the crew entered E-0, Reactor Trip or Safety Injection, and then transitioned to E-2, Faulted Steam Generator Isolation, and that due to a degrading transient, an Orange Path now exists on the Containment Critical Safety Function Status Tree. The operator will be directed to verify Containment Isolation and Heat Removal systems are operating as expected by performing FR-Z.1, Response to High Containment Pressure, starting from Step 1. The operator will be expected to establish Containment Spray flow including flow from the Sodium Hydroxide Tank from at least one train of Containment Spray and manually start the C CNMT Recirculation Fan Cooler. When the operator discovers that one Containment Isolation Valve has failed to close as expected and is unable to take specified response actions **(Alternate Path)** the operator will take alternative action in accordance with ATT-3.0, ATTACHMENT CI/CVI.

JPM G This is a new JPM. The operator will be told that the plant is operating at 100% power, that the Electric Plant is currently in a 50/50 NORMAL lineup, and that RG&E ECC has requested that the plant be placed in a 0/100 lineup on Circuit 7T for scheduled maintenance on offsite Circuit 767 later today. The operator will be directed to establish a 0/100 Electric Plant alignment and de-energize the 12B Station Aux Transformer per O-6.9.2, ESTABLISHING AND/OR TRANSFERRING OFFSITE POWER TO BUS 12A / BUS 12B, Section 6.4. The operator will be expected to transfer 4160V buses from a 50/50 NORMAL Lineup to 0/100 Lineup on Circuit 7T and de-energize the 12B Station Aux Transformer per Section 6.4.2 and 6.4.3 of O-6.9.2.

JPM I This is a Bank JPM. The operator will be told that the D Gas Decay Tank (GDT) is full and approved for release, that the D GDT does NOT have elevated activity and the contents are not from VCT gas and that Procedure S-4.2.5, Release of Gas Decay Tank, is in progress with the procedure signed off through Section 5.0. After being told that the Extra EO has removed the required tagout and completed the necessary paperwork, the operator will be directed to release the D GDT in accordance with S-4.2.5, Release of Gas Decay Tank. The operator will be expected to lineup and release the ‘D’ GDT in accordance with S-4.2.5, Section 6.4 and then stop the release in accordance with S-

4.2.5, Step 4.2 upon an alarm condition on R-14 (**Alternate Path**). This JPM appeared on the 2018 Initial License Exam and was randomly selected for the 2020 Exam.

JPM J This is a Bank JPM. The operator will be told that the plant has experienced a LOCA outside of Containment, which cannot be isolated; and that the crew has entered ECA-1.1, Loss of Coolant Recirculation. The operator will be directed to transfer water from SFP to the RWST using Attachment ATT-18.0, SFP-RWST bypassing the SFP DI and Filter. The operator will be expected to initiate a transfer of water between the Spent Fuel Pool and the RWST bypassing the SFP DI and Filter in accordance with ATT-18.0.

JPM K This is a Bank JPM. The operator will be told that there has been a loss of the 480V safeguards buses, the crew is currently in ECA-0.0, Loss of All AC Power and the TDAFW has been running for one and a half hours with fire water not available. The operator will be directed to align self-cooling to the TDAFW pump per ATT-5.2, ALTERNATE COOLING TO TDAFW PUMP, Part B. The operator will be expected to align self-cooling to the TDAFW pump in accordance with ATT-5.2, ATTACHMENT ALTERNATE COOLING TO TDAFW PUMP within 30 minutes. This is a Time Critical JPM.

Facility:	**Ginna**	Date of Examination:	**10/2020**
Exam Level (circle one):	**SRO(U)**	Operating Test No.:	**N20-1**

Control Room Systems@ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U)			
	System / JPM Title	Type Code*	Safety Function
A.	004 Chemical and Volume Control System [004 A4.07 (3.9/3.7)] Borate for one rod not fully inserted following trip per ES-0.1	S, P, D, A, L	1
B.	008 Component Cooling Water [008 A2.01 (3.3\3.6)] Initiating a Plant Heatup on RHR w/Subsequent Failure of CCW Pump	S, N, A, L	8
C.	006 Emergency Core Cooling System [006 A1.13 (3.5/3.7)] Makeup to the B Accumulator from the RWST	S, D, EN	2
D.	NA		
E.	NA		
F.	NA		
G.	NA		
H.	NA		
In-Plant Systems* 3 for RO; 3 for SRO-I; 3 or 2 for SRO-U			
I.	071 Waste Gas Disposal System [071 A3.03 (3.6/3.8)] Release D Gas Decay Tank	P, D, R, A	9
J.	E04 LOCA Outside Containment [E04 EA2.2 (3.6/4.2)] Perform SFP-RWST Attachment	D, R, E	3
K.	NA		

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

* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	2-3 (3)
(C)ontrol room	
(D)irect from bank	≤ 4 (4)
(E)mergency or abnormal in-plant	≥ 1 (1)
(EN)gineered Safety Feature	≥ 1 (1) (Control Room System)
(L)ow-Power / Shutdown	≥ 1 (2)
(N)ew or (M)odified from bank including 1(A)	≥ 1 (1)
(P)revious 2 exams	≤ 2 (2) (Randomly Selected)
(R)CA	≥ 1 (2)
(S)imulator	

JPM Summary

JPM A This is a Bank JPM. The operator will be told that the plant has experienced a reactor trip and that the crew has completed ES-0.1, REACTOR TRIP RESPONSE, through step 3. After being told that another operator will handle all other MCB operations and alarms not associated with the assigned task, the operator will be directed to perform Step 4 of ES-0.1 (Verify MRPI Indicates – ALL CONTROL AND SHUTDOWN RODS ON BOTTOM). The operator will be expected to determine that one control rod has not fully inserted in accordance with Step 4 of ES-0.1 and attempt to initiate boration via the normal method. When it is discovered that boration cannot be started in accordance with step 4 RNO of ES-0.1 **(Alternate Path)**, the operator will initiate RCS boration in accordance with ER-CVCS.1, Section 6.2.1. This JPM appeared on the 2018 Initial License Exam and was randomly selected for the 2020 Exam.

JPM B This is a new JPM. The operator will be told that the plant is in Mode 5, that the crew has completed Section 6.5 of O-1.1, PLANT HEATUP FROM COLD SHUTDOWN TO HOT SHUTDOWN, the current RCS heatup rate is 11ºF/hour that Chemistry has completed all required RCS sampling and has added Hydrazine and that another operator is standing by to monitor and plot heatup rate. The operator will be directed to initiate an RCS heatup at maximum rate not to exceed 50°F/Hour with a maximum RHR System Total Flow of 1500 gpm using Section 6.6 of O-1.1. The operator will be expected to raise the RCS heatup rate to greater than 35ºF/hour in accordance with section 6.6. of O-1.1, and then respond to a loss of CCW using AP-CCW.3, restarting the A CCW Pump **(Alternate Path)** and stopping the RCPs if necessary.

JPM C This is a Bank JPM. The operator will be told that AR-C-4, ACCUMULATOR 1B (LOOP A) LEVEL 60 % 75, has alarmed. The operator will be directed to refill the B Accumulator to 69% using the C SI Pump. The operator will be expected to start the C SI Pump and raise the level of the B Accumulator to ≈69% in accordance with Section 6.2 of S-16.13, RWST WATER MAKEUP TO THE ACCUMULATORS, without the B Accumulator Level rising to greater than 75%.

JPM I This is a Bank JPM. The operator will be told that the D Gas Decay Tank (GDT) is full and approved for release, that the D GDT does NOT have elevated activity and the contents are not from VCT gas and that Procedure S-4.2.5, Release of Gas Decay Tank, is in progress with the procedure signed off through Section 5.0. After being told that the Extra EO has removed the required tagout and completed the necessary paperwork, the operator will be directed to release the D GDT in accordance with S-4.2.5, Release of Gas Decay Tank. The operator will be expected to lineup and release the ‘D’ GDT in accordance with S-4.2.5, Section 6.4 and then stop the release in accordance with S-4.2.5, Step 4.2 upon an alarm condition on R-14 (**Alternate Path**). This JPM appeared on the 2018 Initial License Exam and was randomly selected for the 2020 Exam.

