Facility: **James A. Fitzpatrick** Scenario No.: **NRC-1** Op-Test No.: **17-2**

Examiners: ______________________ Operators: ______________________

Initial Conditions: The plant is operating at approximately 75% power. IAC C is out of service.

Turnover: Start Service Water pump A, then secure Service Water pump B per OP-42 section G.1.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Swap Service Water Pumps OP-42
2	N/A	R – ATC, SRO	Raise Reactor Power with Recirculation Flow OP-27
3	MC11	C – All	Intake Structure Blockage AOP-56
4	Overrides SW18:B	C – BOP, SRO	EDG D Spuriously Starts; ESW B Fails to Start AOP-77, Technical Specifications
5	RD12	C – ATC, SRO	Control Rod Drift In AOP-27, Technical Specifications
6	AD06:C AD08:C MS16:C	M – All	SRV C Fails Open; SRV C Tailpipe Break in Torus Airspace AOP-36, AOP-1, EOP-2, EOP-4
7	RH01 Remote	C – All	RHR Pumps B and D Trip; RHR Loop A Suction Valve Drifts Closed and Delayed Pump Trips EOP-4
8	Override	C – All	RHRSW to RHR Crosstie Valves Fail Closed EOP-4, EOP-2

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

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-1**		Op-Test No.: **17-2**
1. Malfunctions after EOP entry (1-2) **Events 7 & 8**	2	
2. Abnormal events (2-4) **Events 3, 4, 5**	3	
3. Major transients (1-2) **Event 6**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-2, EOP-4**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) **EOP-2 Emergency Depressurization Leg**	1	
6. Pre-identified critical tasks (≥2)	2	
CRITICAL TASK DESCRIPTIONS: **CT-1: Given a stuck open SRV, the crew will scram the Reactor before Torus water temperature exceeds the Boron Injection Initiation Temperature, in accordance with AOP-36 and EOP-4.** **CT-2: Given the inability to maintain Primary Containment conditions inside the Pressure Suppression Pressure, the crew will perform an Emergency RPV Depressurization, in accordance with EOP-4. Emergency RPV Depressurization must be initiated within 15 minutes of Pressure Suppression Pressure being exceeded.**		

The scenario will begin at approximately 75% power. IAC C is inoperable. The crew will begin by starting Service Water pump A and securing Service Water pump B per OP-42. Next, the crew will raise Reactor power to approximately 80% using Recirculation flow.

An in-flux of lake grass in the intake structure causes high traveling screen differential level and lowering intake level. The crew will execute AOP-56. The crew will lower Reactor power and remove Circulating Water pump C from service. The lower intake flow will alleviate the clogging of the traveling screens.

EDG D will spuriously start and ESW pump B will fail to start, both automatically and manually. The crew will execute AOP-77, Inadvertent Initiation of ECCS or RCIC, and secure EDG D. The SRO will address Technical Specifications.

Control Rod 18-27 will drift full in. The crew will execute AOP-27, Control Rod Malfunction, perform a rapid power reduction, and fully insert the control rod. The SRO will once again address Technical Specifications.

Safety Relief Valve C opens and sticks open. The crew will execute AOP-36, Stuck Open Relief Valve, and determine the valve cannot be closed. The SRO will direct a Reactor scram before Torus temperature reaches 110°F.

The SRO will enter EOP-2, RPV Control, and EOP-4, Primary Containment Control. Additionally, following the scram, the tailpipe of the stuck open SRV will break. This causes Primary Containment pressure to rapidly rise. RHR pumps B and D will trip immediately after starting.

The SRO will direct the Torus and Drywell to be sprayed using RHR loop A. However, the RHR loop A suction valve will drift closed, resulting in lowering suction pressure to RHR pumps A and C, as well as lowered spray flow. Eventually, RHR pumps A and C will trip. RHRSW crossties to RHR will fail closed, preventing alternate spray. This will result in Pressure Suppression Pressure (PSP) to be violated, which requires an Emergency RPV Depressurization. The SRO will direct an Emergency RPV Depressurization and the crew will open 7 SRVs.

The scenario will be terminated when all control rods are inserted, Emergency RPV Depressurization is in progress, and Reactor water level is controlled above 0”.

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-2** Op-Test No.: **17-2**

Examiners: ______________ Operators: ______________

Initial Conditions: The plant is operating at approximately 35% power. Feedwater is in single element control. IAC C is out of service.

Turnover: Swap Feedwater to three element control per OP-2A section G.32.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Swap Feedwater to Three Element Control OP-2A
2	N/A	R – ATC, SRO	Withdraw Control Rods OP-65, OP-26
3	RW01	I – ATC, SRO	Rod Worth Minimizer Fails ARP 09-5-2-1
4	RD11	C – ATC, SRO	Uncoupled Control Rod OP-26, AOP-25
5	SW09:B SW17:C	C – BOP, SRO	RBCLC Pump B Trips; RBCLC Pump C Fails to Automatically Start AOP-11
6	RC03 RC04 RC06	C – BOP, SRO	RCIC Inadvertent Start, RCIC Fails to Trip on Automatic Signal AOP-77, Technical Specifications
7	ED43:A	C – SRO	Loss of Line 3 AOP-72, Technical Specifications
8	Report	M – All	Code Red AOP-70, AOP-1, EOP-2
9	CU10 CU12	I – BOP, SRO	RWCU Fails to Automatically Isolate AOP-70
10	ED43:B	C – All	Loss of Offsite Power AOP-72, AOP-1

