Facility: IPEC Printed: 07/25/2014

Date Of Exam: 05/11/2015

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 & Abnormal Plant Evolutions	1	3	3	3	N/A			3	3	N/A		3	18	0		0	0
	2	1	2	2				1	2			1	9	0		0	0
	Tier Totals	4	5	5				4	5			4	27	0		0	0
2. Plant Systems	1	2	2	3	3	3	2	2	3	2	3	3	28	0		0	0
	2	1	1	1	1	0	1	1	1	1	1	1	10	0	0	0	0
	Tier Totals	3	3	4	4	3	3	3	4	3	4	4	38	0		0	0

3. Generic Knowledge And Abilities Categories	RO 1	RO 2	RO 3	RO 4	RO Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	2	3	2	3	10	0	0	0	0	0

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).

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

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

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

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

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

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

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in 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.

PWR RO Examination Outline

Printed: 07/25/2014

Facility: IPEC

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000008 Pressurizer Vapor Space Accident / 3			X				AK3.05 - ECCS termination or throttling criteria	4.0	1
000009 Small Break LOCA / 3				X			EA1.09 - RCP	3.6	1
000011 Large Break LOCA / 3	X						EK1.01 - Natural circulation and cooling, including reflux boiling	4.1	1
000015/000017 RCP Malfunctions / 4		X					AK2.08 - CCWS	2.6	1
000022 Loss of Rx Coolant Makeup / 2	X						AK1.04 - Reason for changing from manual to automatic control of charging flow valve controller	2.9	1
000025 Loss of RHR System / 4						X	2.4.13 - Knowledge of crew roles and responsibilities during EOP usage.	4.0	1
000026 Loss of Component Cooling Water / 8					X		AA2.03 - The valve lineups necessary to restart the CCWS while bypassing the portion of the system causing the abnormal condition	2.6	1
000027 Pressurizer Pressure Control System Malfunction / 3			X				AK3.03 - Actions contained in EOP for PZR PCS malfunction	3.7	1
000029 ATWS / 1		X					EK2.06 - Breakers, relays, and disconnects	2.9*	1
000054 Loss of Main Feedwater / 4					X		AA2.06 - AFW adjustments needed to maintain proper T-ave. and S/G level	4.0	1
000055 Station Blackout / 6	X						EK1.02 - Natural circulation cooling	4.1	1
000057 Loss of Vital AC Inst. Bus / 6						X	2.4.11 - Knowledge of abnormal condition procedures.	4.0	1
000058 Loss of DC Power / 6			X				AK3.02 - Actions contained in EOP for loss of dc power	4.0	1
000062 Loss of Nuclear Svc Water / 4						X	2.1.39 - Knowledge of conservative decision making practices.	3.6	1
000065 Loss of Instrument Air / 8					X		AA2.05 - When to commence plant shutdown if instrument air pressure is decreasing	3.4*	1
W/E04 LOCA Outside Containment / 3				X			EA1.3 - Desired operating results during abnormal and emergency situations	3.8	1
W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4				X			EA1.2 - Operating behavior characteristics of the facility	3.7	1

PWR RO Examination Outline

Printed: 07/25/2014

Facility: IPEC

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
W/E12 - Steam Line Rupture - Excessive Heat Transfer / 4		X					EK2.2 - Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	3.6	1
K/A Category Totals:	**3**	**3**	**3**	**3**	**3**	**3**	**Group Point Total:**		**18**

PWR RO Examination Outline

Printed: 07/25/2014

Facility: IPEC

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000001 Continuous Rod Withdrawal / 1	X						AK1.19 - Voids coefficient	2.6	1
000003 Dropped Control Rod / 1					X		AA2.03 - Dropped rod, using in-core/ex-core instrumentation, in-core or loop temperature measurements	3.6	1
000005 Inoperable/Stuck Control Rod / 1		X					AK2.03 - Metroscope	3.1*	1
000024 Emergency Boration / 1			X				AK3.01 - When emergency boration is required	4.1	1
000059 Accidental Liquid RadWaste Rel. / 9					X		AA2.02 - The permit for liquid radioactive-waste release	2.9	1
000076 High Reactor Coolant Activity / 9						X	2.2.7 - Knowledge of the process for conducting special or infrequent tests.	2.9	1
W/E07 Inad. Core Cooling / 4				X			EA1.2 - Operating behavior characteristics of the facility	3.2	1
W/E10 Natural Circ. / 4			X				EK3.3 - Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations	3.4	1
W/E14 Loss of CTMT Integrity / 5		X					EK2.1 - Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	3.4	1
K/A Category Totals:	**1**	**2**	**2**	**1**	**2**	**1**	**Group Point Total:**		**9**

PWR RO Examination Outline

Printed: 07/25/2014

Facility: IPEC

ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
003 Reactor Coolant Pump							X					A1.03 - RCP motor stator winding temperatures	2.6	1
004 Chemical and Volume Control						X						K6.26 - Methods of pressure control of solid plant (PZR relief and water inventory)	3.8	1
005 Residual Heat Removal					X							K5.05 - Plant response during "solid plant": pressure change due to the relative incompressibility of water	2.7*	1
006 Emergency Core Cooling											X	2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.3	1
006 Emergency Core Cooling					X							K5.06 - Relationship between ECCS flow and RCS pressure	3.5	1
007 Pressurizer Relief/Quench Tank							X					A1.02 - Maintaining quench tank pressure	2.7	1
008 Component Cooling Water				X								K4.07 - Operation of the CCW swing-bus power supply and its associated breakers and controls	2.6*	1
010 Pressurizer Pressure Control						X						K6.02 - PZR	3.2	1
012 Reactor Protection					X							K5.02 - Power density	3.1*	1
012 Reactor Protection											X	2.3.11 - Ability to control radiation releases.	3.8	1
013 Engineered Safety Features Actuation			X									K3.01 - Fuel	4.4	1
022 Containment Cooling										X		A4.02 - CCS pumps	3.2*	1
022 Containment Cooling		X										K2.01 - Containment cooling fans	3.0*	1
026 Containment Spray		X										K2.02 - MOVs	2.7*	1
026 Containment Spray										X		A4.05 - Containment spray reset switches	3.5	1
039 Main and Reheat Steam				X								K4.06 - Prevent reverse steam flow on steam line break	3.3	1
059 Main Feedwater								X				A2.06 - Loss of steam flow to MFW system	2.7*	1
061 Auxiliary/Emergency Feedwater			X									K3.01 - RCS	4.4	1
062 AC Electrical Distribution									X			A3.04 - Operation of inverter (e.g., precharging synchronizing light, static transfer)	2.7	1

PWR RO Examination Outline

Printed: 07/25/2014

Facility: IPEC

ES - 401 | **Plant Systems - Tier 2 / Group 1** | **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
063 DC Electrical Distribution								X				A2.01 - Grounds	2.5	1
064 Emergency Diesel Generator				X								K4.05 - Incomplete-start relay	2.8	1
064 Emergency Diesel Generator											X	2.2.43 - Knowledge of the process used to track inoperable alarms.	3.0	1
073 Process Radiation Monitoring			X									K3.01 - Radioactive effluent releases	3.6	1
076 Service Water										X		A4.01 - SWS pumps	2.9	1
076 Service Water	X											K1.07 - Secondary closed cooling water	2.5*	1
078 Instrument Air	X											K1.03 - Containment air	3.3*	1
103 Containment								X				A2.05 - Emergency containment entry	2.9	1
103 Containment									X			A3.01 - Containment isolation	3.9	1
K/A Category Totals:	2	2	3	3	3	2	2	3	2	3	3	**Group Point Total:**		28

PWR RO Examination Outline

Printed: 07/25/2014

Facility: IPEC

ES - 401 | **Plant Systems - Tier 2 / Group 2** | **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
001 Control Rod Drive		X										K2.05 - M/G sets	3.1*	1
011 Pressurizer Level Control				X								K4.01 - Operation of PZR heater cutout at low PZR level	3.3	1
014 Rod Position Indication								X				A2.02 - Loss of power to the RPIS	3.1	1
033 Spent Fuel Pool Cooling									X			A3.02 - Spent fuel leak or rupture	2.9	1
034 Fuel Handling Equipment											X	2.4.42 - Knowledge of emergency response facilities.	2.6	1
035 Steam Generator	X											K1.13 - Condensate system	2.7	1
045 Main Turbine Generator							X					A1.06 - Expected response of secondary plant parameters following T/G trip	3.3	1
071 Waste Gas Disposal			X									K3.05 - ARM and PRM systems	3.2	1
079 Station Air										X		A4.01 - Cross-tie valves with IAS	2.7	1
086 Fire Protection						X						K6.04 - Fire, smoke, and heat detectors	2.6	1
K/A Category Totals:	1	1	1	1	0	1	1	1	1	1	1	**Group Point Total:**		10

Generic Knowledge and Abilities Outline (Tier 3)

PWR RO Examination Outline

Printed: 07/25/2014

Facility: IPEC

Form ES-401-3

Generic Category	KA	KA Topic	Imp.	Points
Conduct of Operations	2.1.29	Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	4.1	1
	2.1.44	Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	3.9	1
		Category Total:		**2**
Equipment Control	2.2.23	Ability to track Technical Specification limiting conditions for operations.	3.1	1
	2.2.38	Knowledge of conditions and limitations in the facility license.	3.6	1
	2.2.40	Ability to apply Technical Specifications for a system.	3.4	1
		Category Total:		**3**
Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	1
	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	1
		Category Total:		**2**
Emergency Procedures/Plan	2.4.12	Knowledge of general operating crew responsibilities during emergency operations.	4.0	1
	2.4.23	Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.	3.4	1
	2.4.30	Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.	2.7	1
		Category Total:		**3**

Generic Total: **10**

Facility: IPEC Printed: 07/25/2014

Date Of Exam: 05/11/2015

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

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals" in each K/A category shall not be less than two).

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

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

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

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

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

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

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above; if fuel handling equipment is sampled in 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.

Printed: 07/25/2014

PWR SRO Examination Outline

Facility: IPEC

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 1** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000007 Reactor Trip - Stabilization - Recovery / 1					X		EA2.04 - If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP	4.4	1
000038 Steam Gen. Tube Rupture / 3					X		EA2.10 - Flowpath for charging and letdown flows	3.3	1
000040 Steam Line Rupture - Excessive Heat Transfer / 4						X	2.4.21 - Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.6	1
000056 Loss of Off-site Power / 6						X	2.4.18 - Knowledge of the specific bases for EOPs.	4.0	1
000077 Generator Voltage and Electric Grid Disturbances / 6						X	2.1.39 - Knowledge of conservative decision making practices.	4.3	1
W/E11 Loss of Emergency Coolant Recirc. / 4					X		EA2.1 - Facility conditions and selection of appropriate procedures during abnormal and emergency operations	4.2	1
K/A Category Totals:	0	0	0	0	3	3	**Group Point Total:**		6

PWR SRO Examination Outline

Printed: 07/25/2014

Facility: IPEC

ES - 401 **Emergency and Abnormal Plant Evolutions - Tier 1 / Group 2** **Form ES-401-2**

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	KA Topic	Imp.	Points
000033 Loss of Intermediate Range NI / 7						X	2.1.6 – Ability to manage the control room crew during plant transients.	4.8	1
000036 Fuel Handling Accident / 8						X	2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.4	1
000060 Accidental Gaseous Radwaste Rel. / 9					X		AA2.02 - The possible location of a radioactive-gas leak, with the assistance of PEO, health physics and chemistry personnel	4.0	1
000074 Inad. Core Cooling / 4					X		EA2.02 - Availability of main or auxiliary feedwater	4.6	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**2**	**2**	**Group Point Total:**		**4**

PWR SRO Examination Outline

Facility: IPEC

ES - 401 **Plant Systems - Tier 2 / Group 1** **Form ES-401-2**

Sys/Evol # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	KA Topic	Imp.	Points
003 Reactor Coolant Pump								X				A2.01 - Problems with RCP seals, especially rates of seal leak-off	3.9	1
008 Component Cooling Water											X	2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.7	1
039 Main and Reheat Steam								X				A2.04 - Malfunctioning steam dump	3.7	1
061 Auxiliary/Emergency Feedwater								X				A2.06 - Back leakage of MFW	3.0	1
062 AC Electrical Distribution											X	2.4.40 - Knowledge of SRO responsibilities in emergency plan implementation.	4.5	1
K/A Category Totals:	0	0	0	0	0	0	0	3	0	0	2	**Group Point Total:**		5

PWR SRO Examination Outline

Printed: 07/25/2014

Facility: IPEC

ES - 401 | **Plant Systems - Tier 2 / Group 2** | **Form ES-401-2**

Sys/Evol # / Name	**K1**	**K2**	**K3**	**K4**	**K5**	**K6**	**A1**	**A2**	**A3**	**A4**	**G**	**KA Topic**	**Imp.**	**Points**
002 Reactor Coolant								X				A2.04 - Loss of heat sinks	4.6	1
028 Hydrogen Recombiner and Purge Control								X				A2.02 - LOCA condition and related concern over hydrogen	3.9	1
072 Area Radiation Monitoring											X	2.4.1 - Knowledge of EOP entry conditions and immediate action steps.	4.8	1
K/A Category Totals:	**0**	**0**	**0**	**0**	**0**	**0**	**0**	**2**	**0**	**0**	**1**	**Group Point Total:**		**3**