JPM J This is a Bank JPM. The operator will be told that the plant has experienced a LOCA outside of Containment, which cannot be isolated; and that the crew has entered ECA-1.1, Loss of Coolant Recirculation. The operator will be directed to transfer water from SFP to the RWST using Attachment ATT-18.0, SFP-RWST bypassing the SFP DI and Filter. The operator will be expected to initiate a transfer of water between the Spent Fuel Pool and the RWST bypassing the SFP DI and Filter in accordance with ATT-18.0.

Facility: **Ginna**	Scenario No.: **1**	Op Test No.: **N20-1**	
Examiners:		Operators:	(SRO)
			(RO)
			(BOP)

Initial Conditions:	The plant is at 100% power (EOL). The area has experienced unseasonably hot weather with no precipitation. It is expected to maintain power stable at the current power level throughout the shift.
Turnover:	The following equipment is Out-Of-Service: Containment Recirculation Fan Cooler C (TS 3.6.6 Condition D Action D.1) is OOS for breaker maintenance, and the Condensate Booster Pump A is OOS for thrust bearing replacement. The A RHR Pump has been taken out of service for testing. (TS 3.5.2 Condition A Action A.1).
Critical Tasks:	See Below

Event No.	Malf. No.	Event Type*	Event Description
1	MAL CLG05 ANNARHR 02	C-RO C(TS)-SRO	Failure of A RHR Pump During Surveillance
2	MAL PZR02D	I-RO I(TS)-SRO	Pressurizer Pressure (PT-449) fails HIGH
3	MAL STM04A	C-BOP C-SRO	A ARV Fails OPEN (3411)
4	REM Various Circuit Bkrs	R-RO N-BOP N-SRO	Unscheduled Trip of Transmission Circuits/Downpower
5	MAL FDW07C	C-BOP C-SRO	B FRV fails AS-IS (Manual Control Available)
6	MAL ROD05-K9	M-RO M-BOP M-SRO	Ejected Control Rod
7	MAL TUR02	C-BOP C-SRO	Failure of Turbine to Trip on Rx Trip
8	MAL RPS07A RPS07B	C-RO C-SRO	Failure of A and B SI Pumps to Auto Start
9	MAL RHR01B	NA	B RHR Pumps trips

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

<u>Ginna 2020 NRC Scenario #1</u>

The plant is at 100% power (EOL). The area has experienced unseasonably hot weather with no precipitation. It is expected to maintain power stable at the current power level throughout the shift.

The following equipment is Out-Of-Service: Containment Recirculation Fan Cooler C (TS 3.6.6 Condition D Action D.1) is OOS for breaker maintenance, and the Condensate Booster Pump A is OOS for thrust bearing replacement. The A RHR Pump has been taken out of service for testing. (TS 3.5.2 Condition A Action A.1).

Shortly after taking the watch, the operator will start the A RHR Pump per STP-O-2.2.-COMP-A, “Residual Heat Removal Pump A Comprehensive Test,” and then stop the pump due to a pump seal water cooler failure using the guidance of A-503.1, “Emergency and Abnormal Operating Procedures User’s Guide.” The operator will respond using AP-CCW.2, “Loss of CCW During Power Operation.” The operator will address Technical Specification LCO 3.5.2, “ECCS - MODES 1, 2, and 3.”

Following this, the controlling Pressurizer Pressure Transmitter will fail HIGH, causing the Spray Valves to open. The operator will respond in accordance with AR-F-2, “PRESSURIZER HIGH PRESS 2310 PSI,” and AR-F-10, “PRESSURIZER LO PRESS 2205 PSI,” and enter AP-PRZR.1, “Abnormal PZR Pressure.” AP-PRZR.1 will refer the operator to ER-INST.1, “Reactor Protection Bistable Defeat After Instrumentation Loop Failure,” for the defeat of PT-449. The operator will address Technical Specification LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure From Nucleate Boiling (DNB) Limits;” LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation;” and TR-3.4.3, “Anticipated Transient Without Scram (ATWS) Mitigation.”

Subsequently, the controller for the A SG ARV will fail such that the valve will travel to the fully OPEN position. The operator will respond using A-503.1, “Emergency and Abnormal Operating Procedures Users Guide,” and/or AP-FW.2, “Secondary Coolant Leak,” and take manual control of the ARV-3411 and close the valve.

Then, the off-site transmission circuit 908 will de-energize, and the RG&E Energy Control Center (ECC) will request that Ginna verbally certify that the plant is capable of ramping down to 490 MWe net generation in 14 minutes upon subsequent notification from ECC. The operator will respond in accordance with AR-J-28, “STATION 13A TROUBLE,” enter O-6.9, “Ginna Station Operating Limits for Station 13A Transmission,” and prepare for plant shutdown. After this, the off-site transmission circuit 913 will also de-energize and the ECC will call requesting that the previously agreed to downpower be executed. The operating crew will enter AP-TURB.5, “Rapid Load Reduction,” and lower plant power to 490 MWe.

During the load reduction, a failure of the B FRV to control in AUTO will occur. The operator will respond per AR-G-5, “S/G/ B LEVEL DEVIATION ±7%,” or upon observing an abnormally high level in the B Steam Generator and control the B FRV manually.

After this, Shutdown Bank Control Rod K-9 will be ejected from the core causing a LOCA, and an automatic Rx Trip/SI signal will occur. On the trip the Main Turbine will fail to trip, and the operator will need to manually trip the Turbine. Additionally, the A and the B SI Pumps will fail to start automatically and the B RHR Pump will trip. The operator will be required to manually start both SI Pumps. The operator will enter E-0, “Reactor Trip or Safety Injection,” and transition to E-1, “Loss of Reactor or Secondary Coolant.”

The operator will transition to ECA-1.1, “Loss of Emergency Coolant Recirculation,” due to a loss of both RHR Pumps. The operator will take actions to minimize the inventory loss from the RWST.

The scenario will terminate at Step 10.a RNO of ECA-1.1, after the crew has stopped one SI Pump.

__**Critical Tasks:**__

Manually control PRZR pressure during the failure of PT-449 before the Reactor automatically trips due to low pressurizer pressure

Safety Significance: failure to control PRZR pressure and stop the pressure transient, under the postulated plant conditions, results in an unnecessary transient to the plant and challenge to the Reactor Protection System. Performance of the critical task would stabilize the pressure transient. A failure to stabilize the pressure transient, when able to do so, constitutes a mis-operation or incorrect crew performance which leads to incorrect RCS pressure control.

Manually control the B S/G level during the failure of the B FRV Controller before the Reactor automatically trips due to low S/G level or Feedwater Isolates due to high S/G level

Safety Significance: failure to control B S/G level and stop the level transient, under the postulated plant conditions, results in an unnecessary transient to the plant and challenge to the Reactor Protection System. Performance of the critical task would stabilize the level transient. A failure to stabilize the level transient, when able to do so, constitutes a mis-operation or incorrect crew performance which leads to incorrect RCS temperature and/or pressure control.

Establish flow from at least two SI Pumps before indicating to the US that ATT-27.0 is complete.

Safety Significance: Failure to manually start at least one SI pump under the postulated conditions constitutes mis-operation or incorrect performance in which the crew does not prevent “degraded emergency core cooling system (ECCS) … capacity.” The acceptable results obtained in the FSAR analysis of a small-break LOCA are predicated on the assumption of minimum ECCS pumped injection (i.e. Two SI Pumps). ATT-27.0 is the procedure used to verify that the SI Pumps are automatically started and if not, manually start them. If the control board operator assigned to perform ATT-27.0 reports to the Unit Supervisor that the attachment is complete and at least two SI Pumps are NOT running, the operator has permitted a degraded ECCS condition which should have been corrected and was not.