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

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-2**		Op-Test No.: **17-2**
1. Malfunctions after EOP entry (1-2) **Events 9,10**	2	
2. Abnormal events (2-4) **Events 4,5,6,7**	4	
3. Major transients (1-2) **Event 8**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-2**	1	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set)	0	
6. Pre-identified critical tasks (≥2)	2	
CRITICAL TASK DESCRIPTIONS: **CT-1: Given a Code Red, manually scram the Reactor, in accordance with AOP-70. The scram must be initiated within 15 minutes of Control Room notification of the Code Red.** **CT-2: Given a Code Red and failure of RWCU to automatically isolate, isolate RWCU, in accordance with AOP-70. The isolation must be performed within 15 minutes of the automatic isolation failure.**		

The scenario will begin at approximately 35% power. Feedwater is in single element control. IAC C is inoperable. The crew will begin by swapping Feedwater to three element control per OP-2A. Then the crew will continue the startup by withdrawing control rods.

After the first control rod is withdrawn, the Rod Worth Minimizer will experience a hardware failure that will block both rod withdrawal and insertion. The crew will bypass the Rod Worth Minimizer to allow further control rod withdrawals.

The second control rod to be moved, 42-43, will be uncoupled. When control rod 42-43 is at position 48, it will become apparent that the control rod is uncoupled. The crew will execute AOP-25 to re-couple the control rod and continue with the power ascension.

RBCLC pump B trips and RBCLC pump C fails to automatically start. System pressure will lower. Eventually, system high temperature will develop. The crew will start RBCLC pump A to restore system conditions.

RCIC will inadvertently start and develop an adverse condition that requires an automatic trip, however the automatic trip will fail to occur. The crew will enter AOP-77 and manually secure RCIC. The SRO will determine the Technical Specification impact.

115 KV offsite power Line #3 will de-energize due to a fault. The crew will execute AOP-72 and the SRO will determine the Technical Specification impact.

Security will call the Control Room and report that a ground attack is in progress (Code Red). The crew will execute AOP-70, scram the Reactor and start all EDGs. RCIC will be unavailable for injection due to the earlier event. RWCU will fail to automatically isolate on low Reactor water level. The crew will manually isolate RWCU. The crew will initiate a cooldown using Turbine Bypass Valves.

115 KV offsite power Line #4 will de-energize due to a fault. This will result in a loss of all offsite power. The crew will execute AOP-72 and an override in AOP-1 to close MSIVs due to loss of all Circ Water pumps. The crew will switch pressure control from Turbine Bypass Valves to SRVs and continue the cooldown. The loss of offsite power will also result in a loss of all Condensate and Feedwater. The crew will transition level control to HPCI. Due to heat addition to the Torus from SRVs and HPCI, the crew will initiate Torus Cooling. Security will eventually call the Control Room and report that the threat has been neutralized.

The scenario will be terminated when all control rods are inserted, Reactor pressure is controlled less than 1080 psig, and Reactor water level is controlled above 0”.

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-3** Op-Test No.: **17-2**

Examiners: ______________________ Operators: ______________________

Initial Conditions: The plant is operating at approximately 5% power during a startup. HPCI is out of service and ready to be restored to a standby lineup. IAC C is out of service. Standby Gas Treatment fan B is operating for Containment inerting.

Turnover: Restore HPCI to a standby lineup per OP-15. Then, continue power ascension by withdrawing control rods. Complete the current rod group, then hold for Reactor Engineering.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Restore HPCI to a Standby Lineup OP-15
2	N/A	R – ATC, SRO	Withdraw Control Rods OP-65, OP-26
3	NM08:E	I – ATC, SRO	IRM Fails Inop ARPs, OP-16, Technical Specifications
4	PC04:B	C – BOP, SRO	Standby Gas Treatment Fan B Trips ARP 09-75-2-24(32), OP-20, Technical Specifications
5	EP02 Remote ED23	I – BOP, SRO	Seismic Event, LPCI Inverter Trips AOP-14, ARP 09-8-3-2, Technical Specifications
6	OG03	C – All	Explosion in Air Ejector Discharge ARP 09-6-1-7(15), AOP-4, AOP-1, EOP-2
7	EP01 RH10 Overrides	M – All	Second Seismic Event; RHR Suction Piping Leak; RHR Suction Fails to Isolate EOP-2, EOP-4
8	Overrides	C – ATC, SRO	Feedwater Low Flow Control Valve Fails Closed EOP-2
9	RP03	I – All	MSIVs Spuriously Isolate EOP-2

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

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-3**		Op-Test No.: **17-2**
1. Malfunctions after EOP entry (1-2) **Events 8, 9**	2	
2. Abnormal events (2-4) **Events 5, 6**	2	
3. Major transients (1-2) **Event 7**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-2, EOP-4**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) **EOP-2 Emergency Depressurization Leg**	1	
6. Pre-identified critical tasks (≥2)	2	
CRITICAL TASK DESCRIPTIONS: **CT-1: Given an un-isolable Torus water leak and the inability to maintain Torus water level above 10.75', the crew will initiate a manual HPCI turbine trip, in accordance with EOP-4. HPCI must be tripped before Torus water level lowers below 9.58 feet.** **CT-2: Given an un-isolable Torus water leak and the inability to maintain Torus water level above 9.58', the crew will perform an Emergency RPV Depressurization, in accordance with EOP-2. The depressurization must be initiated before Torus water level lowers below 5.5 feet.**		

The scenario will begin at approximately 5% power during a startup with HPCI and IAC C out of service. Standby Gas Treatment fan B is operating for Containment inerting. The crew will begin by restoring HPCI to service per OP-15. The crew will continue by raising Reactor power by withdrawing control rods. During control rod withdrawals, IRM E will fail. The crew will bypass IRM E and reset a half scram in order to continue with the startup.