Generic Knowledge and Abilities Outline (Tier 3)

PWR SRO Examination Outline

Printed: 07/25/2014

Facility: IPEC

Form ES-401-3

Generic Category	KA	KA Topic	Imp.	Points
Conduct of Operations	2.1.35	Knowledge of the fuel-handling responsibilities of SROs.	3.9	1
	2.1.39	Knowledge of conservative decision making practices.	4.3	1
		Category Total:		**2**
Equipment Control	2.2.35	Ability to determine Technical Specification Mode of Operation.	4.5	1
		Category Total:		**1**
Radiation Control	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personal monitoring equipment, etc.	2.9	1
	2.3.11	Ability to control radiation releases.	4.3	1
		Category Total:		**2**
Emergency Procedures/Plan	2.4.38	Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.	4.4	1
	2.4.45	Ability to prioritize and interpret the significance of each annunciator or alarm.	4.3	1
		Category Total:		**2**

Generic Total: **7**

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information	
Name:	
Date: 5/18/15	Facility/Unit: Indian Point Unit 3
Region: I X II ☐ III ☐ IV ☐	Reactor Type: W X CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results		
RO/SRO-Only/Total Examination Values	____ / ____ / ____	Points
Applicant's Scores	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000007EA204 Ability to determine or interpret the following as they apply to a reactor trip: If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP	
Importance:		4.4	4.6

Question: #1

The reactor was initially at 100% power. A transient has occurred which caused reactor protection setpoints to be reached; however, the reactor does NOT trip. All attempts to trip the reactor from the control room have failed. The crew has entered 3-FR-S.1; Response to Nuclear Power Generation / ATWS. The Reactor Operator is manually inserting control rods and an NPO has been dispatched to locally trip the reactor. Westinghouse ATWS Analysis for various Condition II transients provides the bases for the mitigation strategy implemented in 3-FR-S.1. What are the next two actions required by 3-FR-S.1 and what is the limiting Condition II transient requiring those actions?

a. Trip the Turbine within 60 seconds and Initiate Emergency Boration. Uncontrolled RCCA Bank Withdrawal.
b. Trip the Turbine within 60 seconds and Verify total AFW flow greater than 686 gpm. Loss of Normal Feedwater.
c. Trip the Turbine within 30 seconds and Initiate Emergency Boration. Uncontrolled RCCA Bank Withdrawal.
d. Trip the Turbine within 30 seconds and Verify total AFW flow greater than 686 gpm. Loss of Normal Feedwater.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the requirement for a turbine trip is 60 seconds. The 60 second time criterion is actually for initiation of auxiliary feed. Also plausible for the student to believe that since we are concerned with a loss of subcriticality, that emergency boration is started earlier than verifying aux feed flows. Finally, an Uncontrolled RCCA Bank Withdrawal is a Condition II transient that is analyzed for an ATWS event.
b. Incorrect. Plausible because Plausible because the student may believe that the requirement for a turbine trip is 60 seconds. The 60 second time criterion is actually for initiation of auxiliary feed. Also plausible because the two steps of verifying / tripping the

turbine and verifying aux feed flow > 686 gpm are the next two steps. Finally the Loss of Normal Feedwater is the ATWS event requiring Turbine Trip to maintain SG inventory.

c. Incorrect. Plausible because the first part of the distractor is correct regarding the turbine trip. Also plausible for the student to believe that since we are concerned with a loss of subcriticality, that emergency boration is started earlier than verifying aux feed flows. Finally, an Uncontrolled RCCA Bank Withdrawal is a Condition II transient that is analyzed for an ATWS event.
d. Correct. Based on a loss of normal feedwater ATWS event, it is important to trip the turbine within 30 seconds to maintain SG inventory and verify aux feed flow within 60 seconds. See Westinghouse Owners Group Emergency Response Guideline for FR-S.1. (Steps 2 and 3 of 3-FR-S.1) (Note that Emergency Boration is step 4)

Technical References:	I3LP-ILO-EOPFRS
Proposed References to be provided:	None
Learning Objective:	Objective 11
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	000038EA210 Ability to determine or interpret the following as they apply to a SGTR: Flowpath for charging and letdown flows	
Importance:		3.1	3.3

Question: #2

Given the following conditions:

- The plant has experienced a SGTR coincident with a loss of off-site power.
- All three Diesels started and loaded in an SI plus Blackout Mode.
- The Operators are currently implementing 3-E-3.
- The ruptured SG is isolated.
- The RCS has been cooled down and depressurized to less than the ruptured SG pressure.
- The High Head Safety Injection Pumps have been stopped and placed in AUTO.
- Pressurizer Level is 36%.

What is the current status of Charging and Letdown Flow per the implementation of 3-E-0; Steam Generator Tube Rupture?

a. A Charging Pump is running with the seal injection flow path isolated and letdown is isolated.
b. A Charging Pump is running with the seal injection flow path isolated and letdown in service.
c. A Charging Pump is running with the regenerative heat exchanger flow path isolated and letdown is isolated.
d. A Charging Pump is running with the regenerative heat exchanger flow path isolated and letdown is in service.

Answer: a

Explanation / Justification

a. Correct. A Charging Pump is running per steps 13 and 14 of 3-E-3. Because the Diesels Loaded in SI plus Blackout Mode, the Component Cooling (CCW) pumps are not running. With no component cooling pumps, the RCPs have been tripped and seal injection has been isolated. A Charging Pump can still be run however, with City Water providing pump cooling. However, Letdown will remain isolated because CCW has not been established (See Note at the beginning of Attachment 2; Establishing Letdown – "Letdown should not be placed in service unless charging and CCW have been established."

b. Incorrect. Plausible because the first part of the distractor is correct and the student may believe because PZR Level is 36% (>29%) that letdown flow can be established per step 28 of 3-E-3. However, because component cooling has not been established, letdown will not be placed in service.
c. Incorrect. Plausible because the student may believe that the regenerative heat exchanger flow path is isolated on an SI signal. Also plausible because letdown flow would have been isolated by the Phase A signal. Finally, it is also plausible because the second part of the distractor is actually true, because component cooling has not been established, letdown will not be placed in service.
d. Incorrect. Plausible because the student may believe that the regenerative heat exchanger flow path is isolated on an SI signal. Also plausible because the student may believe because PZR Level is 36% (>29%) that letdown flow can be established per step 28 of 3-E-3. However, because component cooling has not been established, letdown will not be placed in service.

Technical References:	I3LP-ILO-EOPE30
	System Description 10.0; Engineered Safeguards
Proposed References to be provided:	None
Learning Objective:	Objectives 7, 10, & 16
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000402421	

Steam Line Rupture – Excessive Heat Transfer: 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.

Importance:	4.0	4.6

Question: #3

Given the following conditions:

- The Unit was operating at 100% Power when a Steam Line Break occurred downstream of the MSIVs.
- A Reactor Trip and Safety Injection were Automatically initiated,
- The MSIVs failed to close.
- The Operators are presently performing step 5 of 3-ECA-2.1; Uncontrolled Depressurization of All Steam Generators.
- An NPO reports that the 33 SG MSIV has just been closed locally.

The STA is monitoring Critical Safety Function Status Trees and observes the following indications:

- Feedwater flow to each Steam Generator is 100 gpm
- All Steam Generator WR Levels are < 9%
- All Steam Generator Pressures are < 100 psig.
- RCS Pressure is 1150 psig.
- All RCS Cold Leg Temperatures are < 240°F
- Intermediate Range SUR = +.1 DPM

Based on the observations made by the STA, the CRS should direct the crew to:

a. Remain in ECA-2.1 until SI is terminated.
b. Transition to E-2; Faulted Steam Generator Isolation.
c. Transition to FR-P.1; Response to Imminent Pressurized Thermal Shock.
d. Transition to FR-S.1; Response to Nuclear Power Generation.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may recognize that 33 SG MSIV has been closed, but also remember that foldout page for ECA-2.1 states that; E-2 transition criteria does not apply while performing SI termination in ECA-2.1 (steps 5 through 16). However, this large steam line break has challenged RCS Integrity and a transition to FR-P.1 is required.
b. Incorrect. Plausible because the student may recognize that 33 SG MSIV has been closed and assume that 33 SG Pressure is now rising, meeting the E-2 transition criteria.
c. Correct. This large steam line break with failure of the MSIVs to close has challenged RCS Integrity and a transition to FR-P.1 is required because of meeting RED Path criteria per Figure F04-1. SRO candidate should be knowledgeable of the main temperature limits on the curve.
d. Incorrect. Plausible because the student may recognize that Intermediate Range SUR > zero in a transition to FR-S.1. However, although Subcriticality is higher in priority than Thermal Shock, the Subcriticality is only an Orange Path. The Red Path Thermal Shock Path will take priority.

Technical References:	I3LP-ILO-EOPE20
	Critical Safety Function Status Trees
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	Modified IP3 Questions # 2907 & # 24135
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000562418 Loss of Off-Site Power: Knowledge of the specific bases for EOPs	
Importance:		3.3	4.0

Question: #4

A major storm has caused a complete loss of off-site power. Con Ed reported that a widespread system blackout has occurred across Westchester County, restoration will take several days. The Reactor has tripped and all three EDGs have started. However, only 31 and 32 EDGs have loaded in a Blackout Mode. A fault exists on the 5A Bus. The plant is now stable with Natural Circulation Cooling established and the crew has transitioned to 3-ES-0.2; Natural Circulation Cooldown.

What RCS Cooldown restrictions exist and what are the bases for those restrictions?

a. Maintain RCS Cooldown rate < 60°F / HR due to having all CRDM Fans running. Before final RCS Depressurization, a soak time of 27 hours is required due to Indian Point being a top hat upper support plate plant.
b. Maintain RCS Cooldown rate < 25°F / HR due to **NOT** having all CRDM Fans running. Before final RCS Depressurization, a soak time of 27 hours is required due to Indian Point being a top hat upper support plate plant.
c. Maintain RCS Cooldown rate < 60°F / HR due to having all CRDM Fans running. Before final RCS Depressurization, a soak time of 29 hours is required due to Indian Point being a flat upper support plate plant.
d. Maintain RCS Cooldown rate < 25°F / HR due to **NOT** having all CRDM Fans running. Before final RCS Depressurization, a soak time of 29 hours is required due to Indian Point being a flat upper support plate plant.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may not remember that all four CRDM Fans are powered from MCC 38 which is associated with the faulted 5A Bus. 60°F / HR is the rate if all CRDM Fans are operating. Second part of distractor is correct, but for less than all CRDM Fans operating.
b. Correct. Bus 5A which feeds MCC 38 (CRDM Fan Power Supply) is faulted and de-energized, therefore no CRDM Fans are available and a cooldown restriction of 25°F / HR is required. Additionally, Indian Point is a top hat upper support plate plant which requires a 27 hour soak prior to the final RCS depressurization (see ES-0.1, attachment 2, step 20).

c. Incorrect. Plausible because Plausible because the student may not remember that all four CRDM Fans are powered from MCC 38 which is associated with the faulted 5A Bus. 60°F / HR is the rate if all CRDM Fans are operating. The second part of the distractor describes the soak time for a flat upper support plate plant, however, Indian Point is a top hat upper support plate design.
d. Incorrect. Plausible because Bus 5A which feeds MCC 38 (CRDM Fan Power Supply) is faulted and de-energized, therefore no CRDM Fans are available and a cooldown restriction of 25°F / HR is required. The second part of the distractor describes the soak time for a flat upper support plate plant, however, Indian Point is a top hat upper support plate design.