Facility:	**Ginna**	Scenario No.: **2**	Op Test No.: **N20-1**
Examiners:		Operators:	(SRO)
			(RO)
			(BOP)

Initial Conditions:	The plant is at 70% power (BOL). The area has experienced cool weather with no precipitation. The crew is expected to remain at this power level while maintenance is completed during the shift.
Turnover:	The following equipment is Out-Of-Service: Containment Recirculation Fan Cooler C (TS 3.6.6 Condition D Action D.1) is OOS for breaker maintenance, and the B MDAFW Pump (TS 3.7.5 Condition B Action B.1) is OOS for breaker maintenance. The A HDT Pump is OOS due to abnormal noises.
Critical Tasks:	See Below

Event No.	Malf. No.	Event Type*	Event Description
1	MAL CVC12A	C-RO C-SRO	A Charging Pump Trip
2	MAL RCS16	C-RO C(TS)-SRO	Fuel Failure/High RCS Activity
3	MAL GEN02	C-BOP C-SRO	Generator Hydrogen Temperature (Cold Gas) Instrument fails HIGH
4	MAL FDW04B FDW02B	C-BOP C-SRO	B Feed Pump Oil Leak/Trip
5	OVR ROD04C MAL ROD03-G11	R-RO C(TS)-SRO N-BOP	Control Rods fail to Move in AUTO/Stuck Rod
6	MAL CND08	M-RO M-BOP M-SRO	Condensate Header Break
7	MAL EDS04A FDW02A	NA	Loss of Bus 14/A Main Feedwater Pump Trip
8	MAL RPS05A RPS05B	C-RO C-SRO	Automatic Rx Trip fails/No Manual Available
9	REM FDW12	NA	TDAFW Pump trips on Overspeed

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

Ginna 2020 NRC Scenario #2

The plant is at 70% power (BOL). The area has experienced cool weather with no precipitation. The crew is expected to remain at this power level while maintenance is completed during the shift.

The following equipment is Out-Of-Service: Containment Recirculation Fan Cooler C (TS 3.6.6 Condition D Action D.1) is OOS for breaker maintenance, and the B MDAFW Pump (TS 3.7.5 Condition B Action B.1) is OOS for breaker maintenance. The A HDT Pump is OOS due to abnormal noises.

Shortly after taking the watch, the A Charging Pump (Operating in AUTO) will trip. The operator will respond in accordance with AR-G-25, “MOTOR OFF CTR SECT PMPS EXCEPT MAIN & AUX FEED PMPS,” and start the B Charging Pump.

Shortly after this, a fuel failure will occur. The crew will address AR-E-24, “RMS AREA MONITOR HIGH ACTIVITY,” and AR-RMS-9, “R9 LETDOWN LINE MONITOR,” and enter AP-RCS.3, “High Reactor Coolant Activity.” After Chemistry reports that Dose Equivalent Iodine is 13 μCi/gram, the operator will place the 60 GPM letdown orifice in service, in accordance with S-3.2P, “Swapping CVCS Letdown Orifice Valves.” The operator will address Technical Specification LCO 3.4.16, “RCS Specific Activity,” and EPIP 1.13, “Local Radiation Emergency.”

Following this, the Generator Hydrogen Temperature (Cold Gas/TT-2023) Instrument will fail HIGH causing the Temperature Controller to increase cooling flow to the Main Generator, and Generator Gas temperatures to start to decrease. The operator will respond in accordance with AR-H-22, “H2 CLR TEMP HI/BYP VLV < 70% OPEN.” Using AR-H-22, the operator will take manual control of the Generator Hydrogen Temp Controller (TC-2023), as necessary, to control Generator Gas Temperature.

After this, the B Main Feedwater Pump will develop an oil leak, and trip within 60 seconds. The operator will respond using AR-H-4, “MAIN FEED PUMP OIL SYSTEM,” and AR-H-12, “FEED PUMP DC OIL PUMP AUTO START,” and enter AP-FW.1, “Abnormal MFW Pump Flow or NPSH.” The operator will start the A MDAFW Pump and the TD AFW Pump and initiate a downpower in accordance with AP-TURB.5, “Rapid Load Reduction.”

During the downpower, the control rods will fail to move in auto and the operator will need to insert control rods manually. Additionally, Control Rod G-11 will stick in its original position. After the plant stabilizes, the operator will respond in accordance with AR-C-5, “PPCS ROD SEQUENCE OR ROD DEVIATION/PPCS LTOP HI-LOW TEMPERATURE,” and enter AP-RCC.2, “RCC/RPI Malfunction.” The operator will address Technical Specification LCO 3.1.4, “Rod Group Alignment Limits,” and LCO 3.2.4, “Quadrant Power Tilt Ratio.”

After this, a large rupture will occur in the Condensate Header resulting in a loss of feedwater. Simultaneously, a ground fault will occur on Bus 14, isolating the bus and de-energizing the A MDAFW Pump. At the same time, the A Main Feedwater Pump will trip. The automatic Rx Trip system will fail to trip the reactor, and the operator will need to manually trip the reactor from the MCB by de-energizing Bus 13 and 15. The crew will enter E-0, “Reactor Trip or Safety Injection,” and transition to ES-0.1, “Reactor Trip Response.”

The TDAFW Pump will start but then fail to deliver flow. A Red condition will exist on the Heat Sink Critical Safety Function, and the crew will transition to FR-H.1, “Response to Loss of Secondary Heat Sink.” The crew will restore AFW flow to the Steam Generators via ATT-5.1, “Attachment SAFW From SW,” and ATT-22.0, “Attachment Restoring Feed Flow.”

The scenario will terminate at Step 12.b of FR-H.1, after the crew has returned to ES-0.1.

<u>Critical Tasks:</u>

Stabilize the plant during the Main Feedwater Pump Trip before the Reactor must be manually tripped at 20% or automatically trips on a low Narrow Range Steam Generator level

Safety Significance: Failure to control plant power and stabilize the transient, under the postulated plant conditions, results in an unnecessary transient to the plant and challenge to the Reactor Protection System. Performance of the critical task would stabilize the transient. A failure to stabilize the transient, when able to do so (i.e. event initiates from < 75% power), constitutes a mis-operation or incorrect crew performance which leads to incorrect RCS pressure/temperature control.

Manually trip the reactor from the control room before transition to FR-S.1

Safety Significance: Failure to manually trip the reactor from the control room when able to do so (i.e. De-energize Bus 13 and 15) causes a challenge to the subcriticality CSF beyond that irreparably introduced by the postulated conditions and requires that the facility declare an ALERT based on MA3 when the Emergency Classification would have been an Unusual Event based on MU3 had the Critical Task been completed successfully. The failure to complete the Critical Task constitutes an "incorrect performance that necessitates the crew taking action which complicates the event mitigation strategy demonstrating the inability by the crew to recognize and properly mitigate a failure of the automatic actuation of the RPS.

Establish feedwater flow into at least one Steam Generator before RCS Bleed and Feed is required

Safety Significance: Failure to establish feedwater flow into at least one Steam Generator results in the crew having to rely upon the lower-priority action of having to initiate RCS Bleed and Feed to minimize the possibility of core uncovery. Failure to perform this task, when able to do so by placing the Standby Auxiliary Feedwater System in service, constitutes incorrect performance that leads to degradation of the RCS and/or fuel cladding fission product barriers.