Standby Gas Treatment fan B will trip. The crew will swap Standby Gas Treatment trains and the SRO will determine the Technical Specification impact.

A small seismic event will occur. The AC input to the A LPCI inverter will fail. The crew will transfer the A LPCI inverter to the alternate AC supply. The crew will execute AOP-14 in response to the earthquake. The SRO will determine the Technical Specification impact.

A hydrogen explosion will occur in the Steam Jet Air Ejector discharge piping. The crew will enter AOP-4 and insert a manual Reactor scram. The crew will take actions per AOP-1 and/or EOP-2 to stabilize the plant.

A second, larger seismic event will occur. This will cause the suction pipe from the Torus to the RHR system to break. The crew will attempt to isolate the leak, but a valve failure will result in the break being un-isolable. The crew will initiate Torus makeup, but Torus water level will continue to lower. The crew will trip HPCI as Torus water level lowers. The MSIVs will spuriously isolate, limiting the crew’s ability to anticipate Emergency Depressurization. The crew will determine that Torus water level cannot be maintained above 9.58 feet and will perform an Emergency Depressurization.

The scenario will be terminated when all control rods are inserted, an Emergency Depressurization is in progress, and Reactor water level is controlled above zero inches.

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-4** Op-Test No.: **17-2**

Examiners: ______________ Operators: ______________

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Initial Conditions: The plant is operating at approximately 100% power. IAC C is out of service.

Turnover: Perform Core Spray full flow testing per ST-3PA, starting at step 8.7.6.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Perform Core Spray Full Flow Test ST-3PA
2	Override	C – BOP, SRO	Core Spray Pump Overload ARP 09-3-1-31, Technical Specifications
3	TU04	R – ATC, SRO C – BOP	Main Turbine Bearing High Vibration AOP-66
4	ED19:D	C – All	Electrical Fault on 10400 Bus AOP-17, AOP-8, AOP-60, Technical Specifications
5	TU04	C – ATC, SRO	Main Turbine Bearing High Vibration AOP-66, AOP-1
6	RP01AA RP01BA RP09 RD10	M – All	Failure of RPS and ARI to Actuate; Multiple Control Rods Stuck EOP-2, EOP-3
7	SL02 RR13	C – ATC, SRO	First SLC Squib Valve Fails to Fire, Recirculation Pumps Fail to Automatically Trip EOP-3
8	FW19	C – All	Trip of Condensate Pump A, HPCI, and CRD Pump A; RCIC Flow Controller Fails Low AOP-41, EOP-3, EOP-3A
9	RR15:A	C – All	Coolant Leak in Drywell EOP-3, EOP-3A

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

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-4** Op-Test No.: **17-2**		
1. Malfunctions after EOP entry (1-2) **Events 7,8,9**	3	
2. Abnormal events (2-4) **Events 3,4,5**	3	
3. Major transients (1-2) **Event 6**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-2, EOP-4**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) **EOP-3, EOP-3a**	2	
6. Pre-identified critical tasks (≥2)	3	
CRITICAL TASK DESCRIPTIONS: **CT-1: Given a failure to scram with Reactor power above 2.5%, the crew will lower Reactor power by one or more of the following methods, in accordance with EOP-3:** • **Terminating and preventing all RPV injection except SLC, RCIC, and CRD** • **Tripping Recirculation pumps** • **Injecting boron** **The Reactor power reduction must be initiated within five minutes of the start of the failure to scram.** **CT-2: Given a failure to scram, the crew will initiate Control Rod insertion, in accordance with EOP-3. All insertable control rods must be inserted within one hour of the start of the failure to scram.** **CT-3: Given a coolant leak, a loss of high pressure injection systems, the inability to restore and maintain Reactor water level above -19", and multiple stuck control rods, the crew will initiate actions for an Emergency RPV Depressurization, in accordance with EOP-3a. Reactor water level must be restored and maintained above -19" within thirty minutes of lowering below -19".**		

The scenario will begin with the plant operating at approximately 100% power. The crew will begin by performing Core Spray pump testing per ST-3PA. When the Core Spray test valve is opened, an overload condition will develop on Core Spray pump A. The crew will perform ARP actions to lower pump flow and then stop the pump. The SRO will determine the Technical Specification impact.

Main Turbine high vibrations will develop. The crew will enter AOP-66 to address the vibrations. The crew will first lower Main Generator reactive loading, then Reactor power. The vibrations will subside as Reactor power is lowered.