Technical References:	I3LP-ILO-EOPE00, 3-ES-0.2 WOG ERG for ES-0.2
Proposed References to be provided:	None
Learning Objective:	Objectives 7 & 12
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 1 & 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	0000772225 Generator Voltage and Electric Grid Disturbances; Knowledge of the bases in Technical Specifications or limiting conditions for operations and safety limits	
Importance:		3.2	4.2

Question: #5

Given the following:

- The Unit is at 100% Power
- The System Operator has notified the control room that grid frequency is unstable

What plant protection is designed for grid instabilities resulting in lowering frequency, how is the protection achieved, and what is the bases for needing this protection?

a. A single Reactor Coolant Pump Bus Underfrequency condition > P-8 will directly trip the reactor at < 57.5 HZ. This ensures that protection is provided against violating the DNBR limit due to loss of flow in one or more RCS loops from a major network frequency disturbance.
b. A two out of four Reactor Coolant Bus Underfrequency condition will directly trip the reactor at < 57.5 Hz. This ensures that protection is provided anticipating an actual loss of RCS Flow Condition. This is required because an underfrequency condition will slow down the pumps and reduce the pump coastdown time and therefore reduce reactor heat removal capability.
c. A single Reactor Coolant Pump Bus Underfrequency condition > P-8 will cause the tripping open of the associated Reactor Coolant Pump Breaker at < 57.5 Hz. The Reactor will subsequently trip on a Loss of Reactor Coolant Flow condition. Above P-8, loss of flow in any RCS loop will actuate a reactor trip. This ensures that protection is provided against violating the DNBR limit due to loss of flow in one or more RCS loops.
d. A two out of four Reactor Coolant Bus Underfrequency condition will trip all four Reactor Coolant Pump Breakers at < 57.5 HZ. The Reactor will subsequently trip on the Reactor Coolant Pump Breaker Open Position Protection Trip. This ensures that protection is provided against violating the DNBR limit due to loss of flow in two or more RCS loops from a major network disturbance.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the coincidence for underfrequency is a single loop above P-8, similar to the low RCS loop flow trip. The student may also believe that underfrequency conditions are a direct reactor trip. The bases statement is actually correct for a single loop loss of flow condition.
b. Incorrect. Plausible because the underfrequency protection is a two out of four bus logic. The student may also believe that underfrequency conditions are a direct reactor trip. Finally, the bases statement is correct for an underfrequency condition.
c. Incorrect. Plausible because the student may believe that the coincidence for underfrequency is a single loop above P-8, similar to the low RCS loop flow trip. Also plausible because the underfrequency condition is not a direct trip and does cause the opening of the reactor coolant breakers, however underfrequency is a two out of four logic and the reactor coolant pump breaker open signal will trip the reactor, not loss of RCS loop flow as stated. The bases statement is actually correct for a single loop loss of flow condition.
d. Correct. A two out of four Reactor Coolant Bus Underfrequency condition will trip all four Reactor Coolant Pump Breakers at < 57.5 HZ. The Reactor will subsequently trip on the Reactor Coolant Pump Breaker Open Position Protection Trip. This ensures that protection is provided against violating the DNBR limit due to loss of flow in two or more RCS loops from a major network disturbance.

Technical References:	I3LP-ILO-ICRXP
	RPS Instrumentation TS Bases – B 3.3.1 #12
	System Description 28.0; Overall Unit Protection
Proposed References to be provided:	None
Learning Objective:	Objective E-7
Question Source:	Modified IP3 Question # 25121
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 2

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A #	00WE11A201 Ability to determine or interpret the following as they apply to the (Loss of Emergency Coolant Recirculation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations	
Importance:		3.4	4.2

Question: #6

Given the following plant conditions:

- 31 RHR Pump is cleared and tagged for motor replacement
- A LOCA coincident with a complete loss of Off-Site Power has occurred
- Emergency Diesel Generators 31 and 32 have started and loaded in SI plus Blackout Mode
- Emergency Diesel Generator 32 has tripped and operators have been unable to restart.
- When attempting to establish Cold Leg Recirculation per 3-ES-1.3; Transfer to Cold Leg Recirculation, the following alarm was received:
    - Low Head Injection Line Low Flow
- RWST Level is presently 10 feet and lowering
- Containment Pressure is 25 psig and slowly lowering

Which of the following describes the required procedure transition and required actions based on present plant conditions?

a. Transition to 3-FR-Z-1; Response to High Containment Pressure. An Orange Path termination requires suspension of any E-set procedure in progress and transition to the required FRP. Operators will start available Containment Spray Pumps and Containment Fan Cooler Units as necessary.
b. Transition to 3-ECA-1.1; Loss of Emergency Coolant Recirculation. FRPs are NOT implemented during the performance of ECA-1.1 because the strategy for Containment Spray Pump Operation in the FRP does not take into account conserving RWST inventory like ECA-1.1 does. Operators will stop 32 Containment Spray Pump and operate three Containment Fan Cooler Units.
c. Remain in 3-ES-1.3; Transfer to Cold Leg Recirculation because FRPs are NOT implemented during the performance of ES-1.3 as per the NOTE that states; “FRPs should not be implemented until the transfer to cold leg recirculation has been completed.” 3-ES-1.3 will recognize the loss of recirculation pumps and align 32 RHR Pump for recirculation.

d. Transition to 3-ECA-1.3; Loss of Emergency Coolant Recirculation Caused By Sump Blockage. FRPs are NOT implemented during the performance of ECA-1.3 per guidance in OAP-012; EOP Users Guide. Operators will run one Containment Spray Pump and four Containment Fan Cooler Units.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because the student may believe that transition to the Orange Path FRP is appropriate as containment pressure is > 22 psig. Actual transition will depend on whether a containment spray pump is still running and which procedure transition is appropriate. In this case, ECA-1.3; loss of recirculation caused by sump blockage is appropriate and FRPs are not entered from ECA-1.3.
b. Incorrect. Plausible because the student may believe that because the present plant configuration leaves them with no recirculation capability, then transition to ECA-1.1 is appropriate. Also plausible because the second statement is partially true with respect to the strategy of RWST conservation, however, FRPs are entered from ECA-1.1. Finally, the operators will not stop 32 Containment Spray Pump because it is not running due to the loss of 32 EDG.
c. Incorrect. Plausible because FRPs are not implemented during the performance of ES-1.3; Transfer to Cold Leg Recirculation. Also plausible because if the recirculation pumps were not available, ES-1.3 would use Attachment 3 to align RHR for recirculation. However, indication of the "Low Head Injection Line Low Flow" alarm is a foldout page transition to ECA-1.3 as an indication of sump blockage. Additionally due to the loss of 6A bus (32 EDG trip), 32 RHR Pump is also unavailable.
d. Correct. The indication of the "Low Head Injection Line Low Flow" alarm is a foldout page transition to ECA-1.3 as an indication of sump blockage. FRPs are NOT implemented during the performance of ECA-1.3 per guidance in OAP-012; EOP Users Guide. Operators will run one Containment Spray Pump and four Containment Fan Cooler Units. (See ECA-1.3, steps 2 & 3). Note that step 2.a. will attempt to start all available FCUs (only Fan Cooler 35 is unavailable), however per table (step 3.C.) "all" FCUs are not available so one Containment Spray pump is still required whether three or four FCUs are running.

Technical References:	I3LP-ILO-EOPC10 480 Volt AC Power Distribution Simplified Drawing
Proposed References to be provided:	None
Learning Objective:	Objective 5
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A #	0000332225 Loss of Intermediate Range Nuclear Instrumentation: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	
Importance:		3.2	4.2

Question: #7

Given the following plant conditions:

- Plant Startup is in progress IAW 3-POP-1.3; Plant Startup from Zero to 45% Power.
- At 4% Power, N35 Intermediate Range Channel was reading significantly higher than its expected current limit, was declared inoperable and the channel was removed from service IAW 3-SOP-NI-001.
- At 8% Power, N36 Intermediate Range Channel has just failed low.

What actions are required per Technical Specifications and what are the bases for those actions?

a. Power operations may continue based on power level being greater than 5% (MODE 1). In MODE 1, Technical Specification 3.0.3 in not applicable for the Intermediate Range Neutron Flux High Reactor Trip. Greater than 5% power the P-6 interlock has already been met and the Power Range Neutron Flux - Low Setpoint Reactor Trip provides protection from a reactivity addition accident.

b. Immediately suspend operations involving positive reactivity additions and reduce THERMAL POWER to < P-6 within 2 hours. At least one Intermediate Range Channel is required to be operable > P-6 and less than P-10 when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup. Below P-6, the Source Range Neutron Flux Trip provides backup core protection for reactivity accidents.

c. Enter Technical Specification 3.0.3 due to both Intermediate Range Channels being inoperable. Action shall be initiated within 1 hour to place the unit in Mode 3 within 7 hours. In MODE 3, the Intermediate Range Neutron Flux Trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition.

d. Within 1 hour, verify interlock (P-6) is in the required state for the existing unit conditions and because the P-6 interlock has already been met, power shall be raised to greater than P-10 within the next 2 hours. Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux Reactor Trip will be blocked making this function no longer

necessary during a power accession. Above the P-10 setpoint, the Power Range Neutron Flux – High Setpoint Trip provides core protection for a rod withdrawal accident.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because student may believe that TS 3.0.3 would apply for both channels being inoperable, but also believe that applicability is only during startup < 5% power. The P-6 interlock has already been met and the Power Range Neutron Flux - Low Setpoint Reactor Trip does provide primary protection from a reactivity addition accident.
b. Correct. Technical Specification 3.3.1, Action F is applicable if less than one channel is operable. The bases are as stated for TS 3.3.1, Action F.
c. Incorrect. Plausible because student may believe that TS 3.0.3 would apply for both channels being inoperable. Mode 3 bases statement is correct for Mode 3.
d. Incorrect. Plausible because this is the correct stated bases for Technical Specification 3.3.1, Action M if one of more channels of (P-6) interlock are inoperable. The fact that the interlock is in the correct state greater than P-6 setpoint and not required is also correct.

Technical References:	I3LP-ILO-ICEXC Technical Specification 3.3.1, Actions F & M TS 3.3.1 Bases Document
Proposed References to be provided:	None
Learning Objective:	Objectives E-7, 8, & 9
Question Source:	Modified IP3 Question # 8739 (added bases info)
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 2

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A #	0000362123	
		Fuel Handling Incidents: Ability to perform specific system and integrated plant procedures during all modes of plant operation	
Importance:		4.3	4.4

Question: #8

Given the following conditions:

- Refueling Operations are in progress, Full Core Off-Load is being performed.
- A spent fuel assembly has been latched in the manipulator mast and the assembly has just been lifted clear of the reactor vessel flange.
- Visual Observations indicate Refueling Cavity and Spent Fuel Pool Level are both lowering.
- These local visual indications also indicate Refueling Cavity and Spent Fuel Pool are both below Technical Specification Requirements.

The refueling SRO has entered 3-AOP-FH-1; Fuel Damage OR Loss of SFP / Refueling Cavity Level. Which of the following describe the correct initial procedural actions based on the above information?

a. Immediately suspend movement of all irradiated fuel assemblies in accordance with Technical Specifications 3.7.14 and 3.9.6, evacuate all personnel from Containment, initiate Containment Ventilation Isolation (Purge & Pressure Relief Valves Closed), close the fuel transfer canal gate valve and initiate level makeup.
b. Place the suspended fuel assembly in the containment upender, lower and send to the Fuel Storage Building, then close and latch the fuel transfer canal gate valve, inflate the fuel transfer canal gate valve, and evacuate all personnel from the Fuel Storage Building (FSB) and Containment VC).
c. Close and latch the fuel transfer canal gate valve, inflate fuel transfer canal gate seal, place the suspended fuel assembly back in the reactor vessel and evacuate non-essential personnel form the Fuel Storage Building (FSB) and Containment (VC).
d. Place the suspended fuel assembly in the containment upender in the vertical position, evacuate non-essential personnel from the Fuel Storage Building (FSB) and Containment (VC), close the fuel transfer canal gate valve and initiate level makeup.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the student may be focused on the need to immediately suspend movement of irradiated fuel assemblies per the tech spec actions, however, as stated in the tech spec bases; "this does not preclude movement of a fuel assembly to a safe position." Also plausible because if the assembly is left in the manipulator crane mast with level continuing to lower, actions from the "damaged" fuel assembly section requiring Containment Ventilation Isolation and evacuation are very plausible. The need to makeup is also a subsequent step and therefore additionally plausible.
b. Incorrect. Plausible because depending on how fast level is lowering, the upender would be a second choice per the procedure (See Attachment 2, step 2.5), however, the upender would never be sent back to the FSB. Transfer cart needs to be on containment side to facilitate gate valve closure. The remaining steps are correct, except that all personnel are not evacuated, only "non-essential". Plausible because, all personnel are evacuated for a "damaged" assembly.
c. Correct. See steps 4.27 – 4.30 in body of procedure, Attachments 1 & 2.
d. Incorrect. Plausible because depending on how fast level is lowering, the upender would be a second choice per the procedure (See Attachment 2, step 2.5), however, the upender would be then lowered to the "horizontal" position. Closing the gate valve is correct and the need to makeup is also a subsequent step and therefore additionally plausible.

Technical References:	I3LP-ILO-AOPFH1 3-AOP-FH-1
Proposed References to be provided:	None
Learning Objective:	Objectives B & E
Question Source:	Modified – IP3 Question # 18590
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5 & 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A #	000060AA202 Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: The possible location of a radioactive-gas leak, with the assistance of PEO, health physics and chemistry personnel	
Importance:		3.1	4.0

Question: #9

Chapter 14 of the FSAR discusses the Accidental Release of Waste Gases as a potential event. The accidental release of waste gases is analyzed assuming a rupture of tanks that accumulate significant quantities of radioactive gases during operation. Waste Gas Operation Procedural precautions and limitations, along with ODCM Limiting Conditions for Operations have been established to ensure analysis assumptions are met.