Facility:	**Ginna**	Scenario No.: **3**	Op Test No.: **N20-1**
Examiners:		Operators:	(SRO)
			(RO)
			(BOP)

Initial Conditions:	The plant is at 50% power (MOL). The plant was taken to 50% due to a failure of the A MFW Pump. Corrective maintenance has been performed and the A MFW Pump is operating. The area has experienced unseasonably cool weather with no precipitation. The crew is expected to raise power to 100% on this shift.
Turnover:	The following equipment is Out-Of-Service: The A EDG (TS 3.8.1 Condition B Action B.1, B.2, B.3.1 or B.3.2 and B.4) is OOS for fuel pump replacement and is expected to be back in 90 minutes. The A Control Rod Shroud Fan is OOS for breaker maintenance.
Critical Tasks:	See Below

Event No.	Malf. No.	Event Type*	Event Description
1	NA	R-RO N-BOP N(TS)-SRO	Load Ascension/High RWST Boron Concentration
2	MAL GEN03	C-BOP C-SRO	Main Generator Voltage Regulator Fails HIGH
3	MAL CVC05	I-RO I-SRO	Letdown Temperature fails LOW (TI-130)
4	MAL CLG03	C-RO C-SRO	Leak in the NRHX to CCW system
5	MAL EDS01B	C-RO C-BOP C(TS)-SRO	Loss of Offsite circuit 767/ Selected Service Water Pumps fail to start in Auto/Restore 4160 Bus 12B
6	MAL EDS01A GEN04B	M-RO M-BOP M-SRO	B EDG Trips/Loss of 7T Circuit
7	MAL RPS07M RPS07N	C-BOP C-SRO	TDAFW Steam Supply Valves fail to Auto OPEN (3504A and 3505A)
8	MAL RPS07O RPS07R	C-BOP C-SRO	Selected Service Water Pumps fail to start in Auto

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

Ginna 2020 NRC Scenario #3

The plant is at 50% power (MOL). The plant was taken to 50% due to a failure of the A MFW Pump. Corrective maintenance has been performed and the A MFW Pump is operating. The area has experienced unseasonably cool weather with no precipitation. The crew is expected to raise power to 100% on this shift.

The following equipment is Out-Of-Service: The A EDG (TS 3.8.1 Condition B Action B.1, B.2, B.3.1 or B.3.2 and B.4) is OOS for fuel pump replacement and is expected to be back in 90 minutes. The A Control Rod Shroud Fan is OOS for breaker maintenance.

Shortly after taking the watch, the operator will raise power in accordance with O-5.2, "Load Ascension." The operator will address S-3.1, "Boron Concentration Control," to start the load ascension using Alternate Dilute. During the load ascension, Chemistry will call the Control Room and report that the RWST boron concentration is 3063 ppm. The operator will address Technical Specification LCO 3.5.4, "Refueling Water Storage Tank (RWST)."

Subsequently, the Main Generator Voltage Regulator will fail HIGH. The operator will respond in accordance with AR-J-10, "GENERATOR VOLTAGE REGULATOR MANUAL." The operator will reduce generator output voltage using the Base Adjuster and notify RG&E ECC per the NOTE in AR-J-10. The Voltage Regulator will remain in MANUAL throughout the remainder of the scenario.

Following this, the Letdown Temperature indication (TI-130) will fail low causing the Temperature Control Valve TCV-130 to close, and actual letdown temperature to rise. The operator will respond in accordance with A-503.1, "Emergency and Abnormal Operating Procedures Users Guide," Section 5.3.A.5, and should re-open TCV-130 in manual to restore normal letdown temperature. The operator will address AR-A-12, "NON-REGEN HX LETDOWN OUT HI TEMP 145°F," and the TCV-130 Controller will remain in MANUAL throughout the remainder of the scenario.

Then, a 30 gpm leak will develop on the Non-Regenerative Heat Exchanger. The operator will respond in accordance with AR-E-16, "RMS PROCESS MONITOR HIGH ACTIVITY," and AR-RMS-17, "R-17 COMPONENT COOLING," which will direct the crew to AP-CCW.1, "Leakage Into the Component Cooling Loop." Per AP-CCW.1, letdown flow through the NRHX (and the leak) will be isolated and Excess Letdown will be placed in service.

After this, a loss of offsite power circuit 767 will occur, resulting in the loss of 4160 Bus 12B. The crew will respond per AP-ELEC.1, "Loss of 12A and/or 12B Busses," and restore power to 12B in accordance with ER-ELEC.1, "Restoration of Offsite Power," using circuit 7T. During the electrical transient, the Selected Service Water Pumps will fail to automatically start, and the operator will manually start one additional service water pump. The operator will address Technical Specification LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4," and Technical Requirements Manual, TR-3.8.1, "Offsite Power Sources."

Next, the B EDG will trip and the 7T line will be lost, resulting in a Station Blackout. The operator will enter ECA-0.0, "Loss of All AC Power." On the loss of power, the TDAFW Pump will fail to automatically start, and the operator will need to manually start the pump.

While in ECA-0.0 (at Step 13), the A EDG will become available. The operator will start the A EDG per Step 7 RNO actions and restore power to 480V vital busses 14 and 18. Upon restoration of power to Bus 18 the A SWP will fail to start automatically, requiring the operator to manually start the SWP. Following power restoration to Bus 16, recovery actions will continue with Step 34. The crew will likely transition to ECA-0.2, "Loss of All AC Power Recovery With SI Required," based on low Pressurizer level. On the other hand, there is a possibility that the crew could transition to ECA-0.1, “Loss of All AC Power Recovery Without SI Required.”

The scenario will terminate upon completion of ECA-0.2, if the RCP Seal Injection Needle Valves are closed with no Charging Pump running. If these valves are not closed, the scenario should continue in E-1, Loss of Reactor or Secondary Coolant, until the point of Charging Pump start (Step 11). If the crew transitions to ECA-0.1, rather than ECA-0.2, terminate at Step 5 of ECA-0.1, after the crew has restored Charging flow.

<u>**Critical Tasks:**</u>

Establish the minimum required AFW flowrate to the Steam Generators before Wide Range Level in BOTH Steam Generators lowers to 50 inches (100 inches Adverse Containment)

Safety Significance: Failure to establish the minimum required AFW flowrate results in adverse consequences and/or a significant degradation of the mitigative capability of the plant. If AFW flow is not sufficient during a Station Blackout, then cooldown of the RCS cannot be accomplished, and the result would be increased thermal stress on the RCP Seals, and increase the likelihood of increased leakage. Failure to perform this task, when able to do so (i.e. manually opening TDAFW Pump Steam Supply Valve), results in mis-operation that unnecessarily reduces the margin to safety of the plant.

Manually start a Service Water Pump such that the EDG does not fail because of damage caused by engine overheating

Safety Significance: Failure to manually start a SW Pump will result in the ONLY AC Power Source running loaded without cooling water. Running the EDG loaded without cooling water will lead to a condition that can lead to engine failure due to overheating. Thus, failure to perform this task results in mis-operation that leads to degraded emergency power capacity and leads to further challenges to the RCPs seals.

Isolate RCP Seal Injection Before a Charging Pump Starts or is Started if operating in ECA-0.2; or Isolate RCP Seal Injection Before a Charging Pump Starts or is Started AND Isolate CCW to the Thermal Barrier Before a CCW Pump Starts or is Started if operating in ECA-0.1

Safety Significance: Failure to isolate RCP seal injection before starting a charging pump; or isolate CCW flow to the Thermal Barriers before starting a CCW pump, under the postulated plant conditions, can result in unnecessary and avoidable degradation of the RCS fission-product barrier, specifically at the point of the RCP seals, especially if RCPs are subsequently started. Additionally, failure to perform the critical task results in significant degradation in the mitigative capability of the plant in that the RCPs are not available for subsequent event recovery actions (except for a RED path condition on the core cooling CSF).

Facility: **Ginna**	Scenario No.: **5**	Op Test No.: **N20-1**	
Examiners: ____________	Operators:	____________ (SRO)	
____________		____________ (RO)	
____________		____________ (BOP)	

Initial Conditions:	The plant is at 70% power (BOL). The area has experienced steady Thunderstorms for the past 2 hours, with wind from the West at 10-20 mph, and this is expected to continue throughout the shift. It is expected to raise power to 100% on this shift.
Turnover:	The following equipment is Out-Of-Service: Power Range Channel N44 is removed from service per ER-NIS.3 (TS LCO 3.3.1 Condition A/D/S and Action A.1, D.1 and S.1) and the QPTR Monitor has been declared inoperable. Condensate Booster Pump B is OOS for impeller replacement.
Critical Tasks:	See Below

Event No.	Malf. No.	Event Type*	Event Description
1	NA	R-RO N-BOP N(TS)-SRO	Load Ascension/High Secondary Activity
2	MAL NIS06C ROD1A	C-RO I(TS)-SRO	Power Range N42 Upper Detector drifts/Continuous Inward Rod Motion
3	OVR CND11B	C-BOP C-SRO	Hotwell Level Controller fails to Full Makeup
4	MAL CRC01A CND07B	C-RO C-BOP C-SRO	Circ Water Pump A Trips/Loss of Vacuum
5	MAL STM05A/B	M-RO M-BOP M-SRO	Inadvertent Steam Line Isolation Signal
6	MAL RPS05A/B	C-RO C-BOP C-SRO	Failure of the Reactor to trip from the Control Room/ATWS
7	MAL STM09A/B	C-BOP C-SRO	One S/G Safety Valve on each S/G Lifts and sticks partially OPEN

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

Ginna 2020 NRC Scenario #5

The plant is at 70% power (BOL). The area has experienced steady Thunderstorms for the past 2 hours, with wind from the West at 10-20 mph, and this is expected to continue throughout the shift. It is expected to raise power to 100% on this shift.

The following equipment is Out-Of-Service: Power Range Channel N44 is removed from service per ER-NIS.3 (TS LCO 3.3.1 Condition A/D/S and Action A.1, D.1 and S.1) and the QPTR Monitor has been declared inoperable. Condensate Booster Pump B is OOS for impeller replacement.

Shortly after taking the watch, the operator will raise power in accordance with O-5.2, “Load Ascension.” The operator will address S-3.1, “Boron Concentration Control,” to start the load ascension using Alternate Dilute. During the load ascension, Chemistry will call the Control Room and report that the B Steam Generator Secondary Specific Activity is 0.12 µc/gm. The operator will address Technical Specification LCO 3.7.14, “Secondary Specific Activity,” and a plant shutdown will be required.

Following this, the N42 Power Ranger Upper Detector will drift HIGH. Simultaneously a Continuous Control Rod Insertion will occur. The operator will enter AP-RCC.1, “Continuous Control Rod Withdrawal/Insertion.” Eventually, the operator will enter ER-NIS.3, “PR Malfunction,” however, removing N42 from service per this procedure will result in a plant trip. Subsequently, the N42 Power Ranger Upper Detector will drift LOW. The operator will address Technical Specification LCO 3.3.1, “Reactor Trip System (RTS) Instrumentation,” and implement Technical Specification LCO 3.0.3. The operator will also address Technical Requirement 3.2.3, “Axial Flux Difference (AFD) Monitor.” The control rods will remain in MANUAL for the remainder of the scenario.

Then, the Hotwell Level Controller will fail to Full Makeup. The operator will respond per AR-PPCS-L2022A and B, “CONDENSATE STORAGE TANK A(B) LEVEL,” and manually control the hotwell level for the remainder of the scenario.

Next, the A Circ Water Pump will trip. Simultaneously, a Condenser Expansion Joint low-level condition will occur. The crew will respond in accordance with AR-J-16, “MOTOR OFF CW-EH EMERG OIL SEAL OIL BU,” and enter AP-CW.1, “Loss of a Circ Water Pump,” and/or AP-TURB.4, “Loss of Condenser Vacuum.” The crew will reduce load per AP-TURB.5, “Rapid Load Reduction,” as needed to stabilize Turbine Backpressure.

After this, an inadvertent Steam Line Isolation Signal will occur and both MSIVs will close. The reactor will fail to automatically trip, and the operator will attempt to trip the reactor manually. The crew will enter E-0, “Reactor Trip or Safety Injection.”

The reactor will fail to trip manually from the control room, and the crew will enter FR-S.1, “Response to Reactor Restart/ATWS.” On the trip one S/G Safety Valve on each S/G lifted and stuck partially OPEN.

The crew will successfully de-energize the Rod Drive MG set(s) causing the control rods to drop into the core. Upon completion of FR-S.1, the crew will transition back to E-0, and then to E-2, “Faulted Steam Generator Isolation.” While implementing E-2, the crew will recognize that both S/Gs are faulted and transition to ECA-2.1, “Uncontrolled Depressurization of Both Steam Generators.” The crew may take a pre-emptive action of throttling AFW flow to both S/Gs per A-503.1, “Emergency and Abnormal Operating Procedures Users Guide.”

The scenario will terminate at Step 7 of ECA-2.1, after the crew has stopped the RHR Pumps.

Critical Tasks:

Upon diagnosing an ATWS, manually insert the control rods within 1 minute, and continue insertion until the reactor is tripped or the rods are on the bottom

Safety Significance: failure to insert negative reactivity, under the postulated plant conditions, results in an unnecessary situation in which the reactor power remains higher than it otherwise would if the action is taken. Performance of the critical task would move the reactor power lower to prevent a subsequent and unnecessary challenge to reactor core operational limits. A failure to insert negative reactivity constitutes a mis-operation or incorrect crew performance which leads to incorrect reactivity control. According to Section 15.8.3.2 of the Ginna UFSAR it is assumed that manual rod insertion of at least one bank is initiated within the first 60 seconds of the ATWS event.

Control the AFW flowrate to 50 gpm per SG in order to minimize the RCS Cooldown rate before an extreme challenge (Orange Path) develops to the integrity CSF

Safety Significance: Failure to control the AFW flow rate to the SGs leads to an unnecessary and avoidable extreme challenge to the integrity CSF. Also, failure to perform the Critical Task increases challenges to the SUBCRITICALITY Critical Safety Function which otherwise would not occur. If the action is not taken an unnecessary Orange Path could develop on the RCS Integrity Critical Safety Function requiring a substantial change in the mitigation strategy.

Facility: **Ginna Nuclear Power Plant** Date of Exam: **October 5 - 16, 2020**

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency and Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18	3	3	6
	2	1	1	2				2	2			1	9	3	1	4
	Tier Totals	4	4	5				5	5			4	27	6	4	10
2. Plant Systems	1	2	2	3	3	3	2	3	3	2	3	2	28	2	3	5
	2	1	1	1	1	1	1	1	1	0	1	1	10	1 1	1	3
	Tier Totals	3	3	4	4	4	3	4	4	2	4	3	38	4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 2	3 4	7
					3		3		2		2			2 2	1 2	

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 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' 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 PWR Examination Outline Form ES-401-2

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1					05		EA2.05 – Ability to determine or interpret the following as they apply to a reactor trip: Reactor trip first-out indication. (CFR: 41.7 / 45.5 / 45.6)	3.4	1
000008 (APE 8) Pressurizer Vapor Space Accident / 3	01						AK1.01 – Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Thermodynamics and flow characteristics of open or leaking valves. (CFR: 41.8 / 41.10 / 45.3)	3.2	2
000009 (EPE 9) Small Break LOCA / 3						01.19	G2.1.19 – Ability to use plant computers to evaluate system or component status. (CFR: 41.10 / 45.12)	3.9	3
000011 (EPE 11) Large Break LOCA / 3				01			EA1.01 – Ability to operate and monitor the following as they apply to a Large Break LOCA: Control of RCS pressure and temperature to avoid violating PTS limits. (CFR: 41.7 / 45.5 / 45.6)	3.7	4
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4									
000022 (APE 22) Loss of Reactor Coolant Makeup / 2					04		AA2.04 – Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: How long PZR level can be maintained within limits. (CFR: 43.5 / 45.13)	2.9	5
000025 (APE 25) Loss of Residual Heat Removal System / 4	01						AK1.01 – Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation. (CFR: 41.8 / 41.10 / 45.3)	3.9	6
000026 (APE 26) Loss of Component Cooling Water / 8									
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3		03					AK2.03 – Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners. (CFR: 41.7 / 45.7)	2.6	7
000029 (EPE 29) Anticipated Transient Without Scram / 1			08				EK3.08 – Knowledge of the reasons for the following responses as they apply to the ATWS: Closing the main steam isolation valve. (CFR: 41.5 / 41.10 / 45.6 / 45.13)	3.6	8
000038 (EPE 38) Steam Generator Tube Rupture / 3			06				EK3.06 – Knowledge of the reasons for the following responses as they apply to the SGTR: Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures. (CFR: 41.5 / 41.10 / 45.6 / 45.13)	4.2	9
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4		02					AK2.02 – Knowledge of the interrelations between the Steam Line Rupture and the following: Sensors and detectors. (CFR: 41.7 / 45.7)	2.6	10
000054 (APE 54; CE E06) Loss of Main Feedwater /4			03				AK3.03 – Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Manual control of AFW flow control valves. (CFR: 41.5 / 41.10 / 45.6 / 45.13)	3.8	11
000055 (EPE 55) Station Blackout / 6				07			EA1.07 – Ability to operate and monitor the following as they apply to a Station Blackout: Restoration of power from offsite. (CFR: 41.7 / 45.5 / 45.6)	4.3	12
000056 (APE 56) Loss of Offsite Power / 6						04.06	G2.4.6 – Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	3.7	13

000057 (APE 57) Loss of Vital AC Instrument Bus / 6					18		AA2.18 – Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: The indicator, valve, breaker, or damper position which will occur on a loss of power. (CFR: 43.5 / 45.13)	3.1	14
000058 (APE 58) Loss of DC Power / 6						04.45	G2.4.45 – Ability to prioritize and interpret the significance of each annunciator or alarm. (CFR: 41.10 / 43.5 / 45.3 / 45.12)	4.1	15
000062 (APE 62) Loss of Nuclear Service Water / 4									
000065 (APE 65) Loss of Instrument Air / 8									
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6									
(W E04) LOCA Outside Containment / 3	03						W E04: EK1.3 – Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Outside Containment).. (CFR: 41.8 / 41.10 / 45.3)	3.5	16
(W E11) Loss of Emergency Coolant Recirculation / 4		01					W E11: EK2.1 – Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR: 41.7 45.7)	3.6	17
(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4				01			W E05: EA1.1 – Ability to operate and/or monitor the following as they apply to the (Loss of Secondary Heat Sink): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR: 41.7 / 45.5 / 45.6)	4.1	18
K/A Category Totals:	3	3	3	3	3	3	Group Point Total:		18

ES-401 PWR Examination Outline Form ES-401-2 Emergency and Abnormal Plant Evolutions—Tier 1/Group 2 (RO)									
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1									
000003 (APE 3) Dropped Control Rod / 1									
000005 (APE 5) Inoperable/Stuck Control Rod / 1			02				AK3.02 – Knowledge of the reasons for the following responses as they apply to the Inoperable / Stuck Control Rod: Rod insertion limits. (CFR: 41.5 / 41.10 / 45.6 / 45.13)	3.6	19
000024 (APE 24) Emergency Boration / 1									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2			03				AK3.03 – Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions: False indications of PZR level when PORV or spray valve is open and RCS saturated. (CFR: 41.5 / 41.10 / 45.6 / 45.13)	3.5	20
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7						02.39	G2.2.39 – Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13)	3.9	21
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7									
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8									
000037 (APE 37) Steam Generator Tube Leak / 3									
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9									
000060 (APE 60) Accidental Gaseous Radwaste Release / 9				01			AA1.01 – Ability to operate and/or monitor the following as they apply to the Accidental Gaseous Radwaste: Area radiation monitors. (CFR: 41.7 / 45.5 / 45.6)	2.8	22
000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
000067 (APE 67) Plant Fire On Site / 8									
000068 (APE 68; BW A06) Control Room Evacuation / 8					06		AA2.06 – Ability to determine and interpret the following as they apply to the Control Room Evacuation: RCS pressure. (CFR: 43.5 / 45.13)	4.1	23
000069 (APE 69; W E14) Loss of Containment Integrity / 5									
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4									
000076 (APE 76) High Reactor Coolant Activity / 9		01					AK2.01 – Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors. (CFR: 41.7 / 45.7)	2.6	24
000078 (APE 78*) RCS Leak / 3									
(W E01 & E02) Rediagnosis & SI Termination / 3									
(W E13) Steam Generator Overpressure / 4									

(W E15) Containment Flooding / 5				01			W E15: EA1.1 - Ability to operate and/or monitor the following as they apply to the (Containment Flooding): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. (CFR: 41.7 / 45.5 / 45.6)	2.9	25
(W E16) High Containment Radiation /9									
(BW A01) Plant Runback / 1									
(BW A02 & A03) Loss of NNI-X/Y/7									
(BW A04) Turbine Trip / 4									
(BW A05) Emergency Diesel Actuation / 6									
(BW A07) Flooding / 8									
(BW E03) Inadequate Subcooling Margin / 4									
(BW E08; W E03) LOCA Cooldown—Depressurization / 4									
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4					01		W E10: EA2.1 – Ability to determine and interpret the following as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (CFR: 43.5 / 45.13)	3.2	26
(BW E13 & E14) EOP Rules and Enclosures									
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4	01						W E08: EK1.1 – Knowledge of the operational implications of the following concepts as they apply to the (Pressurized Thermal Shock): Components, capacity,, and function of emergency systems. (CFR: 41.8 / 41.10 / 45.3	3.5	27
(CE A16) Excess RCS Leakage / 2									
(CE E09) Functional Recovery									
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
K/A Category Point Totals:	1	1	2	2	2	1	Group Point Total:		9

ES-401	PWR Examination Outline Plant Systems—Tier 2/Group 1 (RO)													Form ES-401-2
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump					03							K5.03 – Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on T-ave., including the reason for the unreliability of T-ave. in the shutdown loop. (CFR: 41.5 / 45.7)	3.1	28
004 (SF1; SF2 CVCS) Chemical and Volume Control										06		A4.06 – Ability to manually operate and/or monitor in the control room: Letdown isolation and flow control valves. (CFR: 41.7 / 45.5 to 45.8)	3.6	29
					02							K5.02 – Knowledge of the operational implications of the following concepts as they apply to the CVCS: Explosion hazard associated with hydrogen containing systems. (CFR: 41.5 / 45.7)	3.5	30
005 (SF4P RHR) Residual Heat Removal				01								K4.01 – Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following: Overpressure mitigation system. (CFR: 41.7)	3.0	31
006 (SF2; SF3 ECCS) Emergency Core Cooling		02										K2.02 – Knowledge of bus power supplies to the following: Valve operators for accumulators. (CFR: 41.7)	2.5	32
007 (SF5 PRTS) Pressurizer Relief/Quench Tank				01								K4.01 – Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: Quench tank cooling. (CFR: 41.7)	2.6	33
008 (SF8 CCW) Component Cooling Water							01					A1.01 – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW flow rate. (CFR: 41.5 / 45.5)	2.8	34
010 (SF3 PZR PCS) Pressurizer Pressure Control		03										K2.03 – Knowledge of bus power supplies to the following: Indicator for PORV position. (CFR: 41.7)	2.8	35
012 (SF7 RPS) Reactor Protection			01									K3.01 – Knowledge of the effect that a loss or malfunction of the RPS will have on the following: CRDS. (CFR: 41.7 / 45.6)	3.9	36
									06			A3.06 – Ability to monitor automatic operation of the RPS, including: Trip logic. (CFR: 41.7 / 45.5)	3.7	37
013 (SF2 ESFAS) Engineered Safety Features Actuation						01						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the ESFAS: Sensors and detectors. (CFR: 41.7 / 45.5 to 45.8)	2.7	38

022 (SF5 CCS) Containment Cooling							02					A1.02 – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment pressure (CFR: 41.5 / 45.5)	3.6	39
								05				A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Major leak in CCS. (CFR: 41.5 / 43.5 / 45.3 / 45.13)	3.1	40
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray										05		A4.05 – Ability to manually operate and/or monitor in the control room: Containment spray reset switches. (CFR: 41.7 / 45.5 to 45.8)	3.5	41
039 (SF4S MSS) Main and Reheat Steam									02			A3.02 – Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS. (CFR: 41.5 / 45.5)	3.1	42
059 (SF4S MFW) Main Feedwater										11		A4.11 – Ability to manually operate and monitor in the control room: Recovery from automatic feedwater isolation. (CFR: 41.7 / 45.5 to 45.8)	3.1	43
											01. 28	G2.1.28 – Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)	4.1	44
061 (SF4S AFW) Auxiliary/Emergency Feedwater					02							K5.02 – Knowledge of the operational implications of the following concepts as they apply to the AFW: Decay heat sources and magnitude. (CFR: 41.5 / 45.7)	3.2	45
						02						K6.02 – Knowledge of the effect that a loss or malfunction of the following will have on the AFW components: Pumps. (CFR: 41.7 / 45.7)	2.6	46
062 (SF6 ED AC) AC Electrical Distribution							01					A1.01 – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AC distribution system controls including: Significance of D/G load limits. (CFR: 41.5 / 45.5)	3.4	47
063 (SF6 ED DC) DC Electrical Distribution											02. 12	G2.2.12 – Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)	3.7	48
			02									K3.02 – Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: Components using DC control power. (CFR: 41.7 / 45.6)	3.5	49
064 (SF6 EDG) Emergency Diesel Generator	02											K1.02 – Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: D/G cooling water system. (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.1	50

073 (SF7 PRM) Process Radiation Monitoring								02				A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure. (CFR: 41.5 / 43.5 / 45.3 / 45.13)	2.7	51
				01								K4.01 – Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint. (CFR: 41.7)	4.0	52
076 (SF4S SW) Service Water			07									K3.07 – Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads. (CFR: 41.7 / 45.6)	3.7	53
078 (SF8 IAS) Instrument Air	05											K1.05 – Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: MSIV air. (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.4	54
103 (SF5 CNT) Containment								04				A2.04 - Ability to (a) predict the impacts of the following malfunctions or operations on the containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Containment evacuation (including recognition of the alarm). (CFR: 41.5 / 43.5 / 45.3 / 45.13)	3.5	55
053 (SF1; SF4P ICS*) Integrated Control														
K/A Category Point Totals:	2	2	3	3	3	2	3	3	2	3	2	Group Point Total:		28

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

Plant Systems—Tier 2/Group 2 (RO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive										11		A4.11 – Ability to manually operate and/or monitor in the control room: Determination of SDM. (CFR: 41.7 / 45.5 to 45.8)	3.5	56
002 (SF2; SF4P RCS) Reactor Coolant														
011 (SF2 PZR LCS) Pressurizer Level Control		01										K2.01 – Knowledge of bus power supplies to the following: Charging pumps. (CFR: 41.7)	3.1	57
014 (SF1 RPI) Rod Position Indication														
015 (SF7 NI) Nuclear Instrumentation														
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor				02								K4.02 – Knowledge of ITM system design feature(s) and/or interlock(s) which provide for the following: Sensing and determination of location core hot spots. (CFR: 41.7)	3.1	58
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control						01						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the HRPS: Hydrogen recombiners. (CFR: 41.7 / 45.7)	2.6	59
029 (SF8 CPS) Containment Purge														
033 (SF8 SFPCS) Spent Fuel Pool Cooling														
034 (SF8 FHS) Fuel-Handling Equipment														
035 (SF 4P SG) Steam Generator											02.44	G2.2.44 – Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)	4.2	60
041 (SF4S SDS) Steam Dump/Turbine Bypass Control							02					A1.02 – Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure. (CFR: 41.5 / 45.5)	3.1	61
045 (SF 4S MTG) Main Turbine Generator					01							K5.01 – Knowledge of the operational implications of the following concepts as they apply to the MT/B System: Possible presence of explosive mixture in generator if hydrogen purity deteriorates. (CFR: 41.5 / 45.7)	2.8	62
055 (SF4S CARS) Condenser Air Removal														
056 (SF4S CDS) Condensate	03											K1.03 – Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: MFW. (CFR: 41.2 to 41.9 / 45.7 to 45.8)	2.6	63

068 (SF9 LRS) Liquid Radwaste								04				A2.04 – Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation. (CFR: 41.5 / 43.5 / 45.3 / 45.13)	3.3	64
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring			01									K3.01 – Knowledge of the effect that a loss or malfunction of the ARM system will have on the following: Containment ventilation isolation. (CFR: 41.7 / 45.6)	3.2	65
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air														
086 Fire Protection														
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:	1	1	1	1	1	1	1	1	0	1	1	Group Point Total:		10

ES-401 PWR Examination Outline Form ES-401-2
Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (SRO)

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1									
000008 (APE 8) Pressurizer Vapor Space Accident / 3									
000009 (EPE 9) Small Break LOCA / 3									
000011 (EPE 11) Large Break LOCA / 3									
000015 (APE 15) Reactor Coolant Pump Malfunctions / 4					10		AA2.10 – Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on loss of cooling or seal injection. (CFR 43.5 / 45.13)	3.7	76
000022 (APE 22) Loss of Reactor Coolant Makeup / 2									
000025 (APE 25) Loss of Residual Heat Removal System / 4									
000026 (APE 26) Loss of Component Cooling Water / 8						01. 20	G2.1.20 – Ability to interpret and execute procedure steps (CFR: 41.10 / 43.5 / 45.12)	4.6	77
000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3									
000029 (EPE 29) Anticipated Transient Without Scram / 1									
000038 (EPE 38) Steam Generator Tube Rupture / 3					08		EA2.08 – Ability to determine or interpret the following as they apply to a SGTR: Viable alternatives for placing plant in safe condition when condenser is not available. (CFR: 43.5 / 45.13)	4.4	78
000040 (APE 40; BW E05; CE E05; W E12) Steam Line Rupture—Excessive Heat Transfer / 4									
000054 (APE 54; CE E06) Loss of Main Feedwater /4									
000055 (EPE 55) Station Blackout / 6									
000056 (APE 56) Loss of Offsite Power / 6									
000057 (APE 57) Loss of Vital AC Instrument Bus / 6									
000058 (APE 58) Loss of DC Power / 6									
000062 (APE 62) Loss of Nuclear Service Water / 4						04. 08	G2.4.8 – Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13)	4.5	79
000065 (APE 65) Loss of Instrument Air / 8						04. 31	G2.4.31 – Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)	4.1	80
000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6					05		AA2.05 – Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Operational status of offsite circuit. (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)	3.8	81
(W E04) LOCA Outside Containment / 3									
(W E11) Loss of Emergency Coolant Recirculation / 4									

(BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4									
K/A Category Totals:					3	3	Group Point Total:		6

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

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
000001 (APE 1) Continuous Rod Withdrawal / 1					03		AA2.03 – Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place. (CFR: 43.5 / 45.13)	4.8	82
000003 (APE 3) Dropped Control Rod / 1									
000005 (APE 5) Inoperable/Stuck Control Rod / 1									
000024 (APE 24) Emergency Boration / 1									
000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2									
000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7									
000033 (APE 33) Loss of Intermediate Range Nuclear Instrumentation / 7					08		AA2.08 - Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Intermediate range channel operability. (CFR: 43.5 / 45.13)	3.4	83
000036 (APE 36; BW/A08) Fuel-Handling Incidents / 8									
000037 (APE 37) Steam Generator Tube Leak / 3									
000051 (APE 51) Loss of Condenser Vacuum / 4									
000059 (APE 59) Accidental Liquid Radwaste Release / 9									
000060 (APE 60) Accidental Gaseous Radwaste Release / 9									
000061 (APE 61) Area Radiation Monitoring System Alarms / 7									
000067 (APE 67) Plant Fire On Site / 8					13		AA2.13 – Ability to determine and interpret the following as they apply to the Plant Fire on Site: Need for emergency plant shutdown. (CFR: 43.5 / 45.13)	4.4	84
000068 (APE 68; BW A06) Control Room Evacuation / 8									
000069 (APE 69; W E14) Loss of Containment Integrity / 5									
000074 (EPE 74; W E06 & E07) Inadequate Core Cooling / 4						04.21	G2.4.21 – 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.. (CFR: 41.7 / 43.5 / 45.12)	4.6	85
000076 (APE 76) High Reactor Coolant Activity / 9									
000078 (APE 78*) RCS Leak / 3									
(W E01 & E02) Rediagnosis & SI Termination / 3									
(W E13) Steam Generator Overpressure / 4									
(W E15) Containment Flooding / 5									
(W E16) High Containment Radiation /9									
(BW A01) Plant Runback / 1									
(BW A02 & A03) Loss of NNI-X/Y/7									
(BW A04) Turbine Trip / 4									

(BW A05) Emergency Diesel Actuation / 6									
(BW A07) Flooding / 8									
(BW E03) Inadequate Subcooling Margin / 4									
(BW E08; W E03) LOCA Cooldown—Depressurization / 4									
(BW E09; CE A13**; W E09 & E10) Natural Circulation/4									
(BW E13 & E14) EOP Rules and Enclosures									
(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4									
(CE A16) Excess RCS Leakage / 2									
(CE E09) Functional Recovery									
(CE E13*) Loss of Forced Circulation/LOOP/Blackout / 4									
K/A Category Point Totals:					3	1	Group Point Total:		4

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
003 (SF4P RCP) Reactor Coolant Pump											01.32	G2.1.32 – Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)	4.0	86
004 (SF1; SF2 CVCS) Chemical and Volume Control														
005 (SF4P RHR) Residual Heat Removal														
006 (SF2; SF3 ECCS) Emergency Core Cooling											04.30	G2.4.30 – Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (CFR: 41.10 / 43.5 / 45.11)	4.1	87
007 (SF5 PRTS) Pressurizer Relief/Quench Tank								02				A2.02 – Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal pressure in the PRT. (CFR: 41.5 / 43.5 / 45.3 / 45.13)	3.2	88
008 (SF8 CCW) Component Cooling Water														
010 (SF3 PZR PCS) Pressurizer Pressure Control														
012 (SF7 RPS) Reactor Protection														
013 (SF2 ESFAS) Engineered Safety Features Actuation								06				A2.06 – Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent ESFAS actuation. (CFR: 41.5 / 43.5 / 45.3 / 45.13)	4.0	89
022 (SF5 CCS) Containment Cooling														
025 (SF5 ICE) Ice Condenser														
026 (SF5 CSS) Containment Spray														
039 (SF4S MSS) Main and Reheat Steam														
059 (SF4S MFW) Main Feedwater														
061 (SF4S AFW) Auxiliary/Emergency Feedwater														
062 (SF6 ED AC) AC Electrical Distribution														
063 (SF6 ED DC) DC Electrical Distribution														
064 (SF6 EDG) Emergency Diesel Generator											02.37	G2.2.37 – Ability to determine operability and/or availability of safety related equipment. (CFR: 41.7 / 43.5 / 45.12)	4.6	90
073 (SF7 PRM) Process Radiation Monitoring														

076 (SF4S SW) Service Water													
078 (SF8 IAS) Instrument Air													
103 (SF5 CNT) Containment													
053 (SF1; SF4P ICS*) Integrated Control													
K/A Category Point Totals:								2			3	Group Point Total:	5

ES-401												PWR Examination Outline Plant Systems—Tier 2/Group 2 (SRO)		Form ES-401-2
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
001 (SF1 CRDS) Control Rod Drive														
002 (SF2; SF4P RCS) Reactor Coolant														
011 (SF2 PZR LCS) Pressurizer Level Control														
014 (SF1 RPI) Rod Position Indication								04				A2.04 – Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Misaligned rod. (CFR: 41.5 / 43.5 / 45.3 / 45.13)	3.9	91
015 (SF7 NI) Nuclear Instrumentation														
016 (SF7 NNI) Nonnuclear Instrumentation														
017 (SF7 ITM) In-Core Temperature Monitor														
027 (SF5 CIRS) Containment Iodine Removal														
028 (SF5 HRPS) Hydrogen Recombiner and Purge Control														
029 (SF8 CPS) Containment Purge											01.23	G2.1.23 – Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6)	4.4	92
033 (SF8 SFPCS) Spent Fuel Pool Cooling														
034 (SF8 FHS) Fuel-Handling Equipment				01								K4.01 – Knowledge of design feature(s) and/or interlock(s) which provide for the following: Fuel protection from binding and dropping. (CFR: 41.7)	3.4	93
035 (SF 4P SG) Steam Generator														
041 (SF4S SDS) Steam Dump/Turbine Bypass Control														
045 (SF 4S MTG) Main Turbine Generator														
055 (SF4S CARS) Condenser Air Removal														
056 (SF4S CDS) Condensate														
068 (SF9 LRS) Liquid Radwaste														
071 (SF9 WGS) Waste Gas Disposal														
072 (SF7 ARM) Area Radiation Monitoring														
075 (SF8 CW) Circulating Water														
079 (SF8 SAS**) Station Air														
086 Fire Protection														
050 (SF 9 CRV*) Control Room Ventilation														
K/A Category Point Totals:	0	0	0	1	0	0	0	1	0	0	1	Group Point Total:		3

Facility: **Ginna Nuclear Power Plant** Date of Exam: **October 5 – 16, 2020**

Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.1	G2.1.1 – Knowledge of conduct of operations requirements. (CFR: 41.10 / 45.13)	3.8	66		
	2.1.26	G2.1.26 – Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen. (CFR: 41.10 / 45.12)	3.4	67		
	2.1.29	G2.1.29 – Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.. (CFR: 41.10 / 45.1 / 45.12)	4.1	68		
	2.1.37	G2.1.37 – Knowledge of procedures, guidelines, or limitations associated with reactivity management. (CFR: 41.1 / 43.6 / 45.6)			4.6	94
	2.1.41	G2.1.41 – Knowledge of the refueling process. (CFR: 41.2 / 41.10 / 43.6 / 45.13)			3.7	95
	Subtotal			3		2
2. Equipment Control	2.2.13	G2.2.13 - Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)	4.1	69		
	2.2.14	G2.2.14 – Knowledge of the process for controlling equipment configuration or status. (CFR: 41.10 / 43.3 / 45.13)	3.9	70		
	2.2.43	G2.2.43 – Knowledge of the process used to track inoperable alarms. (CFR: 41.10 / 43.5 / 45.13)	3.0	71		
	2.2.5	G2.2.5 – Knowledge of the process for making design or operating changes to the facility . (CFR: 41.10 / 43.3 / 45.13)			3.2	96
	2.2.21	G2.2.21 – Knowledge of pre- and post-maintenance operability requirements. (CFR: 41.10 / 43.2)			4.1	97
	Subtotal			3		2
3. Radiation Control	2.3.14	G2.3.14 – Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)	3.4	72		
	2.3.15	G2.3.15 – Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (CFR: 41.12 / 43.4 / 45.9)	2.9	73		
	2.3.13	G2.3.13 – Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 43.4 / 45.9 / 45.10)			3.8	98
	Subtotal			2		1
4. Emergency Procedures/Plan	2.4.14	G2.4.14 – Knowledge of general guidelines for EOP usage. (CFR: 41.10 / 45.13)	3.8	74		
	2.4.43	G2.4.43 – Knowledge of emergency communication systems and techniques. (CFR: 41.10 / 45.13)	3.2	75		

	2.4.22	G2.4.22 – Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. (CFR: 41.7 / 41.10 / 43.5 / 45.12)			4.4	99
	2.4.40	G2.4.40 – Knowledge of SRO responsibilities in emergency plan implementation. (CFR: 41.10 / 43.5 / 45.11)			4.5	100
	Subtotal			2		2
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
Tier 1 / Group 2	028 **AK3.03** RO Q20	Facility was unable to develop an operationally valid and discriminating RO level question to test the selected K/A ("Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions: *False indication of PZR level when PORV or spray valve is open and RCS saturated*"). (IR 3.5) **8/26/20 – CHIEF EXAMINER randomly re-selected AK2.03 from amongst APE 028, K/A Categories AK1 / AK2 / AK3, to maintain K/A Category balance within the outline:** "Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Controllers and positioners." (IR 2.6)