A sustained electrical fault will occur on 4160 VAC Bus 10400. This results in a half scram on RPS Bus B and a trip of RWR pump B. The crew will execute AOP-17, AOP-8 and AOP-60.

Main Turbine high vibrations will re-develop. The crew will re-enter AOP-66 to address the vibrations. The crew's attempt to lower vibrations by lowering Reactor power will be unsuccessful. The crew will enter AOP-1 and attempt a Reactor scram.

RPS A will fail to process the scram and ARI will also fail to insert control rods. The crew will enter EOP-2 and EOP-3. The ATWS system will fail to automatically trip the Recirculation pumps when required. The crew will lower Recirculation flow to minimum and then trip the Recirculation pumps. The crew will terminate and prevent injection except CRD, SLC, and RCIC. The crew will attempt to inject boron using SLC, however the squib valves will fail to fire when the first SLC pump is started. The second SLC pump will successfully start and inject. The crew will be able to manually insert all but 16 control rods.

Once the majority of control rods are inserted, a coolant leak will develop in the Drywell and multiple high pressure injection sources will be lost. Condensate pump A will trip, resulting in a loss of all Condensate and Feedwater. Condensate pump B is unavailable due to the loss of Bus 10400. Condensate pump C is unavailable following the Main Generator trip due to de-energization of Bus 10700. CRD pump A will trip and CRD pump B is unavailable due to the loss of Bus 10400. Two minutes later, a HPCI trip will occur and seal-in. Another two minutes later, RCIC flow controller will fail low. Reactor water level will lower and not be able to be maintained above -19". The crew will perform an Emergency RPV Depressurization per EOP-3a. The crew will terminate and prevent all injection due to the multiple stuck control rods. Once Reactor pressure lowers below 129 psig, the crew will re-inject and restore Reactor water level above -19".

The scenario will be terminated when the insertable control rods are all inserted, an Emergency RPV Depressurization has been performed, and Reactor water level is controlled above -19".

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-5** Op-Test No.: **17-2**

Examiners: ______________________ Operators: ______________________

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Initial Conditions: The plant is operating at approximately 80% power. L-34 is aligned to the alternate supply following breaker maintenance. Reactor Building ventilation is isolated for maintenance and Standby Gas Treatment fan B is operating.

Turnover: Restore L-34 to the normal supply per OP-46A section G.12.3. The procedure is in progress up to step G.12.3.d. Then, raise Reactor power with Recirculation flow.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Restore L-34 to Normal Power Source OP-46A
2	N/A	R – ATC, SRO	Raise Reactor Power with Recirculation Flow OP-27
3	RD06:A RD09	C – ATC, SRO	Control Rod Drive Pump Trip, One Control Rod Drive Accumulator Low Pressure AOP-69, ARP-09-5-1-43, Technical Specifications
4	ED21:A Override	C – BOP, SRO	Loss of L15, Drywell Cooling Fan Fails to Automatically Start AOP-18A, Technical Specifications
5	Override	C – BOP, SRO	Loss of Steam Packing Exhauster A ARP 09-7-3-43, OP-24D
6	CU07 CU10 CU11 CU12	M – All	RWCU Steam Leak into Reactor Building; RWCU Fails to Isolate Automatically and Manually EOP-5, EOP-2, AOP-1
7	RP01BB RP01AA	I – ATC, SRO	RPS Fails to Scram, ARI Works AOP-1
8	Override	C – All	Bypass Opening Jack Fails to Open Beyond 5% AOP-1, EOP-2

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

Facility: **James A. Fitzpatrick** Scenario No.: **NRC-5**		Op-Test No.: **17-2**
1. Malfunctions after EOP entry (1-2) **Events 7,8**	2	
2. Abnormal events (2-4) **Events 3,4**	2	
3. Major transients (1-2) **Event 6**	1	
4. EOPs entered/requiring substantive actions (1-2) **EOP-2, EOP-5**	2	
5. Entry into a contingency EOP with substantive actions (≥1 per scenario set) **EOP-2 Emergency Depressurization Leg**	1	
6. Pre-identified critical tasks (≥2)	2	
CRITICAL TASK DESCRIPTIONS: **CT-1: Given the plant operating at power with an un-isolable primary system discharging into Secondary Containment and RPS failing to scram the Reactor, the crew will manually initiate ARI, in accordance with EOP-5 and AOP-1. ARI must be manually initiated within 5 minutes of when an area reaching the Maximum Safe Temperature.** **CT-2: Given an un-isolable primary system discharging into Secondary Containment and two areas exceeding Maximum Safe Temperatures, the crew will perform an emergency RPV depressurization, in accordance with EOP-5. The emergency RPV depressurization must be initiated within 15 minutes of when the second Maximum Safe Temperature is exceeded.**		

The scenario will begin with the plant operating at approximately 80% power. Reactor Building ventilation is isolated for maintenance and Standby Gas Treatment fan B is operating. L-34 is aligned to the alternate supply following breaker maintenance. The crew will begin by restoring L-34 to the normal power source per OP-46A. Then, the crew will raise Reactor power with Recirculation flow.

The running Control Rod Drive pump will trip. The crew will enter AOP-69, start the standby pump, and restore normal Control Rod Drive parameters. During this evolution, one Control Rod Drive accumulator will develop a low pressure. The SRO will determine the Technical Specification impact.