To ensure a rupture of a Unit 3 Waste Gas Decay Tank remains within analyzed limits, operations personnel will limit the pressure of the gas decay tank to _____ psig and the total curie activity of the gas decay tank to ≤ _______ curies. Chemistry will ensure the curie content restriction is met by a grab sample and analysis prior to a controlled release, but Radiation Monitor _______ is used to alert operators of the potential for reaching the tank curie limit before sampling has occurred. The bases for the curie radioactive content ensures that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed _____ Rem.

a. 110 psig, 50,000 curies, R-20, 0.5
b. 95 psig, 29,761 curies, R-14, 0.5
c. 95 psig, 50,000 curies, R-14, 1.5
d. 110 psig, 29,761 curies, R-20, 1.5

Answer: a

Explanation / Justification

a. Correct. See System Description for Gaseous Waste Disposal System, section 3.1. Also see ODCM Spec D 3.2.6 and associated bases statement. Also see 3-SOP-WDS-002; Gaseous Waste Disposal System Operation Precautions and Limitations.
b. Incorrect. Plausible because the total curie content number is Unit 2's limit for the same ODCM spec. Student may believe that gas decay tanks swap at 95 psig. 0.5 Rem is

correct total dose number and the student could believe that R-14; Plant Vent Radiogas Monitor would be the appropriate.

c. Incorrect. Plausible because the 50,000 curie content number is correct. Student may believe that gas decay tanks swap at 95 psig. 1.5 Rem is the thyroid dose limit and the student could believe that R-14; Plant Vent Radiogas Monitor would be the appropriate.
d. Incorrect. Plausible because the total curie content number is Unit 2's limit for the same ODCM spec. Also 110 psig and R-20 are correct, while 1.5 Rem is the thyroid dose limit.

Technical References:	I3LP-ILO-GWR001 ODCM D3.2.6 & Bases 3-SOP-WDS-002 P&Ls System Description 5.2, section 3.1, page 23
Proposed References to be provided:	None
Learning Objective:	Objectives 11 & 12
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 2

NOTE:

Waste Gas Operation Procedural precautions and limitations, along with ODCM Limiting Conditions for Operations have been established to ensure analysis assumptions are met. These conditions are ensured by operations personnel verifying proper pressure limits are maintained, chemistry conducting periodic grab samples, and health physics monitoring local radiation and airborne contamination monitors.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A #	000074EA202 Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: Availability of main or auxiliary feedwater	
Importance:		4.3	4.6

Question: #10

Operators are presently implementing 3-FR-C.1; Response to Inadequate Core Cooling, Given the following conditions:

- Core Exit Thermocouples are 1250°F and slowly rising
- RVLIS Full Range = 30% and slowly lowering
- All Auxiliary Feedwater Flow has been lost
- It will take approximately one hour to restore a source of feedwater
- SG WR Levels are 24%, 23%, 22%, and 25% respectively.
- NO support systems are available for the RCPs
- Attempts to start High Head Safety Injection Pumps were unsuccessful

What recovery and mitigation strategy will be implemented **next** and why?

a. Start Reactor Coolant Pumps (RCPs) because in 3-FR-C.1; Response to Inadequate Core Cooling, RCPs are started even if support systems are unavailable. Starting the RCPs will provide forced two phase flow through the core and temporarily improve core cooling.
b. Immediately transition to 3-FR-H.1; Response to Loss of Secondary Heat Sink due to the Red Path on Heat Sink. This procedure will help expedite the restoration of a source of feedwater while monitoring for the need to initiate Bleed and Feed. The RCPs are to remain stopped in FR-H.1.
c. Open all PRZR PORVs, PORV Block Valves and Reactor Vessel Head Vents to depressurize the RCS in accordance with 3-FR-C.1; Response to Inadequate Core Cooling. The RCPs are not started due to insufficient water level in the steam generators to protect the steam generator tubes from creep rupture.
d. Depressurize ALL Intact SGs to 175 psig in accordance with 3-FR-C.1; Response to Inadequate Core Cooling. The rapid secondary depressurization has been shown to be the most effective way to reduce RCS pressure, allowing the SI accumulators to inject. The RCPs are not started because without proper support conditions, damage will occur to the RCPs, preventing any future use.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because if high head safety injection has not been established and a secondary heat sink (AFW) cannot be established, FR-C.1, step 9 RNO ends you to step 18 which determines whether RCPs should be started. The NOTE before step 18 even states; "Normal conditions are desired but NOT required for starting the RCPs. The remaining statement is correct and from the WOG ERG Bases for the NOTE. However, step 18 also requires SG NR Level > 9%, which we do not have.
b. Incorrect. Plausible because a RED Path for Heat Sink does exist. However, we are implementing a higher priority Functional Restoration Procedure (FRP) in 3-FR-C.1; Inadequate Core Cooling.
c. Correct. If high head safety injection has not been established and a secondary heat sink (AFW) cannot be established, FR-C.1, step 9 RNO ends you to step 18 which determines whether RCPs should be started. However, step 18 also requires SG NR Level > 9%, which we do not have. The WOG ERG Bases states that "RCPs are only started in this step if there is sufficient water level in their associated steam generator to protect the steam generator tubes from creep rupture." Step 18, RNO has you open the PORVs, PORV Block Valves, and the Reactor Vessel Head Vents to depressurize the RCS. See Step 18.b RNO.
d. Incorrect. Plausible because the next step in FR-C.1 (step 19) (after the RCP decision step 18) is to depressurize all intact SGs to atmospheric Pressure. However, the proper **next** mitigation strategy (in order) is to depressurize the RCS first using PORVs and Head Vents.

Technical References:	I3LP-ILO-EOPFRC 3-FR-C.1 and WOG ERG Bases
Proposed References to be provided:	None
Learning Objective:	Objectives 9 & 15
Question Source:	Modified (IP3 Questions # 24661 & # 12314)
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	003000A201 Ability to (a) predict the impacts of the following malfunctions or operations on the RCPs; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems with RCP seals, especially rates of seal leak-off	
Importance:		3.5	3.9

Question: #11

Given the following plant conditions:

- 100% Reactor Power
- All Control Systems are in AUTO
- RCP NO. 1 SEAL RETURN HIGH LOW FLOW (COMMON) is in alarm
- 33 RCP #1 Seal return flow has been trending upward for the last three days, presently stable at 5.1 gpm
- 33 RCP Seal Inlet Temperature has been stable at 150°F
- The Reactor Operators are recording RCP Seal Return Flow Data per 3-AOP-RCP-1; Reactor Coolant Pump Malfunction

The ATC Reactor Operator has just notified the CRS that #1 Seal Return low has risen to the maximum indicated flow on the digital recorder. 33 RCP Seal Inlet Temperature is also slowly rising. Which of the following describes how the CRS should direct the crew to respond?

a. Initiate a normal Reactor Shutdown IAW 3-POP-2.1; Operation at Greater Than 45% Power to ensure the 33 RCP can be shut down within 8 hours.
b. Trip the Reactor, Stop 33 RCP, Initiate E-0, Close Spray Valve PC-455H, and when 33 RCS Loop Flow stabilizes at 20-30%, then Close Seal Return Valve 261C.
c. Trip the Reactor, Stop 33 RCP, Close Spray Valve PC-455H, when the pump stops rotating close Seal Return Valve 261C and then, Initiate E-0.
d. Continue to monitor 33 RCP Seal Parameters and if Seal Inlet Temperature rises to > 185°F, then Initiate a normal Reactor Shutdown IAW 3-POP-2.1; Operation at Greater Than 45% Power to ensure the 33 RCP can be shut down within 8 hours.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the #1 Seal Leak-off Flow did not exceed 6 gpm. It is also plausible because various RCP abnormal conditions (vibrations, temperatures, etc.) require a shutdown within 8 hours based on vendor manual recommendations.
b. Correct. Step 4.1 IAAT conditions include the following; "#1 seal return flow > 6gpm (5.99 gpm digital). Student needs to recognize that the maximum indicated flow on the digital recorder is 5.99 gpm. See steps 4.2 through 4.9 and then step 4.12 to close affected RCP Seal Return Valve. Note that E-0 is "initiated" after stopping affected RCP.
c. Incorrect. Plausible because the majority of the distractor is correct. See above, but note that E-0 is "initiated" after stopping affected RCP.
d. Incorrect. Plausible because various RCP abnormal conditions (vibrations, temperatures, etc.) require a shutdown within 8 hours based on vendor manual recommendations.

Technical References:	I3LP-ILO-AOPRCP 3-ARP-009 (pages 38-44) 3-AOP-RCP-1 3-SOP-RCS-001 System Description 1.3
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	Modified IP3 Question # 23459
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	0080001217 Component Cooling Water: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	
Importance:		4.4	4.7

Question: #12

Given the following conditions:

- Plant is operating at 100% power
- Annunciator, PROCESS MONITOR HIGH RAD is in alarm
- Annunciator, THERMAL BARRIER COOLING WATER RETURN HIGH TEMP is in alarm
- Annunciator, THERMAL BARRIER CCW HEADER LOW FLOW is in alarm
- AC-FCV-625 indicates closed

Based on the above plant indications, what procedure does the CRS enter, what are the first mitigations steps, and why are those steps taken?

a. The CRS enters 3-AOP-LICCW-1; Leakage into CCW System and directs mitigation through Attachment 1. Attachment 1 directs the crew to momentarily select OPEN on AC-FCV-625 to determine the affected RCP by observing which RCP has the low thermal barrier delta p.
b. The CRS enters 3-AOP-RCP-1; Reactor Coolant Pump Malfunction. The procedure directs the crew to momentarily select OPEN on AC-FCV-625 to determine the affected RCP by observing which RCP has the low thermal barrier delta p.
c. The CRS enters 3-AOP-CCW-1; Loss of Component Cooling and directs mitigation through Attachment 9. Attachment 9 directs the crew to Split CCW Headers to prevent the spread of contamination and assist in exposing the affected RCP when the AC-FCV-625 is subsequently momentarily opened.
d. The CRS enters 3-AOP-LICCW-1; Leakage into CCW System and directs mitigation through Attachment 9. Attachment 9 directs the crew to Split CCW Headers to prevent the spread of contamination and assist in exposing the affected RCP when the AC-FCV-625 is subsequently momentarily opened.

Answer: a

Explanation / Justification

a. Correct. These alarms and the fact that the AC-FCV-625 is closed are all indications of a RCP Thermal Barrier leak. Therefore, the CRS would enter AOP-LICCW, Leakage into CCW System and utilize Attachment 1 to determine which RCP thermal barrier heat exchanger is leaking and isolate it. Attachment 1 directs the crew to momentarily select OPEN on AC-FCV-625 to determine the affected RCP by observing which RCP has the low thermal barrier delta p. See bases document for AOP-LICCW, step 1.2; "determines the affected RCP by momentarily reinitiating CCW flow and observing which RCP has the low thermal barrier delta p."
b. Incorrect. Plausible because the student may believe that RCP Thermal Barrier leaks are handled in AOP-RCP; RCP Malfunctions. Additionally plausible because the remainder of the mitigation strategy and bases are true.
c. Incorrect. Plausible because the student may believe that RCP Thermal Barrier loss is covered by AOP-CCW; Loss of CCW. Second part of distractor is an action from AOP-LICCW to split the headers if the PROCESS MONITOR HIGH RAD is in alarm and it is caused by R17 (CCW High Activity). The student can easily infer that the alarm was caused by R17, then AOP-LICCW actually has an attachment 9 that splits the headers and the bases document for step 4.5 of the procedure states; "Splitting the headers would help prevent the spread of contamination and assist in exposing the affected component."
d. Incorrect. Plausible because AOP-LICCW is the proper mitigating procedure and the second part of distractor is an action from AOP-LICCW to split the headers if the PROCESS MONITOR HIGH RAD is in alarm and it is caused by R17 (CCW High Activity). The student can easily infer that the alarm was caused by R17, then AOP-LICCW actually has an attachment 9 that splits the headers and the bases document for step 4.5 of the procedure states; "Splitting the headers would help prevent the spread of contamination and assist in exposing the affected component." However, the NOTE before step 4.5 states that "it is NOT necessary to split headers if leak has been identified and is isolable." Clearly based on the given information, a RCP Thermal Barrier Leak exists and splitting the headers does not help the mitigation strategy of Attachment 1 of the procedure. The CRS knowing this would not elect to use Attachment 9 and split the headers.

Technical References:	I3LP-ILO-AOPLIC
	3-ARP-010
	3-ARP-009
	3-AOP-LICCW-1 & Bases
Proposed References to be provided:	None
Learning Objective:	Objectives 1, 2, & 3
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	039000A204 Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioning steam dump	
Importance:		3.4	3.7

Question: #13

The plant has been operating at steady state 100% power for the last 90 days. The ATC Reactor Operator monitoring the control board notices the following indications:

- Reactor Power is 100.5% and slowly rising
- RCS T_{avg} is 569°F and slowly lowering
- Rods are at their fully withdrawn position of 230 steps
- A red light is lit for Group 1 Steam Dump Valves on the flight panel (FCF)

The CRS has entered 3-UC-1; Uncontrolled Cooldown. What is the appropriate CRS crew direction for these indications IAW the procedure?

a. Trip the Reactor, Close the MSIVs, and GO TO 3-E-0; Reactor Trip or Safety Injection.
b. Stop any inward rod motion, stop any turbine load increase in progress, initiate a dilution to restore T_{avg} to 1.5°F of program, and dispatch an operator to manually isolate all three steam dump valves in Group 1.
c. Place rods in Manual, stop any turbine load increase in progress, reduce turbine load to maintain reactor power ≤ 100%, stop any boration in progress, maintain turbine load as necessary to maintain T_{avg} within 4°F of T_{ref}, and dispatch an operator to manually isolate affected valves.
d. Trip the Reactor, Close the MSIVs, and INITIATE 3-E-0; Reactor Trip or Safety Injection, while continuing with the implementation of Attachment 1; Investigation Checklist of 3-UC-1.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because this would be the action for a locally un-isolable steam leak. This is actually steps 4.1 through 4.4 of the procedure. However, control board indications

show a malfunctioning / open high pressure steam dump valve, this could be isolated locally.

b. Incorrect. Plausible because the first two actions are correct per steps 4.6 & 4.8 of the procedure, however the procedure does not direct restoring T_{avg} via a dilution. Although this is still plausible as raising temperature to program will add negative reactivity. Student may also believe that isolation of all three group 1 valves is necessary based on the indication. However, the indicator for group 1 turning red may only be one valve.

c. Correct. Information in the stem indicates a malfunctioning / open high pressure steam dump valve and therefore this would be locally isolable. Rods are placed in Manual per step 4.7 due to rods being at 230 steps, then step 4.8 stops any turbine load increases in progress, step 4.9 reduces turbine load to maintain reactor power ≤ 100%, step 4.10 stops any boration in progress, and step 4.15 maintains turbine load as necessary to maintain T_{avg} within 4°F of T_{ref} (because rods are in manual). Finally, step 4.19 RNO closes any affected valves.

d. Incorrect. Plausible because this would be the action for a locally un-isolable steam leak, except that 3-UC-1 states GO TO 3-E-0, not INITIATE. But also plausible because a number of AOPs do use the term INITIATE so that supplemental actions can be completed. In this case the student would believe that the malfunctioning steam dump valve would still require isolation via 3-UC-1, Attachment 1.

Technical References:	I3LP-ILO-AOPUC1 3-AOP-UC-1 & Bases System Description 18.1
Proposed References to be provided:	None
Learning Objective:	Objective 3.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

NOTE: Need to get proper parameter values for question. Need to understand actual control board indication? Are there green closed lights as well? Original intent was for a partially open steam dump valve, but may have to use a fully open valve which could be equal to approx. 3% power. Should possibly check numbers on simulator? Power & $T_{avg.}$ Could change stem to no indication (between green & red), if it has green light. Otherwise, may need much greater power and lower temperature indications.

SRO: Requires knowledge of specific content, verses knowledge of the procedure's overall mitigative strategy or purpose.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	061000A206 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Back leakage of MFW	
Importance:		2.7	3.0

Question: #14

The Unit is at 100% power. An NPO in the field has reported that the 31 ABFP discharge piping to 32 SG is very hot and some steam is coming from the 31 ABFP glands. The field supervisor has concurred with the NPO's findings and recommends that actions for ABFP potential steam binding be taken in accordance the 3-SOP-AFW-002; Auxiliary Feedwater System Support Procedure.

Procedural actions have been taken to reseat the leaking check valve. Which of the following describes the monitoring actions taken following successful reseating of a leaking check valve IAW 3-SOP-AFW-002; Auxiliary Feedwater System Support Procedure?

a. MONITOR ABFP discharge pipe temperatures every hour until temperature is ambient for at least 4 hours.
b. MONITOR ABFP discharge pipe temperatures every 2 hours until temperature is ambient for at least 8 hours.
c. MONITOR ABFP discharge pipe temperatures every 4 hours until temperature is ambient for at least 12 hours.
d. MONITOR ABFP discharge pipe temperatures every 6 hours until temperature is ambient for at least 24 hours.

Answer: a

Explanation / Justification

a. Correct. See step 4.5.10 of 3-SOP-AFW-002. MONITOR ABFP discharge pipe temperatures **hourly** **<u>until</u>** temperature is ambient for at least 4 hours.
b. Incorrect. Plausible because the student may believe that two hours is acceptable from an operability standpoint and effectively utilizes available resources. Also plausible because the eight hours is within the normal once per shift checks that are done as part of the regular conventional plant operator rounds (0800 & 2000 readings).

c. Incorrect. Plausible because the student may believe that increasing normal surveillance of once per shift per the conventional operator logs to three times per shift is effective and prudent from an operability standpoint. Also the 12 hours would be exactly one shift.
d. Incorrect. Plausible because the student may believe that increasing normal surveillance of once per shift per the conventional operator logs to twice per shift is effective and prudent from an operability standpoint. Also plausible that these increased readings would be carried for two days to ensure continued operability prior to returning to the normal frequency.

Technical References:	I3LP-ILO-AFW001 3-SOP-AFW-002 System Description 21.2
Proposed References to be provided:	None
Learning Objective:	Objective 3e
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A #	0620002440 AC Electrical Distribution System: Knowledge of SRO responsibilities in emergency plan implementation	
Importance:		2.7	4.5

Question: #15

Given the following conditions:

- The Unit is at 100% power.
- The Shift Manager is incapacitated due to an illness.
- 32 EDG is cleared and tagged for cylinder liner replacement.

A major grid disturbance has resulted in a Reactor Trip and a Complete Loss of Offsite Power. All 138KV and 13.8KV feeds to Unit 3 are unavailable. Con Ed has stated that grid restoration times are presently unknown, but greater than 4 hours. All available equipment has functioned per design and Natural Circulation Cooling has been established.

As the Control Room Supervisor classify the event, and state what personnel can relieve you as the Emergency Director.

a. UNUSUAL EVENT; Loss of all offsite AC power to 480 V safeguards buses for ≥ 15 minutes. The On-Call Emergency Director or the Plant Operations Manager.
b. ALERT; AC power capability reduced to minimum design for ≥ 15 minutes, such that any additional single failure would result in less than minimum designed safeguards equipment. The On-Call Emergency Director or the Plant Operations Manager.
c. ALERT; AC power capability reduced to minimum design for ≥ 15 minutes, such that any additional single failure would result in less than minimum designed safeguards equipment. The On-Call Emergency Director only.
d. UNUSUAL EVENT; Loss of all offsite AC power to 480 V safeguards buses for ≥ 15 minutes. The On-Call Emergency Director only.

Answer: a

Explanation / Justification

a. Correct. In accordance with IP-EP-120; Emergency Classification this event would be classified as an UNUSUAL EVENT (SU1.1). However, since references are not being provided, the basic SRO knowledge of the Emergency Plan and the classification thresholds

should allow the student to answer this question. UE events are the lowest category and indicate a **potential** degradation of the level of safety. ALERT events involve actual or **substantial** degradation of the level of safety. In this case, a loss of of-site power has occurred, but this is within design analysis (accidents analyzed with a loss of of-site power) and minimum safeguards equipment is available with two AC buses energized from the diesel generators. Additionally the plant is stable with natural circulation established. Therefore, no more than an UNUSUAL EVENT classification is justified. With regard to who can relieve the CRS, IP-EP-210; Central Control Room and IP-EP-120 both state; "... until relieved by the On-Call Emergency Director or other qualified Emergency Director (Plant Operations Manager)."

b. Incorrect. Plausible because the student may believe that because they have lost all off-site power and additionally lost one EDG that event escalation to an ALERT is warranted. Also plausible because IP-EP-120, SA 1.1 states; "... reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in loss of all AC power to safeguards buses" and the stated reason in the distractor uses similar language. However, again the plant is stable and the minimum design requirements are presently met (two AC buses energized from the diesel generators). Second part of distractor regarding Emergency Director relief is correct.

c. Incorrect. Plausible because Plausible because the student may believe that because they have lost all off-site power and additionally lost one EDG that event escalation to an ALERT is warranted. Also plausible because IP-EP-120, SA 1.1 states; "... reduced to a single power source for ≥ 15 minutes such that any additional single failure would result in loss of all AC power to safeguards buses" and the stated reason in the distractor uses similar language. However, again the plant is stable and the minimum design requirements are presently met (two AC buses energized from the diesel generators).

d. Incorrect. Plausible because the first part of the distractor is correct, UNUSUAL EVENT is the proper classification. However, with regard to who can relieve the CRS, IP-EP-210; Central Control Room and IP-EP-120 both state; "... until relieved by the On-Call Emergency Director or other qualified Emergency Director (Plant Operations Manager)."

Technical References:	IPEC Emergency Plan (Classification Definitions) IP-EP-120; Emergency Classification IP-EP-210; Central Control Room IP-EP-AD13; EAL Bases Tech Spec 3.8.1 & Bases
Proposed References to be provided:	None
Learning Objective:	N/A
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A #	002000A204 Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of heat sinks	
Importance:		4.3	4.6

Question: #16

Given the following conditions:

- 33 Auxiliary Feedwater Pump is cleared and tagged for motor replacement.
- PORV PCV-455C is inoperable due to seat leakage and PORV Block Valve RC-MOV-535 is closed.
- The Reactor has tripped from 100% power due to a loss of main feedwater.
- 480 VAC Bus 3A supply breaker has tripped open due to a bus fault.
- 32 Auxiliary Feedwater Pump has tripped on overspeed.
- Operators have transitioned to 3-FR-H.1; Response to Loss of Secondary Heat Sink.
- Initial attempts to restore a secondary heat sink have been unsuccessful and the operators have initiated "Bleed and Feed" in accordance with 3-FR-H.1; Response to Loss of Secondary Heat Sink.

Based on present plant conditions and procedural guidance for "Bleed and Feed", what mitigative actions will be directed by the CRS and will they be successful?

a. Two Charging Pumps running, Three HHSI Pumps running, and both PORV & Associated Block Valves open. Continue attempts to establish secondary heat sink in any SG. Bleed and Feed will be successful, sufficient feed and bleed flow exists to permit RCS heat removal.
b. Two Charging Pumps running, Two HHSI Pumps running, one PORV & Associated Block Valve open, All Reactor Vessel Head Vent Valves open. GO TO Attachment 3, Establishing Feedwater Flow from Secondary Plant. Bleed and Feed may not be successful, if core decay heat exceeds RCS bleed and feed heat removal capability, significant RCS inventory depletion will occur.
c. Two Charging Pumps running, Two HHSI Pumps running, and both PORV & Associated Block Valves open. Continue attempts to establish secondary heat sink in any SG.

Bleed and Feed will be successful, sufficient feed and bleed flow exists to permit RCS heat removal.

d. Two Charging Pumps running, Three HHSI Pumps running, one PORV & Associated Block Valve open, All Reactor Vessel Head Vent Valves open. GO TO Attachment 3, Establishing Feedwater Flow from Secondary Plant. Bleed and Feed may not be successful, if core decay heat exceeds RCS bleed and feed heat removal capability, significant RCS inventory depletion will occur.

Answer: a

Explanation / Justification

a. Correct. Due to the loss of 3A 480 VAC Bus, 32 Charging Pump and 31 ABFP are unavailable, therefore procedurally two charging pumps and three HHSI pumps will be running. The HHSI pumps are on 480VAC Buses 2A, 5A, & 6A. Both PORVs and their associated Block Valves are available because TS 3.4.11 only requires the PORV Block Valve be closed with power available to it for a leaking PORV. The power supplies for the PORV Block Valves are 5A and 6A, not 3A. In accordance with the procedure (3-FR-H.1) and its bases two charging pumps and three HHSI pumps are adequate (feed) to allow opening both PORVs and their associated Block valves (Bleed). Bleed and Feed will be successful, sufficient feed and bleed flow exists to permit RCS heat removal.
b. Incorrect. Plausible because the student may believe that 480VAC Bus 3A supplied both a Charging Pump and a HHSI Pump. Also plausible for the student to believe that power was not available to the one PORV Block Valve, making that PORV not available. Both PORVs and their associated Block Valves are available because TS 3.4.11 only requires the PORV Block Valve be closed with power available to it for a leaking PORV. The power supplies for the PORV Block Valves are 5A and 6A, not 3A. Finally the remainder of the distractor is correct for a condition with only one PORV open.
c. Incorrect. Plausible because the student may believe that 480VAC Bus 3A supplied both a Charging Pump and a HHSI Pump. The HHSI pumps are on 480VAC Buses 2A, 5A, & 6A. Also plausible because the remainder of the distractor is correct, both PORVs are available and the Bleed and Feed would be successful.
d. Incorrect. Plausible because due to the loss of 3A 480 VAC Bus, 32 Charging Pump and 31 ABFP are unavailable, therefore procedurally two charging pumps and three HHSI pumps will be running. The HHSI pumps are on 480VAC Buses 2A, 5A, & 6A. Also plausible for the student to believe that power was not available to the one PORV Block Valve, making that PORV not available. Both PORVs and their associated Block Valves are available because TS 3.4.11 only requires the PORV Block Valve be closed with power available to it for a leaking PORV. The power supplies for the PORV Block Valves are 5A and 6A, not 3A. Finally the remainder of the distractor is correct for a condition with only one PORV open.

Technical References:	I3LP-ILO-EOPFRH WOG ERG for FR-H.1 3-FR-H.1 and Deviation Document Technical Specification 3.4.11 & Bases
Proposed References to be provided:	None
Learning Objective:	Objective 7.0
Question Source:	New

Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A #	039000A204 Malfunctions or operations on the HRPS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: LOCA condition and related concern over hydrogen	
Importance:		3.5	3.9

Question: #17

Analysis has shown that left unchecked, total hydrogen generated following a design bases event can reach the lower flammability limit in 21 days. Hydrogen Recombiners are designed to reduce hydrogen concentration following a LOCA.

The Emergency Operating Procedures place Hydrogen Recombiners in service if measured hydrogen concentration is between ____ % by volume and ____ % by volume. This is because a hydrogen burn becomes a threat to containment integrity when hydrogen concentration exceeds ____ % by volume in dry air.

a. 0.5, 2.4, 3.0.
b. 2.0, 3.0, 3.0.
c. 2.4, 3.0, 4.0.
d. 0.5, 4.0, 4.0.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because hydrogen recombiners are placed in service greater than 0.5% hydrogen concentration by volume in dry air. Also 3-SOP-CB-007; H_2 Recombiner Operation states that they should be placed in service prior to reaching 2.4% by volume to provide adequate time to prevent H_2 concentration from reaching 3.0% by volume.
b. Incorrect. Plausible because student may believe that 2% is the threshold for placing recombiners in service and that the recombiner design capability of 3% is also the lower flammability limit.
c. Incorrect. Plausible because 3-SOP-CB-007; H_2 Recombiner Operation states that they should be placed in service prior to reaching 2.4% by volume to provide adequate time to prevent H_2 concentration from reaching 3.0% by volume. Also plausible because 4% is the lower flammability limit.

d. Correct. 3-ES-1.3; Transfer to Cold Leg Recirculation, states to place the Hydrogen Recombiners in service between 0.5% and 4.0% hydrogen concentration by volume in dry air. Also the lower flammability limit is 4.0%.

Technical References:	I3LP-ILO-VCPAH2 3-SOP-CB-007 3-ES-1.3 & Bases System Description 10.9
Proposed References to be provided:	None
Learning Objective:	Objective 5.0
Question Source:	Modified Bank Questions # 9083 & 9089
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 5

SRO: Requires knowledge of specific content, verses knowledge of the procedure's overall mitigative strategy or purpose.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A #	0720001241 Area Radiation Monitoring: Knowledge of EOP entry conditions and immediate action steps	
Importance:		4.6	4.8

Question: #18

Given the following conditions:

- A small break LOCA resulted in a Safety Injection.
- ECCS operating problems caused some core damage.
- The operating crew restored ECCS flow and is now in 3-ES-1.2; POST-LOCA COOLDOWN AND DEPRESSURIZATION.
- The STA has reviewed the Critical Function Status Trees and has informed the CRS that there is no Red or Orange Path Condition.
- However, the STA has stated that a Yellow Condition exists due to Containment Radiation Levels and states that Functional Restoration Procedure 3-FR-Z.3; Response to High Containment Radiation Level is the appropriate procedure.

What is the Containment Radiation Level that causes the Containment Status Tree to be in a Yellow Condition and is the CRS required to implement the actions of 3-FR-Z.3?

a. ≥ 2 R/HR. No, the CRS is allowed to decide whether or not to implement any Yellow condition FRP.
b. ≥ 3 R/HR. Yes, the CRS will implement the actions of 3-FR-Z.3 concurrently with 3-ES-1.2 or after completion of 3-ES-1.2, but they must complete the actions of 3-FR-Z.3 before the EOP network is exited.
c. ≥ 3 R/HR. No, the CRS is allowed to decide whether or not to implement any Yellow condition FRP.
d. ≥ 2 R/HR. Yes, the CRS will implement the actions of 3-FR-Z.3 concurrently with 3-ES-1.2 or after completion of 3-ES-1.2, but they must complete the actions of 3-FR-Z.3 before the EOP network is exited.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the “Alert” setpoint for Containment High Range Area Radiation Monitors R-25 and R-26 is 2 R/HR. Also plausible because the second part of the distractor is correct per OAP-012, step 4.3.18.
b. Incorrect. Plausible because in accordance with 3-F-0.5; Containment Status Tree, if Containment Radiation is ≥ 3 R/HR a Yellow Path condition exists. The Status Tree states to GO TO 3-FR-Z.3 for those radiation conditions. Also plausible because although the student knows that Yellow Conditions don’t require immediate transition to the FRP from the procedure in effect, they may believe that all valid Yellow conditions must be resolved prior to exiting the EOP network.
c. Correct. In accordance with 3-F-0.5; Containment Status Tree, if Containment Radiation is ≥ 3 R/HR a Yellow Path condition exists. Also see OAP-012, step 4.3.18; “The CRS/SM is allowed to decide whether or not to implement any YELLOW condition FRP.”
d. Incorrect. Plausible because the “Alert” setpoint for Containment High Range Area Radiation Monitors R-25 and R-26 is 2 R/HR. Also plausible because although the student knows that Yellow Conditions don’t require immediate transition to the FRP from the procedure in effect, they may believe that all valid Yellow conditions must be resolved prior to exiting the EOP network.

Technical References:	I3LP-ILO-EOPFRZ 3-FR-Z.3 & Deviation Document 3-F-0.5; Containment Status Tree OAP-012; EOP Users Guide
Proposed References to be provided:	None
Learning Objective:	Objectives 6 & 8
Question Source:	Modified Bank (Questions # 17079 & # 2997)
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 4 & 5

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		COO
	K/A #	1940002135 Knowledge of the fuel-handling responsibilities of SROs	
Importance:		2.2	3.9

Question: #19

In accordance with 3-REF-001-GEN, SECTION 1.1; Refueling Administration, which of the following refueling activities requires that the Refueling SRO be present and in a location that allows direct observation?

I. Reactor Vessel Head Stud De-tensioning and Tensioning.
II. Any movement of the Reactor Vessel Head.
III. Any movement of the Reactor Vessel Upper or Lower Internals.
IV. Control Rod Latching, Unlatching and Drag testing.
V. Any movement of Fuel into or out of the Reactor Vessel.

a. I, II, III, IV, V.
b. II, III, IV, V.
c. II, IV, V.
d. IV, V.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that the refueling procedure requires RSRO oversight of all refueling activities. This is also plausible because stud de-tensioning and tensioning determines MODE change activities as well.
b. Correct. See 3-REF-001-GEN, SECTION 1.1, STEP 1.4.1.2.F. RSRO is not required to be present and in a location that allows direct observation or stud tensioning or de-tensioning.
c. Incorrect. Plausible because the student may believe that only the activities involving core alterations require a RSRO present. Student may believe that the internals are a special case required by the procedure because of special radiological implications (extreme high dose).
d. Incorrect. Plausible because the student may believe that only the activities involving core alterations require a RSRO present. Student could also believe that the RSRO was not required for "ALL" (ANY) vessel head and internals moves.

Technical References:	I3LP-ILO-FHD001 3-REF-001-GEN, Section 1.1
Proposed References to be provided:	None

Learning Objective:	Objectives P (E-16) & Q (E-17)
Question Source:	New
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 7

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		COO
	K/A #	1940001215 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	
Importance:		2.9	3.9

Question: #20

Given the following conditions:

- The Unit is in MODE 1.
- The Shift is manned to the minimum composition in accordance with EN-OP-115; Conduct of Operations.
- The Shift has 4 hours remaining.
- The RO (ATC) has become ill and must leave the site for emergency medical treatment.

Which ONE (1) of the following describes the requirements regarding the shift composition and the MINIMUM required action in this situation IAW EN-OP-115 and Technical Specifications?

a. The RO may leave the site immediately. A replacement must arrive within 1 hour.
b. The RO may leave the site immediately. A replacement must arrive within 2 hours.
c. Responsibilities of the RO may be turned over to the BOP for the remainder of the shift.
d. The CRS may assume the responsibilities of the RO. The STA may perform the duties of the CRS until normal shift relief.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the student may believe that Tech Spec 5.2.2.b requires replacement within 1 hour. Both Tech Spec 5.2.2.b and EN-OP-115 require replacement within 2 hours.
b. Correct. Both Tech Spec 5.2.2.b and EN-OP-115, Attachment 9.4 require replacement within 2 hours.
c. Incorrect. Plausible because the student may believe that Tech Spec 5.2.2.b requires replacement within 4 hours. Both Tech Spec 5.2.2.b and EN-OP-115 require replacement within 2 hours.
d. Incorrect. Plausible because this may be possible if a field supervisor is available and is both SRO Licensed and STA qualified. However, the question stem states minimum composition IAW EN-OP-115, which states that a Field Support Supervisor is not required (0).

Technical References:	I3LP-ILO-ADMIN? Technical Specification 5.2.2.b EN-OP-115, Attachment 9.4
Proposed References to be provided:	None
Learning Objective:	Objective?
Question Source:	Bank – Question # 16705
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 1

SRO knowledge: The required actions or not meting administrative controls listed in Technical Specifications (TS) Section 5 or 6, depending on the facility (e.g., shift staffing requirements).

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		EC
	K/A #	1940002237 Ability to determine operability and/or availability of safety related equipment	
Importance:		3.6	4.6

Question: #21

The plant is at 100% power with the following configuration:

- 1-2-3 is the Essential Service Water Header
- 35 Service Water Pump (SWP) is tagged out for maintenance
- 32 Component Cooling Water Pump (CCWP) is tagged for maintenance

At 1200, 32 EDG is declared inoperable when its pre-lube pump is found off and its lube oil and jacket water temperatures are both below operability values.

In accordance with the Technical Specifications, what must be done?

a. No additional equipment needs to be declared inoperable. 33 EDG must be returned to an operable status within 72 hours or a shutdown is required.
b. Immediately declare 33 CCWP and 36 SWP inoperable. Be in MODE 3 in 7 hours and MODE 5 in 37 hours.
c. Immediately declare 33 CCWP, 33 SWP, and 36 SWP inoperable. 33 EDG must be returned to an operable status within 72 hours or a shutdown is required.
d. 33 CCWP and 36 SWP must be declared inoperable within 4 hours of when 33 EDG was declared inoperable. Once the pumps are declared inoperable, then be in MODE 3 in 6 hours and MODE 4 in 12 hours.

Answer: a

Explanation / Justification

a. Correct. LCO 3.7.9 states that three pumps are required for essential service water header and that two pumps are required for nonessential header. In this case we lose one essential header pump and two nonessential header pumps, both of these being 72 hour action statements. This is the same as the EDG spec and therefore we would not "cascade" nor are we below design bases requirements. LCO 3.7.8 states that two component cooling loops shall be operable, each loop consisting of a single pump and single heat exchanger. Therefore, with only one loop operable, this would be in a 72 hour action statement as well. Again bounded by the Diesel spec and within design bases requirements. Minimum design bases equipment is met with 2 EDGs, 2 essential service water pumps, 1 nonessential

service water pump, and 1 component cooling water pump. Therefore only the 72 hour EDG action statement time is required as redundant equipment remains operable.

b. Incorrect. Plausible because student may believe that redundant equipment is not operable and that Tech Spec 3.0.3 is applicable. They may believe that they are below minimum equipment required as defined in OAP-034.
c. Incorrect. Plausible because these three pumps are supported by the 32 EDG for an emergency power supply. The student may believe that if the emergency power supply is inoperable, then the pumps are automatically declared inoperable and that we do "cascade" tech specs and enter all three technical specifications. Still plausible for the student to believe the 72 hour action time, because we are still meeting minimum design requirements.
d. Incorrect. Plausible because the student may believe that redundant equipment is inoperable and that EDG Tech Spec actions A3 and F apply. Tech Specs are not being provided as a reference, so student may believe that actions can't be complied with and therefore MODE 3 in 6 hours and MODE 4 in 12 hours per 3.8.1.F.

Technical References:	I3LP-ILO-EDSEDG
	Tech Spec 3.7.8 & Bases
	Tech Spec 3.7.9 & Bases
	Tech Spec 3.8.1 & Bases
	OAP-034; Safety Function Determination Process
Proposed References to be provided:	None
Learning Objective:	Objectives 7 & 8
Question Source:	Bank – Question # 24834
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 2

NOTE: SRO level because requires application of required actions and bases knowledge. Also actions are greater than 1 hour and require knowledge of below the line information.

Need to have Indian Point review for accuracy, updated a bank question and changed to match current revision of Technical Specifications.

No references provided because knowledge of operability requirements (minimum design bases), Tech Spec LCOs, and Tech Spec Bases are enough to answer question.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		RC
	K/A #	1940001235 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	
Importance:		4.4	4.6

Question: #22

Given the following conditions:

- The Unit is at 100% Power.
- R15 SJAE EXH has actuated on the Radiation Monitoring Control Cabinet (RMCC)
- 3-AOP-SG-1; Steam Generator Tube Leak has been entered and is presently being implemented by the crew.
- R-15; Air Ejector Exhaust Gas Activity Monitor has a rising trend.
- R-19; Steam Generator Blowdown Liquid Activity Monitor has a rising trend.
- Chemistry has been requested to perform 0-CY-2450; Primary to Secondary Leak.
- The crew has initiated Attachment 1 of 3-AOP-SG-1 and has determined an initial estimated Primary-to-Secondary Leakrate of 30 gpd.

Listed below are the first (3) three estimated leakrate determinations recorded by the crew on Attachment 2; Leak Rate Log.