An electrical fault will develop on Bus L-15. The crew will execute AOP-18A. One Drywell Cooling fan will fail to auto-start. The BOP will start the Drywell Cooling fan. Multiple important loads will be lost, including RHR A Containment spray/cooling valves, Core Spray A injection valves, RWCU isolation valve 12MOV-15, and SLC pump A. The SRO will determine the Technical Specification impact.

The running Steam Packing Exhauster will trip. The crew will respond to the ARP to start the standby Steam Packing Exhauster, and re-establish adequate system parameters.

RWCU will develop a steam leak. This will cause high area temperatures in the Reactor Building. RWCU will fail to automatically isolate. The crew will be able to close 12MOV-18, however 12MOV-15 will fail to close due to earlier loss of power, preventing isolation of the steam leak. The crew will execute EOP-5. The crew will scram the Reactor and enter AOP-1 and EOP-2. RPS will fail to scram the Reactor, but ARI will successfully insert control rods. As Reactor Building area temperatures approach max safe levels, the crew will attempt to anticipate Emergency Depressurization by rapidly lowering Reactor pressure with Turbine Bypass Valves. The Bypass Opening Jack will begin to open a Turbine Bypass Valve, but will be limited to the first 5% of demand. This will limit how quickly Reactor pressure can be lowered. Once two max safe temperatures are exceeded, the crew will perform an Emergency Depressurization.

The scenario will be terminated when all control rods are inserted, the Emergency RPV Depressurization is in progress, and Reactor water level is controlled above 0”.

Facility:	James A. Fitzpatrick	Date of Examination:	Jan 2019
Examination Level:	RO	Operating Test Number:	17-2

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, R	Re-activation of RO License K/A 2.1.4 (3.8), OP-AA-105-102
Conduct of Operations	N, R	Perform Circulating Water Temperature Check with EPIC Out of Service K/A 2.1.7 (4.4), ST-40C
Equipment Control	D, R	Determine Tagout Boundary For RBCLC Pump Work K/A 2.2.13 (4.1), EN-OP-102
Radiation Control	N, S	Perform Daily Check of Radiation Monitors K/A 2.3.5 (2.9), ST-40X
Emergency Plan		

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

* Type Codes and Criteria:

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

Facility:	James A. Fitzpatrick	Date of Examination:	Jan 2019
Examination Level:	SRO	Operating Test Number:	17-2

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, R	Re-activation of SRO License K/A 2.1.4 (3.8), OP-AA-105-102
Conduct of Operations	N, R	Perform Circulating Water Temperature Check with EPIC Out of Service, Determine Reporting Requirement K/A 2.1.7 (4.7), ST-40C
Equipment Control	D, R	Perform Technical Specification Evaluation and LCO Tracking for Inoperable Turbine Bypass Valves K/A 2.2.22 (4.7), COLR, TS 3.7.6
Radiation Control	P, D, R 17-1 NRC	Determine Actions for Inoperable Stack Radiation Monitor K/A 2.3.11 (4.3), ODCM
Emergency Plan	M, R	Determine Emergency Classification and Initiate Event Notification K/A 2.4.40 (4.5), EP-AA-1014, IAP-2, EP-CE-114-100

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

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

Facility:	James A. Fitzpatrick	Date of Examination:	Jan 2019
Exam Level:	RO / SRO-I	Operating Test Number:	17-2

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

System/JPM Title	Type Code*	Safety Function
a. Reactor Protection System / Reset ARI and Scram, Perform Post Scram Reset Control Rod Position Check K/A 212000 A4.14 (3.8/3.8), AOP-1	P, D, EN, S 16-1 NRC	7
b. A.C. Electrical Distribution / Transfer Bus 10100 from Reserve to Normal Using Single Meter Voltage Match Method K/A 262001 A4.04 (3.6/3.7), OP-46A	P, D, A, S 16-1 NRC	6
c. Reactor Core Isolation Cooling System / Initiate RCIC for Injection with Speed Failure K/A 217000 A4.01 (3.7/3.7), OP-19	M, A, EN, S	2
d. Standby Gas Treatment System / Manually Isolate Reactor Building Ventilation, Low Reactor Building D/P K/A 261000 A4.06 (3.3/3.6), OP-51A, OP-20	D, A, EN, S	9
e. Main and Reheat Steam System / Re-open MSIVs With RPV Pressurized, Steam Leak Develops K/A 239001 A4.04 (3.8/3.7), EP-9	M, A, L, S	3
f. Main Turbine Generator and Auxiliary Systems / Emergency Main Turbine Shutdown with High Vibrations K/A 245000 A4.06 (2.7/2.6), OP-9	N, A, L, S	4
g. Component Cooling Water System / Isolate RBCLC Supply to the Drywell K/A 400000 A4.01 (3.1/3.0), EP-12	D, S	8
h. Recirculation System / Start Reactor Recirculation Pump **(RO Only)** K/A 202001 A4.01 (3.7/3.7), OP-27	M, L, S	1

In-Plant Systems:* 3 for RO, 3 for SRO-I, and 3 or 2 for SRO-U		
i. Recirculation Flow Control System / Local Manual Operation of RWR MG Set Speed K/A 202002 2.1.30 (4.4/4.0), OP-27	N, R	1
j. Emergency Generators (Diesel/Jet) / Supply Cooling Water to EDGs A and C from ESW Pump B K/A 295003 AA1.02 (4.2/4.3), OP-22	D, E	6
k. Standby Liquid Control System / Lineup SLC Test Tank for Injection K/A 295031 EA1.08 (3.8/3.9), EP-8	D, R, E	2

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

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

<table>
<tr><td colspan="20">Facility: J.A. FitzPatrick Date of Exam: 1/14/2019</td></tr>
<tr><td rowspan="2">Tier</td><td rowspan="2">Group</td><td colspan="12">RO K/A Category Points</td><td colspan="6">SRO-Only Points</td></tr>
<tr><td>K1</td><td>K2</td><td>K3</td><td>K4</td><td>K5</td><td>K6</td><td>A1</td><td>A2</td><td>A3</td><td>A4</td><td>G*</td><td>Total</td><td colspan="2">A2</td><td colspan="2">G*</td><td colspan="2">Total</td></tr>
<tr><td rowspan="3">1. Emergency and Abnormal Plant Evolutions</td><td>1</td><td>4</td><td>3</td><td>4</td><td colspan="3" rowspan="3">N/A</td><td>3</td><td>3</td><td colspan="2" rowspan="3">N/A</td><td>3</td><td>20</td><td colspan="2">3</td><td colspan="2">4</td><td colspan="2">7</td></tr>
<tr><td>2</td><td>1</td><td>2</td><td>1</td><td>1</td><td>1</td><td>1</td><td>7</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td></tr>
<tr><td>Tier Totals</td><td>5</td><td>5</td><td>5</td><td>4</td><td>4</td><td>4</td><td>27</td><td colspan="2">4</td><td colspan="2">6</td><td colspan="2">10</td></tr>
<tr><td rowspan="3">2. Plant Systems</td><td>1</td><td>3</td><td>1</td><td>3</td><td>2</td><td>2</td><td>3</td><td>3</td><td>3</td><td>2</td><td>2</td><td>2</td><td>26</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">5</td></tr>
<tr><td>2</td><td>1</td><td>1</td><td>1</td><td>2</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>1</td><td>12</td><td>0</td><td>1</td><td colspan="2">2</td><td colspan="2">3</td></tr>
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<tr><td colspan="5" rowspan="2">3. Generic Knowledge and Abilities Categories</td><td colspan="2">1</td><td colspan="2">2</td><td colspan="2">3</td><td colspan="2">4</td><td rowspan="2">10</td><td>1</td><td>2</td><td>3</td><td>4</td><td colspan="2" rowspan="2">7</td></tr>
<tr><td colspan="2">3</td><td colspan="2">3</td><td colspan="2">2</td><td colspan="2">2</td><td>2</td><td>2</td><td>2</td><td>1</td></tr>
</table>

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

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points, and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the outline. Systems or evolutions that do not apply at the facility should be deleted with justification. Operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible. Sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7. The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics’ IRs for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. If fuel-handling equipment is sampled in a category other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2. (Note 1 does not apply.) Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

G* Generic K/As

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

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

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions—Tier 1/Group 1 (RO/SRO)		Form ES-401-1
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295003 (APE 3) Partial or Complete Loss of AC Power / 6						X	G2.4.18 Knowledge of the specific bases for EOPs.	4.0	
295016 (APE 16) Control Room Abandonment / 7					X		AA2.06 Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : Cooldown rate	3.5	
295018 (APE 18) Partial or Complete Loss of CCW / 8					X		AA2.04 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : System flow	2.9	
295021 (APE 21) Loss of Shutdown Cooling / 4						X	G2.1.30 Ability to locate and operate components, including local controls.	4.0	
295023 (APE 23) Refueling Accidents / 8					X		AA2.04 Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : Occurrence of fuel handling accident	4.1	
295025 (EPE 2) High Reactor Pressure / 3						X	G2.1.31 Ability to locate control room switches, controls, and indications, and todetermine that they correctly reflect the desired plant lineup.	4.3	
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1						X	G2.2.37 Ability to determine operability and/or availability of safety related equipment.	4.6	
295001 (APE 1) Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4				X			AA1.06 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Neutron monitoring system	3.3	
295003 (APE 3) Partial or Complete Loss of AC Power / 6					X		AA2.01 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Cause of partial or complete loss of A.C. power	3.4	
295004 (APE 4) Partial or Total Loss of DC Power / 6						X	G2.4.11 Knowledge of abnormal condition procedures.	4.0	
295005 (APE 5) Main Turbine Generator Trip / 3	X						AK1.01 Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP :Pressure effects on reactor power	4.0	
295006 (APE 6) Scram / 1	X						AK1.03 Knowledge of the operational implications of the following concepts as they apply to SCRAM : Reactivity control	3.7	
295016 (APE 16) Control Room Abandonment / 7			X				AK3.03 Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT : Disabling control room controls	3.5	
295018 (APE 18) Partial or Complete Loss of CCW / 8						X	G2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	3.8	