	Current Leakrate (gpd)
Estimate # 1	30 gpd
Estimate # 2	32 gpd
Estimate # 3	35 gpd

How often does the estimated leakrate need to be calculated and what radiation monitors are being used to calculate the estimated leakrate?

a. 15 minutes. Radiation Monitors R-15; Air Ejector Exhaust Gas Activity and R-19; Steam Generator Blowdown Liquid Activity.
b. 2 hours. Radiation Monitors R-15; Air Ejector Exhaust Gas Activity and R-19; Steam Generator Blowdown Liquid Activity.
c. 2 hours. Radiation Monitors R-15; Air Ejector Exhaust Gas Activity and R-63A&B; Gross Failed Fuel Detector.
d. 15 minutes. Radiation Monitors R-15; Air Ejector Exhaust Gas Activity and R-63A&B; Gross Failed Fuel Detector.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because first part of distractor is correct, Attachment 2 of 3-AOP-SG-1 states; "If leak increases ≥ 30 gpd but < 75 gpd take data & determine leakrate every 15 minutes." Also plausible because the student may believe that R-19; SG Blowdown is used in addition to R-15; Air Ejector for determining the estimated leakrate.
b. Incorrect. Plausible because if the leakrate is ≥ 30 gpg but < 75 gpd and stable for 1 hour (≤ 5 gpd increase in 1 hour), then the procedure has you take the readings every 2 hours. Because the initial estimate is ≥ 30 gpd, 15 minute readings are required and the rate of increase is actually 8 – 12 gpd/hr. Also plausible because the student may believe that R-19; SG Blowdown is used in addition to R-15; Air Ejector for determining the estimated leakrate.
c. Incorrect. Plausible because if the leakrate is ≥ 30 gpg but < 75 gpd and stable for 1 hour (≤ 5 gpd increase in 1 hour), then the procedure has you take the readings every 2 hours. Because the initial estimate is ≥ 30 gpd, 15 minute readings are required and the rate of increase is actually 8-12 gpd/hr. Also plausible because the second part of the distractor is correct, R-15 and R-63A&B are used in the estimated leakrate calculation.
d. Correct. Attachment 2 of 3-AOP-SG-1 states; "If leak increases ≥ 30 gpd but < 75 gpd take data & determine leakrate every 15 minutes." The readings provided have therefore been taken every 15 minutes and the delta leakrate would then be 8-12 gpd/hr and also above the threshold for taking data every 2 hours. Finally in accordance with Attachment 1, estimated leakrate calculation method 1, R-15 and R-63A&B are used in the estimated leakrate calculation.

Technical References:	I3LP-ILO-AOPSG
	I3LP-ILO-RMSPRM
	3-AOP-SG-1 & Bases
	3-ARP-040
	3-SOP-RM-10
	System Description 12.0
Proposed References to be provided:	None
Learning Objective:	Objective 2
	Objective B (2)
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 4 & 5

NOTE: SRO knowledge based on the analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures. SRO knowledge of content of the procedure versus knowledge of the procedure' overall mitigative strategy or purpose and when to implement attachments and appendices, including how to coordinate these items with procedure steps.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		RC
	K/A #	1940002311	
		Ability to control radiation releases	
Importance:		3.8	4.3

Question: #23

The Unit is in MODE 5 following planned shutdown into a scheduled refueling outage. The Unit had a continuous run at power for the last 6 months. The Outage Manager has requested Operations initiate a release permit for a VC Purge in accordance with 3-SOP-WDS-013. Chemistry has completed VC grab sample and provided operations with both Noble Gas and Iodine Activity numbers.

Shift Operations has determined that the Calculated Release Rate for Noble Gas is greater than the IP-SMM-CY-001 Annual Release Rate. It has been determined that the Purge will require the use of the Instantaneous Release Rate Limit. Who is required to authorize the use of this limit in accordance with 3-SOP-WDS-013?

a. Unit Operations Manager.
b. Chemistry Manager.
c. General Manager – Plant Operations.
d. Site Vice President.

Answer: c

Explanation / Justification

a. Incorrect. Plausible because the Unit Operations Manager is required to authorize the use of the Quarterly Average Release Rate Limit.
b. Incorrect. Plausible because the student may believe that since the RPM has approval authority over the Radioactive Effluents Control Program (IP-SMM-CY-001) that he would be authorized to provide approval for using a higher release rate.
c. Correct. See step 2.2 and step 4.3.9 of 3-SOP-WDS-013.
d. Incorrect. Plausible because in accordance with the Radioactive Effluents Program Procedure (IP-SMM-CY-001), step 6.2.1.4; "the Instantaneous Release Rate Limit is a SITE limit, and applies to the total release rate from all operating units on site." Based on this definition, the student may believe that using the Instantaneous Limit requires Site Vice President's approval.

Technical References:	I3LP-ILO-GWR001
	3-SOP-WDS-013
	IP-SMM-CY-001
Proposed References to be provided:	None

Learning Objective:	Objectives 6e & 10h
Question Source:	Modified Bank – Question # 24958
Question Cognitive Level:	Knowledge
10CFR Part 55 Content:	55.43 (b) 4

NOTE: SRO knowledge based knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose. Topic is related to 55.43 (b) (4); process for gaseous/liquid release approvals, i.e., release permits.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		EOP-EP
	K/A #	1940002438 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required	
Importance:		2.4	4.4

Question: #24

Given the following conditions:

- A Site Area Emergency has been declared due to a LOCA outside Containment.
- The LOCA is into the PAB building and a pathway to the environment exists.
- Limited makeup to the RWST is available.
- An operator has volunteered to go to the PAB building to locally isolate the leak.
- The EOF has been fully staffed and operational for an hour.
- This action will result in a significant reduction in the offsite dose, protecting a large population.

Using the Emergency Exposure Guidelines listed in the Indian Point Energy Center Emergency Plan and Form EP-6; Emergency Exposures Authorization, what is the maximum emergency exposure this volunteer operator may receive while performing this action and who must approve?

a. 10 Rem TEDE with SM approval.
b. 25 Rem TEDE with SM approval.
c. 10 Rem TEDE with ED approval.
d. 25 Rem TEDE with ED approval.

Answer: d

Explanation / Justification

a. Incorrect. Plausible because 10 Rem is the dose limit for protecting valuable property and the student may believe that 25 Rem is only for **life saving** (**bolded** on Form EP-6). Also plausible because the Shift Manager is initially the Emergency Director (ED). However, he would have been relieved by the on duty Emergency Director (ED) at the EOF.
b. Incorrect. Plausible because 25 Rem is the correct dose limit for life saving or <u>the protection of large populations.</u> Also plausible because the Shift Manager is initially the Emergency Director (ED). However, he would have been relieved by the on duty Emergency Director (ED) at the EOF.

c. Incorrect. Plausible because 10 Rem is the dose limit for protecting valuable property and the student may believe that 25 Rem is only for **life saving** (**bolded** on Form EP-6). Also plausible because the second part of the distractor is correct.
d. Correct. See Form EP-6 and below technical references. "Volunteers may be authorized up to 25 Rem for life saving or the protection of large populations."

Technical References:	IP-EP-115, Forms Form EP-6; Emergency Exposure Authorization IPEC E-Plan, 13-01, Page K-1; Emergency Exposure Guidelines IPEC E-Plan, 13-01, Pages B-3 through B-6 IP-EP-210, Central Control Room IP-EP-220, Technical Support Center IP-EP-250, EOF
Proposed References to be provided:	None
Learning Objective:	N/A
Question Source:	Bank – Indian Point 2 – 2014 NRC Exam, #98
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 4

NOTE: Need to resolve whether the Emergency Plant Manager (EPM) is more appropriate to authorize the emergency exposure because it is for personnel on-site. See references above, they state that EPM authorizes for on-site and ED authorizes for off-site. Also note these ae common procedures and this is a Bank question from IP2 – 2014 NRC Exam, they used ED as correct answer. See page 10 in IP-EP-220, Attachment 9.1; Emergency Plant Manager Checklist and pages 15 & 16 in IP-EP-250, Attachment 9.1; Emergency Director Checklist.

Exam Outline Cross Reference:	Level:	RO	SRO
	Tier #		3
	Group #		EOP-EP
	K/A #	1940002445 Ability to prioritize and interpret the significance of each annunciator or alarm	
Importance:		4.1	4.3

Question: #25

Given the following initial plant conditions:

- Unit is at 100% Power.
- Rod Power Supply M-G Set 32 is cleared and tagged for motor replacement.
- Individual Rod Position Indicating System is powered from backup power supply MCC-39.

A failure of the Station Auxiliary Transformer has just occurred. 32 and 33 EDGs have started and loaded in Blackout Mode. The Control Room receives a number of alarms; including the following annunciator alarms that the BOP Operator believes to be most relevant to the loss of the Station Auxiliary Transformer and requiring action:

- STATION AUX XFMR LOCKOUT RELAY TRIPPED
- 6900 V STATION AUX BREAKER TRIP 52ST5, 52ST6
- MOTOR CONTROL CENTER #39 52/MCC9 AUTO TRIP

Following the CRS's assessment of plant conditions and prioritization of the above annunciators, which of the following procedures should be implemented by the CRS?

a. 3-AOP-ROD-1; Rod Control and Indications System Malfunction.
b. 3-AOP-138KV-1; Loss of Power to 6.9KV Bus 5 and/or 6.
c. 3-ARP-011; MOTOR CONTROL CENTER #39 52/MCC9 AUTO TRIP.
d. 3-E-0; Reactor Trip or Safety Injection.

Answer: b

Explanation / Justification

a. Incorrect. Plausible because the loss of Bus 5A resulted in the loss of MCC-39 which was providing power to the IRPIs. This would cause all the IRPIs to be inoperable and require placing rod control in Manual immediately IAW Tech Spec 3.1.7.B. This action is taken per 3-AOP-ROD-1 step 4.136 and proceeding NOTE. However, although this is an immediate action per Tech Specs, step 4.11 of 3-AOP-138KV-1 will perform the same action and that procedure will also handle the bigger issue of loss of off-site power to buses 5A & 6A.

b. Correct. The CRS should recognize that given the existing plant conditions, 3-AOP-138KV-1 will provide the best overall mitigation. Step 4.11 of 3-AOP-138KV-1 will handle the immediate action required for rods in Manual per Tech Spec 3.1.7.B, but will also handle the bigger issue of loss of off-site power to buses 5A & 6A.
c. Incorrect. Plausible because the loss of Bus 5A resulted in the loss of MCC-39 which was providing power to the IRPIs. This would cause all the IRPIs to be inoperable and require placing rod control in Manual immediately IAW Tech Spec 3.1.7.B. Student may believe that prioritizing the loss of MCC-39 alarm response procedure is the highest priority action. However, this alarm response does not reference Tech Spec 3.1.7.B nor does it require rods to be placed in manual. This is more than likely because MCC-39 is the backup power supply to IRPIs. Again, Step 4.11 of 3-AOP-138KV-1 will handle the immediate action required for rods in Manual per Tech Spec 3.1.7.B, but will also handle the bigger issue of loss of off-site power to buses 5A & 6A.
d. Incorrect. Plausible because the student may believe that the above transient has resulted in a Reactor Trip or requires a Reactor Trip be initiated. The stem of the question states that the IRPIs are being powered from MCC-39, the student may believe that a Reactor Trip is required due to not having any IRPI indications. More plausible is that the student may believe that the plant has experienced a loss of power to both Rod Control M-G Sets. Note that 32 M-G set was tagged for motor replacement in the stem, the student may believe that 31 M-G Set is powered by either bus 5A or 6A. Actually, 32 M-G Set is powered from Bus 6A (already O/S) and 31 M-G Set is powered from Bus 2A which is being powered via the Unit Auxiliary Transformer (Main Generator).

Technical References:	I3LP-ILO-AOP138 and Bases
	3-ARP-011; PANEL SHF – ELECTRICAL
	3-AOP-ROD-1 and Bases
	3-SOP-RC-001
	System Description 27.1
Proposed References to be provided:	None
Learning Objective:	Objective 1.0
Question Source:	New
Question Cognitive Level:	Comprehension
10CFR Part 55 Content:	55.43 (b) 5

NOTE: Question to be reviewed by Indian Point to ensure complete accuracy. May want to run condition on simulator and possibly add additional alarms to prioritize.