295019 (APE 19) Partial or Complete Loss of Instrument Air / 8		X					AK2.14 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Plant air systems	3.2	
295021 (APE 21) Loss of Shutdown Cooling / 4						X	G2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	3.8	
295023 (APE 23) Refueling Accidents / 8			X				AK3.02 Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS : Interlocks associated with fuel handling equipment	3.4	
295024 High Drywell Pressure / 5			X				EK3.04 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : Emergency depressurization	3.7	
295025 (EPE 2) High Reactor Pressure / 3	X						EK1.06 Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: Pressure effects on reactor water level	3.5	
295026 (EPE 3) Suppression Pool High Water Temperature / 5				X			EA1.03 Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature monitoring	3.9	
295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only) / 5		X					EK2.01 Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell spray: Mark-I&II	3.7	
295030 (EPE 7) Low Suppression Pool Water Level / 5				X			EA1.01 Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: ECCS systems (NPSH considerations): Plant-Specific	3.6	
295031 (EPE 8) Reactor Low Water Level / 2					X		EA2.01 Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: Reactor water level	4.6	
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown / 1					X		EA2.01 Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Reactor power	4.2	
295038 (EPE 15) High Offsite Radioactivity Release Rate / 9			X				EK3.01 Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Implementation of site emergency plan	3.6	
600000 (APE 24) Plant Fire On Site / 8	X						AK1.01 Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire Classifications by type	2.5	
700000 (APE 25) Generator Voltage and Electric Grid Disturbances / 6		X					AK2.03 Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Sensors, detectors, indicators	3.0	
K/A Category Totals:	4	3	4	3	3/3	3/4	Group Point Total:		20/7

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295015 (APE 15) Incomplete Scram / 1						X	G2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	4.2	
295032 (EPE 9) High Secondary Containment Area Temperature / 5						X	G2.4.41 Knowledge of the emergency action level thresholds and classifications.	4.6	
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9					X		EA2.01 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : Ventilation radiation levels	4.2	
295009 (APE 9) Low Reactor Water Level / 2						X	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.	2.7	
295017 (APE 17) Abnormal Offsite Release Rate / 9			X				AK3.01 Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE : System isolations	3.6	
295020 (APE 20) Inadvertent Containment Isolation / 5 & 7					X		AA2.04 Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION : Reactor pressure	3.9	
295032 (EPE 9) High Secondary Containment Area Temperature / 5	X						EK1.02 Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Radiation releases	3.6	
295033 (EPE 10) High Secondary Containment Area Radiation Levels / 9		X					EK2.02 Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following: Process radiation monitoring system	3.8	
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9				X			EA1.03 Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : Secondary containment ventilation	4.0	
295036 (EPE 13) Secondary Containment High Sump/Area Water Level / 5		X					EK2.01 Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL and the following: Secondary containment equipment and floor drain system	3.1	
K/A Category Point Totals:	1	2	1	1	1/1	1/2	Group Point Total:		7/3

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

Plant Systems—Tier 2/Group 1 (RO/SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode								X				A2.02 Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Pump trips	3.5	
215004 (SF7 SRMS) Source Range Monitor								X				A2.04 Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Up scale and downscale trips	3.7	
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor											X	G2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.	4.2	
218000 (SF3 ADS) Automatic Depressurization											X	G2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	4.6	
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff											X	G2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.4	
203000 (SF2, SF4 RHR/LPCI) RHR/LPCI: Injection Mode									X			A3.09 Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including: Emergency generator load sequencing	3.6	
205000 (SF4 SCS) Shutdown Cooling				X								K4.05 Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: Reactor cooldown rate	3.6	
206000 (SF2, SF4 HPCIS) High-Pressure Coolant Injection					X							K5.07 Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM : System venting: BWR-2,3,4	2.8	
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray		X										K2.01 Knowledge of electrical power supplies to the following: Pump power	3.0	
211000 (SF1 SLCS) Standby Liquid Control								X				A2.07 Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve closures	2.9	
211000 (SF1 SLCS) Standby Liquid Control					X							K5.06 Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM : Tank level measurement	3.0	

212000 (SF7 RPS) Reactor Protection						X						K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM : Nuclear boiler instrumentation	3.5	
215003 (SF7 IRM) Intermediate-Range Monitor							X					A1.01 Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: Detector position	3.4	
215004 (SF7 SRMS) Source-Range Monitor								X				A2.02 Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: SRM inop condition	3.4	
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor							X					A1.04 Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: SCRAM and rod block trip setpoints	4.1	
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor						X						K6.07 Knowledge of the effect that a loss or malfunction of the following will have on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM : Flow converter/comparator network: Plant-Specific	3.2	
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling			X									K3.01 Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Reactor water level	3.7	
218000 (SF3 ADS) Automatic Depressurization								X				A2.06 Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: ADS initiation signals present	4.2	
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff			X									K3.14 Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: Recirculation system: Plant-Specific	3.0	
239002 (SF3 SRV) Safety Relief Valves											X	G2.2.43 Knowledge of the process used to track inoperable alarms.	3.0	
259002 (SF2 RWLCS) Reactor Water Level Control	X											K1.02 Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: Main steam flow	3.2	
261000 (SF9 SGTS) Standby Gas Treatment				X								K4.03 Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Moisture removal	2.5	