Answer Key

2015 U3 NRC Exam

1	**D**	26	**C**	51	**B**	1	**D**
2	**B**	27	**B**	52	**C**	2	**A**
3	**D**	28	**B**	53	**A**	3	**C**
4	**D**	29	**A**	54	**A**	4	**B**
5	**C**	30	**B**	55	**C**	5	**D**
6	**D**	31	**A**	56	**B**	6	**D**
7	**B**	32	**C**	57	**A**	7	**B**
8	**C**	33	**B**	58	**A**	8	**C**
9	**C**	34	**C**	59	**D**	9	**A**
10	**C**	35	**A**	60	**B**	10	**C**
11	**A**	36	**D**	61	**A**	11	**B**
12	**A**	37	**D**	62	**B**	12	**A**
13	**C**	38	**C**	63	**D**	13	**C**
14	**B**	39	**C**	64	**B**	14	**A**
15	**C**	40	**A**	65	**A**	15	**A**
16	**B**	41	**D**	66	**C**	16	**A**
17	**B**	42	**C**	67	**D**	17	**D**
18	**D**	43	**B**	68	**A**	18	**C**
19	**C**	44	**A**	69	**D**	19	**B**
20	**A**	45	**A**	70	**D**	20	**B**
21	**B**	46	**B**	71	**C**	21	**A**
22	**D**	47	**C**	72	**A**	22	**D**
23	**D**	48	**B**	73	**D**	23	**C**
24	**A**	49	**D**	74	**C**	24	**D**
25	**D**	50	**D**	75	**A**	25	**B**

Facility: Indian Point 3	Date of Examination: 5/11/15
Examination Level: RO X SRO ☐	Operating Test Number: 1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	N R	Perform a Shutdown Margin Calculation 2.1.37 – Knowledge of procedures, guidelines, or limitations associated with reactivity management. RO – 4.3 (45.6)
Conduct of Operations	N R	Hydrogen Recombiner Operation Determine Required Heater Power IAW 3-SOP-CB-007. 2.1.25 – Ability to interpret reference materials, such as graphs, curves, tables, etc. RO – 3.9 (45.12)
Equipment Control	N/A	N/A
Radiation Control	N R	Determine Radiological Conditions, RWP Requirements, and Potential Dose 2.3.7 – Ability to comply with radiation work permit requirements during normal or abnormal conditions. RO – 3.5 (45.10)
Emergency Procedures/Plan	D S	Perform Initial Unusual Event Notification 2.4.39 – Knowledge of RO responsibilities in emergency plan implementation. RO – 3.9 (45.11)

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

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

Facility: Indian Point 3	Date of Examination: 5/11/15
Examination Level: RO ☐ SRO X	Operating Test Number: 1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	N R	Review a Shutdown Margin Calculation 2.1.37 – Knowledge of procedures, guidelines, or limitations associated with reactivity management. SRO – 4.6 (45.6)
Conduct of Operations	D R or S	Determine Isolation Boundaries for CCW Leak Using Plant Prints 2.2.41 - Ability to obtain and interpret station electrical and mechanical drawings. SRO – 3.9 (45.13)
Equipment Control	N R	Review completed Service Water Pump Surveillance. 2.2.12 – Knowledge of surveillance procedures. SRO – 4.1 (45.13)
Radiation Control	D R	Review / Authorize a Liquid Waste Release 2.3.6 – Ability to approve release permits. SRO – 3.8 (45.10)
Emergency Procedures/Plan	D R or S	Determine Protective Action Recommendations 2.4.44 – Knowledge of emergency plan protective action recommendations. SRO – 4.4 (45.11)

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

* Type Codes & Criteria:

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

Facility: Indian Point 3 Date of Examination: 5/11/15

Exam Level: RO ☐ SRO-I X SRO-U ☐ Operating Test No.: 1

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U, including 1 ESF)

System / JPM Title	Type Code*	Safety Function
a. Respond to Turbine 1st Stage Pressure Transmitter Failure (412A) IAW 3-INST-1 and subsequent T_{avg} recovery with manual rods (Continuous rod motion) IAW 3-AOP-ROD-1. (use 41% Power IC)	A, D, S	1
b. Place Excess Letdown In Service	D, S	2
c. Depressurize the RCS following a SGTR (using PORVs) IAW E-3.	A, D, S	3
d. RHR Pump Trip while on shutdown cooling IAW 3-AOP-RHR-1	L, S	4P
e. Swap Essential Service Water Pumps (Check valve on O/S pump sticks open) (ensure pressure does not drop below trip setpoint)	A, N, S	4S
f. Verify Containment Spray Operation IAW E-0, step 9. (Manually initiate and stop RCPs)	A, D, S	5
g. Pressurizer Pressure Channel fails low, trip bistables.	D, S	7

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

System / JPM Title	Type Code*	Safety Function
i. Locally Start and Synchronize for Parallel Operation # 31 EDG IAW 3-SOP-EL-001.	D	6
j. Perform a Radioactive Liquid Release (Monitor Tank) IAW 3-SOP-WDS-014	A, N, R	9
k. Local Charging Pump Operation (sect. 4.10) from RWST with CCW **not** available, requiring alignment to city water cooling (sect. 4.11) IAW 3-SOP-ESP-001.	A, E, M, R	8

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

* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(EN)gineered safety feature	- / - / ≥1 (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	

Facility: Indian Point 3 Date of Examination: 5/11/15

Exam Level: RO X SRO-I ☐ SRO-U ☐ Operating Test No.: 1

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U, including 1 ESF)

System / JPM Title	Type Code*	Safety Function
a. Respond to Turbine 1st Stage Pressure Transmitter Failure (412A) IAW 3-INST-1 and subsequent T_{avg} recovery with manual rods (Continuous rod motion) IAW 3-AOP-ROD-1. (use 41% Power IC)	A, D, S	1
b. Place Excess Letdown In Service	D, S	2
c. Fill an Accumulator with a SI Pump (RO Only)	D, S	2
d. Depressurize the RCS following a SGTR (using PORVs) IAW E-3.	A, D, S	3
e. RHR Pump Trip while on shutdown cooling IAW 3-AOP-RHR-1	L, S	4P
f. Swap Essential Service Water Pumps (Check valve on O/S pump sticks open) (ensure pressure does not drop below trip setpoint)	A, N, S	4S
g. Verify Containment Spray Operation IAW E-0, step 9. (Manually initiate and stop RCPs)	A, D, S	5
h. Pressurizer Pressure Channel fails low, trip bistables.	D, S	7

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

i. Locally Start and Synchronize for Parallel Operation # 31 EDG IAW 3-SOP-EL-001.	D	6
j. Perform a Radioactive Liquid Release (Monitor Tank) IAW 3-SOP-WDS-014	A, N, R	9
k. Local Charging Pump Operation (sect. 4.10) from RWST with CCW **not** available, requiring alignment to city water cooling (sect. 4.11) IAW 3-SOP-ESP-001.	A, E, M, R	8

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

* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(EN)gineered safety feature	- / - / ≥1 (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	

Facility: Indian Point 3 Scenario No.: 1 Op-Test No.: 1

Examiners: ______________________ Operators: ______________________

Initial Conditions: Approx. 75% Power, power ascension in progress to 100% power IAW 3-POP-2.1, Attachment 2. 2 Condensate Pumps are in service. Fuel is Conditioned.

Turnover: Continue power ascension to 100% IAW 3-POP-2.1, Attachment 2 Returning from maintenance on 32 MBFP (oil leak). 32 Charging Pump is out of service, tagged due to a packing leak (maintenance has pump disassembled).

Event No.	Malf. No.	Event Type*	Event Description
1		R (ATC) N (BOP) N (CRS)	Raise load IAW 3-POP-2.1
2		I (BOP) I (ATC) TS (CRS)	31 Steam Generator Loop 1B (419B) Steam Flow Channel fails high.
3		TS (CRS)	33 Accumulator Low Pressure Alarm actuates.
4		C (ALL)	32 MBFP trips.
5		C(ALL)*	IF crew does not trip reactor due to 32 MBFP trip, insert MTG Voltage Regulator Failure and within 3 minutes a direct MTG Trip from Buchannan.*
6		M(ALL)	Automatic Turbine / Reactor Trip / Safety Injection. Coincident with the Reactor Trip, #31 SG Safety Valve will fail open.
7		C (CRS) C(BOP)	31 Safety Injection Pump fails to auto start.
8		M(ALL)	31 SG Tube Rupture

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

Facility: Indian Point 3 Scenario No.: 2 Op-Test No.: 1

Examiners: ______________________ Operators: ______________________

Initial Conditions: 100% Reactor Power, Fuel is Conditioned.

Turnover: Unit in a 7 day Technical Specification due to RWST Level Instrumentation. Tech Spec LCO 3.5.4.B in effect (7-day). LIC-921 and LC-923 are inoperable, maintenance is waiting for parts to repair. Parts delivery expected tomorrow. #32 Charging Pump is out of service, tagged due to a packing leak (maintenance has pump disassembled).

Event No.	Malf. No.	Event Type*	Event Description
1		C (ATC) TRM/TS (CRS)	The in service #31 Charging Pump trips. TRM 3.1.C.1 action A and TRM 3.7.B.3
2		C (ALL) TS (CRS)	Loss of RWST Level. Fork lift truck crashes into RWST resulting in level lowering to about 11 feet over 40 minutes.
3		R (ATC) R (CRS) N (BOP)	Tech Spec Required Shutdown due to inoperable RWST (volume). Team commences shutdown. (Call as Operations Manager if necessary to start shutdown)
4		C (ALL)	# 32 RCP Seal Malfunction, high vibrations and #1 Seal degradation.
5		C (ALL)	#32 RCP Seal leak rises to > 6 gpm.
6		C (ATC)	Automatic Turbine Trip fails.
7		M (ALL)	Hot Leg SBLOCA approx. 800 gpm develops.
8		C (BOP)	31 RHR Pump fails to start.
9			Team transitions to E-1, then ES-1.3, then ECA-1.1 due to loss of Emergency Coolant Recirculation.

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

Facility: Indian Point 3 Scenario No.: 3 Op-Test No.: 1

Examiners: ______________________ Operators: ______________________

Initial Conditions: 100% Power, # 32 Component Cooling Water Pump is out of service for Maintenance (Motor replacement)

Turnover: 100% Power, # 32 Component Cooling Water Pump is out of service for Maintenance (Motor replacement) . Maintain 100%

Event No.	Malf. No.	Event Type*	Event Description
1		C BOP/CRS R (ATC) TS (CRS)	#33 Condensate Pump Trips, crew responds IAW 3-AOP-FW-1. Technical Specification for delta I (discuss at minimum after scenario)
2		I (ATC or BOP) TS (CRS)	32 SG Controlling level channel fails low (LT427B).
3		M (ALL)	Steam Leak in Turbine Building.
4		C/M (ALL)	The reactor will not trip from the control room and the crew responds per FR-S-1, ATWS
5		C (BOP)	Following SI actuation, #32 RHR Pump and #33 SI did not start. BOP starts pumps per 3-RO-1 RNO.
6		M (ALL)	Failure of MSIVs to close from Control Room. (2 will be closed locally after AFW is throttled to 100 gpm per ECA-2.1).

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

Facility: Indian Point 3 Scenario No.: 4 Op-Test No.: 1

Examiners: ______________________ Operators: ______________________

Initial Conditions: 100% Power with 31 EDG out of service due to malfunctioning governor.

Turnover: 100% Power with 31 EDG out of service due to malfunctioning governor. Maintain 100% power. 31 Charging Pump I/S.

Event No.	Malf. No.	Event Type*	Event Description
1	MAL-SWS001C	C (BOP)	33 SWP trips
2	MAL-EPS005C	C (ALL)	480V Bus 5A Fault
3	N/A	R (ATC) TS (CRS)	Tech Spec required shutdown
4	MAL-NIS006A	C (ATS) C (CRS) TS (CRS)	Power Range Channel 41 Upper Detector Fails High
5	MAL-EPS001	M(ALL)	Station Blackout
6	MAL-DSG001A	C (BOP)	32 EDG fail to start
7	MAL-PRS003D	C (ATC)	PZR PORV Fails Open (455C fails open after loss of all AC, we need to isolate block valve later)
8	MAL-SWS001E	C (BOP)	SW pump does not auto start after bus energized 32 EDG comes back, 6A bus back. Failure of 36 SW pump to auto start.

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

Facility: Indian Point 3 Scenario No.: 5 – low power Op-Test No.: 1

Examiners: ______________________ Operators: ______________________

Initial Conditions: 4% Reactor Power, Plant startup following a forced outage, power ascension IAW 3-POP-1.3. #32 Component Cooling Water Pump is out of service for Maintenance (Motor replacement). Raise power to 10%, the warm up the MTG and place in service.

Turnover: 4% Reactor Power, Plant startup following a forced outage, power ascension IAW 3-POP-1.3. #32 Component Cooling Water Pump is out of service for Maintenance (Motor replacement). Raise power to 10%, the warm up the MTG and place in service. Fuel is conditioned.

Event No.	Malf. No.	Event Type*	Event Description
1		R ATC N CRS / BOP	Perform power ascension to 10% IAW 3-POP-1.3; Plant Startup from Zero to 45% Power.
2	XMT-RCS052	C ATC / BOP TS CRS	Thot fails high. Crew responds per 3-AOP-INST-1 and Technical Specifications
3	MAL-RCS005 A	M ALL / TS CRS	Initiate RCS Leakage, starting at 23 gpm. Crew to respond IAW 3-AOP-Leak-1.
4	MAL-RCS005 B	M ALL	Raise RCS leakrate to approximately 223 gpm. Crew will trip RX and initiate manual SI.
5		C ALL	AUTO SI Fails, manual initiation required. If manually initiated, then no failure.
6	MAL-RCS001 A	M ALL	LBLOCA
7		C BOP	#31 RHR and #32 SI Pump did not start and #34 Fan Cooler Unit dampers did not reposition.
8		M ALL	Once SI has been verified by the BOP and reset, a loss of Off-Site Power occurs.
9		C ALL	Crew establishes required SI loads on Diesels.

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