262001 (SF6 AC) AC Electrical Distribution										X		A4.01 Ability to manually operate and/or monitor in the control room: All breakers and disconnects (including available switch yard): Plant-Specific	3.4	
262001 (SF6 AC) AC Electrical Distribution			X									K3.05 Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: Off-site power system	3.2	
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)	X											K1.14 Knowledge of the physical connections and/or cause-effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Main steam line radiation monitors: Plant-Specific	2.8	
263000 (SF6 DC) DC Electrical Distribution										X		A4.03 Ability to manually operate and/or monitor in the control room: Battery discharge rate: Plant-Specific	2.7	
264000 (SF6 EGE) Emergency Generators (Diesel/Jet) EDG									X			A3.01 Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Automatic starting of compressor and emergency generator	3.0	
300000 (SF8 IA) Instrument Air	X											K1.03 Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: Containment air	2.8	
300000 (SF8 IA) Instrument Air						X						K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: Service air refusal valve	2.6	
400000 (SF8 CCS) Component Cooling Water							X					A1.01 Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: CCW flow rate	2.8	
400000 (SF8 CCS) Component Cooling Water											X	G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.4	
K/A Category Point Totals:	3	1	3	2	2	3	3	3/ 2	2	2	2/ 3	Group Point Total:		26/5

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

Plant Systems—Tier 2/Group 2 (RO/SRO)

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201006 (SF7 RWMS) Rod Worth Minimizer											X	G2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	4.6	
215001 (SF7 TIP) Traversing In Core Probe								X				A2.02 Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High primary containment pressure: Mark-I&II	3.0	
286000 (SF8 FPS) Fire Protection											X	G2.2.38 Knowledge of conditions and limitations in the facility license.	4.5	
202002 (SF1 RSCTL) Recirculation Flow Control									X			A3.03 Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Scoop tube operation: BWR-2,3,4	3.1	
215001 (SF7 TIP) Traversing In-Core Probe				X								K4.01 Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation: Mark-I&II	3.4	
215002 (SF7 RBMS) Rod Block Monitor								X				A2.02 Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss or reduction in recirculation system flow (flow comparator): BWR-3,4,5	3.3	
223001 (SF5 PCS) Primary Containment and Auxiliaries										X		A4.09 Ability to manually operate and/or monitor in the control room: SPDS/CRIDS/ERIS/GDS: Plant-Specific	2.5	
234000 (SF8 FH) Fuel-Handling Equipment						X						K6.02 Knowledge of the effect that a loss or malfunction of the following will have on the FUEL HANDLING EQUIPMENT: Reactor manual control system: Plant-Specific	2.8	
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary	X											K1.06 Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: Component cooling water systems	2.6	
259001 (SF2 FWS) Feedwater		X										K2.01 Knowledge of electrical power supplies to the following: Reactor feedwater pump(s): Motor-Driven-Only	3.3	
268000 (SF9 RW) Radwaste							X					A1.01 Ability to predict and/or monitor changes in parameters associated with operating the RADWASTE controls including: Radiation level	2.7	
272000 (SF7, SF9 RMS) Radiation Monitoring											X	G2.2.40 Ability to apply Technical Specifications for a system.	3.4	
286000 (SF8 FPS) Fire Protection				X								K4.02 Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Automatic system initiation	3.3	

288000 (SF9 PVS) Plant Ventilation					X							K5.01 Knowledge of the operational implications of the following concepts as they apply to PLANT VENTILATION SYSTEMS : Airborne contamination control	3.1	
290001 (SF5 SC) Secondary Containment			X									K3.01 Knowledge of the effect that a loss or malfunction of the SECONDARY CONTAINMENT will have on following: Off-site radioactive release rates	4.0	
K/A Category Point Totals:	1	1	1	2	1	1	1	1/1	1	1	1/2	Group Point Total:		12/3

Facility: J. A. FitzPatrick		Date of Exam: 1/14/2019				
Category	K/A #	Topic	RO		SRO-only	
			IR	#	IR	#
1. Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements	3.8			
	2.1.19	Ability to use the plant computer to evaluate system or component status	3.9			
	2.1.23	Ability to perform specific system and integrated plant procedures during all modes of plant operation	4.3			
	2.1.27	Knowledge of system purpose and/or function			4.0	
	2.1.32	Ability to explain and apply all system limits and precautions			4.0	
	Subtotal			3		2
2. Equipment Control	2.2.1	Ability to perform pre-startup procedures for the facility, including those controls associated with plant equipment that could affect reactivity	4.5			
	2.2.6	Knowledge of the process for making changes to procedures	3.0			
	2.2.7	Knowledge of the process for conducting special or infrequent tests	2.9			
	2.2.36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions of operations			4.2	
	2.2.37	Ability to determine operability and/or availability of safety related equipment			4.6	
	Subtotal			3		2
3. Radiation Control	2.3.15	Knowledge of radiation monitoring systems	2.9			
	2.3.7	Ability to comply with radiation work permit requirements during normal or abnormal conditions	3.5			
	2.3.4	Knowledge of radiation exposure limits under normal and emergency conditions			3.7	
	2.3.6	Ability to approve release permits			3.8	
	Subtotal			2		2
4. Emergency Procedures/Plan	2.4.11	Knowledge of abnormal condition procedures	4.0			
	2.4.26	Knowledge of facility protection requirements including fire brigade and portable firefighting equipment usage	3.1			
	2.4.13	Knowledge of crew roles and responsibilities during EOP usage.			4.6	
	2.4.					
	2.4.					
	2.4.					
	Subtotal			2		1
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection