Exelon Generation.

10 CFR 55.40

NMP2L 2596
September 4, 2015

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

Nine Mile Point Nuclear Station, Unit 2
Renewed Facility Operating License No. NPF-69
NRC Docket No. 50-410

Subject: Nine Mile Point Unit 2 Initial License Examination Outlines

Reference (1) Letter from D. Jackson (NRC) to P. Orphanos (NMPNS), dated June 11, 2015, Senior Reactor and Reactor Operator Initial License Examinations - Nine Mile Point Nuclear Station, Unit 2

As discussed in Reference (1), arrangements have been made for the administration of license examinations at Nine Mile Point Unit 2 during the week of November 30, 2015. The examinations are being prepared based on the guidelines in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 10.

Enclosed are the approved examination outlines for the Unit 2 Senior Reactor and Reactor Operator initial license examinations. The written examination outlines were developed using an electronic random outline generator developed by Western Technical Services, Inc.

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

Should you have any questions regarding the information in this submittal, please contact Ryan Hamilton, Operations Training Manager, at (315) 349-1385.

Sincerely,

Jeffrey W. Gerber
Director Site Training, Nine Mile Point Nuclear Station
Exelon Generation Company, LLC

JWG/BTV

Enclosure: Examination Outlines for the Unit 2 Senior Reactor and Reactor Operator Initial License Examinations

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

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COMMITMENTS IDENTIFIED IN THIS CORRESPONDENCE
• None

Posting Requirements for Responses – NOV/Order **NO**

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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 that are not included on the outline should be added. Refer to section D.1.b of ES-401, for guidance regarding 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 KAs 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/A's

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IR) 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 10CFR55.43

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

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
295021 Loss of Shutdown Cooling / 4					X		AA2.05 - Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : Reactor vessel metal temperature	3.5	76
295004 Partial or Total Loss of DC Pwr / 6					X		AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Cause of partial or complete loss of D.C. power	3.6	77
295030 Low Suppression Pool Water Level / 5					X		EA2.02 - Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Suppression pool temperature	3.9	78
295019 Partial or Total Loss of Inst. Air / 8						X	2.4.18 - Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.	4.0	79
295018 Partial or Total Loss of CCW / 8						X	2.2.42 - Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications	4.6	80
295038 High Off-site Release Rate / 9						X	2.4.8 - Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOP's.	4.5	81
295025 High Reactor Pressure / 3					X		EA2.05 - Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Decay heat generation.	3.6	82
295018 Partial or Total Loss of CCW / 8	X						AK1.01 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Effects on component/system operations	3.5	39
700000 Generator Voltage and Electric Grid Disturbances	X						AK1.02 - Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Over-excitation.	3.3	40
295006 SCRAM / 1	X						AK1.03 - Knowledge of the operational implications of the following concepts as they apply to SCRAM : Reactivity control	3.7	41
295021 Loss of Shutdown Cooling / 4		X					AK2.05 - Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: Fuel pool cooling and cleanup system	2.7	42
600000 Plant Fire On-site / 8		X					AK2.01 - Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors, detectors and valves	2.6	43
295026 Suppression Pool High Water Temp. / 5		X					EK2.06 - Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Suppression pool level	3.5	44
295037 SCRAM Conditions Present and Reactor Power			X				EK3.08 - Knowledge of the reasons for the following responses as they apply to	3.6	45

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

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
Above APRM Downscale or Unknown / 1							SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : ATWS circuitry: Plant-Specific		
295038 High Off-site Release Rate / 9			X				EK3.03 - Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Control room ventilation isolation: Plant-Specific	3.7	46
295025 High Reactor Pressure / 3			X				EK3.02 - Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE : Recirculation pump trip: Plant-Specific	3.9	47
295016 Control Room Abandonment / 7				X			AA1.02 - Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : Reactor/turbine pressure regulating system	2.9	48
295005 Main Turbine Generator Trip / 3				X			AA1.04 - Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP : Main generator controls	2.7	49
295023 Refueling Acc Cooling Mode / 8				X			AA1.01 - Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS : Secondary containment ventilation	3.3	50
295030 Low Suppression Pool Water Level / 5					X		EA2.04 - Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Drywell/ suppression chamber differential pressure: Mark-I&II	3.5	51
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4					X		AA2.06 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Nuclear boiler instrumentation	3.2	52
295004 Partial or Total Loss of DC Pwr / 6					X		AA2.04 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : System lineups	3.2	53
295028 High Drywell Temperature / 5						X	2.4.1 - Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps.	4.6	54
295003 Partial or Complete Loss of AC / 6						X	2.2.37 - Equipment Control: Ability to determine operability and/or availability of safety related equipment.	3.6	55
295024 High Drywell Pressure / 5						X	2.4.9 - Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	3.8	56
295031 Reactor Low Water Level / 2	X						EK1.01 - Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL : Adequate core cooling	4.6	57
295019 Partial or Total Loss of Inst. Air / 8		X					AK2.04 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Reactor water cleanup	2.8	58

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

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
K/A Category Totals:	4	4	3	3	3/4	3/3	Group Point Total:		20/7

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

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
295033 High Secondary Containment Area Radiation Levels / 9					X		EA2.03 - Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Cause of high area radiation	4.2	83
295036 Secondary Containment High Sump/Area Water Level / 5						X	2.4.41 - Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications.	4.6	84
500000 High CTMT Hydrogen Conc. / 5					X		EA2.02 - Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Oxygen monitoring system availability	3.5	85
295020 Inadvertent Cont. Isolation / 5 & 7	X						AK1.01 - Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION : Loss of normal heat sink	3.7	59
295012 High Drywell Temperature / 5		X					AK2.02 - Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell cooling	3.6	60
295010 High Drywell Pressure / 5			X				AK3.01 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : Drywell venting	3.8	61
295009 Low Reactor Water Level / 2				X			AA1.04 - Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL : Reactor water cleanup	2.7	62
295029 High Suppression Pool Water Level / 5					X		EA2.03 - Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Drywell/containment water level	3.4	63
295035 Secondary Containment High Differential Pressure / 5						X	2.2.39 - Equipment Control: Knowledge of less than one hour technical specification action statements for systems.	3.9	64
295034 Secondary Containment Ventilation High Radiation / 9	X						EK1.0 - Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : Radiation Releases	4.1	65
K/A Category Totals:	2	1	1	1	1/2	1/1	Group Point Total:		7/3

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
203000 RHR/LPCI: Injection Mode								X				A2.14 - Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Initiating logic failure	3.9	86
211000 SLC								X				A2.04 - Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadequate system flow	3.4	87
217000 RCIC											X	2.1.23 – Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.4	88
262002 UPS (AC/DC)											X	2.2.40 – Ability to apply Technical Specifications for a system.	4.7	89
300000 Instrument Air											X	2.1.20 – Conduct of Operations: Ability to interpret and execute procedure steps.	4.6	90
300000 Instrument Air	X											K1.04 - Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: Cooling water to compressor	2.8	1
263000 DC Electrical Distribution	X											K1.04 - Knowledge of the physical connections and/or cause- effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Ground detection	2.6	2
215004 Source Range Monitor		X										K2.01 - Knowledge of electrical power supplies to the following: SRM channels/detectors	2.6	3
209001 LPCS		X										K2.03 - Knowledge of electrical power supplies to the following: Initiation logic	2.9	4
218000 ADS			X									K3.01 - Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Restoration of reactor water level after a break that does not depressurize the reactor when required	4.4	5

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
203000 RHR/LPCI: Injection Mode			X									K3.03 - Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following: Adequate core cooling	4.6	6
211000 SLC				X								K4.05 - Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Dispersal of boron upon injection into the vessel	3.4	7
209002 HPCS				X								K4.02 - Knowledge of HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) design feature(s) and/or interlocks which provide for the following: Prevents over filling reactor vessel: Plant-Specific.	3.4	8
215003 IRM					X							K5.01 - Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : Detector operation	2.6	9
264000 EDGs					X							K5.06 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : Load sequencing	3.4	10
212000 RPS						X						K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM : D.C. electrical distribution	2.8	11
217000 RCIC						X						K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Suppression pool water supply	3.5	12
259002 Reactor Water Level Control							X					A1.04 - Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor water level control controller indications	3.6	13
400000 Component Cooling Water							X					A1.01 - Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: CCW flow rate	2.8	14

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
261000 SGTS								X				A2.11 - Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High containment pressure	3.2	15
239002 SRVs								X				A2.03 - Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV	4.1	16
262001 AC Electrical Distribution									X			A3.01 - Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Breaker tripping	3.1	17
215005 APRM / LPRM									X			A3.02 - Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Full core display	3.5	18
223002 PCIS/Nuclear Steam Supply Shutoff										X		A4.04 - Ability to manually operate and/or monitor in the control room: System indicating lights and alarms	3.5	19
262002 UPS (AC/DC)										X		A4.01 - Ability to manually operate and/or monitor in the control room: Transfer from alternative source to preferred source	2.8	20
205000 Shutdown Cooling											X	2.4.21 - Emergency Procedures / Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.0	21
300000 Instrument Air											X	2.2.44 - Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	22

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
215003 IRM	X											K1.06 - Knowledge of the physical connections and/or cause- effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: APRM SCRAM signals: Plant-Specific	3.9	23
223002 PCIS/Nuclear Steam Supply Shutoff				X								K4.02 - Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Testability	2.7	24
262001 AC Electrical Distribution										X		A4.03 - Ability to manually operate and/or monitor in the control room: Local operation of breakers	3.2	25
212000 RPS		X										K2.01 - Knowledge of electrical power supplies to the following: RPS motor-generator sets	3.2	26
K/A Category Totals:	3	3	2	3	2	2	2	2/2	2	3	2/3	Group Point Total:	26/5	

Nine Mile Point Unit 2
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q #
241000 Reactor/Turbine Pressure Regulator								X				A2.23 - Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine high eccentricity	2.6	91
259001 Reactor Feedwater											X	2.4.30 - Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.	4.1	92
219000 RHR/LPCI: Torus/Pool Cooling Mode											X	2.4.47 - Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	4.2	93
239001 Main and Reheat Steam	X											K1.05 - Knowledge of the physical connections and/or cause- effect relationships between MAIN AND REHEAT STEAM SYSTEM and the following: Moisture separator reheaters: Plant-Specific	2.8	27
256000 Reactor Condensate		X										K2.01 - Knowledge of electrical power supplies to the following: System pumps	2.7	28
286000 Fire Protection			X									K3.03 - Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on following: Plant protection	3.6	29
241000 Reactor/Turbine Pressure Regulator				X								K4.10 - Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature(s) and/or interlocks which provide for the following: Turbine Shell Warming: EHC Only	2.5	30
245000 Main Turbine Generator and Auxiliary Systems					X							K5.02 - Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Turbine operation and limitations	2.8	31
214000 RPIS						X						K6.02 - Knowledge of the effect that a loss or malfunction of the following will have on the ROD POSITION INFORMATION SYSTEM : Position indication probe	2.7	32

Nine Mile Point Unit 2
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q #
215001 Traversing In-core Probe							X					A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the TRAVERSING IN-CORE PROBE controls including: Radiation levels: (Not-BWR1)	2.8	33
272000 Radiation Monitoring								X				A2.11 - Ability to (d) predict the impacts of the following on the RADIATION MONITORING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Leakage and/or breaks from contaminated systems to atmosphere or to other process systems	3.4	34
233000 Fuel Pool Cooling/Cleanup									X			A3.02 - Ability to monitor automatic operations of the FUEL POOL COOLING AND CLEAN-UP including: Pump trip(s)	2.6	35
290001 Secondary CTMT										X		A4.12 - Ability to manually operate and/or monitor in the control room: Reactor building differential pressure: Plant-Specific	3.3	36
202001 Recirculation											X	2.4.31 - Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures	4.2	37
201001 CRD Hydraulic											X	2.1.23 - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.3	38
K/A Category Totals:	1	1	1	1	1	1	1	1/1	1	1	2/2	Group Point Total:		12/3

Facility:	Nine Mile Point Unit 2	Date: 11/30/15				
Category	K/A #	Topic	RO		SRO-Only	
			IR	Q#	IR	Q#
1. Conduct of Operations	2.1.43	Ability to use procedures to determine the effects on reactivity of plant changes, such as RCS temperature, secondary plant, fuel depletion, etc.			4.3	94
	2.1.25	Ability to interpret reference materials, such as graphs, curves, tables, etc.			4.2	98
	2.1.13	Knowledge of facility requirements for controlling vital / controlled access.	2.5	66		
	2.1.28	Knowledge of the purpose and function of major system components and controls.	4.1	67		
	Subtotal			2		2
2. Equipment Control	2.2.38	Knowledge of conditions and limitations in the facility license.			4.5	95
	2.2.23	Ability to track Technical Specification limiting conditions for operations.			4.6	99
	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	4.6	68		
	2.2.22	(Equipment Control: Knowledge of limiting conditions for operations and safety limits.	4.0	69		
	2.2.12	Knowledge of surveillance procedures.	3.7	74		
	Subtotal			3		2
3. Radiation Control	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personell monitoring equipment, etc.			2.9	96
	2.3.13	Knowledge of Radiological Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.	3.4	70		
	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	71		

	Subtotal			2		1
4. Emergency Procedures / Plan	2.4.47	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.			4.2	97
	2.4.28	Knowledge of procedures relating to a security event.			4.1	100
	2.4.16	Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, AOP's and SAMG's.	3.5	72		
	2.4.28	Knowledge of procedures relating to a security event.	3.2	73		
	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.0	75		
	Subtotal			3		2
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
2/2	Question 30 241000 Reactor/Turbine Pressure Regulator K4.16 Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature(s) and/or interlocks which provide for the following: Reactor Cooldown	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. Randomly re-sampled KA 241000 K4.10 - Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature(s) and/or interlocks which provide for the following: Turbine Shell Warming: EHC Only
2/2	Question 36 209001 Secondary Containment A4.12 - Ability to manually operate and/or monitor in the control room: Surveillance testing: Plant-Specific	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. Randomly re-sampled KA 209001 A4.01 - Ability to manually operate and/or monitor in the control room: Reactor building differential pressure: Plant-Specific
2/1	Question 6 218000 Automatic Depressurization System K3.03 Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following: Automatic depressurization logic	Question concept overlaps with question #5 concept for the randomly sampled KA. Randomly re-sampled KA 203000 K3.04 - Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following: Adequate core cooling
2/1	Question 22 300000 Instrument Air 2.2.40 - Equipment Control: Ability to apply technical specifications for a system	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. Randomly re-sampled KA 300000 2.2.44 - Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.
2/2	Question 31 201004 RSCS K5.03 - Knowledge of the operational implications of the following concepts as they apply to ROD SEQUENCE Group notch control limits and rod density: BWR-4,5	System has been retired, unable to write a question for the randomly sampled KA. Randomly re-sampled KA 203000 K5.02 - Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): Core cooling methods

2/2	Question 37 202001 Recirculation 2.4.41 - Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. Randomly re-sampled KA 202001 2.4.31 – Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures
1/1	Question 54 295028 High Drywell Temperature/5 2.4.49 - Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. Randomly re-sampled KA 295028 2.4.1 – Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps
1/1	Question 55 295003 Partial or Complete Loss of AC / 6 2.2.3 - Equipment Control: (multi-unit license) Knowledge of the design, procedural, and operational differences between units	NMP2 is a standalone unit. Randomly re-sampled KA 295003 2.2.37 – Equipment Control: Ability to determine operability and/or availability of safety related equipment.
1/2	Question 65 295034 Secondary Containment Ventilation High Radiation / 9 EK1.01- Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Personnel protection	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. Randomly re-sampled KA 295034 EK1.02 – Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Radiation Releases.
3	Question 69 2.2.4 Suppression Pool High Water Temp. / 5 2.2.4 - Equipment Control: (multi-unit license) Knowledge of the design, procedural, and operational differences between units	NMP2 is a standalone unit. Randomly re-sampled KA 295003 2.2.22 – Equipment Control: Knowledge of limiting conditions for operations and safety limits.
2/2	Question 31 201004 RSCS K5.03 - Knowledge of the operational implications of the following concepts as	Randomly sampled system does not exist at this facility

	they apply to ROD SEQUENCE Group notch control limits and rod density: BWR-4,5	Randomly re-sampled KA 245000 K5.02 – Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Turbine operation and limitations
1/1	Question 80 295018 Partial or Total Loss of CCW / 8 2.2.39 - Equipment Control: Knowledge of less than one hour technical specification action statements for systems.	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. Knowledge of less than one hour technical specification action statements is RO level of knowledge per NUREG1021 ES401, attachment 2. Randomly re-sampled KA 295018 2.2.42– Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications
1/1	Question 81 295038 High Off-site Release Rate / 9 2.4.20 - Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes.	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. Randomly re-sampled KA 295038 2.4.8 – Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOP's.
2/1	Question 88 217000 RCIC 2.1.27 - Conduct of Operations: Knowledge of system purpose and / or function.	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. The question can be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location per NUREG1021 ES401, attachment 2. Randomly re-sampled KA 217000 2.1.23 – Ability to perform specific system and integrated plant procedures during all modes of plant operation.
2/1	Question 90 300000 Instrument Air 2.1.30 - Conduct of Operations: Ability to locate and operate components, including local controls.	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. The question can be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location per NUREG1021 ES401, attachment 2. Randomly re-sampled KA 300000 2.1.20 – Conduct of Operations: Ability to interpret and execute procedure steps.
3	Question 98 2.1.44 2.1.44 - Conduct of Operations: Knowledge of RO duties in the control room during fuel handling such as responding to alarms from the fuel	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA.

	handling area, communication with fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation.	Randomly re-sampled KA 2.1.25 – Ability to interpret reference materials, such as graphs, curves, tables, etc.
1/2	Question 89 262002 UPS (AC/DC) 2.4.46 - Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions.	An operationally relevant question at the appropriate license level could not be written for the randomly sampled KA. There is not enough of a direct tie between the UPS system and the Emergency Procedures / Plan. Randomly re-sampled KA 262002 2.2.40 – Ability to apply Technical Specifications for a system.

Facility: Nine Mile Point Unit 2
Examination Level: RO

Date of Examination: November 2015
Operating Test Number: LC2 14-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	P, D, R 2012 NRC	Perform Jet Pump Surveillance N2-OSP-LOG-D001
Conduct of Operations	D, R	Determine Core Thermal Power N2-REP-11, KA 2.1.45 (4.3)
Equipment Control	D, R	Describe How to Defeat HPCS Level 8 Interlock N2-EOP-6.20, HPCS Electrical Prints, KA 2.2.15 (3.9)
Radiation Control	N, R	Determine Radiological and Heat Stress Requirements - Valve leak in RWCU Pump Room RP-AA-460, RP-AA-203, KA 2.3.7 (3.5)
Emergency Procedures/Plan		

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

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

Facility: Nine Mile Point Unit 2	Date of Examination: November 2015
Examination Level: SRO	Operating Test Number: LC2 14-1

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D, R	Review Daily Checks Log – Jet Pumps N2-OSP-LOG-D001, N2-OP-29, KA 2.1.18 (3.8)
Conduct of Operations	P, R 2014 NRC	Determine Plant Impact for Inoperable Unit Cooler N2-OP-53E, Technical Specifications, KA 2.1.32 (4.0)
Equipment Control	D, R	Review Service Water Pump and Valve Operability Test N2-OSP-SWP-Q002, KA 2.2.12 (4.1)
Radiation Control	N, R	Determine Radiological and Heat Stress Requirements - Valve leak in RWCU Pump Room RP-AA-460, RP-AA-203, KA 2.3.7 (3.6)
Emergency Procedures/Plan	M, S	Post Scenario Event Classification EP-CE-111, KA 2.4.29 (4.4)

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

* Type Codes & Criteria:

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

Facility: Nine Mile Point Unit 2	Date of Examination: December 2015
Exam Level: RO/SRO-I	Operating Test No.: LC2 14-1 NRC

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

System / JPM Title	Type Code*	Safety Function
a. Refuel One Rod Out Interlock Test K/A 201002 A3.01 (3.2/3.1), N2-OSP-RMC-W@002	N, A, S, L	1
b. Restore SDC to Service K/A 205000 A4.01 (3.7/3.7) N2-OP-31, N2-SOP-31	M, A, S, L	4
c. Transfer RCIC Lineup Post-Scram for Pressure Control K/A 217000 A4.07 (3.9/3.8), N2-EOP-RPV, N2-OP-35	N, A, S, EN	2
d. Augment RPV Pressure Control Using MSL Drains (RO Only) K/A 239001 A4.02 (3.2/3.2) N2-EOP-6.27	D, S, L	3
e. Perform Operating Checks on a DBA H2 Recombiner K/A 223001 A4.13 (3.4/3.4) N2-OP-62	D, A, S, L	5
f. Energizing 2ENS*SWG103 from the Div II EDG & 2NNS-SWG015 from 2ENS*SWG103 K/A 262001 A4.01 (3.4/3.7) N2-SOP-3	P, D, S 2012 NRC	6
g. Temper Service Water Using Circ Water K/A 202002 A2.03 (2.9/3.0) N2-OP-11	D, A, S	8
h. Insert Substitute Rod Position Information in the RWM K/A 201006 A4.06 (3.2/3.2) N2-OP-95A	D, S	7

In-Plant Systems* (3 for RO); (3 for SRO-I)

System / JPM Title	Type Code*	Safety Function
i. Align Fire Water to RHS B K/A 203000, A2.02 (3.5/3.5) N2-EOP-6.6	P, D, E, R 2014 NRC	2
j. Transfer UPS-2A to maintenance and shutdown K/A 262002 A3.01 (2.8/3.1) N2-OP-71D	D, R	6
k. Recover Offgas after Automatic Shutdown K/A 271000, A.2.10 (3.1/3.3) N2-OP-42	D, R	9

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

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

Pairings:

A alone
B alone
C then D
E alone
F alone
G alone
H alone

Facility: Nine Mile Point Unit 2 Scenario No.: NRC-1 Op-Test No.: LC2 14-1

Examiners: ______________ Operators: ______________

Initial Conditions: The plant is operating at approximately 3% power with a reactor startup in progress. IRM 'D' is out of service and bypassed. EHC pump 'B' is out of service.

Turnover: Continue reactor startup and raise power to 10%

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R-ATC, SRO	The crew will assume the watch and continue the startup by withdrawing rods per N2-OP-101A. **N2-OP-101A**
2	RD07	C-ATC, SRO	Stuck Control Rod **N2-OP-30**
3	NM07	I-ATC, SRO TS-SRO	IRM downscale failure **ARPs, TS 3.3.1.1**
4	RC14	C-BOP, SRO TS-SRO	RCIC keep full pump trip **ARPs, TS 3.5.1**
5	IA02A IA04A IA04B	C-BOP, SRO	Trip of Instrument Air Compressor A **N2-SOP-19**
6	RH13A	C-BOP, SRO TS-SRO	Inadvertent initiation of Division 1 ECCS systems. **N2-OP-31, TS 3.5.1, TS 3.8.1**
7	MT01 FW08	C-ATC, SRO	Seismic Event causes FWLC failure. **N2-SOP-90, N2-SOP-6**
8	RC12	M (All)	RCIC Steam Leak in Reactor Building **N2-EOP-RPV, N2-EOP-SC**
9	Overrides	C-BOP, SRO	RCIC Isolation valves fail to close leading to degrading Secondary Containment conditions. **N2-EOP-SC, N2-EOP-C2**

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

Facility: **Nine Mile Point Unit 2** Scenario No.: **NRC-1**		Op-Test No.: **LC2 14-1**
1. Total malfunctions (5-8) **Events 2, 3, 4, 5, 6, 7, 8, 9**	8	
2. Malfunctions after EOP entry (1-2) **Event 9**	1	
3. Abnormal events (2-4) **Events 5, 7**	2	
4. Major transients (1-2) **Event 8**	1	
5. EOPs entered/requiring substantive actions (1-2) **N2-EOP-RPV, N2-EOP-SC**	2	
6. EOP contingencies requiring substantive actions (0-2) **N2-EOP-C2**	1	
7. Critical tasks (2-3)	3	

CRITICAL TASK DESCRIPTIONS:	**CRITICAL TASK JUSTIFICATION:**
CT-1.0: Given a trip of the running instrument air compressor and a failure of the lag and backup air compressors to automatically start, the crew will take action to manually start the lag or backup air compressor.	*This task is identified as critical because without operator action to start the lag or backup air compressor, instrument air header pressure will degrade until the reactor scrams due to low RPV level and/or loss of scram air header pressure.*
CT-2.0, Given secondary containment temperatures approaching a maximum safe value in one area, the crew will initiate a manual reactor scram IAW N2-EOP-RPV	*This task is identified as critical because without operator action to scram, the reactor will continue to provide energy to the RCIC steam line break and cause increased secondary containment temperatures and radiation levels.*
CT-3.0A Given secondary containment temperatures approaching or above maximum safe values in one area, the crew will open 5 main turbine bypass valves IAW N2-EOP-RPV	*This task is identified as critical because without operator action to depressurize the reactor, secondary containment integrity, the integrity of equipment located in the secondary containment, and continued safe operation of the plant cannot be assured. Note: The crew may choose to wait until two or more areas are above maximum safe values before depressurizing the reactor. If the crew chooses to depressurize the reactor via the SRVs, then CT-3.0A does not have to be evaluated.*
CT-3.0B Given secondary containment temperatures above maximum safe values in two areas, the crew will open 7 ADS valves IAW N2-EOP-C2	*This task is identified as critical because without operator action to depressurize the reactor, secondary containment integrity, the integrity of equipment located in the secondary containment, and continued safe operation of the plant cannot be assured. Note: The crew may choose to "anticipate blowdown" and depressurize the reactor to the main condenser. If the crew chooses to depressurize the reactor to the main condenser and are successful in preventing two areas from exceeding the maximum safe temperatures, then CT-3.0B does not have to be evaluated.*

SCENARIO SUMMARY

The crew will take the shift at ~3% power. The RO will raise power using rods. While withdrawing rods, a control rod will stick. The crew will take action to raise drive water pressure per N2-OP-30. Raising drive water pressure will free the stuck rod and allow the startup to continue. After power has been sufficiently raised, an IRM will fail downscale. The crew will respond per the ARPs and bypass the IRM. The CRS will declare the IRM inoperable and evaluate TS 3.3.1.1. Once the IRM is bypassed, the RCIC keep full pump will trip on motor electric fault followed shortly by a high point vent low alarm. The crew will respond per the ARPs and shut the RCIC trip throttle valve. The CRS will declare RCIC inoperable and evaluate TS 3.5.1.

Once the CRS has evaluated TS 3.5.1, an electrical fault will occur on Instrument Air Compressor A. Compressors B and C will fail to auto start. The crew will take action per N2-SOP-19 and manually start either Compressor B or C **(CRITICAL TASK)** to restore air header pressure.

After the loss of instrument air, an electrical failure will cause a spurious initiation of Division 1 ECCS systems. The crew will take action to shutdown the Division 1 ECCS systems. The CRS will evaluate TS 3.5.1 and 3.8.1. Following the inadvertent initiation of Division 1 ECCS systems, a seismic event occurs. The event will cause an unisolable RCIC steam leak and a FWLC failure. The crew will take action per N2-EOP-SC and enter N2-EOP-RPV to manually scram the reactor **(CRITICAL TASK)**. RPV level control will be complicated by the FWLC failure. Due to the RCIC steam leak, Secondary Containment conditions will continue to degrade requiring the crew to either anticipate RPV blowdown per N2-EOP-RPV, or perform a blowdown per N2-EOP-C2 **(CRITICAL TASK)**. The scenario may be terminated when the RPV is being depressurized.

Facility: Nine Mile Point Unit 2 Scenario No.: NRC-2 Op-Test No.: LC2 14-1

Examiners: ______________ Operators: ______________

Initial Conditions: The plant is operating at approximately 65% power with a plant shutdown in progress. IRM 'D' is out of service and bypassed. EHC pump 'B' is out of service.

Turnover: Place the "C" heater drain pump in recirculation mode IAW N2-OP-8, Section G.1. Lower power using recirculation flow to 58%.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N-BOP, SRO	Place heater drain pumps in recirculation mode. **N2-OP-101D, N2-OP-8 G.1.0**
2	N/A	R-ATC, SRO	Continue shutdown using Recirc flow **N2-OP-101C**
3	RD05	C-ATC, SRO TS-SRO	Control Rod drift. **ARPs, N2-SOP-8, T.S. 3.1.3**
4	CS01B	C-BOP, SRO TS-SRO	Inadvertent initiation of HPCS due to high drywell pressure. **N2-OP-33, T.S. 3.5.1**
5	CW09 CW26	C-BOP, SRO	Lowering intake level due to clogged strainer **N2-SOP-11**
6	FW03A FW03B	M-All	Loss of all high pressure feedwater pumps requiring scram **N2-SOP-06, N2-EOP-RPV**
7	RD12	C-ATC, SRO	Loss of RDS for injection. **N2-EOP-RPV, N2-SOP-101C**
8	RR20 RH14A	M-All	A LOCA occurs. The Division 1 ECCS system fails to automatically initiate. **N2-EOP-PC, N2-EOP-6**
9	RH10B	C-All	2RHS*MOV25B will stick shut, preventing drywell sprays. **N2-EOP-PC, N2-EOP-6, N2-EOP-C2**

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

Facility: **Nine Mile Point Unit 2** Scenario No.: **NRC-2**		Op-Test No.: **LC2 14-1**
1. Total malfunctions (5-8) **Events 3, 4, 5, 6, 7, 8, 9**	7	
2. Malfunctions after EOP entry (1-2) **Event 7, 8, 9**	3	
3. Abnormal events (2-4) **Events 3, 5, 6, 7**	4	
4. Major transients (1-2) **Event 6, and 8**	2	
5. EOPs entered/requiring substantive actions (1-2) **N2-EOP-RPV, N2-EOP-PC**	2	
6. EOP contingencies requiring substantive actions (0-2) **N2-EOP-C2**	1	
7. EOP Based Critical tasks (2-3)	2	
CRITICAL TASK DESCRIPTIONS:		**CRITICAL TASK JUSTIFICATION:**
CT-1.0: Given service water intake bay level less than 234 ft and a failure of 2SWP*MOV77A & 77B to automatically open, the crew will take action to manually open 2SWP*MOV77A & 77B per N2-SOP-11.		*This task is identified as critical because without operator action, the plant will lose its ultimate heat sink.*
CT- 2.0: Given a LOCA in the Drywell with a failure of Feedwater and CRD pumps, the crew will inject with preferred and alternate injection systems to restore and maintain RPV water level above -14 inches, in accordance with N2-EOP-RPV.		*This task is identified as critical because without operator action, adequate core cooling, through submergence, would be lost, which would result in damage to fuel cladding.*
CT- 3.0: Given Suppression Chamber Pressure unable to be restored and maintained within the Pressure Suppression Limit, the crew will open 7 SRV's IAW N2-EOP-C2.		*This task is identified as critical because without operator action, the primary containment pressure suppression function would continue to degrade and would not be able to accept a full blowdown of the reactor.*

SCENARIO SUMMARY

The plant is operating at ~65% power and is the process of shutting down for a scheduled refueling outage. The crew will place the "C" heater drain pump in recirculation mode. The crew will then continue the shutdown by lowering recirculation flow. After a significant power reduction a control rod will drift out of the core. The SRO will direct the control rod be inserted and disarmed then enter Technical Specification 3.1.3.

Once the control rod drift is addressed the crew must respond to an inadvertent HPCS initiation that will require securing HPCS and placing it in pull-to lock. The SRO reviews T.S. 3.5.1 for HPCS being inoperable. After these Technical Specifications are determined, Service Water intake clogging will occur causing Service Water intake bay level to lower. The crew will take action per N2-SOP-11 and attempt to clean the traveling screens. Intake bay will continue to lower to 234 feet. The intake bay bypass valves 2SWP*MOV77A/B will fail to automatically open requiring the crew to take manual action to open the valves **(CRITICAL TASK)**. Once MOV77A and B are open, intake bay level will recover.

Once Service Water intake bay level is restored, a loss of all feed pumps will occur. The loss will require the crew to place the Mode Switch in shutdown. A LOCA will occur. RPV level will be controlled using alternate level control systems in accordance with N2-EOP-RPV **(CRITICAL TASK)**. The LOCA will also cause Primary Containment (PC) parameters to degrade and the crew will enter N2-EOP-PC to stabilize PC parameters. Malfunctions in the Division 1 RHS systems will prevent RHS A from being used for primary containment control and the crew will be required to use RHS B to spray the suppression chamber. As PC conditions continue to degrade, the crew will attempt to spray the drywell using RHS B. While the crew is attempting to align drywell sprays, 2RHS*MOV25B (Drywell Spray Valve) will stick shut. Plant Operators will be dispatched in an attempt to manually open MOV25B. While the POs are attempting to manually open MOV25B, primary containment parameters will continue to degrade. The CRS will determine that Suppression Chamber Pressure cannot be restored and maintained within the Pressure Suppression Limit and will enter N2-EOP-C2 and direct 7 ADS valve be opened. The crew will open 7 SRV's and blowdown the reactor **(CRITICAL TASK)**. The scenario may be terminated once 7 SRV's are opened.

Facility: Nine Mile Point Unit 2 Scenario No.: NRC-3 Op-Test No.: LC2 14-1

Examiners: ______________ Operators: ______________

Initial Conditions: The plant is operating at approximately 90% power. IRM 'D' is out of service and bypassed. EHC pump 'B' is out of service.

Turnover: Perform a live Bus Transfer of 2NNS-SWG013 to 2NNS-SWG012 per N2-OP-71B, Sect H.6.0. Then, raise power to 95% using recirculation flow.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N-BOP, SRO	Perform a live Bus Transfer of 2NNS-SWG013 to 2NNS-SWG012 **N2-OP-71B, Sect. H.6.0**
2	N/A	R-ATC, SRO	The crew will raise reactor power to 95% using recirculation flow. **N2-OP-101D**
3	NM19A	I-BOP, SRO, TS-SRO	RBM "A" Inop requires bypassing **ARPs, N2-OP-92, Tech Spec 3.3.2.1**
4	RR10A	C-ATC, SRO, TS-SRO	Recirculation Pump "A" trips. **N2-SOP-29, Tech Specs 3.4.1**
5	MC01	C-All	Loss of condenser vacuum requires a reactor scram **N2-SOP-9, N2-SOP-101C**
6	RD17Z	M-All	Failure of Control Rods to Insert During the Scram **N2-EOP-RPV, N2-EOP-C5**
7	MS04	M-ATC, SRO	Steam Leak in the Drywell **N2-EOP-PC, N2-EOP-6**
8	RR27	C-All	Loss of Reactor Water Level Indication **N2-EOP-C4**

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

Facility: **Nine Mile Point Unit 2** Scenario No.: **NRC-3**		Op-Test No.: **LC2 14-1**
1. Total malfunctions (5-8) **Events 3, 4, 5, 6, 7, 8**	6	
2. Malfunctions after EOP entry (1-2) **Event 7, 8,**	2	
3. Abnormal events (2-4) **Events 4, 5**	2	
4. Major transients (1-2) **Events 6, 7**	2	
5. EOPs entered/requiring substantive actions (1-2) **N2-EOP-RPV, N2-EOP-PC**	2	
6. EOP contingencies requiring substantive actions (0-2) **N2-EOP-C5**	1	
7. EOP Based Critical tasks (2-3)	2	

CRITICAL TASK DESCRIPTIONS:	**CRITICAL TASK JUSTIFICATION:**
CT-1.0: Given a failure of the reactor to scram with power <4%, the crew will ensure the reactor will remain shutdown without boron in accordance with N2-EOP-C5	*This task is identified as critical because without operator action, the plant continuing to produce power will present challenges to the plant during a failure to scram. Inserting control rods or injecting boron will lower power. Inserting control rods will ultimately provide stable, long-term core shutdown conditions.*
CT- 2.0: Given the plant with water level unknown, execute N2-EOP-C4, RPV Flooding, in accordance with N2-EOP-RPV	*This task is identified as critical because without operator action, adequate core cooling cannot be assured.*

SCENARIO SUMMARY

The plant is operating at ~90% power. The crew will perform a live bus transfer of 2NNS-SWG013 to 2NNS-SWG012 per OP-71B, Sect H.6.0.

Once the bus transfer is completed the crew will take the shift and raise reactor power to 95% using recirculation flow. After the reactivity maneuver, the "A" RBM will fail inop requiring bypassing the RBM and entering Tech Spec 3.3.2.1.

After these T.S. are addressed, the A Recirculation Pump will trip the Crew will enter N2-SOP-29 and take actions including inserting 4 cram rods. The SRO will enter Technical Specifications 3.4.1 for single loop operation.

After the Tech Specs are addressed a condenser vacuum leak will occur and efforts made IAW N2-SOP-09 will not be successful. When the reactor is scrammed several groups of control rods will fail to fully insert and reactor power drops below 4%. The SRO will direct the RO to insert the control rods IAW N2-EOP-06, Attachment 14 (**CRITICAL TASK**). When the reactor scrams a steam leak will occur inside the drywell. The steam leak will result in rising Drywell temperature and pressure. When drywell temperature rises, all RPV water level indication will fail upscale. RPV Flooding is required and the RPV will have to be depressurized to flooding the RPV to the Main Steam Lines **(CRITICAL TASK**).

When Suppression Chamber pressure exceeds 10 psig, the crew will spray the Drywell with RHR to stay below Pressure Suppression Pressure Limit.

Facility: Nine Mile Point Unit 2 Scenario No.: NRC-4 Op-Test No.: LC2 14-1

Examiners: ______________________ Operators: ______________________

Initial Conditions: The plant is operating at approximately 95% power. IRM 'D' is out of service and bypassed. EHC pump 'B' is out of service.

Turnover: The crew will start RHR Loop C in full flow test mode. Then raise reactor power to 100% using recirc flow.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N-BOP, SRO	Start RHR in full flow test mode **N2-OP-31**
2	N/A	R-ATC, SRO	Increase power to 100%
3	RR07B	C-ATC, SRO	APRM Flow signal fails Hi **N2-SOP-97 Tech Spec 3.3.2**
4	NM11D	TS-SRO	APRM fails Hi – same RPS channel **Tech Spec 3.3.1**
5	CW01A CW10E	C-BOP, SRO TS-SRO	Service Water Pump 1A trips. While starting the standby pump, the associated discharge valve will fail to automatically open requiring the operator to manually open the valve. The CRS will declare the pump inoperable and evaluate TS 3.7.1. **ARP's, N2-OP-11, TS 3.7.1**
6	AD05B	C-BOP, SRO	One Non-ADS SRV fails open **N2-SOP-34**
7	PC12	M-All	Suppression Pool rupture results in loss of inventory in the suppression pool, requires scram and eventual blowdown. **N2-EOP-RPV N2-EOP-PC**
8	AD08A AD08C	C-ATC, SRO	Failure of the ADS pushbuttons to actuate all 7 ADS valves. **N2-EOP-C2**

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

Facility: **Nine Mile Point Unit 2** Scenario No.: **NRC-4**		Op-Test No.: **LC2 14-1**
1. Total malfunctions (5-8) **Events 3, 4, 5, 6, 7, 8**	6	
2. Malfunctions after EOP entry (1-2) **Event 8**	1	
3. Abnormal events (2-4) **Events 3, 6**	2	
4. Major transients (1-2) **Event 7**	1	
5. EOPs entered/requiring substantive actions (1-2) **N2-EOP-RPV, N2-EOP-PC**	2	
6. EOP contingencies requiring substantive actions (0-2) **N2-EOP-C2**	1	
7. EOP Based Critical tasks (2-3)	2	
CRITICAL TASK DESCRIPTIONS:		**CRITICAL TASK JUSTIFICATION:**
CT-1.0: Given an un-isolable leak in the suppression pool, exceeding makeup capacity, the crew will scram the reactor in accordance with N2-EOP-PC.		*This task is identified as critical because lowering suppression pool water level challenges the pressure suppression function of the Primary Containment. Continued Reactor operation is not allowed with an inoperable Primary Containment. A Reactor scram also allows subsequent mitigating actions, such as Reactor cooldown and/or blowdown.*
CT- 2.0 Given a lowering suppression pool level, the crew will enter and execute N2-EOP-C2, RPV Blowdown, prior to suppression pool level reaching 192 feet.		*This task is identified as critical because without operator action to blowdown the RPV prior level reaching 192 feet, the primary containment pressure limit could be exceeded due to a loss of pressure suppression capability concurrent with pressure control via SRVs.*

SCENARIO SUMMARY

The scenario begins at about 95% power. IRM 'D' is out of service and bypassed. EHC Pump B is out of service for maintenance. The crew will be required to start RHR Loop C in full flow test mode per N2-OP-31 Section H.14.0. After the evolution, power must be raised to 100%.

One APRM flow signal will fail high requiring actions to bypass the APRM and the SRO will address TS. Another APRM will then fail requiring an entry to a TS LCO. Once TS are addressed, Service Water Pump 1A will trip on motor electric fault. The crew will take action to start a standby service water pump per N2-OP-11. When starting the standby pump, the discharge valve will fail to automatically open requiring the crew to manually open the valve. The SRO will evaluate tech spec 3.7.1.

After these T.S. are addressed one Non-ADS SRV will fail open placing the plant in an uncontrolled cooldown situation. The crew will take the required actions per N2-SOP-34 and close the valve. The pressure transient caused by this stuck open SRV will have impacted the suppression pool causing a suppression pool leak. The crew will take action and attempt to refill the suppression pool The leak will cause flooding alarms in the RB requiring entry into N2-EOP-SC. The lowering suppression pool level will require the crew to enter N2-EOP-RPV and scram the reactor (**CRITICAL TASK**), and N2-EOP-C2 and blowdown the reactor, **(CRITICAL TASK)**. The blowdown will be complicated by a failure of 2 of the 7 ADS valves to open and the crew will be required to open 2 additional SRVs.

Facility: Nine Mile Point Unit 2 Scenario No.: NRC-5 Op-Test No.: LC2 14-1

Examiners: ______________________ Operators: ______________________

Initial Conditions: The plant is operating at approximately 92% power. A momentary loss of power signal caused lockup of LV-10A last shift. IRM 'D' is out of service and bypassed. EHC pump 'B' is out of service.

Turnover: Reset LV-10A Lockup IAW N2-SOP-06, Attachment 1. After the valve has been reset and is back in automatic, raise reactor power to 100% using recirc flow.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N-BOP, SRO	Reset a LV-10A Lockup and place the valve back in automatic **N2-SOP-6**
2	N/A	R-ATC, SRO	Raise reactor power to 100% **N2-OP-101D**
3	CU08	I-BOP, SRO TS-SRO	RWCU fails to automatically isolate on RWCU flow mismatch caused by cleanup RWCU non-regen heat exchanger tube leak. **ARPs, T.S. 3.3.6.1**
4	RD18	C-ATC, SRO TS-SRO	CRD P1A suction filter clog causes pump trip. After the pump is restarted, a Control Rod Drive Accumulator will fail to recharge. **N2-SOP-30, T.S. 3.1.5**
5	ED04D	C-BOP, SRO	Loss of NNS-SWG014 switchgear. Restore CRD, and other lost loads. **N2-SOP-03, SOP-30. SOP-68, SOP-19, SOP-97**
6	Override s MS13	M-All	EHC Regulator slow failure causes Reactor Pressure to lower, crew must scram and shut MSIVs **N2-SOP-23, N2-SOP-101C, N2-EOP-RPV**
7	ED02A (B) MS03	C-All	Loss of all offsite power with Div II EDG failing to start. Crew will manually start Div II EDG. **N2-SOP-03**
8	RR20	M-All	LOCA in Drywell **N2-EOP-RPV, N2-EOP-PC**

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

Facility: **Nine Mile Point Unit 2** Scenario No.: **NRC-5**		Op-Test No.: **LC2 14-1**
1. Total malfunctions (5-8) **Events 3, 4, 5, 6, 7, 8**	6	
2. Malfunctions after EOP entry (1-2) **Events 7, 8**	2	
3. Abnormal events (2-4) **Events 4, 5, 6, 7**	4	
4. Major transients (1-2) **Event 8**	1	
5. EOPs entered/requiring substantive actions (1-2) **N2-EOP-RPV, N2-EOP-PC**	2	
6. EOP contingencies requiring substantive actions (0-2)	0	
7. EOP Based Critical tasks (2-3)	2	
CRITICAL TASK DESCRIPTIONS:		**CRITICAL TASK JUSTIFICATION:**
CT-1.0: Given a lowering RPV pressure, the crew will; (1) scram the reactor before exceeding a reactor core safety limit and (2) close the appropriate number of MSIVs before exceeding the allowable cooldown rate IAW N2-EOP-RPV.		*This task was identified as critical because without operator action to manually scram the reactor, as pressure lowers below 785 psig the reactor would be violating a safety limit. Additionally, as pressure continues to lower, operator action is needed to shut an appropriate number of MSIVs to prevent exceeding the 100°F cooldown rate.*
CT-2.0: Given a steam leak in the drywell, the crew will spray the drywell prior to exceeding the PSP limit IAW N2-EOP-6, Attachment 22		*This task is identified as critical because without operator action to spray the drywell, PSP would be exceeded which would limit the ability of the primary containment to accept a full reactor blowdown.*

SCENARIO SUMMARY

The plant is operating at ~92% power with IRM 'D' out of service and bypassed. EHC Pump B is also out of service for maintenance. Feed Reg Valve LV-10A locked. The valve locked up last shift when a momentary loss of signal occurred. Procedure N2-SOP-06, Feedwater Failures, was entered; power lowered to 92% and RPV level was stabilized. The valve is ready to be reset IAW N2-SOP-06, Attachment 1. The crew is directed to reset Feed Reg Valve LV-10A and place the valve back in automatic.

Once the Feed Reg Valve is back in automatic, the crew will restore Reactor power to 100% using Recirc Flow. After power has been raised, a heat exchanger tube leak in Reactor Water Cleanup will result in a high differential flow. The expected automatic isolation will fail. The BOP is expected to recognize the failure and manually isolate the system IAW associated ARPs. The SRO is expected to refer to Tech Specs for the instrument/isolation failure.

Next, the CRD P1A suction filter will clog causing a trip of the CRD pump and low pressure alarms on three accumulators. When the standby CRD suction filter is placed in service and a CRD pump is started the low pressure on one accumulator will NOT clear. Report from the field indicates accumulator pressure at 910 psig.. The SRO must enter T.S. 3.1.5 for the inoperable accumulator.

After the Technical Specifications are addressed, a loss of offsite power to SWG014 Switchgear occurs when breaker 14-2 fails open, the crew will take action per N2-SOP-3, N2-SOP-19, N2-SOP-30, N2-OP-60, and N2-SOP-97 to stabilize the plant.

Once the loss of switchgear is addressed a malfunction in the EHC pressure regulator system causes a slow reduction in reactor pressure. The crew will manually scram the reactor **(CRITICAL TASK)** unless plant parameters cause an automatic scram. As reactor pressure lowers the MSIVs will fail to automatically isolate. The operators must diagnose the failure of the MSIVs to isolate and manually close the MSIVs to stabilize reactor pressure **(CRITICAL TASK)** and execute N2-EOP-RPV.

Following the scram a loss of all off-site power will occur. The crew will be in a blackout condition because the Div II Diesel will fail to automatically start. The crew must manually start the diesel **(CRITICAL TASK)**. Then a steam leak will occur in the drywell raising drywell pressure and requiring suppression pool sprays. As drywell and suppression pool pressure continue to rise the crew must initiate drywell sprays to mitigate the rising drywell pressure **(CRITICAL TASK)**.

ADAMS MASTER EXAM FILE PACKAGE

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Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K1.04
	Importance Rating	2.8

Instrument Air

Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: Cooling water to compressor

Proposed Question: #1

The plant is operating at rated conditions with the following:

- A complete loss of CCP Mini-Loop cooling water to the Instrument Air system occurs.

What is the impact on the Instrument Air Compressors?

	The operating Compressor trips on...	Backup Compressors...
A.	low cooling water flow.	are blocked from starting.
B.	high outlet air temperature.	are blocked from starting.
C.	low cooling water flow.	start and eventually trip.
D.	high outlet air temperature.	start and eventually trip.

Proposed Answer: D

Explanation: The compressors trip on high outlet air temperature. There is no interlock to prevent the lagging and backup compressors from starting. The backup air compressors will start and eventually also trip on high outlet air temperature.

A. Plausible– There is no low cooling water flow interlock. Plausible because there is a low cooling water flow alarm. The backup air compressors will start and eventually also trip on high outlet air temperature.
B. Plausible – The backup air compressors will start and eventually also trip on high outlet air temperature.
C. Plausible – There is no low cooling water flow interlock. Plausible because there is a low cooling water flow alarm.

Technical Reference(s): ARP 851259, Vision Objective #73379

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-278001C01 RBO-5

Question Source: Bank 2010 NRC 26

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K1.04
	Importance Rating	2.6

DC Electrical Distribution

Knowledge of the physical connections and/or cause- effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: Ground detection

Proposed Question: #2

The plant is operating at rated conditions with the following:

- Annunciator 852108, DIV 1 EMER BUS BYS 002A 125 VDC SYSTEM TROUBLE alarms
- Computer point BYSIC05, 125VDC DIV 1 BAT 2A GROUND, caused the alarm
- The operator taking grounds on the bus places the selector switch to POS then to NEG with the following results:
  - POS 85 Volts
  - NEG 25 Volts

Which ground indication(s), if any, require(s) action in accordance with N2-OP-74A?

A. Positive Bus Only

B. Negative Bus Only

C. Both Positive and Negative Busses

D. Neither Positive or Negative Busses

Proposed Answer: A

Explanation: Annunciator 852108 alarms due to a battery ground at +/- 75 Volts. The associated ARP directs the operator to N2-OP-74A, section H.14. Per section H.14, a value of 75VDC or greater for either positive or negative ground readings indicates a ground exists and action is required.

B. Plausible – There is a negative ground indication. However, it is below the threshold which requires action in accordance with N2-OP-74A.
C. Plausible – Positive grounds do require action. However, the negative grounds do not.
D. Plausible – No action is required for the negative ground indication. However, positive grounds are above the threshold.

Technical Reference(s): N2-ARP-852108, N2-OP-74A

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-263000C01, RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 K2.01
	Importance Rating	2.6

Source Range Monitor

Knowledge of electrical power supplies to the following: SRM channels/detectors

Proposed Question: #3

The plant is in Mode 4 when 600V distribution panel 2NJS-PNL600 is de-energized.

Which one of the following is the affect of this loss on the Source Range Monitors (SRM)?

A. SRM Channels A and C de-energize immediately.

B. SRM Channels A and C de-energize after several hours.

C. SRM Channels B and D de-energize immediately.

D. SRM Channels B and D de-energize after several hours.

Proposed Answer: D

Explanation: 2NJS-PNL600 powers chargers 2BWSCHGR3B1 and 2BWS-CHGR3D1. Although both chargers are lost, the SRMs remain energized via 2BWS-BAT3B/3D. These batteries are sized to maintain voltage for up to 4 hours before voltage begins to decay. If the chargers are not restored, SRMs will eventually de-energize.

A. Plausible – 2NJS-PNL600 supplies chargers which feed 2 SRM channels. However, the chargers associated 2BWS-BAT3B/3D supply SRMs B and D. SRM A and C are supplied by BWS-PNL300A, which is fed by 2NJS-PNL-500. Plausible if the candidate does not understand the AC and 24/48 VDC system configuration.
B. Plausible – 2NJS-PNL600 supplies chargers which feed 2 SRM channels. However, the chargers associated 2BWS-BAT3B/3D supply SRMs B and D. SRM A and C are supplied by BWS-PNL300A, which is fed by 2NJS-PNL-500. Plausible if the candidate does not understand the AC and 24/48 VDC system configuration.
C. Plausible – SRM B and D are affected but do not immediately de-energize because on the loss of power to the 24/48 VDC battery chargers, 2BWS-BAT3B/3D supply power and are sized to maintain voltage for 4 hours. Plausible in that if the battery chargers are not available, the SRMs would immediately de-energize.

Technical Reference(s): N2-OP-73B, 24/48 VOLT D.C. DISTRIBUTION system description, page 2. N2-ELU-01, Attachment 92 (page 3 of 5) and EE-MO1F.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-215002C01, Source Range & Intermediate Range Monitoring Systems, RBO-04, System & Component Power Supplies

Question Source: Bank

Question History: 2010 NRC #4

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K2.03
	Importance Rating	2.9

LPCS

Knowledge of electrical power supplies to the following: Initiation logic

Proposed Question: #4

The plant is operating at rated conditions with the following:

- A small steam line break in the drywell occurs
- The mode switch is placed in shutdown due to rising drywell pressure
- Simultaneous with the event, a power supply failure occurs that results in the following annunciator:
  - 601401, "DIVISION I LPCS SYSTEM INOPERABLE"
- When drywell pressure reaches 1.68 psig, Division 1 low pressure ECCS fails to initiate
- LPCI A / LPCS MANUAL INITIATION white initiation light remains extinguished
- RHR 'A' is needed for suppression chamber sprays
- 2RHS*P1A is manually started from 2CEC*PNL601 and placed in suppression chamber spray

Which of the following describes the potential source of power that was lost AND with the loss of power, if LPCS injection were to be needed, how can the LPCS system be lined up to inject to the RPV?

	Power source	Method to lineup the LPCS system for RPV injection
A.	2BYS*PNL201A	Start 2CSL*P1, PMP 1 by placing the pump control switch to start and manually open 2CSL*MOV104, PMP 1 INJECTION VLV from 2CEC*PNL601
B.	2BYS*PNL201B	Manually initiate Division 1 Low Pressure ECCS by arming and depressing the LPCI A / LPCS MANUAL INITIATION pushbutton on 2CEC*PNL601 and verify system response
C.	2BYS*PNL201A	Manually initiate Division 1 Low Pressure ECCS by arming and depressing the LPCI A / LPCS MANUAL INITIATION pushbutton on 2CEC*PNL601 and verify system response
D.	2BYS*PNL201B	Start 2CSL*P1, PMP 1 by placing the pump control switch to start and manually open 2CSL*MOV104, PMP 1 INJECTION VLV from 2CEC*PNL601

Proposed Answer: A

Explanation: For the first part of the answer, 2BYS*PNL201A (DC Power) supplies power to LPCS system initiation logic. With 2BYS*PNL201A de-energized, no automatic or manual initiation of LPCS will occur. See N2-SOP-04, Attachment 2, section 2BYS*PNL201A:

ATTACHMENT 2
LOADS AFFECTED BY LOSS OF 2BYS*SWG002A

2BYS*PNL201A

Page 7 of 10

CUB/ FUSE	LOAD	CKT #	ESK/VENDOR PRINT	ACTION ON LOSS OF POWER
F1	Div I Annunciators	2IHAA01	10IHA10	Loss of Division I Annunciators
F2	CSL System	2CSLN07	807E171TY SH2	Loss of CSL System Auto/Manual Initiations. Annunciator 601401, DIVISION I LPCS SYSTEM INOPERABLE, in alarm

For the second part of the answer, per N2-SOP-04 attachment 2, manual initiation is not available, so the LPCS system would have to be aligned manually from control room panel 601.

B. Plausible – Both the first and second part of the distracter are incorrect, but plausible. For the first part of the distracter, 2BYS*PNL201B supplies DC power which provides power to Div II RHS Auto/Manual Initiations. The candidate could think that logic power for the initiation logic for CSL comes from DC power from 2BYS*PNL201B since CSL is not a residual heat removal system and does not follow the divisional power assignments. For the second part of the distracter, per N2-SOP-04 attachment 2 (above) with 2BYS*PNL201A lost, both auto and manual initiation is lost. If the candidate does not recognize that the logic power is DC power this choice could be plausible.
C. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter, per N2-SOP-04 attachment 2 (above) with 2BYS*PNL201A lost, both auto and manual initiation is lost. If the candidate does not recognize that the logic power is DC power this choice could be plausible.
D. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part of the distracter, 2BYS*PNL201B supplies DC power which provides power to Div II RHS Auto/Manual Initiations. The candidate could think that logic power for the initiation logic for CSL comes from DC power from 2BYS*PNL201B since CSL is not a residual heat removal system and does not follow the divisional power assignments.

Technical Reference(s): N2-SOP-04, attachment 2, "LOADS AFFECTED BY LOSS OF 2BYS*SWG002" (2BYS*PNL201A) page 7 of 10, N2-OP-32, section F.3.0.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-209001C01, RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K3.01
	Importance Rating	4.4

Automatic Depressurization System / 3

Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Restoration of reactor water level after a break that does not depressurize the reactor when required

Proposed Question: #5

The plant is operating at rated conditions with the following:

- A fault occurs on Division I 125 VDC Bus 2BYS*SWG002A
- A coolant leak in Drywell causes Drywell pressure to slowly rise to 8 psig
- Both Feedwater Pumps trip during the scram transient and none can be started
- RPV Pressure is 950 psig and slowly lowering
- RPV water level is 20 inches and lowering
- HPCS Initiation Seal In white light is OFF and pump is NOT running
- RCIC Initiation Seal In light is OFF and system is NOT running

Which one of the following actions is required to prevent RPV water level from lowering to the Top of Active Fuel (TAF) in accordance with N2-EOP-RPV?

A. Manually initiate RCIC and recover level using RCIC injection.

B. Manually initiate HPCS and recover level using HPCS injection.

C. Lower RPV pressure to 500 to 600 psig using SRV keylock switches on 2CEC*PNL601, then recover level with Condensate Booster Pump injection.

D. WAIT until level drops to 17.8 inches, then open 7 ADS valves at 2CEC*PNL601 or 2CEC*PNL628 and recover level using Low Pressure ECCS injection.

Proposed Answer: B

Explanation:

There are no conditions provided in the question that would prevent a successful HPCS injection, except that the pump is not running. The pump can be started, which will inject immediately and prevent level from lowering to TAF. The loss/malfunction of ADS is the loss of power to ADS logic and solenoids for Division I.

Loss or malfunction of this system will have the following effects on the following parameters: Restoration of reactor water level after a break that does not depressurize the reactor when required. With the RPV pressurized water level control would be completely dependent on high pressure injection sources such as Feedwater, CSH, RCIC, and RDS. If the MSIVs are open, or can be opened, consideration can also be given to depressurizing using the bypass valves.

A. is incorrect because with loss of 2BYS*SWG002A (Division I 125VDC) RCIC logic power and power to operate components required to realign for injection is not available. If power was available, this would be a successful strategy for recovering water level.

C. is incorrect because with loss of 2BYS*SWG002A (Division I 125VDC) SRV "C" solenoids do not have electrical power to open the SRVs to lower pressure. If power was available, this would be a successful strategy for recovering water level.

D. is incorrect because the center (alternate level control) leg of N2-EOP-RPV does not direct an RPV Blowdown until level reaches -14 inches (TAF), not 17.8 inches. With loss of 2BYS*SWG002A (Division I 125VDC) Division I ADS logic power and "A" solenoids do not have electrical power to open the SRVs from either P601 or P628. Relying on the low pressure injection systems under these conditions will not prevent level from reaching -14 inches (TAF).

Technical Reference(s): N2-EOP-RPV flowchart, N2-SOP-04 Attachment 2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-218000C01, RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	203000 K3.04
Importance Rating	4.2

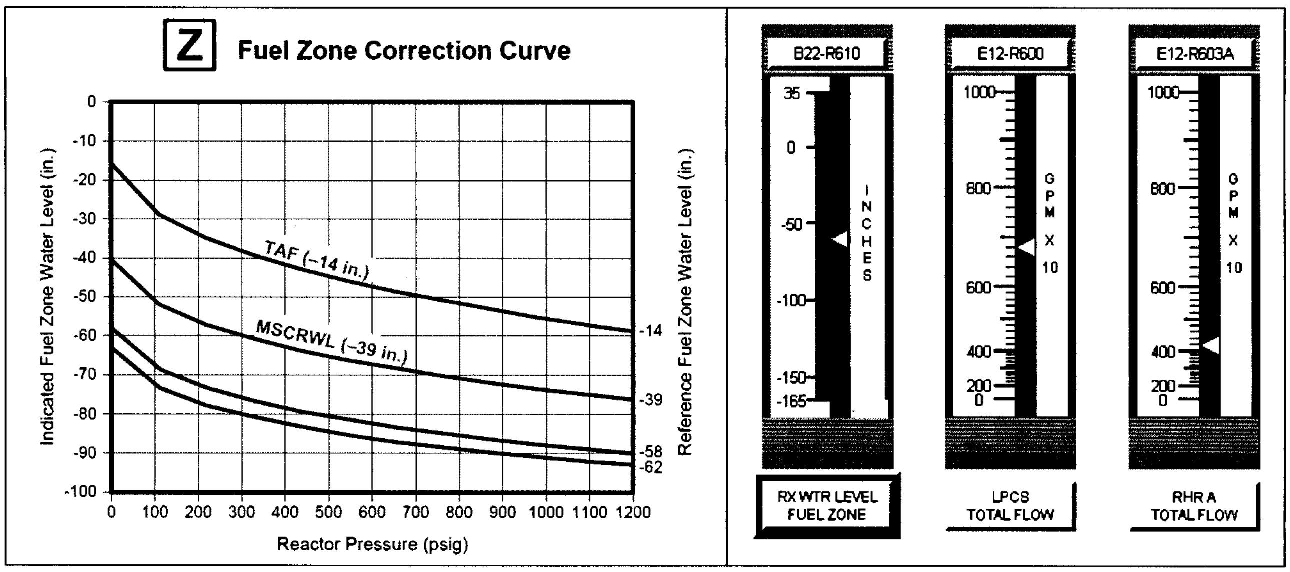
RHR/LPCI: Injection Mode

Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on following: Adequate core cooling

Proposed Question: #6

A LOCA is in progress with the following:

- The reactor is scrammed and all control rods are inserted
- RCIC and Division 2 low pressure ECCS pumps failed to initiate and cannot be started manually
- All Division 1 low pressure ECCS pumps have responded as designed
- RPV pressure is 100 psig following an RPV blowdown.
- No other systems are available to inject



Using the given fuel zone correction curve and control room panel indications, which one of the following describes the current method and mechanism of adequate core cooling?

	Method	Mechanism
A.	Core Spray Cooling	Boiling heat transfer only
B.	Core Spray Cooling	Submergence and spray flow
C.	Steam Cooling	Boiling heat transfer only

D.	Steam Cooling	Submergence and spray flow

Proposed Answer: B

Explanation: Per NER-2M-039, Rev 8.0, section 2.0 Definitions; Adequate Core Cooling is heat removal from the reactor sufficient to prevent rupturing the fuel clad. Three viable mechanisms for establishing adequate core cooling are used in the EOPs—core submergence, core spray cooling, and steam cooling.

Submergence is the preferred method of core cooling and exists when RPV water level is above the top of the active fuel (TAF). In this state heat removal is by boiling heat transfer.

Core Spray Cooling is provided when design core spray flow requirements are satisfied (at least design core spray flow from either HPCS or LPCS) and RPV water level is at or above the elevation of the jet pump suctions. ***The covered portion of the core is then cooled by submergence while the uncovered portion is cooled by the spray flow.***

Steam cooling is the third method of core cooling and exists when RPV water level is below TAF and steam flow production in the covered portion of the fuel bundle is sufficient to cool the uncovered portion of the fuel bundle.

Steam cooling is relied upon only if RPV water level cannot be restored and maintained above TAF, cannot be determined, or must be intentionally lowered below TAF. The core is adequately cooled by steam if the steam flow across the uncovered length of each fuel bundle is sufficient to maintain the hottest peak clad temperature below the appropriate limiting value—1500°F if makeup can be injected, 1800°F if makeup cannot be injected. The covered portion of the core remains cooled by boiling heat transfer and generates the steam which cools the uncovered portion.

When RPV water level cannot be restored and maintained above TAF in EOP-RPV and when RPV water level is intentionally lowered to TAF in EOP-C5, ***adequate steam flow is established by maintaining RPV water level above the Minimum Steam Cooling RPV Water Level (MSCRWL).***

In this question, RPV level can be read on the fuel zone level indicator as -60 inches, Using the fuel zone level correction curve with reactor pressure given at 100 psig, actual RPV level is between -39 and -58 inches, but above -62 inches. This is below the Minimum Steam Cooling Water Level. LPCS spray flow can be read on E12-R600 meter as 6800 gpm which meets N2-EOP-RPV step L-17 requirements for adequate core cooling:

⚠ **Operating ECCS or RCIC with suppression pool water level below El. 195 ft may cause system damage.**

Restore and maintain RPV water level above -14 in. (Fig Z) using Preferred Injection Systems (Detail E1).

➨ OK to augment with Alternate Injection Systems if needed (Detail E2).

IF	THEN
You cannot restore and maintain RPV water level above -39 in. (Fig Z) with Preferred Injection Systems	Restore and maintain RPV water level at or above -62 in. (Fig Z) with at least 6350 gpm Core Spray loop flow (Core Spray Cooling).
You cannot restore and maintain Core Spray Cooling	1 Restore and maintain RPV water level above -39 in. (Fig Z) with <u>all</u> available injection sources. 2. IF...... you <u>still</u> cannot restore and maintain level above -39 in. (Fig Z). THEN... **FLOOD THE DRYWELL:** Exit <u>all</u> EOPs → Enter <u>all</u> SAPs

L-17

A. Plausible – Core spray cooling is correct for the method of adequate core cooling, the candidate could mistakenly conclude that since some degree of boiling is still occurring in the vessel that this answer choice is correct.
C. Plausible – The candidate could think that steam cooling is occurring with the given conditions and forget that the requirement is to be above MSCRWL in order for this method

to be employed. Also the candidate could mistakenly conclude that since some degree of boiling is still occurring in the vessel that this answer choice is correct.

D. Plausible – The candidate could think that steam cooling is occurring with the given conditions and forget that the requirement is to be above MSCRWL in order for this method to be employed. The mechanism is correct.

Technical Reference(s): NER-2M-039, Rev 8.0, section 2.0 Definitions

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01, Objective N2-EOP-RPV-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)5

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K4.05
	Importance Rating	3.4

SLC

Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Dispersal of boron upon injection into the vessel

Proposed Question: #7

2SLS*P1A, "PMP 1A" control switch on 2CEC*PNL601 is taken to the 'PUMP A RUN' position.

Which one of the following describes the SLS and WCS System response?

Suction Valve, 2SLS*MOV1A, receives an open signal, ...

A. Squib Valve 2SLS*VEX3A fires, RWCU Outboard Isolation valve receives a close signal, 2SLS*P1A starts when suction valve is full open. Boron is injected inside of the shroud and is sprayed across the core region.

B. Squib Valve 2SLS*VEX3A fires, RWCU Inboard Isolation valve receives a close signal, 2SLS*P1A starts when suction valve is full open. Boron is injected into the down comer area and is mixed with reactor coolant.

C. Squib Valves 2SLS*VEX3A and 2SLS*VEX3B fire, RWCU Outboard Isolation valve closes, 2SLS*P1A starts when suction valve full open. Boron is injected inside of the shroud and is sprayed across the core region.

D. Squib Valves 2SLS*VEX3A and 2SLS*VEX3B fire, RWCU Inboard Isolation valve closes, 2SLS*P1A starts when suction valve full open. Boron is injected into the down comer area and is mixed with reactor coolant.

Proposed Answer: A

Explanation: The "A" squib fire, RWCU isolates and the "A" pump starts when the suction valve is fully open. Per USAR section 9.3.5.2, "The liquid is pumped into the HPCS line downstream of the inboard containment isolation check valve. The sodium pentaborate solution is discharged radially over the top of the core through the HPCS sparger.

B. Plausible – Both the first part and the second part of the distracters are incorrect, but plausible. In the first part of the distracter, the candidate could confuse which valves (inboard or outboard) are associated with which division. For the second part of the distracter, the candidate could think that the feedwater sparger is used to disperse boron.
C. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. In the first part of the distracter, only the valve divisionally associated with the started pump receives a signal to fire, but the candidate could confuse the logic and believe that both squibs fire since the downstream piping is cross connected.
D. Plausible – Both the first part and the second part of the distracters are incorrect, but plausible. In the first part of the distracter, only the valve divisionally associated with the started pump receives a signal to fire, but the candidate could confuse the logic and believe that both squibs fire since the downstream piping is cross connected. For the second part of the distracter, the candidate could think that the feedwater sparger is used to disperse boron.

Technical Reference(s): N2-OP-36A, section B. System Description, USAR section 9.3.5.2.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-211000C01, RBO-05

Question Source: Bank

Question History: 2009 NRC 23

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209002 K4.02
	Importance Rating	3.4

HPCS

Knowledge of HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) design feature(s) and/or interlocks which provide for the following: Prevents over filling reactor vessel: Plant-Specific.

Proposed Question: #8

A plant scram has occurred from rated conditions with the following:

- RPV water level dropped below Level 2, then recovered.
- Subsequently, RPV water level has exceeded the Level 8 setpoint and is rising slowly.

Which of the following indicates the current status of the HPCS system?

	HPCS Pump	Injection Valve	Minimum Flow Valve
A.	Running	Open	Closed
B.	Running	Closed	Open
C.	Off	Closed	Closed
D.	Off	Closed	Open

Proposed Answer: B

Explanation: HPCS will initiate on Level 2 resulting in pump running and injection valve opening. When level 8 is reached, the pump will continue running, the injection valve will go closed, and the min flow valve will open, allowing flow to the suppression pool.

A. Plausible – HPCS pump is running. However, plausible if Level 8 valve automatic position is not understood.
C. Plausible – Flow from HPCS is stopped, but the pump remains running. If the examinee believes the HPCS pump trips, both valves going shut is plausible.
D. Plausible – Flow from HPCS is stopped, but the pump remains running. Also, LPCS min flow valve normal position is open.

Technical Reference(s): N2-OP-33, section B. System Description

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-209002C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K5.01
	Importance Rating	2.6

IRM

Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : Detector operation

Proposed Question: #9

A reactor startup is in progress with the following:

- All Intermediate Range Monitors (IRMs) fully inserted and on range 5.
- Then, IRM channel "B" fails downscale.

Which one of the following is true if IRM "B" is bypassed?

A. All trips (rod blocks, scram signals, and annunciator inputs) associated with IRM "B" will be defeated.

B. The IRM "B" detector will not be able to be withdrawn.

C. The IRM "B" detector can be withdrawn but will result in a rod block unless the reactor mode switch is first placed in RUN.

D. The IRM "B" downscale rod block will continue to be in effect until IRM "B" is placed in range 1.

Proposed Answer: A

Explanation: USAR paragraph 4.11 section 7.6.2.4.3.1 states "The NMS system is designed to permit a channel to be bypassed for maintenance or calibration without initiating protective action at the system level. Four IRMs supply RPS trip signals to RPS Division 1 or 3, and the four other IRMs supply scram trips to RPS Division 2 or 4. (A trip of Division 1 or 3 coincident with a trip of Division 2 or 4 will result in a full scram.) One IRM in each set of four may be bypassed. Removing more than one from each set of four will result in an IRM INOP condition, which is an automatic trip of the related RPS logic."

USAR paragraph 4.7, section 7.6.2.4.3.1 states that "With one IRM channel bypassed in the group of IRM channels feeding RPS Division 1 or 3, one channel bypassed in the group of IRM channels feeding RPS Division 2 or 4, or one APRM/OPRM channel bypassed, and an additional single random failure in a control system requiring protective action, the remaining redundant protection channels (six IRM or three APRM/OPRM) will provide protective action with one-out-of-two twice logic for the IRMs.

B. Plausible – The candidate may think that bypassing the IRM would bypass the trip circuitry signals sent to RPS as well as the drive motor circuitry for the associated IRM, however this is not the case, bypassing the IRM only bypasses the trip circuitry signals sent to RPS.

C. Plausible – USAR section 7.6.2.4.1 states "A rod block is initiated if the IRM detectors are not fully inserted in the core unless the reactor mode switch is in the RUN position", however since the IRM is bypassed, no trip or rod block outputs signals will be sent.

D. Plausible – USAR section 7.6.2.4.1 states "A rod block is initiated if the IRM detectors are not fully inserted in the core unless the reactor mode switch is in the RUN position", however since the IRM is bypassed, no trip or rod block outputs signals will be sent.

Technical Reference(s): USAR section 7.6.2.4.3.1, paragraph 4.7 & 4.11

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-215002C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K5.06
	Importance Rating	3.4

EDGs

Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : Load sequencing

Proposed Question: #10

The plant is operating at rated conditions with the Div I EDG INOPERABLE due to maintenance when the following occurs:

Time mm:ss	Event
00:00	A loss of 115KV Line 6 occurs.
00:45	A LOCA in the drywell causes drywell pressure to rise above 1.68 psig.

Which one of the following identifies proper sequencing for 2RHS*P1A and 2RHS*P1B with the conditions listed above?

	2RHS*P1A starts at time...	2RHS*P1B starts at time...
A.	00:46	00:46
B.	00:46	00:50
C.	00:50	00:46
D.	00:50	00:50

Proposed Answer: C

Explanation: The question stem states that the division 1 EDG is inoperable due to maintenance. With the division 1 EDG inoperable and out of service, the only impact of this would be if there was a loss of Line 5. At time 00:00 when the loss of line 6 occurs, the division II EDG immediately starts, comes up to speed and closes its output breaker at 00:10. At time 00:45 the LOCA event occurs. The LOOP signal for division II is still sealed in so the LOCA signal causes essentially a LOOP/LOCA signal which results in the following sequencing (T=0 is when the LOCA signal occurs):

- T=1 second, RHS*P1B starts (for this question the time would be 00:46)

For division I, there was no loss of power signal so, the LOCA signal alone will cause the following sequencing (T=0 is when the LOCA signal occurs):

- T=5 seconds, RHS*P1A starts (for this question the time would be 00:50)

A. Plausible - The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part of the distracter, the candidate may not understand the logic and think that the loss of offsite power signal generated from the loss of Line 6 affects the division II logic. In this case the thought process would be that both divisions see a LOOP/LOCA signal and the start time for RHS*P1A be 00:46.
B. Plausible - Both the first part and the second part of the distracters are incorrect, but plausible. For the first part of the distracter, the candidate may not understand the logic and think that the loss of offsite power signal generated from the loss of Line 6 affects the division II logic. In this case the thought process would be that both divisions see a LOOP/LOCA signal and the start time for RHS*P1A be 00:46. For the second part of the distracter, the candidate may not recall that the division II EDG has already started on the LOOP signal. In this case the thought process would be that RHS*P1B would start 5 seconds later at 00:50.
D. Plausible - The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the candidate may not recall that the division II EDG has already started on the LOOP signal. In this case the thought process would be that RHS*P1B would start 5 seconds later at 00:50.

Technical Reference(s): 05ENS-021 sheet 2, 05ENS-022 sheet 2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-264000C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	212000 K6.04
Importance Rating	2.8

RPS

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM : D.C. electrical distribution

Proposed Question: #11

The plant is operating at rated conditions with the following:

- Scram Pilot Valve (2RDS*SOV139) for fully withdrawn control rod 10-15 is stuck and will not reposition during a scram.
- Division II emergency DC bus 2BYS*SWG002B is de-energized.
- A full reactor scram occurs.

Control rod 10-15...

A. will NOT insert.

B. will insert simultaneously with the other control rods at normal speed.

C. will insert simultaneously with the other control rods at slower than normal speed.

D. insertion will be delayed, once insertion commences rod will insert at normal speed.

Proposed Answer: D

Explanation: The loss of DC will still leave one backup scram valve available to depressurize the header. Normally the scram pilot just has to depressurize a small segment of pipe at each HCU. The backup scram valve needs to depressurize the entire header. This will result in a delay in commencing rod motion. Once the scram inlet and outlet reposition, the rod will insert at normal speed

A. Plausible – Rod will insert when the scram air header is depressurized via the "A" Backup scram valve. Only one back up scram valve is required to perform this function.
B. Plausible – The backup scram valve needs to depressurize the entire header. This will result in a delay in commencing rod motion.
C. Plausible – Control rod insertion will be delayed because the individual scram valve fails to open. It will insert at normal speed when air is bled from the scram valves by the backup scram valve.

Technical Reference(s): LP #2101-212000C01, RPS, Page 42 for description of scram pilot valve operation and how only one backup scram valve is needed. Page 57 for the power supply to the backup scram valves.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-212000C01, RBO-4/11, Vision Objectives 75962 & 75963

Question Source: Bank

Question History: 2010 Audit 11

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 K6.03
	Importance Rating	3.5

RCIC

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Suppression pool water supply

Proposed Question: #12

The plant is operating at rated conditions with the following:

- RCIC is in a standby condition.
- Leaking valves cause Suppression Pool Level to increase such that annunciator 601717, "SUPPRESSION POOL LEVEL HIGH" alarms.

Which one of the following describes the response of the RCIC system to these conditions?

Note:
- 2ICS*MOV129, ICS Pump Suction Isolation from CST
- 2ICS*MOV136, ICS Pump Suction Isolation from Suppression Pool

A. Upon receipt of the High Suppression Pool Water Level alarm, 2ICS*MOV129 from the CST will close, and then 2ICS*MOV136 from the Suppression Pool will open.

B. Upon receipt of the High Suppression Pool Water Level alarm, 2ICS*MOV136 from the Suppression Pool will open, and then 2ICS*MOV129 from CST will close.

C. RCIC suctions will remain in standby configuration until a low CST Level condition occurs. The RCIC suctions will then transfer with 2ICS*MOV136 from the Suppression Pool opening, followed by the 2ICS*MOV129 from the CST closing.

D. RCIC suctions will remain in standby configuration until a low CST Level condition occurs. The RCIC suctions will then transfer with 2ICS*MOV129 from CST closing followed by the 2ICS*MOV136 from the Suppression Pool opening.

Proposed Answer: C

Explanation: RCIC suctions will remain in standby configuration until a low CST Level condition occurs. The RCIC suctions will then transfer with 2ICS*MOV136 from the Suppression Pool opening, followed by the 2ICS*MOV129 from the CST closing.
The correct answer properly describes that RCIC will not realign for the given conditions until a CST low level condition is detected, then 2ICS*MOV136 from the Suppression Pool opening followed by 2ICS*MOV129 from the CST closing.
USAR section, II.K.3.22 "RCIC SUCTION SOURCE," The modification of the RCIC system allows automatic switchover of pump suction from the CST to the suppression pool if the RCIC pump suction pressure falls to a preset low level. Two pressure transmitters are used to detect low pressure at the RCIC pump suction. If either transmitter senses low pressure (indicating low CST level), pump suction is automatically transferred to the suppression pool. These are different transmitters/trip units from those that activate switchover for the HPCS system. The CST suction valve will be signaled to close upon opening of the suppression pool suction valve.

A. Plausible – Candidate could believe that the logic for RCIC suction source swap over is initiated by Suppression pool level and also believe that 2ICS*MOV129 closes first then 2ICS*MOV136 opens
B. Plausible – Candidate could believe that the logic for RCIC suction source swap over is initiated by Suppression pool level, the order of valve operation is correct
D. Plausible – The first part of the distracter is correct and the candidate could think that 2ICS*MOV129 closes first then 2ICS*MOV136 opens

Technical Reference(s): USAR section, II.K.3.22 "RCIC SUCTION SOURCE", N2-OP-35 System Description

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-217000C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A1.04
	Importance Rating	3.6

Reactor Water Level Control

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor water level control controller indications

Proposed Question: #13

The plant is operating at rated conditions with the following:

- Reactor water level is 183" and stable.
- The Feedwater master controller, 2FWS-HIC1600, is in AUTO.
- The individual Feedwater level control valve controllers, 2FWS-HIC1010A and 2FWS-HIC1010B, are in AUTO.

Then, Reactor power is lowered to 90% and plant conditions stabilize.

Which one of the following describes the effect of depressing the Feedwater master controller, 2FWS-HIC1600, MAN pushbutton at this time?

Feedwater flow will __(1)__. Feedwater flow may be adjusted by depressing the __(2)__ OPEN/CLOSE pushbuttons.

A. (1) rapidly rise
(2) 2FWS-HIC1600

B. (1) rapidly rise
(2) 2FWS-HIC1010A or 2FWS-HIC1010B

C. (1) remain near the current value
(2) 2FWS-HIC1600

D. (1) remain near the current value
(2) 2FWS-HIC1010A or 2FWS-HIC1010B

Proposed Answer: C

Explanation: Although Reactor power has been lowered by 10%, 2FWS-HIC1600 manual signal automatically tracks the current Feedwater flow. This feature allows a smooth transfer while placing the controller in MAN without any need for controller nulling. A small mismatch between steam and feed flow at the time of transfer may necessitate manual adjustments. In the given conditions, this would be performed by depressing the OPEN/CLOSE pushbuttons on 2FWS-HIC1600. The OPEN/CLOSE pushbuttons on 2FWS-HIC1010A or 2FWS-HIC1010B will not cause a flow change while those controllers are still in AUTO.

A. Plausible – Feedwater flow will remain near the current value, because 2FWS-HIC1600 manual signal automatically tracks the auto signal.
B. Plausible – Feedwater flow will remain near the current value, because 2FWS-HIC1600 manual signal automatically tracks the auto signal. The OPEN/CLOSE pushbuttons on 2FWS-HIC1010A or 2FWS-HIC1010B will not cause a flow change while those controllers are still in AUTO.
D. Plausible – The OPEN/CLOSE pushbuttons on 2FWS-HIC1010A or 2FWS-HIC1010B will not cause a flow change while those controllers are still in AUTO.

Technical Reference(s): N2-OP-3 Section F.8.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-259002-RBO-5

Question Source: Bank

Question History: 2012 NRC

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 A1.01
	Importance Rating	2.8

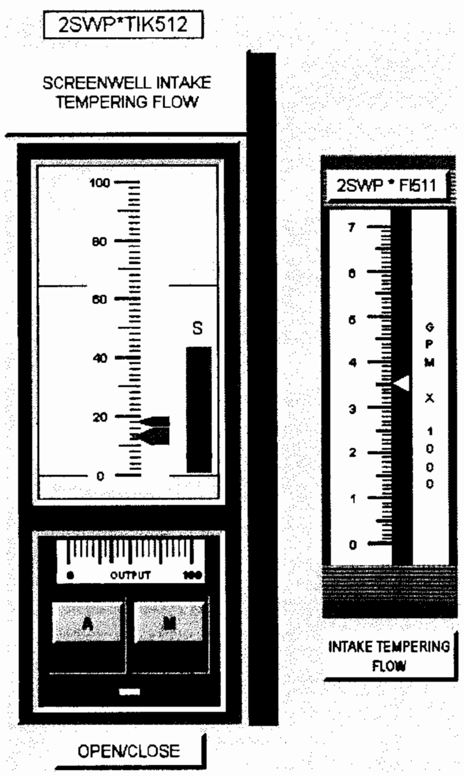
Component Cooling Water

Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: CCW flow rate

Proposed Question: #14

The plant is operating at rated conditions with the following:

- Circ. Water tempering of Service Water is in progress per N2-OP-11, SERVICE WATER SYSTEM
- Chemistry determines that intake bay temperature is too warm and wants operations to LOWER intake bay temperature
- Indications on 2CEC*PNL601 for Service Water tempering are as follows:



Which one of the following correctly completes the following statement?

In order to lower the intake bay temperature, the RO should __(1)__ the INTAKE TEMPERING FLOW by taking the SLIDER on 2SWP*TIK512 to the __(2)__ .

(1)	(2)

A.	RAISE	LEFT
B.	RAISE	RIGHT
C.	LOWER	LEFT
D.	LOWER	RIGHT

Proposed Answer: D

Explanation: Per N2-OP-11, Section F.9.0, the notes within the section indicate that the 2SWP-TIK5112 controller works in reverse of most controllers. In this case, 100% output is a fully closed tempering valve. When the controller output is at 0%, then the valve is fully open. In order to get the valve to go farther closed, you have to take the controller slider to the right.

The more tempering flow you have, the warmer the intake bay will be. Since the question stem wants you to lower the intake bay temperature, then you have to lower the tempering flow.

A. Plausible – The candidate could confuse the concept of tempering and believe that raising tempering flow by opening the tempering valve would introduce more cool water to the intake bay
B. Plausible – The candidate could confuse the concept of tempering and believe that raising tempering flow by closing (since the controller direction works in reverse) the tempering valve would introduce more cool water to the intake bay
C. Plausible – The candidate could get the concept that less tempering flow is required, but miss that the controller positioned works in reverse.

Technical Reference(s): N2-OP-11, Section F.9.0

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-276000C01, RBO-10

Question Source: Bank (Vision ID 104246)

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 A2.11
	Importance Rating	3.2

SGTS

Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High containment pressure

Proposed Question: #15

The plant has scrammed from rated conditions with the following:

- There is a small LOCA in the DRYWELL.
- Drywell pressure indication is unavailable.
- There is no indication of a bypass path in the containment.
- Suppression chamber pressure is 1.7 psig

Based on the above information, DRYWELL pressure is approximately _____(1)_____ AND The Stand-by Gas Treatment system_____(2)_____, per N2-EOP-PC.

A. (1) 1.7 psig
(2) is running in the desired containment venting lineup

B. (1) 6.7 psig
(2) is running in the desired containment venting lineup

C. (1) 1.7 psig
(2) requires operator action to establish a containment venting lineup

D. (1) 6.7 psig
(2) requires operator action to establish a containment venting lineup

Proposed Answer: D

Explanation: There is differential of approximately 5 psig between drywell and suppression chamber pressure due to the submergence of the downcomers. Suppression chamber pressure of 1.7 psig correlates to approximately 6.7 psig, which is above the isolation setpoint for Reactor Building Ventilation. When RB Vent isolates, SBGT starts automatically. Even though GTS automatically initiates, N2-EOP-PC directs the use of OP-61A, section H.1 to use GTS to vent the containment to try to maintain primary containment pressure. This requires manual operator action.

A. Plausible – 1.7 psig would be the pressure in the drywell if a bypass pathway were present. GTS would be running at 1.7 psig in the drywell after auto initiation on a LOCA signal at 1.68 psig. However, the auto initiation lineup is to maintain negative DP in the reactor building, not containment venting.
B. Plausible – 6.7 psig is correct. GTS would be running at 6.7 psig in the drywell after auto initiation on a LOCA signal at 1.68 psig. However, the auto initiation lineup is to maintain negative DP in the reactor building, not containment venting.
C. Plausible – 1.7 psig would be the pressure in the drywell if a bypass pathway were present. Operator action is required to establish containment venting with GTS.

Technical Reference(s): N2-EOP-PC, N2-OP-61A

Proposed references to be provided to applicants during examination: None

Learning Objective: 2102-223001C01, RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 A2.03
	Importance Rating	4.1

SRVs

Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Stuck open SRV

Proposed Question: #16

The plant is operating at rated conditions with the following:

- Safety Relief Valve (SRV) 2MSS*PSV120 inadvertently opens.
- All fuses are removed for the affected SRV, per N2-SOP-34, Stuck Open Safety Relief Valve.

Assuming the SRV shuts, what is the effect on Total Steam Flow INDICATION and an acceptable method to confirm the SRV is shut, per N2-SOP-34?

	Total Steam Flow Indication	Method to Confirm SRV is Shut
A.	Lowers	Closed indication on ERF computer
B.	Rises	Closed indication on ERF computer
C.	Lowers	Closed indication on panel P601
D.	Rises	Closed indication on panel P601

Proposed Answer: B

Explanation: Initially with the SRV open, total steam flow leaving the reactor remains the same as EHC closes down on the control valves an equivalent amount to control pressure. This causes a drop in total steam flow indication since the SRVs are upstream of the flow transmitters. When the SRV shuts total steam flow will rise and per USAR, section II.D.3, "Unit 2 has two means of SRV position indication. The primary means is an acoustic monitoring system that monitors SRV tailpipe noise. This system is IE and is qualified for in-containment use. Valve position is indicated in the control room and is entered into the Emergency Response Facility (ERF) computer. The secondary means of valve lift monitoring is non-IE thermocouples on the SRV tailpipe. This monitoring means is only a backup/verification of valve lift. The output of the acoustic monitoring system drives individual valve open/closed indications on control boards. The system also drives sequence of events (SOE) input computer points. When a valve lifts, the Operator is alerted via the alarm cathode ray tube (CRT) and alarm printer in the main control room." N2-SOP-34 directs the operator to confirm the SRV is shut using indications in Detail 1.

A. Plausible - The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. In the first part of the distracter, the candidate may not understand the EHC response.
C. Plausible - The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. In the second part of the distracter, the candidate may not understand the impact on control room panel indication of pulling all SRV fuses and the panel P601 indication would be the normal way to check the status of SRVs. However, in this case, the examinee needs to recognize power is not available to that indication when fuses are pulled.
D. Plausible - Both the first part and the second part of the distracters are incorrect, but plausible. In the first part of the distracter, the candidate may not understand the EHC response. In the second part of the distracter, the candidate may not understand the impact on control room panel indication of pulling all SRV fuses and the panel P601 indication would be the normal way to check the status of SRVs. However, in this case, the examinee needs to recognize power is not available to that indication when fuses are pulled.

Technical Reference(s): N2-SOP-34, Stuck Open SRV, USAR section II.D.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP34C01, EO-2, Operational Actions and Sequence

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A3.01
	Importance Rating	3.1

AC Electrical Distribution

Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Breaker tripping

Proposed Question: #17

Given the following circuit breakers:

(1) 1-3, Supply to 2NPS-SWG001 from Normal Station Transformer
(2) 1-1, Supply to 2NPS-SWG001 from Reserve Station Transformer A
(3) 3-14, Supply to 2NPS-SWG003 from Normal Station Transformer
(4) 3-1, Supply to 2NPS-SWG003 from Reserve Station Transformer B
(5) 13-6, Supply to 2NNS-SWG014 from 2NPS-SWG001
(6) 15-3, Supply to 2NNS-SWG015 from 2NPS-SWG003

Which one of the following indicates the circuit breakers that are expected to automatically OPEN following a Main Generator trip (Generator Lockout Trip) from rated conditions?

A. (1), (3)

B. (2), (4)

C. (1), (3), (5), (6)

D. (2), (4), (5), (6)

Proposed Answer: A

Explanation: On a main generator trip, a fast transfer occurs, transferring 2NPS-SWG001(003) from the Normal Station Transformer to the Reserve Station Transformers. Supply Breakers from the normal Station Transformer will automatically open.

B. Plausible – The supply breakers from the reserve transformers will automatically operate following a main generator trip. However, they will close, not open.
C. Plausible – 2NNS-SWG014(015) are expected to re-energize following a main generator trip. They will re-energize from their normal supply. The normal supply breakers for these busses will not trip.
D. Plausible – 2NNS-SWG014(015) are expected to re-energize following a main generator trip. They will re-energize from their normal supply. The normal supply breakers for these busses will not trip.

Technical Reference(s): N2-OP-68 Attachment 1-4, N2-ARP-852614, Drawing EE-MO1A

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-262001C01, RBO 5

Question Source: New

Question History:

Question Cognitive Level: Memory Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	215005 A3.02
Importance Rating	3.5

APRM / LPRM

Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Full core display

Proposed Question: #18

A manual scram has been attempted. Given the following three separate sets of conditions:

(1) All control rod position indication is lost on the full core display. All APRMs indicate downscale.

(2) Three control rods remain at position 02 and all other control rods fully insert. All APRMs indicate downscale.

(3) All control rods indicate fully inserted on the full core display. All APRM indications are lost.

Which one of the following describes the ability of an Operator to determine that the Reactor will remain shutdown without boron in these conditions, in accordance with EOP Bases and OP-NM-101-111-1001, Transient Mitigation Guidelines (TMG)?

An Operator can make the determination that the Reactor will remain shutdown without boron...

A. in condition (2) only.

B. in condition (3) only.

C. in conditions (1) and (2) only.

D. in conditions (2) and (3) only.

Proposed Answer: B

Explanation: OP-NM-101-111-1001, Transient Mitigation Guidelines (TMG) provides two situations where an Operator is empowered to determine that the Reactor will remain shutdown without boron:

- All rods are fully inserted (position 00) or
- Only 1 rod is withdrawn beyond position 00 AND All other rods are fully inserted

APRMs being downscale are not enough information to make this determination. Neither condition (1) or (2) fit the allowed situations to determine the Reactor will remain shutdown without boron.

A. Plausible – Both conditions provide evidence that the scram resulted in significant negative reactivity insertion, however neither condition meets the criteria of OP-NM-101-111-1001.
C. Plausible – Conditions (1) and (2) do show enough negative reactivity was inserted to make APRMs indicate downscale, however APRM indication is NOT enough to make the shutdown determination.
D. Plausible – Condition (2) does show that control insertion was almost completely successful, however the criteria of OP-NM-101-111-1001 are NOT met.

Technical Reference(s): OP-NM-101-111-1001, Transient Mitigation Guidelines (TMG) Attachment 2, section 2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01, N2-EOP-RPV-CE-02

Question Source: Modified 2013 Audit

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	223002 A4.04
Importance Rating	3.5

PCIS/Nuclear Steam Supply Shutoff

Ability to manually operate and/or monitor in the control room: System indicating lights and alarms

Proposed Question: #19

The plant is starting up during the winter months with the following:

- RPV pressure is 850 psig and stable.
- Main Steam tunnel temperatures are:

Instrument	Location	Temp.
E31-N604A	MSL TUNNEL EL 263 FT	170°F
E31-N604B	MSL TUNNEL EL 263 FT	172.5°F
E31-N604C	MSL TUNNEL EL 263 FT	160°F
E31-N604D	MSL TUNNEL EL 263 FT	157.5°F

Which one of the following describes the expected alarms and indications on the 2CEC*PNL602 panel?

Annunciator ____(1)____ will be in alarm.

Inboard and Outboard MSIVs, 2MSS*AOV6A-D and 2MSS*AOV7A-D will indicate ___(2)___ .

Inboard MSL Drain 2MSS*MOV111 AND/OR Outboard MSL Drains (2MSS*MOV112 AND MOV208) will indicate ___(3)___.

A. (1) 602228, MN STEAM LINE PIPE TUNNEL TEMP HI-HI/DIFF TEMP HI
(2) SHUT
(3) OPEN

B. (1) 602234, MN STEAM LINE PIPE TUNNEL DIFF TEMP HIGH
(2) SHUT
(3) SHUT

C. (1) 602228, MN STEAM LINE PIPE TUNNEL TEMP HI-HI/DIFF TEMP HI
(2) OPEN
(3) SHUT

D. (1) 602234, MN STEAM LINE PIPE TUNNEL DIFF TEMP HIGH
(2) OPEN
(3) SHUT

Proposed Answer: A

Explanation: Annunciator 602228 alarms when a Trip Unit E31-N604A - D exceeds 163°F. The differential temperature high annunciator uses a different temperature sensor and would only cause an alarm (E31-N615 set for 56°F). With Logic inputs A and B tripped, ONLY the MSIVs will isolate, the MSL drains would remain open.

B. Plausible – The differential temperature high annunciator uses a different temperature sensor and would only cause an alarm (E31-N615 set for 66.7°F). With Logic inputs A and B tripped, ONLY the MSIVs will isolate, the MSL drains would remain open.
C. Plausible – With Logic inputs A and B tripped, ONLY the MSIVs will isolate, the MSL drains would remain open.
D. Plausible – The differential temperature high annunciator uses a different temperature sensor and would only cause an alarm (E31-N615 set for 56°F). With Logic inputs A and B tripped, ONLY the MSIVs will isolate (green indication), the MSL drains would remain open.

Technical Reference(s): N2-ARP-602228, N2-OP-83 attachment 3

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-239001C01, RBO-05

Question Source: Bank

Question History: 2012 Audit

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 A4.01
	Importance Rating	2.8

UPS (AC/DC)

Ability to manually operate and/or monitor in the control room: Transfer from alternative source to preferred source

Proposed Question: #20

The plant is operating at rated conditions with the following:

- Annunciator 852503, UPS 1A SYSTEM TROUBLE, alarms in the Control Room.
- Approximately thirty (30) seconds later, annunciator 852503 clears.
- The Control Room Operators determined the alarm was due to a large 50 amp load being started on 2VBB-UPS1A.

Which one of the following describes the electrical supply currently supplying power to 2VBB-UPS1A and the required operator actions, per N2-ARP-852500?

	Electrical Bus Supplying Power	Required Operator Actions
A.	Normal	Secure the large load
B.	Alternate	Secure the large load
C.	Normal	Clear local alarms
D.	Alternate	Clear local alarms

Proposed Answer: C

Explanation: ARP 852503 has a note that states when a large load (40 amps) is started on UPS 1A, the UPS may transfer to its maintenance supply until current stabilizes. The UPS will then automatically transfer back to its normal power supply. If a system trouble alarm is annunciated and then clears AND it is known that a large load has been added to or removed from the UPS, no action would be required except the clearing of local alarms.

A. Plausible – The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the UPS is on the normal supply. It's reasonable to assume the large load placed the UPS in an undesirable condition and would warrant securing the load. However, this is not required.
B. Plausible – Both the first and second part of the distracter is incorrect, but plausible. For the first part of the distracter, the annunciator originally alarmed because the UPS was on an alternate power supply. The candidate could think that UPS1A shifted to its alternate supply and is therefore still being fed from its alternate supply. For the second part of the distracter, the UPS is on the normal supply. It's reasonable to assume the large load placed the UPS in an undesirable condition and would warrant securing the load. However, this is not required.
D. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part of the distracter, the annunciator originally alarmed because the UPS was on an alternate power supply. The candidate could think that UPS1A shifted to its alternate supply and is therefore still being fed from its alternate supply.

Technical Reference(s): N2-ARP-852503

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-262002C01, RBO-03

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 2.4.21
	Importance Rating	4.0

Shutdown Cooling

Emergency Procedures / Plan: 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.

Proposed Question: #21

The reactor is in Mode 4 with the following:

- RHR Loop "A" is in Shutdown Cooling
- RHR Loop "B" is in Suppression Pool Cooling

Then a LOCA causes RPV level to lower to Level 1.

Which of the following describes the RHR system response?

A. RHR Loop A continues to operate in shutdown cooling and RHR B pumps trips.

B. RHR Loop A continues to operate in shutdown cooling and RHR Loop B realigns to the injection mode.

C. RHR pump A trips. RHR Loop B realigns to the injection mode and RHR B pump continues to run.

D. RHR pumps A and B trip. RHR Loop B realigns to the injection mode and RHR pump B restarts.

Proposed Answer: C

Explanation: A RPV level 159.3 will cause a group 5 isolation. A group 5 isolation will cause the RHR Shutdown cooling valves to close. Without a suction source the "A" pump will trip. The 2RHS*MOV38B will go shut on a LOCA signal (Level 1) and the "B" RHR pump will continue to run and will inject when the 2RHS*MOV24B permissives are met.

A. Plausible – A Group 5 isolation will occur when RPV level lowers to < 159.3 inches causing a SDC isolation. RHR Pump B will continue to run in the injection mode.
B. Plausible – A Group 5 isolation will occur when RPV level lowers to < 159.3 inches causing a SDC isolation.
D. Plausible – RHR B will not trip. No trip signals are generated when RHR B realigns for injection as its suction path is not impacted by the realignment.

Technical Reference(s): N2-OP-83 attachment 3, Vision Objective #73502

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-205000C01, RBO-5

Question Source: Bank

Question History: 2012 Audit

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 G2.2.44
	Importance Rating	4.2

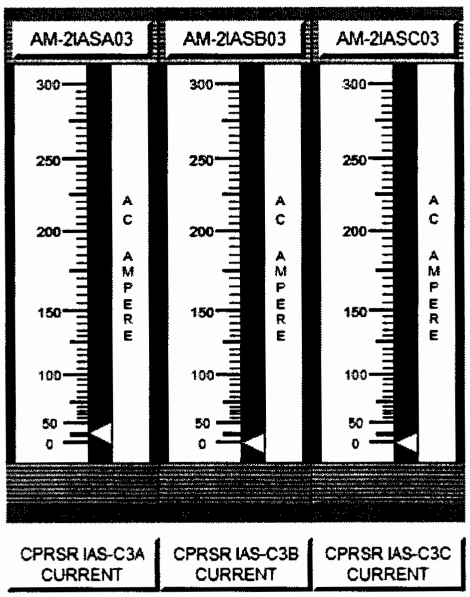
Instrument Air System/ 3

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. (CFR: 41.5 / 43.5 / 45.12)

Proposed Question: #22

The plant is operating at rated conditions. During a panel walkdown, the following conditions are observed:

- Instrument Air Compressor 2IAS-C3A is operating
- INSTRUMENT AIR COMPRESSOR SELECTOR switch is in ABC position.
- Instrument Air Header Pressure is 110 psig and slowly lowering
- Instrument Air Compressor amperage indications are as follows:



Which one of the following identifies the status of 2IAS-C3A and the required actions?

A. 2IAS-C3A is fully LOADED. Start 2IAS-C3B then place INSTRUMENT AIR COMPRESSOR SELECTOR switch in BCA position.

B. 2IAS-C3A is fully LOADED. Place INSTRUMENT AIR COMPRESSOR SELECTOR switch in BCA position then start 2IAS-C3B.

C. 2IAS-C3A is UNLOADED. Start 2IAS-C3B then place INSTRUMENT AIR COMPRESSOR SELECTOR switch in BCA position.

D. 2IAS-C3A is UNLOADED. Place INSTRUMENT AIR COMPRESSOR SELECTOR

switch in BCA position then start 2IAS-C3B.

Proposed Answer: D

Explanation:
The compressor is loaded and unloaded by discharge air pressure signal, which changes based on the rate of air consumption, and maintains the air pressure within a pre-selected band (115 psig - 125 psig). With IAS pressure below 115 psig, the operating compressor should be loaded and running current should be at about 170 amps. Since pressure is low and the compressor is unloaded, as indicated by a current reading of only 30 amps, action is needed to restore IAS header pressure. Entry into N2-SOP-19 is required and for a degraded compressor, another compressor is manually started by repositioning the selector switch and starting the compressor at P851. If the selector switch is not selected to the correct position, the compressor will not start when control switch is placed in start position.

A. Plausible – Incorrect, based on the ammeter reading of 30 amps, the compressor is NOT loaded. The candidate may not recall what fully loaded IAS compressor amps are and think that with some amps indicated, the compressor is loaded. If amps were 170 amps (compressor loaded), this would be a correct answer.
B. Plausible – Incorrect, based on the ammeter reading of 30 amps, the compressor is NOT loaded. The candidate may not recall what fully loaded IAS compressor amps are and think that with some amps indicated, the compressor is loaded.
C. Plausible – Incorrect, because if the selector switch is not first selected to the correct position, the compressor will not start when control switch is placed in start position. The candidate may get the sequence of actions wrong and think that this is the correct answer.

Technical Reference(s): 2101-278001C01, N2-SOP-19, N2-OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-278001C01, RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K1.06
	Importance Rating	3.9

IRM

Knowledge of the physical connections and/or cause- effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: APRM SCRAM signals: Plant-Specific

Proposed Question: #23

The plant is in a reactor startup with the following:

- Reactor power is 4%
- Reactor mode switch is in “Startup”
- IRM ‘E’ is bypassed due to equipment malfunction
- APRM #2 is bypassed due to a critical self test fault

In the event of a power excursion, which one of the following describes the neutron monitoring system(s) that will generate a scram signal that will trip RPS?

	IRM’s	APRM’s
A.	Yes	Yes
B.	No	No
C.	No	Yes
D.	Yes	No

Proposed Answer: A

Explanation: NMP2 USAR section 7.2.1.2.1 states that the reactor mode switch determines whether IRM trips are effective in initiating a reactor scram. With the reactor mode switch in REFUEL, STARTUP, or SHUTDOWN, an IRM upscale or inoperative trip signal actuates a NMS trip of the RPS. Only one of the IRM channels must trip to initiate a NMS trip of the associated RPS trip channel. Additionally it states that in addition to the IRM upscale trip, an APRM trip function with a setpoint of 15-percent power is active when the reactor mode switch is in the STARTUP position.

B. Plausible – The candidate could think that with two instruments bypassed that a trip signal would not be generated and sent to RPS

C. Plausible – The candidate could think that since reactor power is indicating in the power range that the IRM trips are no longer in effect

D. Plausible – The candidate could think that since the mode switch is still in startup that the APRM trips are not in effect

Technical Reference(s): NMP2 USAR, section 7.2.1.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-215002C01, RBO-05
2101-215003C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K4.02
	Importance Rating	2.7

PCIS/Nuclear Steam Supply Shutoff

Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Testability

Proposed Question: #24

The Standby Liquid Control System (SLS) quarterly operability test on 2SLS*P1A is about to be performed.

Which one of the following identifies how the automatic Reactor Water Cleanup System (WCS) isolation is avoided during this test?

A. WCS isolation signal is bypassed when using the SLS pump TEST switch to start the SLS pump.

B. WCS isolation bypass switches are placed in the BYPASS position prior to starting the SLS pump.

C. WCS system is shutdown and containment isolation valves closed prior to starting the SLS pump.

D. WCS containment isolation valve power supply breakers are opened prior to starting the SLS pump.

Proposed Answer: A

Explanation: Initiation of SLS would normally cause a group 7 isolation. Use of the TEST Switch for either pump bypasses the interlocks that open the MOV1A/B valves, fire the squibs, and isolate the respective divisional WCS isolation valve. This allows operation of the pump to circulate the contents of the test tank for surveillance testing.

B. Plausible – The candidate could think that there is a separate switch that bypasses the WCS isolation signal when in reality there is not. The only way to prevent this isolation during the test is by using the TEST switch for SLS.
C. Plausible – The candidate may think that shutting down the WCS system and closing the containment isolation valves in a controlled shutdown manner would allow the logic testing to check for continuity up to and including the containment isolation valve closure signal.
D. Plausible – There are surveillances that require the associated breaker power supplies opened to prevent inadvertently isolating systems during surveillance testing. The candidate could think that this is how the test is performed for the WCS system since isolation of the system would impact the core thermal power calculation.

Technical Reference(s): N2-OSP-SLS-Q001 and SLS GE Print 807E161TY (0007.223-001-005Q) Sheet 4

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-211000C01, RBO-5

Question Source: Bank

Question History: 2013 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A4.03
	Importance Rating	3.2

AC Electrical Distribution

Ability to manually operate and/or monitor in the control room: Local operation of breakers

Proposed Question: #25

The plant is performing a reactor startup with the following:

- Preparations are underway for rolling the main turbine
- 2ARC-P1A, "AIR REMOVAL PMP 1A" is in service
- It is desired to make 2ARC-MOV15A throttleable for rub mitigation concerns
- An Equipment Operator is in the field performing the actions to make 2ARC-MOV15A throttleable

Per N2-OP-9 Attachment 4, "Making 2ARC-MOV15A(B) Throttleable", which one of the following describes the correct control room indications for 2ARC-MOV15A during this evolution?

	Control room light indication while lead lifting evolution is being conducted	Control Room light indication when lead lifting evolution is complete
A.	Green light OFF Red light ON	Green light OFF Red light ON
B.	Green light OFF Red light ON	Green light ON Red light ON
C.	Green light OFF Red light OFF	Green light OFF Red light ON
D.	Green light OFF Red light OFF	Green light ON Red light ON

Proposed Answer: C

Explanation: Per N2-OP-9 Attachment 4, for making making 2ARC-MOV15A(B) Throttleable the first step is to open the power supply breaker for 2ARC-MOV15A, then lift and tape seal in circuitry leads, and then the final step is to reclose the power supply breaker. While the power supply breaker is opened, no light indication will be present for 2ARC-MOV15A on control room panel 851. All breaker light indication will remain off until the power supply breaker is reclosed at which point the red light indication only will illuminate since 2ARC-P1A was in service prior to the performance of attachment 4.

A. Plausible – The candidate could think that either the breaker is not opened to perform this evolution and therefore the red light would remain energized or may think that breaker control power comes from a different source and that the red light would remain energized while 2ARC-MOV15A power supply breaker (2NHS-MCC006-2A) is open.
B. Plausible - The candidate could think that either the breaker is not opened to perform this evolution and therefore the red light would remain energized or may think that breaker control power comes from a different source and that the red light would remain energized while 2ARC-MOV15A power supply breaker (2NHS-MCC006-2A) is open. For the second part of the distracter, the candidate could think that once the seal in circuitry has been broken by lifting leads, that the valve would show the red and green lights lit indicating that the valve was in a throttleable mode.
D. Plausible – The first of the distracter are correct. For the second part of the distracter, the candidate could think that once the seal in circuitry has been broken by lifting leads, that the valve would show the red and green lights lit indicating that the valve was in a throttleable mode.

Technical Reference(s): N2-OP-9, Attachment 4

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-255000C01, RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K2.01
	Importance Rating	3.2

RPS

Knowledge of electrical power supplies to the following: RPS motor-generator sets

Proposed Question: #26

The plant is operating at rated conditions with the following:

- 2NJS-US5 de-energizes due to a bus fault.

Which one of the following describes (1) the RPS component which de-energizes and (2) the effect of the bus fault on RPS?

A. (1) 2RPM-MG1A
(2) "A" logic de-energizes

B. (1) 2RPM-MG1A
(2) "A" scram solenoids de-energize

C. (1) 2VBB-UPS3A
(2) "A" logic de-energizes

D. (1) 2VBB-UPS3A
(2) "A" scram solenoids de-energize

Proposed Answer: B

Explanation: 2RPM-MG1A is powered from 2NJS-US5 and will lose power on a bus fault.

A. Plausible– 2RPM-MG1A does de-energize, however the logic is powered from the UPS, not the MG. The candidate might think that the logic is powered from the MG.
C. Plausible – 2NJS-US5 is the alternate power supply to 2VBB-UPS3A, therefore the UPS would remain energized. The candidate might think that the loss of power would cause the loss of the UPS.
D. Plausible – 2NJS-US5 is the alternate power supply to 2VBB-UPS3A, therefore the UPS would remain energized. The candidate might think that the loss of power would cause the loss of the UPS.

Technical Reference(s): EE-1AL, EE-MO001C, 807E166TY (0007.225.001.038E) Sheet 4A

Proposed references to be provided to applicants during examination: None

Learning Objective: 2102-212000C01, RBO-04

Question Source: Modified – 2013 Audit #3

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 K1.05
	Importance Rating	2.8

Main and Reheat Steam

Knowledge of the physical connections and/or cause- effect relationships between MAIN AND REHEAT STEAM SYSTEM and the following: Moisture separator reheaters: Plant-Specific

Proposed Question: #27

The plant is operating at rated conditions with the following:

- 851411, REHEATER DRN RCVR TK6A/6B WATER LEVEL HIGH, alarms
- Reheater Drain Receiver TK6A level determined to be greater than 8 inches above centerline
- 851401, REHEATER SYSTEM TROUBLE, alarms
- Reheater Drain Receiver TK6A level determined to be greater than 18 inches above centerline (TK6A WATER LEVEL HIGH-HIGH)

Which one of the following describes the response of the Moisture Separator Reheat steam supply valves below?

2MSS-AOV92A, “Reheater 1A Steam Supply Valve”
2MSS-AOV92B, “Reheater 1B Steam Supply Valve”
2MSS-MOV9A, “Reheater Steam Control Valve PV28A Drain Valve”
2MSS-MOV9B, “Reheater Steam Control Valve PV28B Drain Valve”

	2MSS-AOV92A	2MSS-AOV92B	2MSS-MOV9A	2MSS-MOV9B
A.	Closed	Open	Closed	Closed
B.	Closed	Closed	Closed	Closed
C.	Closed	Closed	Open	Open
D.	Open	Open	Open	Closed

Proposed Answer: C

Explanation: On a Hi Hi level in EITHER drain tank, the steam supplies are isolated and the drain valves are opened to both reheaters (prevents unbalancing the LP turbines.

A. Plausible – The steam supply to the 1B reheater is also close along with both drain valves opening.
B. Plausible if the applicant thinks only the steam valves respond to high tank level. The drain valves for both heaters open draining the steam supply header to the main condenser.
D. Plausible if the applicant thinks only the drain valve for the associated tank will open on high level. The steam supply valves close for both reheaters.

Technical Reference(s): N2-ARP-851401, Reheater System Trouble.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-239001C01, RBO-05

Question Source: Bank

Question History: 2010 Audit 35

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	256000 K2.01
	Importance Rating	2.7

Reactor Condensate

Knowledge of electrical power supplies to the following: System pumps

Proposed Question: #28

The plant is operating at rated conditions with the following:

- All three Condensate Booster Pumps are in service
- A bus fault occurs on 2NPS-SWG003
- Annunciator 852530, 13.8 KV NPS 003 Electrical Fault, is in alarm

Which one of the following describes (1) which Condensate Booster Pump(s) remain(s) available for use and (2) the required operator action in accordance with N2-SOP-29?

A. (1) 2CNM-P2A, only
(2) Perform a manual reactor scram

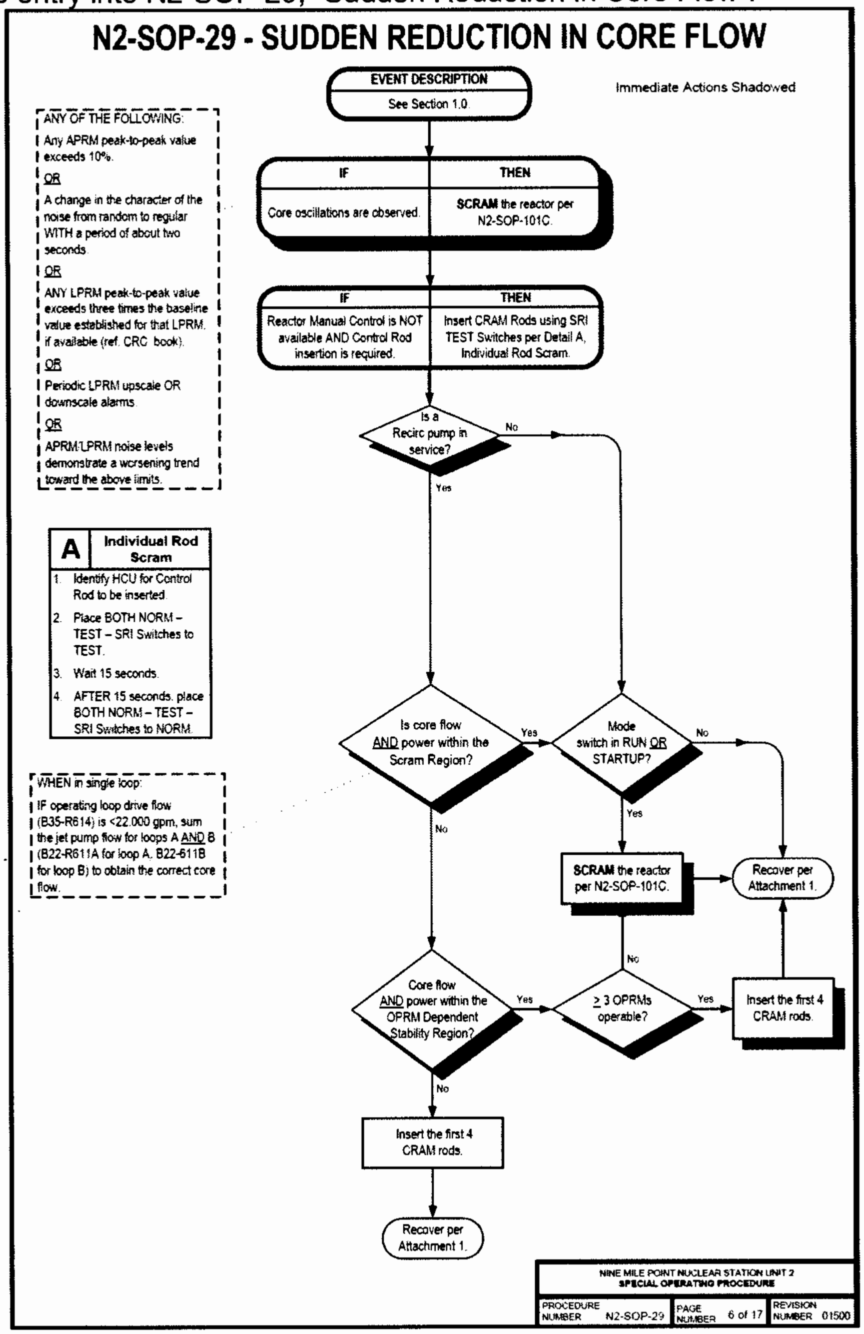
B. (1) 2CNM-P2A and 2CNM-P2C, only
(2) Perform a manual reactor scram

C. (1) 2CNM-P2A, only
(2) Insert the first 4 CRAM rods

D. (1) 2CNM-P2A and 2CNM-P2C, only
(2) Insert the first 4 CRAM rods

Proposed Answer: D

Explanation: P2A is powered from 2NPS-SWG001. P2C can be powered from either 2NPS-SWG001 or 2NPS-SWG003, therefore both pumps remain available for use. When 2NPS-SWG003 is lost, the 'B' reactor recirculation pump will trip ('A' RCS Pump will remain running). This will require entry into N2-SOP-29, "Sudden Reduction in Core Flow":



A. Plausible – Both parts of the distracter are incorrect, but plausible. In the first part of the distracter, one power supply to P2C has been lost, however it remains available from its second power supply. For the second part of the distracter, significant plant loads have been lost due to the bus fault, however a reactor scram is not yet required to ensure plant safety.

B. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter, significant plant

loads have been lost due to the bus fault, however a reactor scram is not yet required to ensure plant safety.

C. Plausible – The first part of the distracter is incorrect, but plausible the second part of the distracter is correct. For the first part of the distracter, one power supply to P2C has been lost, however it remains available from its second power supply.

Technical Reference(s): N2-OP-3, N2-SOP-29

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-256000C01, RBO-03

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	286000 K3.03
	Importance Rating	3.6

Fire Protection

Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on following: Plant protection

Proposed Question: #29

One zone (loop) of Relay Room Underfloor Area fire detection is inoperable and has been placed in Alarm Only at 2CEC-PNL849. This area is protected by cross zone (two loop) detection.

Which one of the following identifies the impact of this failure if an actual fire occurs in the room?

The remaining zone (loop) will ....

A. ONLY actuate control room alarms. Fire suppression must be manually actuated.

B. actuate local AND control room alarms. Fire suppression must be manually actuated.

C. ONLY actuate control room alarms. Fire suppression will be automatically actuated.

D. actuate local AND control room alarms. Fire suppression will be automatically actuated.

Proposed Answer: B

Explanation: Per N2-OP-47, Section B, page 14 (system description), If a fire occurs in a zone and the fire is determined to be an actual fire but the zone is in Alarm Only, the suppressant may be discharged manually from the control room or locally. Also, this zone is designated a cross zone area and N2-OP-47 states that "Zones designated by X are cross zones which contain two fire detection loops. One detector sensing a fire will bring in an alarm and no suppression. When a detector from the other loop senses a fire it will bring in an alarm and activate the fire suppression system." In this case with one of the cross zones in alarm only, you will not be able to get one detector in each loop to sense the fire and will therefore not get automatic suppression.

A. Plausible – Suppression must be manually actuated. Alarms will still occur locally from the remaining operating zone.
C. Plausible – Alarms occur locally and in the control room, suppression requires both
D. Plausible – Alarms occur locally and in the control room, however requires one detector in both zones for auto actuation

Technical Reference(s): N2-OP-47, Sect B, pg 14

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-286000C01, RBO-05

Question Source: Bank

Question History: 2012 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 K4.16
	Importance Rating	3.3

Reactor/Turbine Pressure Regulator

Knowledge of REACTOR/TURBINE PRESSURE REGULATING SYSTEM design feature(s) and/or interlocks which provide for the following: Turbine Shell Warming: EHC Only

Proposed Question: #30

A plant startup is in progress with the following:

- Main Turbine SHELL WARMING is in progress
- Turbine Steam Chest Pressure is 109 psia and rising slowly

Which one of the following actions is required to restore Steam Chest Pressure to within shell warming limits and the basis for this action per N2-OP-21, Main Turbine?

At the MAIN STOP VALVE POSITION DEMAND FOR CHEST/SHELL WARMING controls, press the ...

A. INCREASE pushbutton to prevent lengthening required soak time.

B. INCREASE pushbutton to prevent an automatic reactor scram.

C. DECREASE pushbutton to prevent lengthening required soak time.

D. DECREASE pushbutton to prevent an automatic reactor scram.

Proposed Answer: D

Explanation: Per N2-OP-21, E.3.0 Shell Warming. Per step E.3.19 pressure is to be maintained between 75 and 100 psia. 109 psia must be recognized as being above the high end limit and that the DECREASE pushbutton is used to throttle less steam flow through MSV #2 to restore pressure to the proper band.

A. Plausible – INCREASE will result in pressure approaching and exceeding the first stage pressure that will un-bypass the reactor scram on MSV position.
B. Plausible – INCREASE will result in pressure approaching and exceeding the first stage pressure that will un-bypass the reactor scram on MSV position.
C. Plausible – DECREASE is used to regain margin to prevent exceeding the first stage pressure that will un-bypass the reactor scram on MSV position. Soak time lengthening is not required. If pressure was low (below 75 psia for more than 15 minutes) the soak time would have to be lengthened).

Technical Reference(s): N2-OP-21, section E.3.16

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-241000C01, RBO-05

Question Source: Bank

Question History: 2005 NRC 61

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	245000 K5.02
	Importance Rating	2.8

Main Turbine Generator and Auxiliary Systems

Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Turbine operation and limitations

Proposed Question: #31

A plant startup is in progress with the following:

- The Main Turbine is being rolled in accordance with N2-OP-21, Main Turbine System.
- Turbine speed is currently 1800 RPM.

Then...

- Annunciator 851140, TURBINE GENERATOR VIBRATION HIGH, alarms.

Which one of the following describes the requirements to trip the Turbine and break Main Condenser vacuum, in accordance with ARP 851140?

The Turbine should be tripped if any vibration level is ≥ __(1)__ mils for 15 minutes and Main Condenser vacuum should be broken if vibration levels are projected to go above __(2)__ mils following the Turbine trip.

	(1)	(2)
A.	7	12
B.	7	30
C.	10	12
D.	10	30

Proposed Answer: D

Explanation: ARP 851140 requires tripping the Turbine if vibrations are above 10 mils for 15 minutes. ARP 851140 requires breaking Main Condenser vacuum if vibrations are projected to exceed 30 mils following the Turbine trip.

A. Plausible – 7 mils is the annunciator setpoint. 12 mils is a vibration value that requires immediately tripping the Turbine.
B. Plausible – 7 mils is the annunciator setpoint.
C. Plausible – 12 mils is a vibration value that requires immediately tripping the Turbine.

Technical Reference(s): ARP 851140

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-245000-RBO-10

Question Source: Bank

Question History: 2013 Audit 31

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b) 10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	214000 K6.02
	Importance Rating	2.7

RPIS

Knowledge of the effect that a loss or malfunction of the following will have on the ROD POSITION INFORMATION SYSTEM : Position indication probe

Proposed Question: #32

The plant is operating at rated conditions with the following:

- The Position Indication Probe (PIP) connector for control rod 30-31 has disconnected.

The monitoring of which of the following Control Rod 30-31 indications will be affected based on the given conditions?

1. Control Rod position
2. Control Rod Temperature
3. Control Rod HCU pressure

A. 1, 2 and 3

B. 1 and 2 ONLY

C. 2 and 3 ONLY

D. 1 and 3 ONLY

Proposed Answer: B

Explanation: RPIS monitors both control rod position and temperature. A thermocouple is installed in the position indicator tube at the top to monitor drive temperature.

A. Plausible – The candidate may think that the position indicator probe is the only electrical connection to each control rod, so this connection would be the only method available to gain information about each control rod including temperature, pressure and position.

C. Plausible – The candidate may think that the position indicator probe is a probe used to determine pressure and temperature only. Typically probes are not used for position indication and the candidate may think the reed switches alone accomplish the control rod position indication.

D. Plausible – Control rod position is correct. The candidate may think that control rod Temperatures are measured using separate thermocouples and pressure instruments are part of the PIP.

Technical Reference(s): N2-OP-96, section B.3.0

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-201002C01, RBO-03

Question Source: Bank

Question History: LaSalle 2010 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215001 A1.01
	Importance Rating	2.8

Traversing In-core Probe

Ability to predict and/or monitor changes in parameters associated with operating the TRAVERSING IN-CORE PROBE controls including: Radiation levels: (Not-BWR1)

Proposed Question: #33

The plant is shutdown for refueling with the following:

- Personnel are working in the Drywell when a Traversing In- Core Probe (TIP) trace is initiated.

Which of the following hazards is created by this action?

A. High radiation exposures to personnel in the Drywell

B. Damage to the TIP machine if fuel movement is in progress

C. High radiation exposures to personnel on the Refuel Floor

D. Excessive drywell leakage when the TIP isolation valve is opened

Proposed Answer: A

Explanation: Per N2-OP-96, section F, "While operating TIP machines, high radiation hazards can exist in the following places:

- TIP Drive Room (Rx Bldg. El. 250)
- TIP Shield Room (Rx Bldg. El. 250)
- Drywell (If open for personnel access)"

B. Plausible – The TIP machine may be damaged if work was being performed on the LPRM strings, but fuel movement will have no effect.
C. Plausible – There would be no change of radiation on the Refuel Floor because of water shielding in the reactor and reactor cavity.
D. Plausible – The TIP system is a closed system there is no pathway for water to leak from the vessel into the drywell.

Technical Reference(s): N2-OP-94, section F

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-215001C01, RBO-10

Question Source: Bank

Question History: 2009 NRC 71

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)9

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	272000 A2.11
	Importance Rating	3.4

Radiation Monitoring

Ability to (a) predict the impacts of the following on the RADIATION MONITORING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Leakage and/or breaks from contaminated systems to atmosphere or to other process systems

Proposed Question: #34

The plant is operating at rated conditions with the following:

- The seals on the 'A' Reactor Water Cleanup Pump fail.
- Annunciator 851254, PROCESS AIRBORNE RADN MON ACTIVATED, is in alarm.
- DRMS indicates 2HVR*CAB-32A-1 and 2HVR*CAB-32B-1 are in alarm.
- The Reactor Building is isolated.
- The Emergency HVR Recirc Units are in service.

Which one of the following describes (1) the radiation monitor(s) that can be used to monitor Reactor Building airborne contamination levels and (2) the required operator actions for this condition in accordance with N2-ARP-851254?

A. (1) 2HVR-RE229
(2) Verify Group 9 Isolation has occurred

B. (1) 2HVR-RE229
(2) Verify GTS Train has initiated

C. (1) 2HVR*RE32A and B
(2) Verify Group 9 Isolation has occurred

D. (1) 2HVR*RE32A and B
(2) Verify GTS Train has initiated

Proposed Answer: B

Explanation: ARP 851254 states that with the reactor building isolated, HVR-229 is used to monitor RB Vent and the RE32's are not valid. The first step in the ARP is to verify the automatic response has occurred, which includes verifying GTS Train in operation.

A. Plausible – RE229 is the correct monitor. However, a group 9 isolation would be caused by the GTS effluent radiation monitor.
C. Plausible – RE32s are normally used to monitor below refuel floor RB ventilation radiation when the reactor building is not isolated.
D. Plausible – RE32s are normally used to monitor below refuel floor RB ventilation radiation when the reactor building is not isolated.

Technical Reference(s): N2-ARP-851254 Auto Response #1, Operator Action #3

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-272000C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	233000 A3.02
	Importance Rating	2.6

Fuel Pool Cooling/Cleanup

Ability to monitor automatic operations of the FUEL POOL COOLING AND CLEAN-UP including: Pump trip(s)

Proposed Question: #35

Which one of the following is the effect of a Loss of Coolant Accident (LOCA) coincident with a total Loss of Offsite Power (LOOP) on Fuel Pool temperatures and what actions are required to re-establish temperature control?

	Fuel Pool Temperature	To Re-establish Temperature Control
A.	Lowers	Manually close the Filter/Demin Bypass Valve, 2SFC-HIC113 only.
B.	Rises	Wait 70 seconds then restart the Fuel Pool cooling only mode using CCP to the SFC Heat Exchanger.
C.	Lowers	Manually close the Filter/Demin Bypass Valve, 2SFC-HIC113 and place the Filter Demins back in service.
D.	Rises	Wait 70 seconds then restart the Fuel Pool Cooling Pump and line up Service Water Backup Cooling to the SFC heat exchanger.

Proposed Answer: D

Explanation: On a Loss of Coolant Accident (LOCA), the running Pump will continue to run but a Pump cannot be started or restarted during the first 70 seconds after receipt of the LOCA signal. If a Loss of Offsite Power (LOOP) and a LOCA occur simultaneously, the running Pumps will trip. A Pump may be manually restarted after 70 seconds using N2-SOP-03. The Reactor Building Closed Loop Cooling Water supply to the SFC Heat Exchanger is also lost on a LOOP/LOCA and Service Water Backup Cooling to the SFC Heat Exchanger must be manually lined up by the operator.

A. Plausible - Fuel Pool temperature rises because the SFP pumps trip on a LOOP/LOCA and must be manually restarted. Filter/Demin Bypass Valve, 2SFC-HIC113 initially fails closed on a momentary loss of power. The valve will then return to its normal position

B. Plausible - To re-establish cooling Service Water Backup Cooling must be line-up.

C. Plausible - Fuel Pool temperature rises because the SFP pumps trip on a LOOP/LOCA and must be manually restarted. Filter/Demin Bypass Valve, 2SFC-HIC113 initially fails closed on a momentary loss of power. The valve will then return to its normal position because this valve is always controlled in manual. Placing the filter/demins in service will not affect fuel pool cooling temperatures.

Technical Reference(s): N2-SOP-38 attachment 1 section 1.5, 05ENS021 sheet 1 and 2, 05ENS022 sheet 1 and 2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-233000C01, RBO-02

Question Source: Bank

Question History: 2010 NRC 33

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	290001 A4.12
	Importance Rating	2.8

Secondary CTMT

Ability to manually operate and/or monitor in the control room: Reactor building differential pressure: Plant-Specific

Proposed Question: #36

The plant is operating at rated conditions with a normal Reactor Building ventilation configuration. Reactor Building differential pressure is -0.6 inches WG when the following sequence of events occur:

Time:

1300: Drywell pressure rises to 14 psig
1300: GTS Trains A and B automatically start
1310: GTS Train A is shutdown by placing Train A Initiation control switch in Auto After Stop
1315: GTS Train B Fan trips due to a motor fault and cannot be restarted

Which one of the following describes the impact of these events on Reactor Building differential pressure (DP) immediately after the GTS Train B Fan trip and the actions required, per N2-OP-61B, to restore RB differential pressure?

	Reactor Building DP	Actions Required
A.	Becomes less negative	GTS Train A must be manually restarted
B.	Becomes less negative	Confirm automatic restart of GTS Train A
C.	Remains the same	GTS Train A must be manually restarted
D.	Remains the same	Confirm automatic restart of GTS Train A

Proposed Answer: B

Explanation: GTS Train A restarts because the high Drywell pressure (>1.68 psig) initiation signal is still present and with no running GTS train RB differential pressure will degrade toward 0. When pressure reaches -0.25 inches WG, the manually shutdown train will automatically restart.

A. Plausible – Manual restart of Train A is NOT required to restore DP. The train will auto restart.
C. Plausible – With no running GTS train RB differential pressure will degrade toward 0 and not be maintained above -0.25 inches. The actions to defeat high drywell pressure interlocks and restart HVR are only authorized by EOP-SC. If dp never reached 0, then EOP-SC would not be entered.
D. Plausible – With no running GTS train RB differential pressure will degrade toward 0 and not be maintained above -0.25 inches.

Technical Reference(s): N2-OP-61B, section 'B' System Description

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-261000C01, RBO-05

Question Source: Bank

Question History: 2010 NRC 15

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)5

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202001 2.4.31
	Importance Rating	4.2

Recirculation

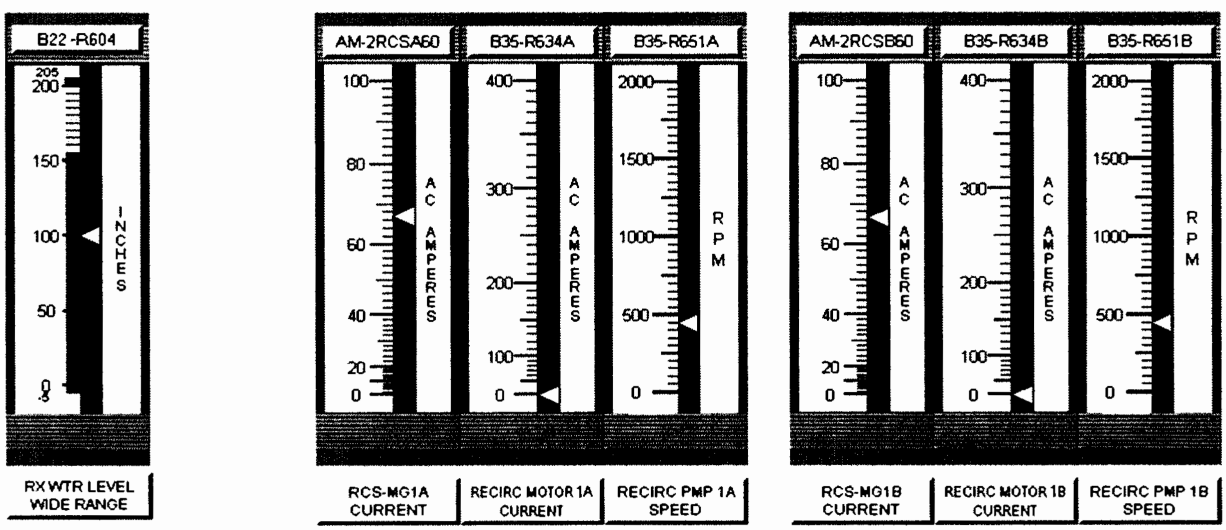
Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures

Proposed Question: #37

The plant is operating at rated conditions with the following:

- A spurious reactor scram occurs
- Several control rods fail to insert
- Reactor power is indicating 10% on the APRM's
- N2-EOP-C5 has been entered
- The crew has terminated and prevented all RPV injection except boron, CRD and RCIC and is in the process of lowering reactor water level to at least 100 inches in accordance with N2-EOP-C5
- As reactor water level is lowered reactor power has dropped to 2% on the APRM's

A few moments later, the following indications are observed:



Which one of the following describes the required action for the reactor recirculation pumps and the correct method to perform the action?

	Action	Method
A.	Downshift the RCS Pumps	Place RECIRC PUMP 1A MOTOR A BRKR 5A & 5B control switches to "PTL/Stop" position and release

B.	Trip the RCS Pumps to off	Place LOW FREQ MG SET DRIVE MOT BRKR 1A & 1B control switches to "Stop" position and release
C.	Downshift the RCS Pumps	Place LOW FREQ MG SET DRIVE MOT BRKR 1A & 1B control switches to "Transfer to MG" position and release
D.	Trip the RCS Pumps to off	Place RECIRC PUMP 1A MOTOR A BRKR 5A & 5B control switches to "PTL/Stop" position and release

Proposed Answer: B

Explanation: In the given, the wide range level indicator shows RPV level below 108.8 inches (~100 inches) and the RCS pump speed and current indicators show that the recirc. pumps are running in slow speed. The recirc. pumps should have downshifted to slow (low frequency MG) at 159.3 inches (which is where they are currently) and they both should have tripped to off when wide range level reached 108.8 inches. In this case, per N2-EOP-C5 step L-9, the RCS pumps should be verified tripped.

1. Lower RPV water level by terminating and preventing all RPV injection except boron, CRD, and RCIC.
   - Ignore any power or level oscillations.
2. Let level drop at least to 100 in.
   - Verify RCS pumps downshift below 159.3 in.
   - Verify RCS pumps trip below 108.8 in.
3. IF...... suppression pool temperature is above 110°F,
   THEN...let level continue to drop until:
   - Power drops below 4%
     OR
   - Level drops to -14 in. (Fig Z)
     OR
   - All SRVs stay closed and drywell pressure stays below 1.68 psig
4. Record the final level: ________________

L-9

The correct method to trip the RCS pumps is contained in N2-EOP-HC, attachment 1 section 2.0.

2.0	Trip Reactor Recirc Pumps
2.1	IF pumps are running in fast speed, place the following control switches to PTL AND STOP release:
	• RECIRC PUMP 1A MOTOR A BRKR 5A .... ☐
	• RECIRC PUMP 1B MOTOR A BRKR 5B .... ☐
2.2	IF pumps are running in slow speed, place the following control switches to STOP AND release:
	• LOW FREQ MG SET DRIVE MOT BRKR 1A .... ☐
	• LOW FREQ MG SET DRIVE MOT BRKR 1B .... ☐

A. Plausible – The candidate could believe that based on the given indication, that the RCS pumps are running in fast speed and if this is the case the correct action per N2-EOP-HC would be to place RECIRC PUMP 1A MOTOR A BRKR 5A & 5B control switches to "PTL/Stop" position and release.
C. Plausible – The candidate could believe that based on the given indication, that the RCS pumps are running in fast speed and believe that the action would be to initiate the downshift by placing the LOW FREQ MG SET DRIVE MOT BRKR 1A & 1B control switches to "Transfer to MG" position and release.
D. Plausible – The first part of the distracter is correct, however with the RCS pumps running in slow speed, the correct action per N2-EOP-HC would be to place LOW FREQ MG SET DRIVE MOT BRKR 1A & 1B control switches to "Stop" position and release. Using the 5A & 5B breakers would not work since the RCS pump motors are running using the 1A(B) and 2A(B) breakers.

Technical Reference(s): N2-EOP-HC, attachment 1 section 2.0, N2-EOP-C5, step L-9

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01, Objective # N2-EOP-C5-CE-02

Question History: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201001 2.1.23
	Importance Rating	4.3

CRD Hydraulic

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: #38

The plant is operating at rated conditions with the following:

- Control rod 14-35 is currently at position 12 and has been determined to be mechanically stuck
- No other HCU's are isolated

Which one of the following describes the preferred isolation action per N2-OP-30, for control rod 14-35?

	Type of Isolation	Cooling Water Status
A.	Electrical	with cooling water
B.	Electrical	without cooling water
C.	Hydraulic	with cooling water
D.	Hydraulic	without cooling water

Proposed Answer: C

Explanation: For stuck control rods it is required that a hydraulic isolation be conducted. N2-OP-30, section F.8.2 states "A withdrawn control rod that is stuck should be disarmed hydraulically, preferably with cooling water flow (hydraulically disarming a withdrawn control rod that is stuck is required per Technical Specification 3.1.3)". Also from Tech Spec bases "The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram and normal insert and withdraw pressure prevents damage to the CRDM or reactor internals. The control rod isolation method should also ensure cooling water to the CRD is maintained."

A. Plausible – The operator could mistakenly think that either electric or hydraulic isolation is appropriate for the given condition.
B. Plausible – The operator could mistakenly think that either electric or hydraulic isolation is appropriate for the given condition. Additionally, N2-OP-30 states "Isolating an HCU without cooling water should be performed when reactor water temperature is less than 200°F, or isolating an HCU without cooling water when reactor water temperature is greater than 200°F should be minimized (less than one shift duration) or seal degradation could occur."
D. Plausible – Hydraulic isolation is the correct isolation however, cooling water should not be isolated. N2-OP-30 states "Isolating an HCU without cooling water should be performed when reactor water temperature is less than 200°F, or isolating an HCU without cooling water when reactor water temperature is greater than 200°F should be minimized (less than one shift duration) or seal degradation could occur."

Technical Reference(s): N2-OP-30 section F.8.2 Note

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-201001C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)6

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK1.01
	Importance Rating	3.5

Partial or Total Loss of CCW / 8

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Effects on component/system operations

Proposed Question: #39

The plant is operating at 70% power with the following:

- A complete loss of reactor building closed loop cooling water requires a reactor scram to be inserted.

Which of the following describes the reason for the required reactor scram?

The scram is directed because of the loss of cooling to the .....

A. Control Rod Drive (RDS) Pump seals.

B. Residual Heat Removal (RHR) Pump seal coolers.

C. Reactor Recirculation (RCS) Pump motors and seals.

D. Reactor Water Clean-Up (WCS) Non-Regenerative Heat Exchangers.

Proposed Answer: C

Explanation: Per USAR Table 9.2-3, the Reactor recirculation motor winding coolers, motor bearing coolers, and pump seal coolers have a combined heat load of 3.7×10^6 BTU/hr. Without cooling water, this heat load will result in rapidly rising temperatures on the recirculation pump motor windings, motor bearings and recirculation pump seals. Per N2-OP-29, section H.3.0 "The loss of CCP to the pump motor bearing or winding cooler will be a limiting factor in continuing pump operation." If the recirculation pumps have to be tripped at power, a reactor scram will be required.

A. Plausible – Loss of cooling to the Control Rod Drive (CRD) Pump seals will result in eventual damage to the RDS pump seals, but would not require the reactor to be scrammed. However the operator may think that the loss of CRD Pump seals will result in a loss of CRD pumps which would require a reactor scram.
B. Plausible – Loss of cooling to the Residual Heat Removal (RHR) Pump seal coolers will not require the reactor to be scrammed since the residual heat removal pumps are not required to be running to remain at power.
D. Plausible – Loss of cooling to the Reactor Water Clean-Up (RWCU) Non-Regenerative Heat Exchangers will not require the reactor to be scrammed, but instead would require RWCU be removed from service. This is plausible since the operator may think that since this is the largest heat load at power for the CCP system that loss of cooling would require a scram.

Technical Reference(s): Unit #2 USAR Table 9.2-3, N2-OP-29 section H.3.0 note, N2-SOP-13 flowchart

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-208000C01, RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)6

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AK1.02
	Importance Rating	3.3

Generator Voltage and Electric Grid Disturbances

Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Over-excitation.

Proposed Question: #40

The plant is operating at rated conditions with the following:

- A grid disturbance has been experienced.
- Main Generator reactive load is 300 MVARs to the Bus.
- Main Generator hydrogen pressure is 76 psig.

Which one of the following is correct for these conditions, in accordance with N2-OP-27, "Generator H2 and CO2 System" and N2-OP-68, "Main Generator"?

A. Main Generator parameters are satisfactory.

B. Raise Main Generator hydrogen pressure.

C. With Power Control concurrence, adjust the AC Voltage Regulator control switch to lower MVARs.

D. With Power Control concurrence, either adjust the AC Voltage Regulator control switch to lower MVARs or raise Main Generator hydrogen pressure.

Proposed Answer: C

Explanation: Explanation: N2-OP-27 directs Main Generator hydrogen pressure to be controlled 75-77 psig. N2-OP-68 limits Main Generator reactive load to a maximum of 233 MVARs to the Bus when at this hydrogen pressure. Therefore, reactive load must be lowered using the AC Voltage Regulator.

A. Plausible – 300 MVARs is well within the scale of the N2-OP-68 Estimated Capability Curve, however it is NOT within the additional caution limiting reactive load to 233 MVARs to the Bus.
B. Plausible – Reactive loading limitations do rise as hydrogen pressure is raised, however operation is already within the N2-OP-27 guidance for hydrogen pressure.
D. Plausible – Reactive loading limitations do rise as hydrogen pressure is raised, however operation is already within the N2-OP-27 guidance for hydrogen pressure.

Technical Reference(s): N2-OP-27 F1.1, N2-OP-68 Section F.1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-245001C01, RBO-10

Question Source: NMP Audit Exam Bank Question #57

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AK1.03
	Importance Rating	3.7

SCRAM / 1

Knowledge of the operational implications of the following concepts as they apply to SCRAM : Reactivity control

Proposed Question: #41

The plant is operating at rated conditions with the following:

- A manual reactor scram is performed
- An RPS circuit failure has occurred which fails to energize time delay relays associated with Scram reset prevention

What are the operational implications of this failure if the SCRAM were to be manually reset before 10 seconds has elapsed?

A. All control rods may fail to fully insert.

B. The Scram Air header may not re-pressurize.

C. Reactor water level may not recover above the scram setpoint.

D. The Scram Discharge Volume vent and drain valves may not close.

Proposed Answer: A

Explanation: Per N2-OP-97, The purpose of the ten second time delay is to allow sufficient time for the rods to scram prior to allowing for the scram to be reset.

B. Plausible – The candidate could think that with time delay relays K16A through D failing to energize that the scram logic would not allow the scram valves to reset and would therefore prevent the Scram Air header from re-pressurizing.
C. Plausible – The candidate may think that with time delay relays K16A through D failing to energize that the low RPV water level contacts in the RPS circuitry would not allow the FWLC system to energize the setpoint setdown circuitry while below L-3 (159.3").
D. Plausible – The candidate may think that with the time delay relays failing to energize that the SDV vent and drains may not receive a close signal directly from the manual; Scram.

Technical Reference(s): N2-OP-97, P&L 6.0, GE print 807E166TY

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-212000C01, RBO-05

Question Source: NMP NRC Bank question RO #41

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)6

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AK2.05
	Importance Rating	2.7

Loss of Shutdown Cooling / 4

Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: Fuel pool cooling and cleanup system

Proposed Question: #42

The plant is in Mode 5 with the following:

- The Reactor Cavity is flooded and the gates are removed
- A loss of shutdown cooling has occurred
- The crew has entered N2-SOP-31R, Refueling Operations Alternate Shutdown Cooling

In accordance with N2-SOP-31R which one of the following describes the system operating configuration when the Residual Heat Removal System (RHS) is aligned to the Spent Fuel Pool Cooling (SFC) Assist Mode?

A. SFC system is required to be shutdown and isolated

B. RHS pump placed in series with the SFC pump thereby raising SFC system flow

C. RHS suction is directly from the fuel pool and returns through the RHS diffusers

D. One loop of RHS, and the opposite loop of SFC are placed in service for cooling

Proposed Answer: D

Explanation: Connections between the Spent Fuel Pool Cooling (SFC) System and the "A" and "B" RHR loop permit the "A" or "B" RHR heat exchanger to cool the spent fuel pool. Isolation valves are provided in each line in both the RHR and SFC systems. This RHR feature is used whenever an abnormally high heat load is placed on spent fuel pool cooling or a loss of shutdown cooling occurs. The arrangement is established and secured by operator action. Operation of RHS in SFC assist mode requires that opposite loops be utilized. For example if RHS A loop is the only available loop of RHS, then SFC B must be the SFC loop in operation. This is due to RHS A returning to SFC A loop and RHS B returning to SFC B loop.

A. Plausible – The candidate may not recall that SFC piping is necessary for RHS to operate in alternate shutdown cooling. Additionally during the loss of shutdown cooling RHS will be operated in parallel with one SFC subsystem to provide maximum cooling.
B. Plausible – The candidate may not recall the system lineup and think that the RHS pump can be placed in series with the SFC pump.
C. Plausible – RHS returns to the SFC loop returns.

Technical Reference(s): N2-SOP-31R, attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP31RC01, Objective # N2-SOP-31R-CE-02

Question Source: NMP Audit Exam Bank Question RO #44

Question History: Bank # 33392

Question Cognitive Level: Memory Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AK2.01
	Importance Rating	2.6

Plant Fire On-site / 8

Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors, detectors and valves

Proposed Question: #43

The plant is operating at rated conditions with the following:

- A fire is detected for a FPW pre-action area in the diesel generator building
- Motor Driven Fire Pump, 2FPW-P2, is running maintaining system pressure at 150 psig
- Diesel-Driven Fire Pump, 2FPW-P1, is in standby with its control switch in Auto
- A fault on 2NNS-SWG013 occurs and the bus de-energizes

Which one of the following describes the final configuration of the Fire Protection Water System?

	Motor-Driven Fire Pump (2FPW-P2)	Diesel-Driven Fire Pump (2FPW-P1)
A.	Remains in operation	Remains in standby
B.	Remains in operation	Starts on loss of power to 2ATX-XS3
C.	Trips	Starts on low system pressure
D.	Trips	Starts on loss of power to 2ATX-XS3

Proposed Answer: A

Explanation: Motor-Driven Fire Pump, 2FPW-P2 is supplied from NNS-SWG012 which in turn is normally fed from NNS-SWG011. NNS-SWG011 is not impacted by the NNS-SWG013 fault and the Motor-Driven Pump remains running. Since the Motor-Driven Pump remains running, pressure is maintained and the diesel driven pump remains in standby.

This meets the K/A since the question hinges on the concept that the pre-action valve will stroke open on a fire detected signal.

B. Plausible – The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the candidate could confuse the normal vs. alternate power supply to NNS-SWG012 and think that power would be lost to the electric fire pump and the DG fire pump would start due to the loss of logic signals to the DG Fire pump start circuitry.
C. Plausible – Both the first and second part of the distracter are incorrect, but plausible. For the first part of the distracter, NNS-SWG012 can also be fed from NNS-SWG013. If the candidate thinks that NNS-SWG012 is normally fed from NNS-SWG013 and that NNS-SWG012 has an auto transfer feature (it does not) this might be construed as the response following the initial de-energizing of the pump and subsequent re-energizing by NNS-SWG011. For the second part of the distracter the candidate could think that the pre-action piping volume rise and the trip of the motor driven pump would cause header pressure to lower and the DG fire pump to start.
D. Plausible – Both the first and second part of the distracter are incorrect, but plausible. For the first part of the distracter, NNS-SWG012 can also be fed from NNS-SWG013. If the candidate thinks that NNS-SWG012 is normally fed from NNS-SWG013 and that NNS-SWG012 has an auto transfer feature (it does not) this might be construed as the response following the initial de-energizing of the pump and subsequent re-energizing by NNS-SWG011. For the second part of the distracter, the candidate could confuse the normal vs. alternate power supply to NNS-SWG012 and think that power would be lost to the electric fire pump and the DG fire pump would start.

Technical Reference(s): EE-MO001A

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-286000C01, RBO-04

Question Source: NMP Audit Exam Bank Question RO #28

Question History: 2010 Audit Bank # 17778 – audit 2008

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EK2.06
	Importance Rating	3.5

Suppression Pool High Water Temp. / 5

Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Suppression pool level

Proposed Question: #44

The plant is operating at rated conditions with the following:

- Suppression Pool Water Level is slowly lowering and is currently at 196 ft.
- N2-EOP-PC has been entered
- Actions are being taken to fill the suppression pool using the HPCS pump
- Suppression chamber airspace is currently 30°F warmer than suppression pool water temperature

Which one of the following describes the indicated average suppression pool temperature reading as compared to the actual average suppression pool temperature?

A. Lower, due to CMS temperature elements becoming uncovered

B. Lower, due to CMS temperature element submergence

C. Higher, due to CMS temperature elements becoming uncovered

D. Higher, due to CMS temperature element submergence

Proposed Answer: C

Explanation: Suppression Pool Temperature Detector Locations are listed in Detail "X" of N2-EOP-PC. When suppression pool level is below elevation 197 ft. all of the temperature elements will be uncovered and will not respond to suppression pool temperature changes.

A. Plausible – Temperature would not indicate lower since the suppression chamber airspace was given to be 30°F warmer than suppression pool water temperature so as the temperature elements uncover, indicated suppression pool temperature would get higher because the suppression pool temperature detectors would indicate suppression chamber airspace temperature.
B. Plausible – Temperature would not indicate lower since the suppression chamber airspace was given to be 30°F warmer than suppression pool water temperature so as the temperature elements uncover, indicated suppression pool temperature would get higher because the suppression pool temperature detectors would indicate suppression chamber airspace temperature.
D. Plausible – Temperature would indicate higher since the suppression chamber airspace was given to be 30°F warmer than suppression pool water temperature. However, the temperature elements will uncover, because they are located at elevation 199 ft & 197 ft.

Technical Reference(s): N2-EOP-PC, Detail "X", NER-2M-039, Rev 08.00 step SPT-6 page 3-106

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-223001C01, RBO-03

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)5

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK3.08
	Importance Rating	3.6

SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1

Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : ATWS circuitry: Plant-Specific

Proposed Question: #45

The plant is operating at rated conditions with the following:

- APRM #1 is bypassed due to a failed computer module in the APRM chassis
- A spurious closure of all MSIV's occurs
- Reactor Pressure peaked at 1071 psig
- RPS fails to trip automatically
- RRCS automatically initiates and the operator-at-the-controls manually initiates RRCS per US direction
- ARI functions, but limited control rod movement occurs
- 3 minutes following the start of the transient reactor power remains greater than 4%

Which one of the following describes the plant response to this transient?

A. SLS is injecting only

B. RCS pumps have tripped to off and SLS is injecting only

C. RCS pumps are operating in slow speed, FWS runback has occurred and SLS is injecting

D. RCS pumps have tripped to off, FWS runback has occurred and SLS is injecting

Proposed Answer: D

Explanation: Per N2-OP-36B, High pressure alone will:

- Initiate an ARI scram,
- Immediately trip the 60 Hz circuit breakers and initiate transfer of the Recirculation Pumps to LFMG (low speed) operation.
- After 25 seconds, if the APRM channels are not downscale or are INOP, the RRCS System trips the 15 Hz circuit breakers to complete the RPT.
- It also initiates a feedwater runback and the feedwater min flow valves fail open which are both sealed in for 25 seconds.
- After an additional 73 seconds with the APRM channels still not downscale or are INOP, the RRCS System initiates the SLS System which isolates the WCS System.

A. Plausible – The candidate could think that the manual RRCS initiation that was performed would only result in an immediate ARI scram, then after 98 seconds, if the APRM channels are not downscale or INOP, the SLS System would initiate and that manual initiation does not cause an RPT or feedwater runback.

B. Plausible – The candidate could confuse the plant response to that of a low reactor water level RRCS response.

C. Plausible – The candidate could think that the high pressure alone will, in addition to an ARI scram, immediately trip the 60 Hz circuit breakers and initiate transfer of the Recirculation Pumps to LFMG (low speed) operation and forget that after an additional 25 seconds, if the APRM channels are not downscale or are INOP, the RRCS System trips the 15 Hz circuit breakers to complete the RPT.

Technical Reference(s): N2-OP-36B

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-294008C01, RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)6

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EK3.03
	Importance Rating	3.7

High Off-site Release Rate / 9

Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Control room ventilation isolation: Plant-Specific

Proposed Question: #46

The plant is operating at rated conditions with the following:

- A plant transient has occurred that has resulted in a high off-site release rate.

Which one of the following identifies the source(s) that will cause an isolation of the control room ventilation system and the reason for the isolation?

	Source(s)	To protect personnel in the _______ from receiving excessive radiation exposure from the outside air source
A.	Either 2HVC*RE18A **OR** B only	control & relay rooms
B.	Both 2HVC*RE18A **AND** C	control & relay rooms
C.	Both 2HVC*RE18A **AND** C	control, relay & remote shutdown rooms
D.	Either 2HVC*RE18A **OR** B only	control, relay & remote shutdown rooms

Proposed Answer: B

Explanation: USAR section 7.3.1.1.10 and N2-OP-53A section H.1.0 state that the control building special filter train will automatically start when either 2HVC*RE18A & C OR 2HVC*RE18B & D exceed their alarm setpoint or have failed. The reason for this isolation is discussed in USAR section 9.4.1.2.2. It is to protect and preserve control room habitability for the control room staff during high supply air radiation levels.

A. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part of the distracter, the candidate could recall that either of the 'A' or "C" radiation elements in alarm would cause the isolation of control building ventilation.
C. Plausible – The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the candidate could think that the control room and "associated areas" as discussed in the USAR include the remote shutdown rooms.
D. Plausible – Both the first and second part of the distracter is incorrect, but plausible. For the first part of the distracter, the candidate could recall that either of the 'A' or "C" radiation elements in alarm would cause the isolation of control building ventilation. For the second part of the distracter, the candidate could think that the control room and "associated areas" as discussed in the USAR include the remote shutdown rooms.

Technical Reference(s): USAR section 7.3.1.1.10 and N2-OP-53A section H.1.0, USAR section 9.4.1.2.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-288003C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EK3.02
	Importance Rating	3.9

High Reactor Pressure / 3

Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE : Recirculation pump trip: Plant-Specific

Proposed Question: #47

The plant is operating at rated conditions with the following:

- A transient occurs that causes reactor pressure to rise to 1080 psig.
- Both Reactor Recirculation Pumps have tripped off.

Which one of the following describes the reason (Bases) for the trip of the Recirculation pumps?

A. It adds additional negative reactivity by increasing the voiding in the core

B. It increases core inlet subcooling which reduces Reactor power

C. It moves the boiling boundary up the fuel channel which adds negative reactivity

D. It overcomes the rise in power caused by the moderator temperature increase due to the rising RPV pressure.

Proposed Answer: A

Explanation: Per Nine Mile Point Unit 2 USAR, section 15.8.3.3, the purpose of the RPT is to reduce core flow and create core voids to decrease power generation, thus limiting any power or pressure disturbance.

B. Plausible – The candidate could think that by tripping the recirculation pumps, less flow will be pushed through the vessel. This lower flow would allow the water to be cooler since it is not being forced through the core at such a high flow rate. The candidate could then think that with this increase in core inlet subcooling the temperature and enthalpy of the water would be much lower and would require more energy from the core to heat up. This cooler water would then cause reactor power to lower.
C. Plausible – The candidate could think that when the boiling boundary moves the up the fuel channel that boiling is occurring closer to the core inlet. The candidate could confuse the reference point of moving up the fuel bundle. If this were the case more of the core would be voided which would add negative reactivity.
D. Plausible – The candidate could mistakenly think that reactor pressure is rising since it was stated in the stem of the question. This rise in reactor pressure would then cause a rise in moderator temperature. This rise reactor pressure would cause actual bundle power to rise. So the candidate could believe that tripping the recirculation pumps would counteract the affects of this mechanism.

Technical Reference(s): Unit 2 USAR, section 15.8.3.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-202001C01, RBO-05

Question Source: Columbia Bank Question #12 October 2009

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)1 & (b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AA1.02
	Importance Rating	2.9

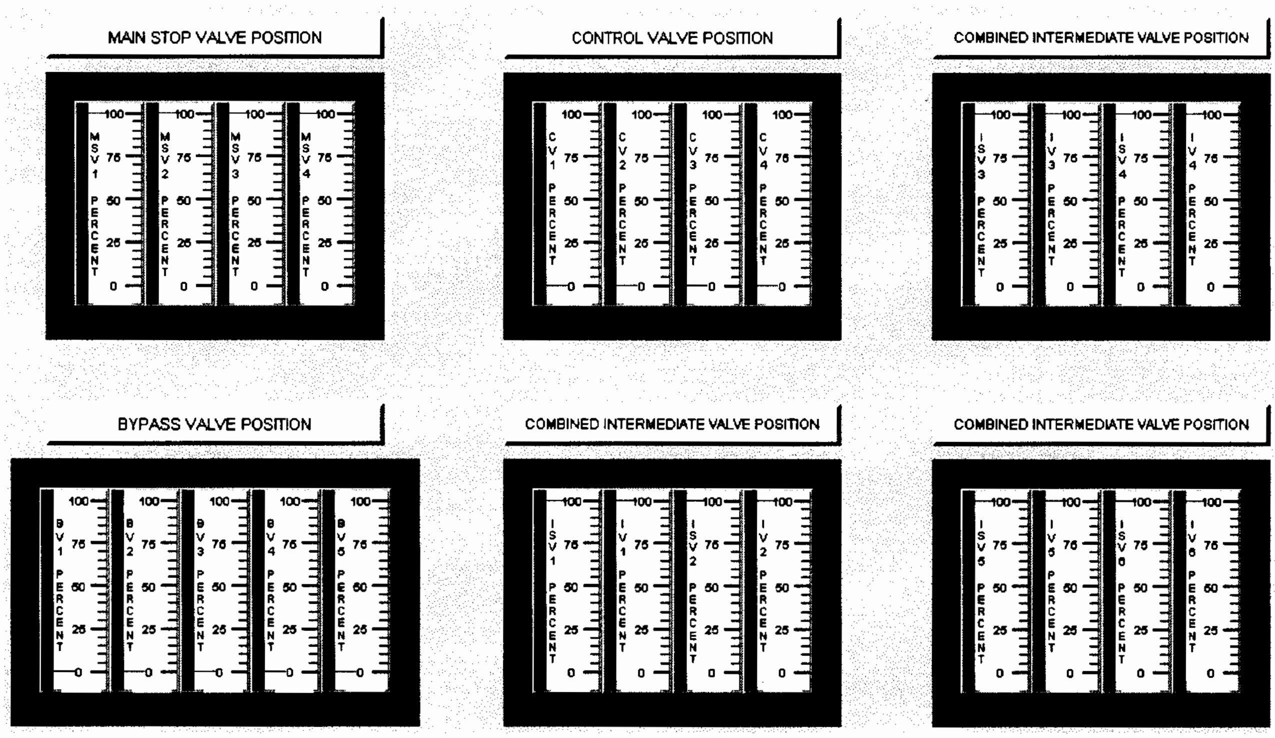
Control Room Abandonment / 7

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : Reactor/turbine pressure regulating system

Proposed Question: #48

The plant is operating at rated conditions with the following:

- A control room evacuation is required
- N2-SOP-78 requires the main turbine tripped
- The main turbine is tripped from 2CEC*PNL852
- The following indications are present after the turbine trip pushbuttons are depressed:



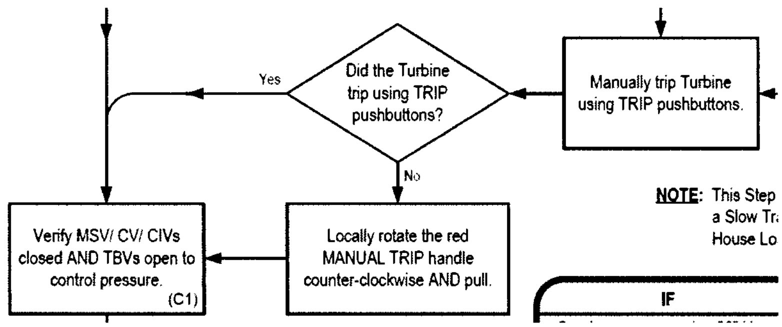
Which one of the following describes the current main turbine status and the next required action?

	Turbine tripped	Required Action
A.	Yes	Initiate condenser neck spray
B.	Yes	Verify Turbine Bypass Valves are controlling pressure

C.	No	Fast close the outboard MSIV's only
D.	No	Locally trip the main turbine from the front standard

Proposed Answer: D

Explanation: Based on the indications provided, the main turbine is not tripped. For the given panel indications, 100% means full open and conversely 0% means full closed. The main stop valves and combined intermediate valves all show 100% open indicating that the turbine is not tripped. The turbine control valves indicate closed since the mode switch has been placed in shutdown and reactor is no longer making power. Per N2-OP-78, the Operator at the controls (OATC) is required to verify the main turbine tripped. N2-SOP-78 Upper Turbine building operator is directed to trip the main turbine from the front standard. N2-OP-21 also specifies the following:



Therefore the next required step would be to trip the main turbine locally at the front standard using the manual trip handle.

A. Plausible –The candidate could think that since the turbine control valves indicate 0% that the main turbine is tripped and an appropriate action to be taken per N2-SOP-21 would be to initiate condenser neck spray.
B. Plausible – The candidate could think that since the turbine control valves indicate 0% that the main turbine is tripped and with the TBVs indicating 0% the action to verify that the Turbine Bypass Valves are controlling pressure could be chosen by the candidate.
C. Plausible – The first part of the distracter is true, and the candidate could think that closing the MSIV's would be an appropriate action since it would isolate steam flow to the main turbine.

Technical Reference(s): USAR 7.2.1.2.2, USAR Table 14.2-231, N2-SOP-78 page 6 flow chart, N2-SOP-21

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP21C01, Objective # N2-SOP-21-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AA1.04
	Importance Rating	2.7

Main Turbine Generator Trip / 3

Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Main generator controls

Proposed Question: #49

The plant is operating at rated conditions with the following:

- The reactor was manually scrammed
- The main turbine has tripped
- MWe is indicating approximately zero
- The main generator has failed to trip
- Main condenser vacuum is 28 inches Hg

Which one of the following describes the first action to be performed to address the failure of the main generator to trip in accordance with N2-SOP-21, Turbine Trip?

A. Immediately open R230 and R925 from 2CEC*PNL852 and then verify 41E and 41M open.

B. Transfer house loads from the normal station transformer to the reserve transformer.

C. Verify the main generator voltage regulator is in auto, lower MVAR to near zero, then open 345 KV motor operated disconnect switch MDS1/233N from 2CEC*PNL852.

D. Shift the main generator voltage regulator to manual, lower MVAR to near zero and verify that the main generator trips on reverse power.

Proposed Answer: D

Explanation:

NOTE: This Step will cause a Slow Transfer of House Loads

IF	THEN
Condenser vacuum is <26" Hg AND MWe is about 0 for 90 sec, OR Condenser vacuum is ≥26" Hg AND MWe is about 0 for 5 min,	1. Open R230 AND R925. 2. Verify 41-M AND 41-E are open. 3. Verify 2YXC-MDS1, 345 KV MODS MDS1/233N, is open

1. Using MAIN GENERATOR DC VOLTAGE REGULATOR control switch, null the Main Generator DC Voltage Regulator.
2. Place MANUAL-AUTO SELECT switch in MAN.
3. Using MAIN GENERATOR DC VOLTAGE REGULATOR control switch, reduce MAIN GENERATOR MEGAVARS to zero.
4. Confirm Main Generator field amps are <2200 DC amps.
5. IF the Generator fails to trip on reverse power, transfer House Loads per Attachment 1.

Per N2-SOP-21, section 5.5, Following a Turbine trip, trip of the Main Generator should be verified. The Main Generator may operate in a motoring condition (Generator gross MWe reaching zero) within the following guidelines:

Limits for Motoring the Generator With Exciter Field Breaker Closed	
Motor Time	Plant Condition
90 Seconds	Condenser Vacuum < 26" Hg
5 Minutes	Condenser Vacuum ≥ 26" Hg

If Main Generator output breakers R925 and R230 have not tripped shortly following the Main Generator reaching a motoring condition, initiation of a Main Generator breaker reverse power trip will be attempted by shifting Main Generator Voltage Regulator control to Manual and lowering generator MVARs to near zero with Generator load near zero. If the Main Generator fails to trip on reverse power after adjusting MVARs to near zero, breakers R925 and R230 must be manually opened to avoid damage to the Main Turbine (overheating of exhaust hood and last stage buckets), using the above table for guidance on allowable motoring durations. If the Main Generator Output Breakers (R230 and R925) must be opened manually, a Slow Transfer of Normal Station Service will occur.

A. Plausible –One of the secondary actions is to manually open R230 and R925 if the reverse power trip fails
B. Plausible –The transfer of house loads is a required action if the main generator fails to trip on reverse power
C. Plausible –The action is to reduce MVAR to zero, however the main generator voltage regulator will have to be shifted to manual to accomplish this action. Also, the

preferred way to trip the main generator under the given circumstances is to have it trip on reverse power.

Technical Reference(s): N2-SOP-21, section 5.5

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP21C01, Objective # N2-SOP-21-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AA1.01
	Importance Rating	3.3

Refueling Acc Cooling Mode / 8

Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Secondary containment ventilation

Proposed Question: #50

The plant is operating at rated conditions with the following:

- A small leak occurs in the spent fuel pool
- Spent fuel pool level begins to lower
- N2-SOP-39, "Refuel Floor Events" has been entered
- 2HVR*CAB14A-1, "Reactor Building Above Refueling Floor 'A' Gaseous Radiation Monitor" level reaches yellow "Alert" level on DRMS
- 2HVR*CAB14B-1 "Reactor Building Above Refueling Floor 'B' Gaseous Radiation Monitor" level reaches red "Alarm" on DRMS
- No reactor building ventilation control component failures occurred

Which one of the following describes the expected reactor building ventilation indications on 2CEC*PNL870 & 871 in response to the event without any operator action?

	2HVR*AOD 1, 9, 10 closed	2HVR*UC413(s) running	GTS Train(s) running
A.	'A' & 'B'	'A' & 'B'	'A' & 'B'
B.	'A' only	'A' only	'A' only
C.	'A' & 'B'	'B' only	'A' & 'B'
D.	'B' only	'B' only	'B' only

Proposed Answer: C

Explanation: Per N2-OP-52, section H.1.0 upon receipt of a reactor building above/below refueling floor exhaust radiation high alarm (2HVR*CAB14A-1, 2HVR*CAB14B-1, 2HVR*CAB32A-1 or 2HVR*CAB32B-1, the following occurs:

- 2HVR*UC413B starts (413B is the lead unit). Only one Unit Cooler should start unless a control component failure has occurred which in the given it states no control component failures occurred. (OP-52, Precaution & Limitation 18.0)
- All reactor building isolation dampers AOD 1A & B, 9A & B, and 10A & B close
- Both standby gas treatment system filter trains start

A. Plausible - The candidate may think that the gaseous channel reaching the "red" (alarm) level on either HVR*CAB14 will cause both HVR*UC413's to start along with both 'A' & 'B' dampers to close and both GTS trains to start.

B. Plausible - The candidate may think that the gaseous channel "yellow" (alert) level is the trip setpoint. The candidate may think that the gaseous channel reaching the "yellow" (alert) level on only HVR*CAB14A will cause only the 'A' dampers to close, only the 'A' GTS train to start and only the HVR*UC413A to start.

D. Plausible - The candidate may think that the gaseous channel reaching the "red" (alarm) level on only HVR*CAB14B will cause only the 'B' dampers to close, only the 'B' GTS train to start and only the HVR*UC413B to start.

Technical Reference(s): N2-OP-52, section H.1.0, N2-ARP-851254

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-288001C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EA2.04
	Importance Rating	3.5

Low Suppression Pool Water Level / 5

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Drywell/ suppression chamber differential pressure: Mark-I&II

Proposed Question: #51

The plant is operating at rated conditions with the following:

- A leak in the suppression pool has occurred
- Suppression pool level is lowering

Which one of the following describes the effect on drywell (DW) to suppression chamber (SC) differential pressure (D/P), as suppression pool water level lowers?

A. DW to SC D/P rises because DW pressure remains relatively unchanged, but suppression chamber pressure lowers as the suppression chamber gas volume rises due to the loss of suppression pool water inventory.

B. DW to SC D/P lowers because DW pressure remains relatively unchanged, but suppression chamber pressure rises due to expansion of the air and non-condensable gases.

C. DW to SC D/P remains the same because the drywell to suppression chamber vacuum breakers open to equalize pressure as suppression chamber pressure rises due to expansion of the air and non-condensable gases. So, although DW and SC pressures are changing, the D/P between them remains the same.

D. No effect because the loss of suppression pool inventory changes the suppression chamber gas volume, however the change is negligible and therefore does not affect DW to SC D/P. So, neither DW nor SC pressures change.

Proposed Answer: A

Explanation: As suppression pool level lowers, the effective volume of the air space above the water level increases. This air space volume increase results in a drop in measured suppression chamber pressure. 2CMS*PI168 and 2CMS*PI7A both measure suppression chamber pressure at the 224 ft elevation in the containment (Through containment penetration z337-1). This is above the normal suppression pool water level of ~200ft. Drywell pressure remains relatively unchanged due to the containment design while the pressure suppression pressure function is intact. With suppression chamber pressure lowering and drywell pressure remaining relatively unchanged, the DW to SC D/P will rise.

B. Plausible – The candidate may think that when suppression pool water level lowers the air space gas will expand causing D/P to lower.

C. Plausible – The candidate may think that the Drywell to Suppression chamber vacuum breakers will open to equalize the pressure between the two chambers.

D. Plausible – The candidate may think that the loss of Suppression pool level will not effect suppression chamber pressure at all.

Technical Reference(s): USAR, section 6.2.5.2.2, Table 6.2-3, PID 82B

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-223001C01, RBO-11

Question Source: New

Question History: New Question

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)9

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AA2.06
	Importance Rating	3.2

Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Nuclear boiler instrumentation

Proposed Question: #52

The plant is operating at 70% power with the following:

- A nozzle (rams head) separates from its associated riser on Reactor Recirculation Loop 'A'

Based on this failure, which one of the following describes what the reactor operator would observe?

A. A rise in indicated Total Core Flow on 2CEC*PNL 603 recorder

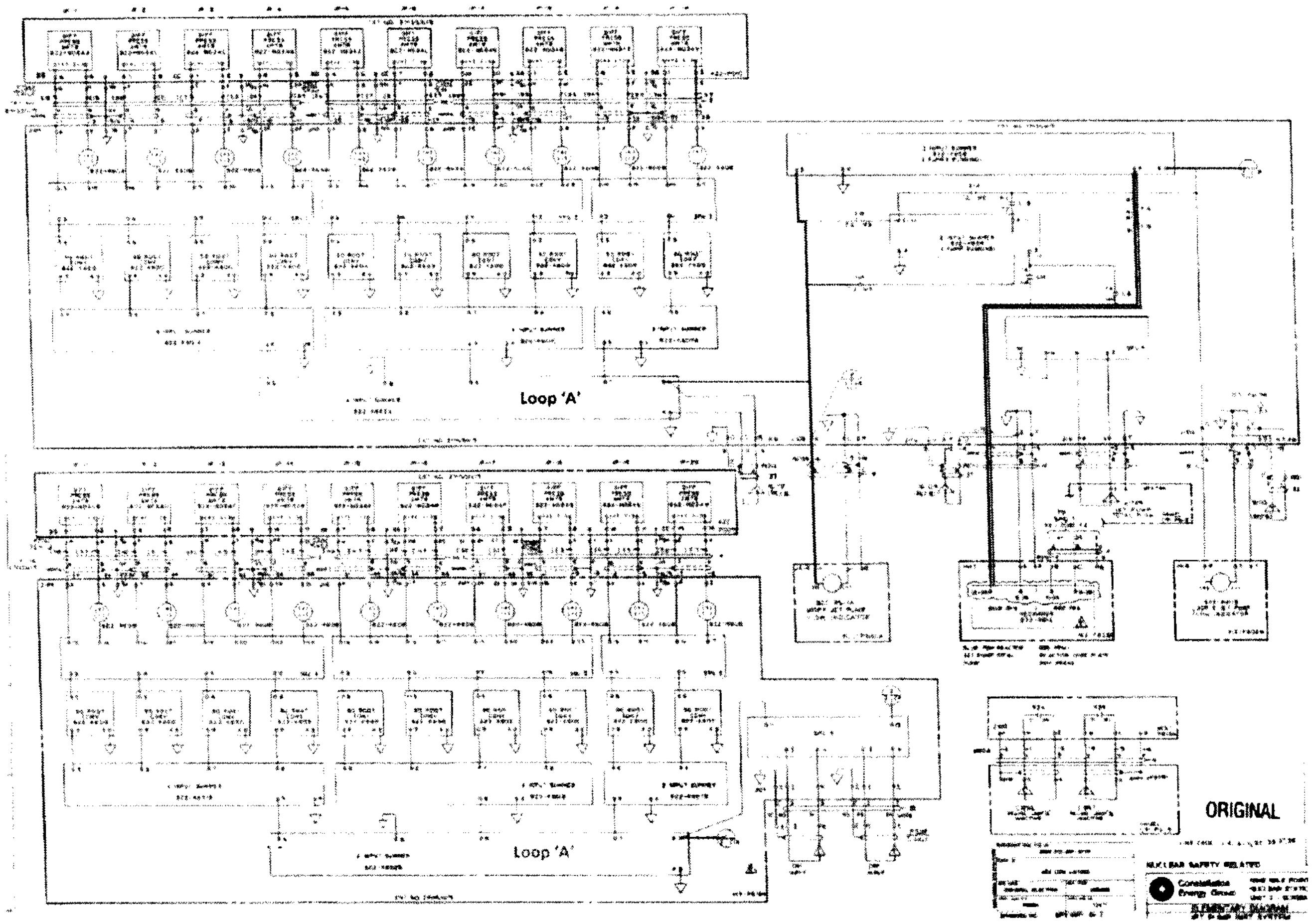
B. A rise in noise indication on the failed jet pump's differential pressure indicator

C. A rise in indicated flow on Loop 'A' Jet Pump Flow Indicator on 2CEC*PNL602

D. A rise in indicated flow on Recirc Loop 'A' Flow recorder on 2CEC*PNL 602

Proposed Answer: D

Explanation: The Recirculation Loop 'A' Flow recorder on control room panel 602 measures "drive flow" upstream of the jet pumps. This "drive flow" will increase due to less resistance to recirculation flow since two of the ten jet pumps on the 'A' RCS loop are no longer restricting flow.



Print: 807E156TY Sheet 2

A. Plausible – Total core flow is measured using the sum of all 10 of the jet pump D/P cells on RCS loop 'A' and all 10 from RCS loop 'B'. (See print above) With this failure, D/P in two of the 10 jet pumps on RCS loop 'A' would be zero therefore total core flow indication would lower. The candidate may confuse the inputs to each of the flow indicators.
B. Plausible – Jet pump d/p indication will be less noisy on the failed jet pump due to the lack of flow. The candidate may think that with the failure, more flow turbulence in the area of the failure would result in more indicated noise.
C. Plausible – Loop 'A' Jet Pump Flow indication on control room panel 602 is measured using the sum of all 10 of the jet pump D/P cells on RCS loop 'A'. (See print above) With this failure, D/P in two of the 10 jet pumps on RCS loop 'A' would be zero therefore Loop 'A' Jet Pump Flow indication on control room panel 602 would lower. The candidate may confuse the inputs to each of the flow indicators.

Technical Reference(s): 807E156TY, Sheet 2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: 2006 COLUMBIA GENERATING STATION REACTOR OPERATOR WRITTEN EXAMINATION, QUESTION # 1, NOVEMBER 2006

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)5

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AA2.04
	Importance Rating	3.2

Partial or Total Loss of DC Pwr / 6

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : System lineups

Proposed Question: #53

Given the following:

- An Extended Station Blackout is in progress
- RCIC is currently injecting to the RPV

If all DC power will soon be lost, powering which of the following buses will prevent a RCIC trip and allow CONTINUED RCIC injection?

A. 2EJS*US1

B. 2EJS*US3

C. 2NJS-US5

D. 2NJS-US6

Proposed Answer: A

Explanation: Loss of 2VBA*UPS2A/C will cause the loss of 2VBS*PNL101A and resultant loss of power to circuit number 2ICSN20. This will cause the automatic speed control (Ramp/Flow signal) to be defeated thereby causing the RCIC turbine to trip on overspeed. Bus 2EJS*US1 ultimately provides normal and alternate power to 2VBA*UPS2A/C. Restoring power to 2EJS*US1 will allow restoring 2VBA*UPS2A/C and 2VBS*PNL101A. Restoration 2EJS*US1 will also restore power to 2BYS*CHGR2A1 and 2BYS*CHGR2A2 which will in turn allow restoration to 2BYS*SWG002A. This would restore normal DC power to 2VBA*UPS2A/C.

B. Plausible – 2EJS*US3 provides power to 2VBA*UPS2B/D, 2VBS*PNL301B and 2BYS*SWG002B, however loss of these panels have minimal effects on RCIC operation. Loss of some indications will occur, but RCIC will continue to operate.
C. Plausible – 2NJS-US5 provides power to 2BYS-SWG001A, however loss of this panel will have minimal effects on RCIC operation. Loss of some indications and controls will occur, but RCIC will continue to operate.
D. Plausible – 2NJS-US6 provides power to 2BYS-SWG001B, however loss of this panel will have minimal effects on RCIC operation. Loss of some indications and controls will occur, but RCIC will continue to operate.

Technical Reference(s): AE-100C (Page 48), AE-100C App. A (Page 20), EE-MO1E, EE-MO1F

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-263000C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295028 2.4.1
Importance Rating	4.6

High Drywell Temperature / 5

Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps.

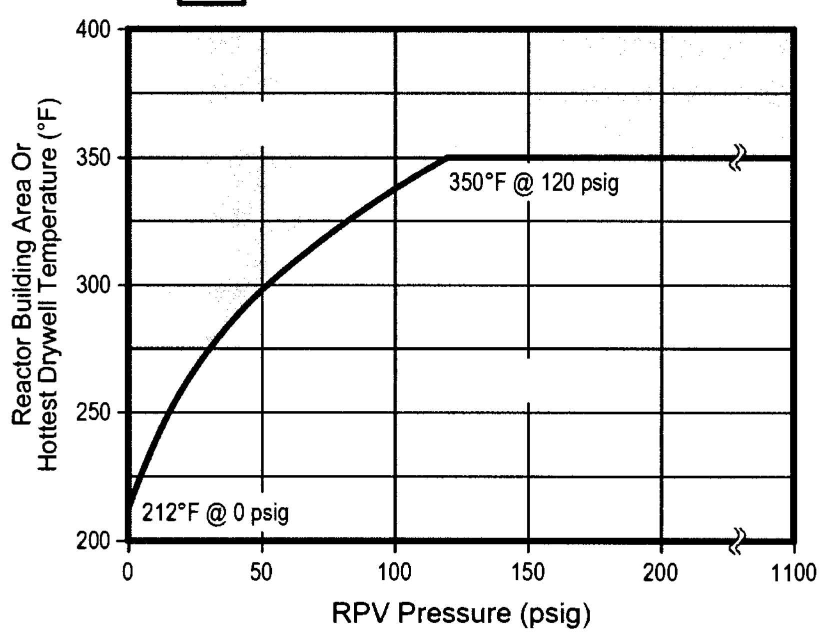
Proposed Question: #54

The plant is in a reactor startup with the following:

- Reactor pressure is 80 psig
- RPV level is stable and being controlled at 185 inches
- A steam leak developed that raised drywell temperature and the leak is now isolated
- All available drywell cooling is currently being placed in service per N2-OP-60
- The following Drywell temperature values are stable:

Div 1 DW Temp Elevation	Temperature	Div 2 DW Temp Elevation	Temperature
244’	120°F	244’	118°F
255’	170°F	253’	160°F
268’	180°F	262’	180°F
283’	250°F	282’	248°F
289’	270°F	296’	272°F
307’	280°F	310’	280°F

B RPV Saturation Temperature



Which one of the following describes the EOP entry condition requirements and the effect on RPV level instrumentation?

	EOP entry condition met?	RPV level instrumentation concern?
A.	Yes	Yes
B.	Yes	No
C.	No	Yes
D.	No	No

Proposed Answer: B

Explanation: Temperatures are indicated on the regulatory guide 1.93 instrumentation on Panel 873 & 875 in the control room. N2- EOP –PC entry condition on high drywell temperature is average drywell temperature above 150°F. The average of the given drywell elevation temperatures is 212°F. This is above 150°F which is a required N2-EOP-PC entry. The hottest drywell temperature allowed that would keep plant operation below the RPV saturation line is 325°F for the given reactor pressure of 80 psig. The current hottest drywell pressure was given at 290°F so per Figure 'B' of N2-EOP-PC, RPV level instrumentation would not be of concern.

A. Plausible – An EOP entry condition is met (N2- EOP –PC entry condition on high drywell temperature is average drywell temperature above 150°F), however the hottest drywell temperature allowed that would keep plant operation below the RPV saturation line is 325°F for the given reactor pressure of 80 psig. The current hottest drywell pressure was given at 290°F. This could be picked as an answer if the candidate decides that a conservative action need be taken since the containment volume may continue to heat up even with the leak isolated as drywell cooling mixes the air in the drywell.

C. Plausible – This could be picked as an answer if the candidate sees the given temperatures and notices that there are indicated temperatures below 150°F also, the hottest drywell temperature allowed that would keep plant operation below the RPV saturation line is 325°F for the given reactor pressure of 80 psig. The current hottest drywell pressure was given at 290°F. This could be picked as an answer if the candidate decides that a conservative action need be taken since the containment volume may continue to heat up even with the leak isolated as drywell cooling mixes the air in the drywell.

D. Plausible – This could be picked as an answer if the candidate sees the given temperatures and notices that there are indicated temperatures below 150°F. The second part of this distracter is correct. The hottest drywell temperature allowed that would keep plant operation below the RPV saturation line is 325°F for the given reactor pressure of 80 psig. The current hottest drywell pressure was given at 290°F so per Figure 'B' of N2-EOP-PC, RPV level instrumentation would not be of concern.

Technical Reference(s): N2-EOP-PC flowchart, TS bases 3.6.1.5, Applicable safety analysis section.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01, Objective # N2-EOP-PC-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 2.2.37
	Importance Rating	3.6

Partial or Complete Loss of AC / 6

Equipment Control: Ability to determine operability and/or availability of safety related equipment.

Proposed Question: #55

The plant is operating at rated conditions with the following:

- 2ENS*SWG101 and 2ENS*SWG102 are being powered from reserve transformer 'A'
- 2ENS*SWG103 is being powered from reserve transformer 'B'
- Breaker R60 fails and trips open
- All equipment responds and functions as designed

Which one of the following describes the safety related equipment that is OPERABLE?

	2ENS*SWG101	2ENS*SWG102	2ENS*SWG103
A.	Yes	No	No
B.	No	Yes	Yes
C.	Yes	Yes	Yes
D.	No	No	No

Proposed Answer: C

Explanation: Breaker R60 supplies power from TB2 to off-site 115KV line 6. When breaker R60 trips the one qualified circuit between the offsite transmission network and the onsite Class 1E AC Electric Power Distribution System per TS 3.8.1 is inoperable, however in this case the Division 1 (2ENS*SWG101) and Division 2 (2ENS*SWG103) AC electrical power distribution switchgears still have power and would therefore remain operable and available per TS 3.8.8. 2ENS*SWG101 is being powered from off-site 115KV line 5 and 2ENS*SWG103 is being powered from EGS*EG3.
Per TS 3.8.8 LCO bases, "OPERABLE AC electrical power distribution subsystems require the associated buses to be energized to their proper voltages"

A. Plausible – The candidate could think that with R60 open, 2ENS*SWG103 would be inoperable and unavailable. 2ENS*SWG101 is considered both operable and available, however ENS*SWG103 is as well since it is being powered from its respective EDG.
B. Plausible – The candidate could think that R60 supplies off-site 115 KV line 5 and therefore think that 2ENS*SWG101 would be inoperable and unavailable. 2ENS*SWG103 is considered both operable and available because it is being powered from its respective EDG, however ENS*SWG101 is as well since it is unaffected.
D. Plausible – The candidate could think that since a loss of off-site powered occurred with the source stemming from a breaker in the scriba yard that both off-site power supplies are inoperable and unavailable.

Technical Reference(s): TS Bases 3.8.1 and 3.8.8

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-264000C01, RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 2.4.9
	Importance Rating	3.8

High Drywell Pressure / 5

Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: #56

The plant is in a reactor startup with the following:

- Reactor pressure is 500 psig
- Reactor water level is being maintained in the normal band using 2FWS-LV55B
- FWS pump start preps are being conducted
- Reactor mode switch is in startup
- Condenser air removal pumps are in service maintaining condenser vacuum
- APRM's are indicating approximately 2% power
- A LOCA occurs and drywell pressure rises to 1.7 psig
- The reactor automatically scrams and all control rods insert

Which one of the following describes the appropriate action to conduct for the given set of conditions?

A. Manually trip the main turbine using the TRIP pushbuttons

B. At 2CEC-PNL842 Shutdown the Hydrogen Water Chemistry System

C. Fast close the outboard MSIV's prior to reactor pressure reaching 420 psig

D. Prevent LPCS and LPCI injection prior to reactor pressure reaching 400 psig

Proposed Answer: D

Explanation: With a high drywell pressure signal in, all LP ECCS pumps would start and immediately receive the open permissive for D/P since the reactor is in startup and reactor pressure is already low (500 psig). Per N2-EOP-RPV control step P-1 it is required to "Before RPV pressure drops to 400 psig, prevent LPCS and LPCI injection not needed for core cooling."

PRESSURE

IF	THEN
Drywell pressure is at or above 1.68 psig	<u>Before</u> RPV pressure drops to 400 psig, prevent LPCS and LPCI injection not needed for core cooling.
RPV Blowdown (EOP-C2) is <u>anticipated</u>	Rapidly depressurize the RPV using the main turbine bypass valves. ☛ OK to exceed 100°F/hr cooldown.

P-1

Low pressure ECCS initiate automatically on a high drywell pressure signal of 1.68 psig and begin to inject when RPV pressure decreases below the shutoff head of the pumps, 400 psig (based on the shutoff head of LPCS, the highest press ECCS). If it is not needed for core cooling, the injection is prevented, since it would only complicate efforts to control RPV water level.

A. Plausible – Per N2-SOP-101C immediate action if the main turbine is online AND Power ≤ 4% it is required to Trip Main Turbine per N2-SOP-21. This will require the candidate to understand that the main turbine would not be on-line at this point in the reactor startup.
B. Plausible – Per N2-SOP-101C general action flowchart it is required to shutdown HWC at panel 842. This will require the candidate to understand that HWC would not be in service at this point in the reactor startup.
C. Plausible – Per N2-SOP-101C pressure control flowchart it is required to maintain Cooldown rate <100 degrees per hour (RPV pressure >420 psig for first hour), however in this case reactor pressure was at 500 psig so it would not be required to fast close the outboard MSIV's prior to reactor pressure reaching 420 psig because dropping below 420 psig would not violate the cooldown rate limitation.

Technical Reference(s): N2-EOP-RPV step P-1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOP-RPVC01, N2-EOP-RPV-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK1.01
	Importance Rating	4.6

Reactor Low Water Level / 2

Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL : Adequate core cooling

Proposed Question: #57

The following conditions exist:

- A small break LOCA is in progress.
- Appropriate EOPs have been entered.
- No injection sources are available.
- RPV level is approaching the TAF.

Which one of the following describes why RPV Blowdown is delayed until water level has lowered to the Minimum Zero-Injection RPV Water Level per N2-EOP-C3?

A. RPV Blowdown before this level reduces the time the core remains adequately cooled during these conditions.

B. Steam Cooling will NOT maintain fuel cladding temperature below 1500°F if blowdown is performed with water level above TAF.

C. RPV Blowdown before this level produces insufficient steam mass removal rate for adequate core cooling.

D. Steam cooling will NOT maintain fuel cladding temperature below 1800°F if blowdown is performed with water level above TAF.

Proposed Answer: A

Explanation: Per EOP-C3, (Page 10-14 of EOP Bases Document), it states "opening the SRVs before RPV water level reaches the MZIRWL would reduce the time over which the core remains adequately cooled with no injection".

B. Plausible –The temperature limit in steam cooling is 1800 degrees. This value is not reached until level has lowered to -58". If the blow down is commenced before TAF, temperature will be well below 1800 degrees. However the time available before adequate core cooling is lost will be reduced since the blowdown was performed before the temperature limit was reached.

C. Plausible –The concern is time permitted for adequate cooling with no injection and not steam mass removal rate generated heat.

D. Plausible – -58" is the level at which steam cooling would be unable to preclude clad temperature from reaching 1800 degrees and is the Minimum Zero-Injection RPV Level. It is not the reason you delay a blowdown. TAF is -14 inches.

Technical Reference(s): N2-EOP-C3 bases Rev 08.00 page 3-214

Proposed references to be provided to applicants during examination: None

Learning Objective: LP #2101-EOPC3C01, Steam Cooling EO-1, EOP Flowchart, Function, and Purpose

Question Source: Nine Mile Point Unit 2 2010 NRC RO Written Examination, Question RO #58

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)2

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AK2.04
	Importance Rating	2.8

Partial or Total Loss of Inst. Air / 8

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Reactor water cleanup

Proposed Question: #58

The plant is operating at rated conditions with the following:

- The reactor water cleanup system (WCS) is operating in a 1 pump, 3 filter demin line-up.
- RDS is supplying seal water injection to the WCS pumps
- An instrument air leak occurs that results in a loss of air to WCS air operated valves
- A spurious WCS system isolation occurs
- Control room operators take required ARP actions and throttle open WCS-MOV110

Which one of the following describes the affect on the in service WCS Filter Demineralizer inlet and outlet valves and the associated consequences in accordance with N2-OP-37?

	In service WCS Filter Demineralizer inlet and outlet valves	Associated consequences
A.	Fail Closed	Loss of all WCS input to the process computer Core Thermal Power Calculation
B.	Fail Closed	Over pressurization of the WCS pump suction piping
C.	Fail Open	Potential resin intrusion into the RPV if WCS were to be placed back in service
D.	Fail Open	Operation with the potential to drain the reactor vessel (OPDRV)

Proposed Answer: B

Explanation: Per N2-SOP-19 section 5.3 table:

Component	Fail Position	Plant Impact	Procedure
Condensate Prefilter Inlet & Outlet Valves	As Is	Loss of ability to place Condensate Prefilters in or out of service.	N2-OP-5A
Cond Demin Inlet & Outlet Valves	As Is	Loss of ability to place Cond Demins in or out of service	N2-OP-5
2CNS-LV103 (Cond Normal M/U)	Open	Constant Makeup to the Main Condenser	N2-OP-4
2CNS-LV105 (Cond Spill Valve)	Closed	No reject from the Main Condenser	N2-OP-4
2CNS-AOV304 (Cond Emerg M/U)	Closed	No emergency Makeup to the Main Condenser	N2-OP-4
WCS F/D Inlet & Outlet Valves	Closed	Loss of Filter/Demin Possible WCS Pump trip	N2-OP-37

This effectively isolates the WCS system and per N2-ARP-602307, 309, 310, 313, 319 WCS-MOV110 is required to be verified throttled open.

Automatic Response

- WCS*MOV102, CLEANUP SUCT INBOARD ISOL VLV and/or *MOV112, CLEANUP SUCT OUTBOARD ISOL VLV, fully close.
- WCS-P1A and P1B, CLEANUP RECIRC PMPs, trip due to system isolation valve closure.

Operator Actions

1. **ENTER** N2-EOP-SC.
2. **VERIFY** 2WCS*MOV102 **AND/OR** *MOV112 have fully closed by checking the green valve position indicating lights are lit on 2CEC*PNL602.
3. **VERIFY** 2WCS-P1A **AND** P1B have tripped by checking that each pumps green lights are lit on 2CEC*PNL602 **AND** verifying that alarm 602314, RWCU PUMP 1A/1B AUTO TRIP, actuates.
4. **IF** the RDS System is supplying seal water injection to the WCS pumps, **VERIFY** throttled open WCS-MOV110, CLEANUP DEMIN BYPASS VLV THROTTLE, as required to **PROVIDE** a leak path for pump seal water flow, to **AVOID** over pressurizing pump suction piping.

A. Plausible – The first part of the distracter is correct, however the second part of the distracter is incorrect, but plausible. The second part of the distracter is incorrect per N2-OP-37 Precaution and Limitation 11.0, “The Process Computer's Core Thermal Power Calculation may be slightly inaccurate (up to approximately ±6 MWTH if system flow, inlet temperature, or outlet temperature computer point values fall outside the calculation's acceptable ranges. In this case the last "good" reading will be used for the calculation, whether it is accurate or not. Failure to verify appropriate PPC computer points are in service and functioning correctly may result in invalid inputs to the CTP calculation. This could occur if a sensor fails. If this occurs, the Core Thermal Power Calculation (OD-3) output can be reviewed for any failed sensor inputs. N2-REP-30 should be utilized to adjust the computer points range. Reactor Engineering and Computer Department support is required.

C. Plausible – Both the first and second part of the distracter are incorrect, but plausible. The candidate could think that with a loss of air that the system would be designed to have the inlet and outlet valves remain open so WCS flow remains through the demins thus ensuring that the filter media in the filter demins remains intact so that resin fines do not get released into the RPV.

D. Plausible – Both the first and second part of the distracter are incorrect, but plausible. The candidate could think that with a loss of air that the system would be designed to have the inlet and outlet valves remain open so WCS flow remains through the demins thus ensuring that the filter media in the filter demins remains intact. The candidate could also think that by maintaining a flow path through the Demineralizers with RDS supplying seal water injection to the WCS pumps a loss of water from the RPV could occur.

Technical Reference(s): N2-SOP-19, section 5.3 table, PID-037C & D

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-204000C01, RBO-08

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295020 AK1.01
	Importance Rating	3.7

Inadvertent Cont. Isolation / 5 & 7

Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION : Loss of normal heat sink

Proposed Question: #59

The plant is operating at rated conditions with the following:

- A spurious reactor scram occurs
- Main steam tunnel temperature is 180°F and slowly rising
- Drywell pressure reaches 1.71 psig
- RPV pressure is 1050 psig and rising

Which one of the following systems can be used for Reactor Pressure control?

A. Turbine Bypass Valves

B. Main Steam Line Drains

C. Reactor Core Isolation Cooling

D. Reactor Pressure Vessel Head Vent

Proposed Answer: C

Explanation: With Group 1 and 5 Isolations present, ONLY RCIC may be used for Pressure Control

A. Plausible – Turbine Bypass Valves are listed as the Preferred Pressure Control method in N2-EOP-RPV Control, but they cannot be used due to MSIV / MSL Drain Isolation (Group 1). The candidate may not recall that a group 1 isolation closes the MSIV's.
B. Plausible – Main Steam Line Drains are listed as an Alternate Pressure Control method in N2-EOP-RPV Control, but they cannot be used due to MSIV / MSL Drain Isolation (Group 1). The candidate may not recall that a group 1 isolation closes 2MSS*MOV111, 2MSS*MOV112, 2MSS*MOV208.
D. Plausible – While RPV head vents are used as a pressure control strategy, conditions have not been met to enter N2-EOP-C2 alternate pressure control nor are they procedurally allowed under the given conditions. The candidate could however think that the head vent is an available source that would be able to control reactor pressure.

Technical Reference(s): N2-OP-83, Attachment 1 group 1 & 5

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01, Objective N2-EOP-RPV-CE-02

Question Source: Nine Mile Point Unit 2 Reactor Operator Written Examination March 2008, Question # 61

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b) 5

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	2
K/A #	295012 AK2.02
Importance Rating	3.6

High Drywell Temperature / 5

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell cooling

Proposed Question: #60

The plant has experienced a LOCA, with the following:

- Drywell Pressure is 3.0 psig
- Hottest Drywell Temperature is 275°F
- Drywell Cooling Fans are tripped

Which one of the following describes the effect on restoring Drywell Cooling System (DRS) per EOP support procedures?

A. Can be restored after defeating interlocks

B. Can be restored without defeating interlocks

C. Cannot be restored, restoring CCP flow may result in water hammer

D. Cannot be restored, restarting DRS fans may result in air duct damage

Proposed Answer: C

Explanation: Per N2-EOP-6.24, if DWT is above 250°F, the containment isolation MOVs cannot be reopened, because the water volume in the section of piping between the inboard and outboard isolation valves may have flashed to steam due to the elevated DWT.

A. Plausible - Candidate may not recall the operational design characteristics and focus on the logic of the system and think that with interlocks defeated the system can be restored.
B. Plausible – Candidate may believe that restoration of the system (though partial) can be accomplished by using the "fans only" mode without defeating interlocks.
D. Plausible – Candidate may confuse the guidance that 2DRS-UC3A and UC3B cannot be run at the same time because of possible duct work damage.

Technical Reference(s): N2-EOP-6.24, P&L 5.8

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOP6HCC01, Objective # N2-EOP-6-CE3-A24

Question Source: U2 NRC REACTOR OPERATOR WRITTEN EXAMINATION 2005, Question #22

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295010 AK3.01
	Importance Rating	3.8

High Drywell Pressure / 5

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : Drywell venting

Proposed Question: #61

The plant is operating at 10% power continuing power ascension with the following:

- Annunciator 603140, "DRYWELL PRESSURE HIGH/LOW" alarms
- The alarm was not caused by barometric pressure changes
- No primary system leaks are occurring
- Drywell pressure is currently 0.90 psig and rising slowly

Which one of the following describes the **allowable** vent path and reason why venting is performed under these conditions in accordance with N2-OP-61A, Primary Containment Ventilation, Purge, and Nitrogen System?

	Vent from ______________ through the 2 inch line.	Why
A.	either the drywell or suppression chamber	To keep containment pressure in a specific band to ensure that containment pressures remain within design values during a LOCA
B.	either the drywell or suppression chamber	To keep all air and non-condensable gases out of the containment to prevent exceeding the maximum negative D/P if inadvertent containment spray occurs
C.	the drywell and suppression chamber, simultaneously	To keep all air and non-condensable gases out of the containment to prevent exceeding the maximum negative D/P if inadvertent containment spray occurs
D.	the drywell and suppression chamber, simultaneously	To keep containment pressure in a specific band to ensure that containment pressures remain within design values during a LOCA

Proposed Answer: A

Explanation: Per N2-ARP-603140 operator action #3, "IF Drywell pressure change is NOT due to Barometric change, OR as directed by SM/CRS, perform the following:

a. IF pressure is high, THEN PERFORM N2-OP-61A, Subsection H.1.0."
   Venting is non-emergency and only allowed from either the DW or SC to prevent bypass flowpath.

For the second part of the answer, TS bases for LCO 3.6.1.4 states "A limitation on the drywell and suppression chamber internal pressure of ≥ 14.2 psia and ≤ 15.45 psia is required to ensure that primary containment initial conditions are consistent with the initial safety analyses assumptions so that containment pressures remain within design values during a LOCA and the design value of containment negative pressure is not exceeded during an inadvertent operation of drywell sprays."

B. Plausible – The first part of the distracter is correct and the second part is incorrect, but plausible. For the second part of the distracter the candidate could think that having air and non-condensables in the containment would prevent the design operation of the containment and associated sprays during LOCA conditions.
C. Plausible – Both the first and second part of the distracter is incorrect, but plausible. For the first part of the distractor, the candidate may not recall that this would bypass the pressure suppression function of the primary containment and also might think that since containment temperatures are high and that the high drywell pressure annunciator is lit that the larger vent path should be used. For the second part of the distracter the candidate could think that having air and non-condensables in the containment would prevent the design operation of the containment and associated sprays during LOCA conditions.
D. Plausible – The candidate may not recall that this would bypass the pressure suppression function of the primary containment and also might think that since containment temperatures are high and that the high drywell pressure annunciator is lit that the larger vent path should be used. The second part is correct.

Technical Reference(s): N2-ARP-603140, N2-OP-61A, Section H.1.0, TS bases for LCO 3.6.1.4 (LCO section of bases)

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-223001C01, RBO-10

Question Source: U2 2005 AUDIT REACTOR OPERATOR EXAMINATION, Question #21

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295009 AA1.04
	Importance Rating	2.7

Low Reactor Water Level / 2

Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL : Reactor water cleanup

Proposed Question: #62

The plant has been scrammed due to a steam leak on RCIC:

- N2-EOP-SC has been entered
- Two separate area temperatures have exceeded maximum safe values
- Division I and II ADS have failed to initiate and cannot be initiated manually
- 'A' & 'B' ADS DC solenoid power has been lost
- 'C' SRV DC solenoid power has been lost
- SLS has been injected for level control
- Reactor water level wide range indication is reading 15 inches
- Reactor water level narrow range indication is reading 50 inches

Which one of the following describes a viable method and the required action needed to perform an RPV blowdown from the control room envelope in accordance with N2-EOP-C2, RPV Blowdown?

	System	Required Action
A.	WCS (Recirculation mode)	Defeat all WCS Isolations per N2-EOP-6.11
B.	SRV's (ADS mode)	Operate all 14 keylock switches on panel 2CEC*PNL628 & 631
C.	WCS (Blowdown mode to main condenser)	Break main condenser vacuum by opening condenser vacuum breakers on 2CEC*PNL851
D.	SRV's (Relief mode)	Operate 7 SRV 'C' solenoid keylock switches on 2CEC*PNL601

Proposed Answer: A

Explanation: N2-EOP-C2 states all of the systems that are available for use for conducting an RPV blowdown. Since DC power has been lost to all modes of SRV operation, WCS is the only system choice remaining for use to blowdown the RPV. The question states that SLS was injected for RPV level control so WCS, blowdown mode would not be permissible. WCS recirculation mode is the only allowable choice.

B. Plausible – The candidate would have to recall that the 14 keylock switches on panels 628 & 631 operate the "A" & "B" ADS solenoids and that with DC power lost, operation of these switches would not be possible.

C. Plausible – Though WCS is an allowable system to use for an alternate blowdown system, it is not allowed in this case since boron has been injected for level control and the lineup for this method uses the main condenser. The candidate could think that in order to effectively use this method that main condenser vacuum would be reduced to prevent damage to the main condenser due to the differential pressure between WCS and the main condenser.

D. Plausible – The candidate would have to recall that the keylock switches on panel 601 operate the "C" solenoids and that with DC power lost, operation of these switches would not be possible.

Technical Reference(s): N2-EOP-C2 Detail "O", N2-EOP-6.11 section 6.4

Proposed references to be provided to applicants during examination: None

Learning Objective: 2102-204000C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295029 EA2.03
	Importance Rating	3.4

High Suppression Pool Water Level / 5

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Drywell/containment water level

Proposed Question: #63

The plant is shutdown with the following:

- Suppression Pool level indication is upscale
- Containment flooding with service water is in progress
- Service water intake temperature is 72°F
- 2CMS-PI7A is reading 20 psig
- 2CPS-PI127 is reading 5 psig

Determine containment water level using the information provided below:

NOTE

- "S" pressure is indicated on 2CMS*PI7A, SUPPR CHAMBER PRESS, *(2CEC*PNL601)*.
- "D" pressure is indicated on 2CPS-PI127, PRIMARY CONTMT INLET N2 PRESS, *(2CEC*PNL873)*.
- When injection temperatures are less than or equal to 70°F, water levels are referenced to the 70°F curve of Figure 1a.
- When injection temperatures are greater than 70°F, water levels are referenced to the 210°F curve of Figure 1b.
- The different pressure lines of Figure 1a AND Figure 1b are used as follows:
    - P(0) = 14.2 psia Line - used when "D" pressure is less than 30.0 psia.
    - P(0) = 30.0 psia Line - used when "D" pressure is greater than or equal to 30.0 psia but less than 59.7 psia.
    - P(0) = 59.7 psia Line - used when "D" pressure is greater than or equal to 59.7 psia.

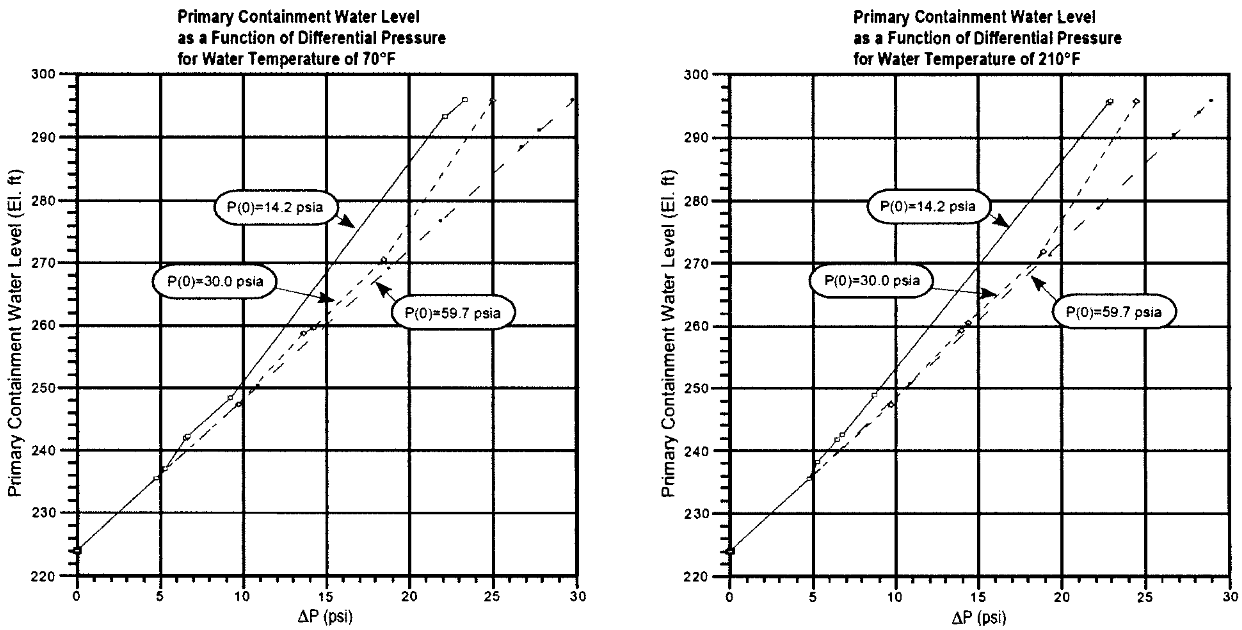


FIGURE 1a .... **FIGURE 1b**

Using Figure 1a OR Figure 1b, determine primary containment water level as a function of ΔP.

Primary containment water level is..........

A. 286 ft

B. 270 ft

C. 268 ft

D. 262 ft

Proposed Answer: B

Explanation: Per N2-EOP-6.23 the following steps are required to determine containment water level:

- Determine the appropriate figure (1a or 1b): Service water intake temperature is given to be 72°F, per bullet 4 of the note this temperature would require the use of figure 1b.
- Determine Differential Pressure: ΔP = S – D (It is not required to convert the pressure values to psia to obtain ΔP), Therefore ΔP = 20 – 5 which equals 15 psi
- Determine the appropriate P(0) pressure line: (It is required to convert the pressure values given from psig to psia to determine the appropriate P(0) pressure line), therefore P(0) pressure line = 2CMS-PI127 pressure + 14.7 which equals 5 +14.7 = 19.7 psia. Per bullet 5 of the note, a value of 19.7 psia would require the use of P(0) 14.2 psia pressure line
- Determine "Y" axis intersection point: Determine where the "X" axis value (ΔP) and the p(0) 14.2 pressure line intersect on figure 1b and correlate the intersection point to the "Y" axis to determine containment water level value. The containment water level value plots out to be 270 ft.

A. Plausible – If figure 1b is used and only the "S" value is used to determine ΔP
C. Plausible – If figure 1a is used which is the wrong figure based on service water injection temperature
D. Plausible – If "S" value vs. "D" value is used to determine P(0)

Technical Reference(s): N2-EOP-6.23 Rev 00001 pages 8 & 11

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-223001C01, RBO-12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295035 2.2.39
	Importance Rating	3.9

Secondary Containment High Differential Pressure / 5

Equipment Control: Knowledge of less than one hour technical specification action statements for systems.

Proposed Question: #64

The plant is in mode 3 with the following:

- Four recently irradiated fuel bundles are being relocated in the spent fuel pool.
- A malfunction of reactor building ventilation pressure controller 2HVR-PDI103 causes reactor building differential pressure to reach -0.20 inches WG.

Which one of the following describes the required technical specification action and the time requirements to complete the action?

	Technical Specification Action	Time requirement to complete the action
A.	Evacuate the refuel floor	Within 1 hour
B.	Suspend the movement of recently irradiated fuel bundles	Within 1 hour
C.	Evacuate the refuel floor	Immediately
D.	Suspend the movement of recently irradiated fuel bundles	Immediately

Proposed Answer: D

Explanation: In this question secondary containment D/P was given to be -0.20 inch WG. This is below the TS allowed value of -0.25 inch WG which makes the secondary containment inoperable. TS 3.6.4.1 states that with Secondary containment inoperable during the movement of recently irradiated fuel assemblies in the secondary containment, it is required to Immediately suspend movement of recently irradiated fuel assemblies in the secondary containment. IMMEDIATE COMPLETION TIME is defined in Tech specs by the following; "When Immediately is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner."

A. Plausible – The candidate could think that with secondary containment ventilation not performing correctly, that the above and below refuel floor process rad monitors would not be functioning properly and that an evacuation of the refuel floor would be required within 1 hour, since the RB is still negative and the ventilation system is still operating.
B. Plausible – The candidate could think that the action to Suspend the movement of irradiated fuel bundles (which is correct) would have to be performed immediately which could be interpreted as within 1 hour.
C. Plausible – The candidate could think that with secondary containment ventilation not performing correctly, that the above and below refuel floor process rad monitors would not be functioning properly and that an evacuation of the refuel floor would be required immediately due to the movement of recently irradiated fuel.

Technical Reference(s): NMP2 Tech Spec bases section 3.6.4.1, NMP2 Tech Specs section 3.0 and 3.6.4.1.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-288001C01, RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295034 EK1.02
	Importance Rating	4.1

Secondary Containment Ventilation High Radiation / 9

Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Radiation releases

Proposed Question: #65

The plant is returning to rated power following operation at reduced power due to the need to make a containment entry with the following:

- A containment low flow nitrogen purge is in progress in accordance with N2-OP-61A
- A steam leak occurs in the secondary containment
- Annunciator 851254, "Process Airborne Radiation Monitor Activated" alarms
- 2HVR*CAB32A-1, "Reactor Building Below Refueling Floor 'A' Gaseous Radiation Monitor" level reaches its trip setpoint
- 2HVR*CAB32B-1 "Reactor Building Below Refueling Floor 'B' Gaseous Radiation Monitor" level reaches its trip setpoint

Which one of the following describes the required action and the potential consequence if the action is not taken?

	Required Action	Potential Consequence
A.	Close 2GTS*AOV101, CONTAINMENT PURGE TO SBGTS ISOL VLV	2GTS*FN1A (B), SBGTS FAN A (B) Trip on high flow
B.	Close 2GTS*SOV102, CONTMT DEPRESSURIZE TO SBGTS ISOL VLV	Ground Level Release
C.	Verify 2GTS*AOV101, CONTAINMENT PURGE TO SBGTS ISOL VLV auto isolation	2GTS*FN1A (B), SBGTS FAN A (B) Trip on high flow
D.	Verify 2GTS*SOV102, CONTMT DEPRESSURIZE TO SBGTS ISOL VLV auto isolation	Ground Level Release

Proposed Answer: B

Explanation: Per N2-OP-61A section F.13 (specifically step F.13.13), containment low flow nitrogen purge utilizes the 2" solenoid operated control valve (GTS*SOV102) to perform the purge. Also, section F.13 of N2-OP-61A states in the caution that "If a purge is in progress and a Rx Bldg Ventilation High Radiation alarm is received, termination of purge is required to ensure GTS is only lined up to the Reactor Building". This is because 2GTS*AOV101 and GTS*SOV102 do not automatically close on reactor building exhaust radiation high. In this case SBGTS will be taking a suction on both the containment and the reactor building. With a steam leak occurring that is pressurizing the reactor building and having SBGTS lined up to both suction paths, it would extend the time it would take for the SBGTS system to draw down the reactor building D/P and could result in a ground level release.

A. Plausible – The candidate could think that 2GTS*AOV101 is used for the low flow purge and also think that 2GTS*AOV101 auto closes on high reactor building exhaust radiation which is does not.
C. Plausible – The candidate could think that 2GTS*AOV101 is used for the low flow purge and could also rationalize that with both suction paths lined up that the GTS fan could trip on high flow.
D. Plausible – The candidate could think that GTS*SOV102 auto closes on high reactor building exhaust radiation, when in fact it does not. The Potential consequence of a ground level release is correct.

Technical Reference(s): N2-OP-61A section F.13 (specifically step F.13.13), USAR section 6.5.1.2.1.1.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-261000C01, RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.13
	Importance Rating	2.5

Knowledge of facility requirements for controlling vital / controlled access.

Proposed Question: #66

The plant is operating at rated conditions with the following:

- A credible insider threat has resulted in activation of the "Two Person Rule" in accordance with EPIP-EPP-10, Security Contingency Event.
- The Unit Supervisor determines that access to the Reactor Building is needed by an EO to immediately operate a valve which is critical to plant operations.
- A security officer is currently available to provide assistance to Operations Dept.
- There are no other operators currently available.
- The EO requests an exception to the "Two Person Rule".

Which one of the following describes the correct response to this request in accordance with EPIP-EPP-10?

An Exception to the "Two Person Rule"...

A. may be approved by the Unit Supervisor.

B. may be approved by the Security Shift Supervisor.

C. may NOT be approved, but a Security Officer may accompany the Operator.

D. may NOT be approved and the EO must wait until another operator becomes available.

Proposed Answer: C

Explanation: The two person rule requires: 1) Equal task qualification levels are not necessary, 2) Partner must remain in line of sight, 3) Partner must have access to the vital area. The security guard meets the requirements for the two man rule.

A. Plausible – The candidate may think that since the need to operate a valve in the plant is required for safe operation that Unit supervisor permission is adequate for the action.
B. Plausible – The candidate may think that since this is a security event and that security actions will be on-going that security shift supervisor permission would be required.
D. Plausible – The candidate may think that the only requirement is that you are paired up with someone when the two person rule is in effect, however the requirement for the two person rule are only that another person qualified to be in the vital area is within line of site. Equal qualifications are not required.

Technical Reference(s): EPIP-EPP-10, section 5.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101- SOP76C01, Objective # N2-SOP-76-CE-03

Question Source: NMP2 LC2 10-01, SRO Question # 94

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.28
	Importance Rating	4.1

Knowledge of the purpose and function of major system components and controls.

Proposed Question: #67

Which one of the following describes the electrical interlock with 2RHS*MOV2A, PMP 1A SDC SUCT VLV, and the reason for that interlock?

If 2RHS*MOV2A is OPEN then...

A. 2RHS*MOV24A, LPCI A Injection Valve, will not open to prevent bypassing the recirculation loop.

B. 2RHS*MOV4A, RHR Pump 1A Minimum Flow Valve, will not open to prevent draining the RPV to the suppression pool.

C. 2RHS*FV38A, RHR A Return to the Suppression Pool, will not open to prevent draining the RPV to the suppression pool.

D. 2RHS*MOV24A, LPCI A Injection Valve AND 2RHS*MOV40A, SDC A Injection valve, cannot both be opened at the same time to prevent running out the RHR pump.

Proposed Answer: C

Explanation: If in SDC none of the suppression pool cooling /spray valves can be opened.

K/A justification: Interlock is a generic RHS interlock. The K/A is sampling function of controls of a system and this generic knowledge concept meets the K/A.

A. Plausible – No such interlock. Plausible in that RHS*MOV24A would bypass the Recirc loop which is the normal path for SDC.
B. Plausible – No such interlock. Plausible in that the min flow valve opening would drain the RPV if in SDC. (Valve is disabled when in SDC).
D. Plausible – No such interlock. Plausible in that both valves being open would create two parallel discharge paths.

Technical Reference(s): ESK-06RHS019, RHR SUPPR. POOL COOLING MOV 2RHS*FV38A
Vision Objective #73502 page 18 (2101-205001C01, RBO-05)

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-205001C01, RBO-5

Question Source: Unit 2 2010 NRC RO Written Examination

Question History:

Question Cognitive Level: Memory Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.2
	Importance Rating	4.6

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Proposed Question: #68

A plant startup and power ascension is in progress with the following:

- Reactor Power is 26%
- Generator output is 250 MWe
- Load Limit setpoint is 250 MWe because of a generator winding issue

Then, annunciator 851150, TURBINE BYPASS VALVE OPEN alarms and #1 Bypass Valve opens.

Which one of the following actions is required to close the #1 Bypass Valve?

A. Insert control rods.

B. Raise Pressure Set.

C. Lower Pressure Set.

D. Lower Bypass Opening Jack setpoint.

Proposed Answer: A

Explanation: With the Load Limit set at 250 MWe the turbine cannot accept any more load. Since the EHC systems wants to open the control valves, but can't because of the load limit the bypass valve is opening to control pressure. The only way to close the bypass valves is to lower reactor power and hence pressure by inserting the control rods.

B. Plausible – Reactor power is providing more power than the generator can accept. Raising or lowering pressure setpoint will change the controlling setpoint but steam supply will still exceed steam demand from the turbine.

C. Plausible – Reactor power is providing more power than the generator can accept. Raising or lowering pressure setpoint will change the controlling setpoint but steam supply will still exceed steam demand from the turbine.

D. Plausible – lowering the Bypass Opening Jack setpoint would not affect BPV position because the pressure regulator is in control. Additionally the Bypass Opening Jack is used to open the bypass valves.

Technical Reference(s): N2-ARP-851150, Turbine Bypass Valve Open, N2-SOP-101D Flowchart

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-248000C01, RBO-10

Question Source: Nine Mile Point Unit 2 2010 NRC RO Written Examination, RO #68

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)7

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.22
	Importance Rating	4.0

Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #69

The plant is operating at reduced power due to rod sequence exchange:

- 3D Monicore MCPR indication is not functioning and is currently displaying “XXXXX”
- Core cycle exposure is 2107.3 MWd/ST
- Tau is 1.0

```
                                          00008069
                                                                        PAGE   1

CORE PARAMETERS              3D MONICORE        26-JUL-2015 09:49 CALCULATED
                             PREDICTOR LOG      26-JUL-2015 09:50 PRINTED
POWER   MWT          2509.                      CASE ID FTFP1050726094926
FLOW    MLB/HR      55.010    CALC RESULTS      RESTART FMLD1050726085454
FPAPDR               0.785                      FIT - FULL CORE
SUBC    BTU/LB      32.58     Keff       1.0000
PR      PSIa      1007.6      XE WORTH % -2.45  LOAD LINE SUMMARY
CORE    MWD/sT   26581.5      XE/RATED   1.01   CORE POWER          72.4%
CYCLE   MWD/sT    2107.3                        CORE FLOW           50.7%
MCPR               XXXXX                        LOAD LINE           98.2%
T STEP HOURS         0.1
CORRECTION FACTOR:  MFLCPR= 1.512  MFLPD= 1.000  MAPRAT= 1.000
OPTION:  PRE_ARTS   DUAL LOOP      MANUAL FLOW   MCPRLIM= 1.610
                  MOST LIMITING LOCATIONS (NON-SYMMETRIC)
MFLCPR   LOC     MFLPD    LOC        MAPRAT   LOC        PCRAT    LOC
1.620   21-40   0.568    7-36- 4    0.604   45-48- 6    0.717    9-38- 3
0.975   11-34   0.548   39-24- 4    0.602   13-46- 6    0.717   31-46-17
0.974   21-38   0.545    9-36- 4    0.602   29-46- 8    0.716    9-34-16
0.974   23-40   0.545   39-22- 4    0.585   39-22- 4    0.715   35-36-18
0.963   19-42   0.536    9-44- 4    0.580   41-22- 5    0.715   11-44- 3
0.958   13-32   0.536    7-38- 4    0.577   21-42- 5    0.714   41-10- 3
0.954   29-48   0.527   41-20- 4    0.575    9-38- 5    0.713   13-34-16
0.952   27-50   0.526   13-30- 7    0.571    9-36- 4    0.713   43-12- 3
0.946    9-32   0.520   29-48- 8    0.568   15-32- 8    0.713   11-32-16
0.944   17-44   0.518   13-48- 6    0.567    9-44- 4    0.713   45-30-16
```

$$\text{MFLCPR} = \frac{(\text{MCPR}_{\text{Limit}})\,(\text{MFLCPR Correction Factor})}{\text{MCPR}_{\text{Actual}}}$$

Using the equation above, which one of the following describes the technical specification (T.S.) Safety Limit status and the applicable Technical Specification(s) requiring action?

Note:

The following delineates the specific LCO titles:

- T.S. 3.2.2, “MINIMUM CRITICAL POWER RATIO (MCPR)”
- T.S. 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR)”

Safety Limit Applicable T.S.

A.	Not violated	3.2.2 and 3.2.3
B.	Not violated	3.2.2 only
C.	Violated	3.2.2 and 3.2.3
D.	Violated	3.2.2 only

Proposed Answer: B

Explanation: Technical Specification (T.S.) safety limit 2.1.1.2 states "With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

- MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

In order to determine MCPR (since the indication has failed for the given 3D Monicore printout) the MFLCPR formula (given) must be used. The equation must be algebraically re-arranged to solve for $MCPR_{Actual}$. When algebraically re-arranged the equation becomes:

$$MCPR_{Actual} = \frac{MCPR_{Limit}\text{ (MFLCPR Flow Correction Factor)}}{\text{MFLCPR}}$$

With given values (provided in the 3D Monicore printout) substituted into the equation the equation becomes:

$$MCPR_{Actual} = \frac{1.610\,(1.512)}{1.620}$$

$$MCPR_{Actual} = 1.503$$

This value of MCPR is above the T.S. safety limit value of ≥ 1.07 and therefore a T.S. safety limit has not been violated. T.S. 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)" LCO states "All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR." The COLR value for MCPR is a given value on the provided 3D monicore printout. The value is listed as MCPRLIM and is indicating a value of 1.610. The calculated value of MCPR is 1.503 which is less than the allowable 1.610, therefore LCO 3.2.2 is not met and would require action to be taken.

A. Plausible – This first part of this distracter is correct and the second part of the distracter is partially correct, however T.S. 3.2.3 would not be applicable. T.S. 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)" LCO states "All LHGRs shall be less than or equal to the limits specified in the COLR." LHGR is monitored by 3D monicore using MFLPD. Since the equation for MFLPD is:

$$\text{MFLPD} = \frac{LHGR_{Actual}}{LHGR_{Limit}}$$

As long as all nodes in the core have LHGR value less than the limit, MFLPD will remain less than 1.0. In the given 3D Monicore printout, all MFLPD values are less than 1.0 so T.S. LCO 3.2.3 would not applicable. This is plausible because some of the thermal limits are designed to have acceptable values less than 1.0 and some are designed to have acceptable values greater than 1.0.

C. Plausible – With MFLCPR greater than 1.0 the candidate may think that a violation of a safety limit has occurred. The second part of the distracter is partially correct, however T.S. 3.2.3 would not be applicable. T.S. 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)" LCO states "All LHGRs shall be less than or equal to the limits specified in the COLR." LHGR is monitored by 3D monicore using MFLPD. Since the equation for MFLPD is:

$$\text{MFLPD} = \frac{LHGR_{Actual}}{LHGR_{Limit}}$$

As long as all nodes in the core have LHGR value less than the limit, MFLPD will remain less than 1.0. In the given 3D Monicore printout, all MFLPD values are less than 1.0 so T.S. LCO 3.2.3 would not applicable. This is plausible because some of the thermal limits are designed to have acceptable values less than 1.0 and some are designed to have acceptable values greater than 1.0.

D. Plausible – With MFLCPR greater than 1.0 the candidate may think that a violation of a safety limit has occurred. The second part of the distracter is correct. T.S. 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)" LCO states "All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR." The COLR value for MCPR is a given value on the provided 3D monicore printout. The value is listed as MCPRLIM and is indicating a value of 1.610. The calculated value of MCPR is 1.503 which is less than the allowable 1.610, therefore LCO 3.2.2 is not met and would require action to be taken.

Technical Reference(s): COLR2-15 figure 2A, T.S. 2.1.1.2 and associated bases, T.S. 3.2.2 and 3.2.3 and associated bases.

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-101000C01, RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.13
	Importance Rating	3.4

Knowledge of Radialogical Safety Procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: #70

Given the following:

- Entry into the TIP Room is required.
- TIP Room is LOCKED and posted as VERY HIGH RADIATION AREA.

Which one of the following identifies who is responsible for controlling access to the key, and the additional approval(s) required for entry per RP-AA-460-001, CONTROLS FOR VERY HIGH RADIATION AREAS?

	Key Control	Additional Approval(s) Required
A.	Radiation Protection Manager	Site VP OR the Plant Manager AND the Radiation Protection Manager
B.	Radiation Protection Manager	Radiation Protection Manager ONLY
C.	Shift Manager	Site VP OR the Plant Manager AND the Radiation Protection Manager
D.	Shift Manager	Radiation Protection Manager ONLY

Proposed Answer: A

Explanation: Per RP-AA-460-001 step 4.2.4.1; Keys designated for VHRA's are to be maintained in a separate cabinet from ALL other keys and controlled by the RPM. Per step 4.3.3.1; Obtain approval of the Radiation Protection Manager or designee and Obtain approval of the Site VP, Plant Manager, or designee prior to entry.

B. Plausible – Per step 4.3.3.1 of RP-AA-460-001; it is required to obtain approval of the Radiation Protection Manager or designee and the Site VP, Plant Manager, or designee prior to entry., not just the Radiation Protection Manager.
C. Plausible – Per RP-AA-460-001 step 4.2.4.1; Keys designated for VHRA's are to be maintained in a separate cabinet from ALL other keys and controlled by the RPM. This distracter is plausible because Radiation Protection Personnel do control keys to Locked High Radiation Areas (RP-AA-460, step 4.2.2.2), however the Radiation Protection Manager is required to control the keys to Very High Locked Radiation Areas.
D. Plausible – Per RP-AA-460-001 step 4.2.4.1; Keys designated for VHRA's are to be maintained in a separate cabinet from ALL other keys and controlled by the RPM. This distracter is plausible because Radiation Protection Personnel do control keys to Locked High Radiation Areas, however the Radiation Protection Manager is required to control the keys to Very High Locked Radiation Areas. Also, per step 4.3.3.1 of RP-AA-460-001; it is required to obtain approval of the Radiation Protection Manager or designee and the Site VP, Plant Manager, or designee prior to entry, not just the Radiation Protection Manager.

Technical Reference(s): RP-AA-460-001, section 4.2.4.1 & 4.3.3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-ADMTPSC44, Objective # GAP-RPP-01-CT-01

Question Source: Modified from the Nine Mile Point Unit 2 2010 NRC SRO #96

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.4
	Importance Rating	3.2

Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: #71

The plant is in a refuel outage with the following:

- A radiation accident has occurred on the refuel floor resulting in serious injury to a worker on the Refuel Platform.
- The worker is still on the Refuel Platform.
- Radiation levels in the area of the injured operator are 4500 mRem/hr.
- Emergency exposure limit for life saving operations has been authorized.
- The individual designated to provide life saving assistance has a current lifetime exposure of 1000 mrem

Which one of the following is the maximum stay time for the individual providing life saving assistance to ensure the limits established in EP-CE-113, Personnel Protective Actions Procedure are not exceeded?

A. 0.88 hrs

B. 1.11 hrs

C. 5.33 hrs

D. 5.55 hrs

Proposed Answer: D

Explanation: Per EP-CE-113, “Emergency exposure limits are exclusive to current occupational exposure.” Therefore, based on correct limit of 25 Rem limit (life saving per EP-CE-113 Table 5.1-1) 25R/4.5R per hr = 5.55 hours.

A. Plausible – This response is based on a limit of a limit of 5,000 mrem, but figures in the individual designated to provide life saving assistance current lifetime exposure of 1000 mrem. 5R-1R/4.5 R per hr =0.88 hours
B. Plausible – This response is based on a limit of a limit of 5,000 mrem.
C. Plausible – This response is based on a limit of a limit of 25,000 mrem, but figures in the individual designated to provide life saving assistance current lifetime exposure of 1000 mrem. 25R-1R/4.5 R per hr = 5.33 hours

Technical Reference(s): EP-CE-113, Table 5.1-1.

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-EP1014C01, Objective # NS-EPL004-EO-03

Question Source: August 2009, RO Question #70

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.16
	Importance Rating	3.5

Emergency Procedures / Plan: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, AOP's and SAMG's.

Proposed Question: #72

The plant is operating at rated conditions with the following:

- An I & C technician inadvertently operates the wrong component during a surveillance and causes both reactor recirculation pumps to trip.
- A reactor scram occurs and all control rods fully insert
- Reactor Pressure is 850 psig and stable
- RPV water level is 155 inches and lowering very slowly
- The condensate and feedwater system is available and operating normally and no operator action has been taken with the condensate and feedwater system

Which one of the following represents the correct reactor water level band for the given conditions and the corresponding level control means?

	Band	Level Control Means
A.	227 to 243 inches	Raise RPV level to specified band using the feedwater pumps
B.	227 to 243 inches	Raise RPV level to specified band by maximizing CRD flow
C.	159.3 to 202.3 inches	Restore and maintain RPV level using preferred injection systems
D.	159.3 to 202.3 inches	Maintain RPV level using both preferred and alternate injection systems

Proposed Answer: C

Explanation: RPV Control, step L-3 states, “Restore and maintain RPV water level between 159.3 in. and 202.3 in. using one or more Preferred Injection Systems (Detail E1).” OP-NM-101-111-1001, section 5.3 states:

5.3. Procedure Hierarchy

> **NOTE**
>
> Implementation of the EDMGs (Extreme Damage Mitigation Guidelines) may be done in parallel with the below procedures.

A. Nothing shall supersede the proper implementation of the EOPs/SAP's. The following describes the operating procedure implementation hierarchy from highest to lowest priority:

1. SAPs (Severe Accident Procedures) including those Procedures directed by SAPs
2. EOPs (Emergency Operating Procedures) including those Procedures directed by EOPs
3. SOPs (Special Operating Procedures)
4. ARPs (Annunciator Response Procedures)
5. OPs (Operating Procedures)

In this case N2-EOP-RPV entry is required due to the scram with reactor water level less than 159.3 inches. Since N2-EOP-RPV is a higher tiered procedure, reactor water should be maintained per the EOP prescribed band of 159.3-202.3 inches. Both reactor recirculation pumps are tripped and per N2-SOP-29, step 6.2.1 it would be required to raise RPV water level to 227 to 243 inches on the Shutdown Range to establish natural circulation, however since the EOP's are a higher tiered document than SOP's the required band would be 159.3-202.3 inches. For the second part of the answer, per N2-EOP-RPV you are allowed to restore and maintain RPV level and would only use preferred injection systems step L-3.

A. Plausible – For the first part of the distracter, the candidate may believe that since the reactor is operating in natural circulation that RPV level should be raised per N2-SOP-29. The second part of the distracter the candidate may think that since the feedwater pumps are available that they would be the quickest way to raise RPV level to prevent stratification.

B. Plausible – For the first part of the distracter, the candidate may believe that since the reactor is operating in natural circulation that RPV level should be raised per N2-SOP-29. For the second part of the distracter the candidate may think that it is appropriate to maximize CRD since it is a preferred injection system.

D. Plausible – The first part of the distracter is correct. For the second part of the distracter the candidate may think that it is acceptable to use alternate injection systems since there is not an ATWS in progress, however the use of alternate systems is only performed when RPV level cannot be restored and maintained between 159.3 and 202.3 inches.

Technical Reference(s): OP-NM-101-111-1001 (section 5.3), N2-EOP-RPV (step L-3), OP-AA-108-112, attachment 3 page 11 of 20

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01, Objective # N2-EOP-RPV-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(b)10

Comments:

Question #73 Is Withheld From Public Disclosure Due To Security Information Contained Therein.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.12
	Importance Rating	3.7

Knowledge of surveillance procedures.

Proposed Question: #74

Primary Containment Isolation Valve testing is being conducted on containment purge AOVs in accordance with N2-OSP-CPS-Q001, Primary Containment Purge System Valve Operability Test.

Which one of the following would be the correct method for determining whether an AOV's closure time was acceptable for a fully open AOV?

	Start the timing when the....	Stop the timing....
A.	control switch is taken to the close position	when the green closed indication illuminates
B.	green closed indication illuminates	when the red open indication extinguishes
C.	control switch is taken to the close position	when the red open indication extinguishes
D.	green closed indication illuminates	two seconds after red open indication extinguishes

Proposed Answer: C

Explanation: N2-OSP-CPS-Q001Sect. 4.2.1 Measuring Valve Stroke times for all valves except Solenoid Operated Valves.

- Measure opening stroke time from the time the control switch is placed to OPEN until the green indicating light de-energizes.
- Measure closing stroke time from the time the control switch is placed to CLOSE until the red indicating light de-energizes.

This simulates the receipt of the PCIS signal and the action of the valve in response.

A. Plausible – This is the timing closed method for a CPS SOV.
B. Plausible – The valve has to leave the full open seat before the intermediate limit switch picks up. This would result in an incomplete timing of the valve.
D. Plausible – The green light will illuminate as soon as the valve goes intermediate.

Technical Reference(s): N2-OSP-CPS-Q001, section 4.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-223003C01, RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.21
	Importance Rating	4.0

Emergency Procedures / Plan: 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.

Proposed Question: #75

The reactor is operating at rated power when annunciator 602116, RECIRC PMP 1B SEAL STAGING FLOW HIGH LOW, alarms. The following indications exist for Reactor Recirc Pump 1B:

- Seal staging flow is high (computer point RCSFC10)
- Lower (#1) Seal cavity pressure is 1020 psig
- Upper (#2) Seal cavity pressure is 830 psig

Which one of the following identifies the status of Reactor Recirc Pump B seals?

A. Only the lower seal is degraded

B. Only the upper seal is degraded

C. Both the upper and lower seals are partially degraded

D. Both the upper and lower seals have completely failed

Proposed Answer: A

Explanation: As the lower seal degrades, its pressure drop goes down, resulting in the upper seal cavity pressure rising toward the lower cavity pressure. Staging flow rises as a result of the degraded lower seal; hence, the annunciator cited in the stem; in this case for HI FLOW.

K/A Justification: This K/A, though it is a generic K/A, asks for specific knowledge about the parameters and logic used to assess the status of reactor coolant system integrity. This question provides indications associated with RCS seals (which is part of the reactor coolant system integrity) and makes the candidate determine reactor coolant system integrity.

A. Plausible – This answer is incorrect because a degraded upper seal would result in its seal cavity pressure lowering (towards Drywell atmospheric pressure), not rising toward lower seal pressure.
B. Plausible – Both seal pressures would drop if they were degraded.
D. Plausible – Both seal pressures would be ~350 psig if they had both completely failed

Technical Reference(s): N2-SOP-29.1, page 6, N2-OP-29 section H.5.0

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP291C01, Objective # N2-SOP-29.1-CE-01

Question Source: NMP NRC LC10-01, Question RO #72

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(b)10

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.05
	Importance Rating	3.5

Loss of Shutdown Cooling / 4

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor vessel metal temperature

Proposed Question: #76

The plant is in Mode 4 during a shutdown. Conditions are as follows:

- Shutdown Cooling has been interrupted due to an equipment malfunction.
- The following Reactor coolant temperature readings have been taken:

Time (hhmm)	Reactor Coolant Temperature (°F)
0000	140
0005	152

Assuming the Reactor coolant temperature trend remains constant, which one of the following describes the procedure that provides the required actions that mitigate these plant conditions and when the plant Mode will change, in accordance with Technical Specifications (TS)?

	Mitigating Procedure Direction	Time of Mode Change (hhmm)
A.	N2-SOP-31R, Refueling Operations Alternate Shutdown Cooling	0025
B.	N2-SOP-31, Loss of Shutdown Cooling	0030
C.	N2-SOP-31, Loss of Shutdown Cooling	0025
D.	N2-SOP-31R, Refueling Operations Alternate Shutdown Cooling	0030

Proposed Answer: C

Explanation: The current heat-up rate is 144°F/hr [(152°F – 140°F)*(60 min/hr)/(5 min)]. This is above the 100°F/hr limit of Technical Specification Surveillance Requirement 3.4.11.1. The plant transitions from Mode 4 to Mode 3 at 200°F. Based on the current heat-up rate, this temperature will be reached at time 0025. In this condition, the Reactor Cavity is not flooded to the level of the Spent Fuel Pool and therefore would not require the use of N2-SOP-31R. N2-SOP-31 is designed to be used when when in shutdown cooling without refueling operations occurring and the reactor cavity is not flooded.

Note: This question meets SRO level guidelines because it requires assessment plant conditions (normal, abnormal, or emergency) and then selection a procedure or section of a procedure to mitigate, recover, or with which to proceed.
K/A justification: This question is directly related to LCO 3.4.11 "RCS Pressure and Temperature (P/T) Limits" which establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

A. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. Candidate could fail to recognize the distinction between N2-SOP-31 and N2-SOP-31R. In this case the reactor is given to be in mode 4. N2-SOP-31R would only be used in Mode 5 with the cavity flooded and the gates removed.
B. Plausible –The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. For the second part of the distracter the candidate may either miscalculate heat-up rate or not know limit. 0030 is the time at which Reactor coolant temperature will reach 212°F and begin to boil. However, the Mode change is at 200°F, not 212°F.
D. Plausible – Both the first part and the second part of the distracters are incorrect, but plausible the candidate could fail to recognize the distinction between N2-SOP-31 and N2-SOP-31R. In this case the reactor is given to be in mode 4. N2-SOP-31R would only be used in Mode 5 with the cavity flooded and the gates removed. For the second part of the distracter, 0030 is the time at which Reactor coolant temperature will reach 212°F and begin to boil. However, the Mode change is at 200°F, not 212°F.

Technical Reference(s): N2-SOP-31, T.S. 3.4.11, Technical Specification Table 1.1-1 and Surveillance Requirement 3.4.11.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-101001C01, RBO-14

Question Source: Bank Previous NRC exam

Question History: 2014 NRC Exam, SRO #78

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295004 AA2.01
	Importance Rating	3.6

Partial or Total Loss of DC Pwr / 6

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Cause of partial or complete loss of D.C. power

Proposed Question: #77

The plant is operating at rated power with the following:

- A planned battery charger swap is in progress
- Operators are in the field removing Division 1 Battery charger 2BYS*CHGR2A1 and placing in service charger 2BYS*CHGR2A2
- Battery bus 2BYS*SWG002A voltage prior to the battery charger swap reads 135 VDC
- When 2BYS*CHGR2A1 breaker B302, "DC OUTPUT" is placed in OFF, 2BYS*SWG002A terminal voltage drops to 125 VDC and annunciator 852108, "DIV I EMER BUS BYS 002A 125VDC SYSTEM TROUBLE" alarms.

Which one of the following describes the OPERABILTY status with the associated reason?

	Division 1 DC Electrical Power subsystem is...	Reason
A.	Inoperable	Battery terminal voltage is less than 130 VDC with the associated battery charger out of service
B.	Inoperable	Incorrect breaker alignment has caused 2BYS*SWG002A terminal voltage to drop below 135 VDC
C.	Operable	Entry into associated Conditions and Required Actions may be delayed for up to 6 hours with the redundant function operable
D.	Operable	2BYS*SWG002A terminal voltage is maintained greater than 120 VDC from the battery

Proposed Answer: A

Explanation: Per T.S. 3.8.4, "The DC electrical power subsystems, each subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the divisions, are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Battery terminal voltage will drop to ~ 125VDC when a charger is removed from the switchgear and SR 3.8.4.1 requires battery terminal voltage verified ≥ 130 V on float charge. Therefore T.S. 3.8.4 LCO must be entered and appropriate required actions taken.

B. Plausible – The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. Per T.S. 3.8.8 bases, OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. The proper voltage is defined in Table B 3.8.8-1:

TYPE	VOLTAGE	DIVISION 1[a]	DIVISION 2[a]	DIVISION 3[a]
DC buses	125 V	Switchgear 2BYS*SWG002A MCC 2DMS*MCCA1 Distribution Panels 2BYS*PNL201A, 2BYS*PNL202A, and 2BYS*PNL204A	Switchgear 2BYS*SWG002B MCC 2DMS*MCCB1 Distribution Panels 2BYS*PNL201B, 2BYS*PNL202B, and 2BYS*PNL204B	Distribution Panel 2CES*IPNL414

The required voltage is 125VDC which is the value given for 2BYS*SWG002A and it is being energized from its respective battery. Therefore, the switchgear meets its minimum voltage and T.S. 3.8.8 would not apply, however the candidate may think that since the annunciator is in alarm that the proper voltage is not met and that the field operator incorrectly operated a breaker locally.

C. Plausible – Both the first part and the second part of the distracters are incorrect, but plausible. For the first part of the distracter, the candidate could think that for a normal evolution battery charger swap that the DC susbsystem would not be inoperable. For the second part of the distracter, typical surveillance notes provide allowance to delay entry into associated Conditions and Required Actions for up to 6 hours while performing required surveillances provided redundant function maintains capability. TS 3.8.4 does not provide this note.

D. Plausible – Both the first part and the second part of the distracters are incorrect, but plausible For the first part of the distracter, the candidate could think that for a normal evolution battery charger swap that the DC susbsystem would not be inoperable. For the second part of the distracter, The candidate could think that since the system is a 120 volt system that it is acceptable to have DC voltage at 120 volts or higher.

Technical Reference(s): Technical specifications 3.8.4 and 3.8.8
Tech Spec Bases, section 3.8.4 and 3.8.8

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-263000C01, RBO-14

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295030 EA2.02
	Importance Rating	3.9

Low Suppression Pool Water Level / 5

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool temperature

Proposed Question: #78

The plant is shutdown following a LOCA. Conditions are as follows:

- Suppression Pool Level is 196.5 feet and steady
- "A" RHS is operating in LPCI mode
- "B" RHS is operating in Suppression Pool Cooling mode

Per NER-2M-039, NMP Unit 2 EOP Bases, which one of the following describes the method for determining suppression pool temperature AND the time limit to restore suppression pool level within Tech Spec limits?

	Method for determining temperature	Time Limit
A.	RHS heat exchanger inlet temperature on "B" RHS.	2 Hours
B.	RHS heat exchanger inlet temperature on "B" RHS.	7 Days
C.	Suppression Pool temperature indication on SPDS.	2 Hours
D.	Suppression Pool temperature indication on SPDS.	7 Days

Proposed Answer: A

Explanation: With Suppression Pool level low, temperature elements will become uncovered and inaccurate. In order to determine SP level with S/P level <197 ft, one method given for determining temperature is the HTX inlet temp provided that the pump is aligned with suction from the suppression pool. For the second part of the question, Per TS (suppression pool water level shall be $\geq$ 199 ft 6 inches and $\leq$ 201 ft) 3.6.2.2 condition A, "Suppression pool water level not within limits" required action A.1, "Restore suppression pool water level to within limits" within 2 hours.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed. With Suppression Pool level low, temperature elements will become uncovered and inaccurate. Therefore the candidate will have to analyze the level in the suppression pool determine if it is below the lowest elevation temperature detector and then determine the proper EOP action to determine actual suppression pool temperature and candidate will have to recall TS specific action time for a given condition.

B. Plausible – The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the candidate could confuse the TS completion times associated with RHR that are 7 day durations with the suppression pool level completion times.

C. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part of the distracter, the candidate could think that the temperature elements remain covered at elevation 196.5', however at this elevation all temperature indicators are uncovered.

D. Plausible – Both the first part and the second part of the distracters are incorrect, but plausible. For the first part of the distracter, the candidate could think that the temperature elements remain covered at elevation 196.5', however at this elevation all temperature indicators are uncovered. For the second part of the distracter, the candidate could confuse the TS completion times associated with RHR that are 7 day durations with the suppression pool level completion times.

Technical Reference(s): NER-2M-039, Rev 08.00, N2-EOP-PC Basis, page 3-106

Proposed references to be provided to applicants during examination: TS 3.6.2.2 No bases

Learning Objective: 2101-EOPPCC01, Objective # N2-EOP-PC-CE-02

Question Source: Bank

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b)2

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295019 2.4.18
	Importance Rating	4.0

Partial or Total Loss of Inst. Air / 8

Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.

Proposed Question: #79

The plant is in a failure to scram condition:

- A manual scram using the reactor mode switch was conducted
- 128 control rods are still not full in
- Concurrently with the event, Instrument Air supply to the SDV Vent and drain valves has been lost
- RPS has tripped
- Scram Pilot valves are de-energized
- Scram valves are open

Which one of the following sections of N2-EOP-6.14 should be directed as a method to insert control rods?

A. 6.5, "Manual Control Rod Insertion"

B. 6.3, "Additional Manual Scram Initiation"

C. 6.2, "Scram Air Header Venting"

D. 6.4, "Individual Control Rod Scrams"

Proposed Answer: A

Explanation: In this situation with the instrument air supply to the SDV Vent and drain valves lost, the SDV vent and drain valves will fail closed. This failure will preclude the use of scrams to get the control rods inserted. Also, with RPS tripped and the scram pilot valves de-energized, actions conducted per N2-EOP-6.14 that address de-energizing the scram pilot solenoid valves would not be effective. Per the EOP bases NER-2M-039, Rev 08.00 pages 3-328 thru 3-330, "*Drive control rods, defeating RWM interlocks if necessary.*
This method is best applied when only a few control rods cannot be inserted, alternate methods are being performed which cannot be performed continuously (such as inserting additional manual scrams), the scram cannot be reset, or individual control rod scrams are not effective. To assist in driving individual control rods, the procedure provides direction to maximize drive pressure by starting an additional CRD pump, shutting the drive water pressure control valve, and opening the flow control valve."

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed.(ES-401, Attachment 2, Figure 2)

B. Plausible – Per NER-2M-039, Rev 08.00, "Reset *the scram, defeating RPS logic trips if necessary, drain the scram discharge volume, and initiate a manual scram.*
This method may be effective where control rods are stuck, reactor pressure and accumulator pressure are not sufficient to effect a full control rod scram, or the scram system functioned but did not result in full control rod insertion. A reactor scram is repeated as long as control rod movement occurs by starting with a drained scram discharge volume and charged accumulators. Scram signals may exist due to other plant conditions such as high drywell pressure. These must be defeated to allow scram reset. However this method is not permissible since the SDV vent and drain valves have failed closed. This is plausible because the candidate could not recognize the significance of the loss of instrument air to the SDV vent and drain valves.

C. Plausible – Per NER-2M-039, Rev 08.00, "*Vent the scram air header,*"
This method is effective only where one or more scram valves did not open and the HCU area is accessible. While this method will not open valves that are mechanically prevented from opening, it will allow those to open which are still being held closed by air pressure. To be effective, the scram air header must be depressurized before the scram discharge volume pressurizes sufficiently (from bypass leakage on control rods which did scram) to prevent further control rod movement..

D. Plausible – Per NER-2M-039, Rev 08.00, "*Individual control rod scrams,*"
This method acts on only a single control rod at a time, but can be repeated quickly for many rods. If the scram can be reset, this method may be more effective than a full core scram because the total available differential pressure of the CRD hydraulic system is applied to the single selected control rod. Since the scram discharge volume vent and drain valves remain open, the maximum differential pressure is applied over the full travel of the control rod. Also, the rate of water loss from the RPV through the control rod drive mechanism is small.

Technical Reference(s): NER-2M-039, Rev 08.00 pages 3-328 thru 3-330, N2-EOP-6.14, N2-SOP-19 section 5.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOP6HCC01, Objective # N2-EOP-6-CE2-A14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295018 2.2.42
	Importance Rating	4.6

Partial or Total Loss of CCW / 8

Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications

Proposed Question: #80

The plant is operating at 100% power.

Which of the following service water conditions would require that the plant be placed in Mode 3 within 12 hours of discovering the condition?

Condition 1: Service water supply header water temperature is 80 degrees with only 2 SW pumps in operation

Condition 2: Service water supply header water temperature is 85 degrees with only 4 SW pumps in operation

Condition 3: Inadvertently closing 2SWP*MOV50A, PMP 1A DISCH HEADER CROSS-TIE ISOL VLV

Condition 4: Breaker for 2SWP*MOV3A, OUTLET TO TURBINE BLDG tripping open

A. Condition 1 only

B. Condition 2 only

C. Condition 1 and 3

D. Condition 2 and 4

Proposed Answer: B

Explanation: With a SW temperature of 85 degrees the Ultimate Heat sink is not operable (see bases page B3.7.1-5). Per TS 3.7.1, condition G, the plant must be in mode 3 within the next 12 hours.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(2) based on requiring the applicant to assess Technical Specification conditions involving > 1 hour action statements. The question also requires knowledge of Technical Specification Bases.

A. Plausible – This condition may eventually lead to requiring the plant to be placed in mode 3 within 12 hours. With SW temperature < 82 degrees, 4 SW pumps are required to be in operation. However, TS 3.7.1, condition F allows 1 hour to restore one of the two required pumps to operation before action is required to place the plant in mode 3 within 12 hours per condition G.
C. Plausible – Condition 1 may eventually lead to requiring the plant to be placed in mode 3 within 12 hours. With SW temperature < 82 degrees, 4 SW pumps are required to be in operation. However, TS 3.7.1, condition F allows 1 hour to restore one of the two required pumps to operation before action is required to place the plant in mode 3 within 12 hours per condition G. Similar to condition 1, TS 3.7.1 allows 1 hour to fix the condition prior to requiring reactor mode change.
D. Plausible – Condition 2 does require the plant to be placed in Mode 3 within 12 hours. (See explanation above) Condition 4 does not require the action of TS 3.7.1 condition G to be immediately executed. 72 hours is allowed to correct the condition prior to requiring a reactor mode change.

Technical Reference(s): TS 3.7.1

Proposed references to be provided to applicants during examination: TS 3.7.1 No Bases

Learning Objective: 2101-276000C01, Service Water, RBO-14, Application of Technical Specifications

Question Source: Bank

Question History: 2011 Audit 81

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b) 2

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295038 2.4.8
	Importance Rating	4.5

High Off-site Release Rate / 9

Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOP's.

Proposed Question: #81

Following a fuel failure the crew entered the following procedures:

- N2-SOP-17, Fuel Failure or High Activity in Rx Coolant or Off-Gas.
- N2-SOP-09, Loss of Condenser Vacuum.
- N2-SOP-101C, Reactor Scram

Following the scram, the off-site release rate exceeds the Alert Emergency Action Level.

Which one of the following actions is required in accordance with NER-2M-039, NMP2 EOP and SAP Basis Document?

A. Enter N2-EOP-RPV, RPV Control, only, and continue in all the SOPs as necessary.

B. Enter N2-EOP-RR, Radioactivity Release Control and continue in all the SOPs as necessary.

C. Enter N2-EOP-RPV, RPV Control, only, and continue with N2-SOP-101C and N2-SOP-17, exit N2-SOP-09.

D. Enter N2-EOP-RR, Radioactivity Release Control and continue with N2-SOP-101C, exit N2-SOP-09 and N2-SOP-17.

Proposed Answer: B

Explanation: Declaring an Alert based on a radioactive release is an entry condition for N2-EOP-RR. Actions prescribed in EOPs take precedence over actions required in other plant procedures; however the other SOP actions are still carried out provided they do not conflict with the EOP.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to analyze and interpret facility conditions as they pertain to the selection of the appropriate abnormal and emergency procedures. Per ES-401, Attachment 2, section II.D.

A. Plausible – Entering N2-SOP-101C may require entry into N2-EOP-RPV. However, declaring an Alert based on a release is an entry condition for N2-EOP-RR.
C. Plausible – Entering N2-SOP-101C may require entry into N2-EOP-RPV. However, declaring an Alert based on a release is an entry condition for N2-EOP-RR. The other SOP actions are still carried out provided they do not conflict with the EOP. There is no requirement to exit any SOPs. Although the applicant may think it is no longer necessary to execute SOP-09 and therefore it can be exited.
D. Plausible – Declaring an Alert based on a radioactive release is an entry condition for N2-EOP-RR. There is no requirement to exit any SOPs. Although the applicant may think it is no longer necessary to execute SOP-09 and therefore it can be exited. The applicant may think SOP-17 actions no longer apply due to entering EOP-RR and therefor it can be exited.

Technical Reference(s): Pg. 3-8 of EOP Bases, N2-EOP-RR

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank

Question History: 2009 NRC 81

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b) 5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295025 EA2.05
	Importance Rating	3.6

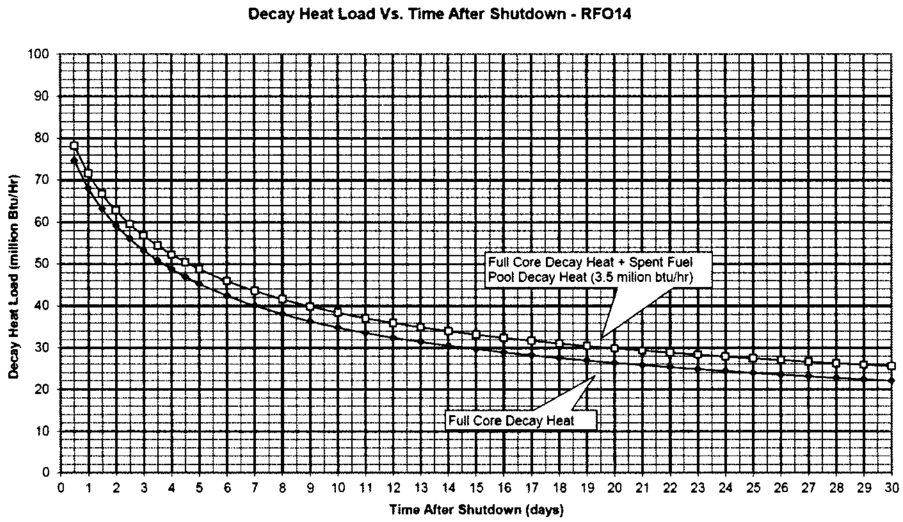
High Reactor Pressure / 3

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Decay heat generation.

Proposed Question: #82

The plant is operating at rated power with the following:

- The plant is operating at the end of core life
- Normal electrical plant lineup
- A loss of the electrical grid occurs
- Simultaneously, with the loss of the electrical grid, the running EHC pump (2TMB-P1A) trips
- 2TMB-P1B fails to auto start and cannot be manually started



Which one of the following describes the required procedure direction to address reactor pressure AND the expected decay heat load 24 hours (1 day) after the event?

	Direct reactor pressure stabilized with a band of 800-1000 psig using…	Decay Heat Load
A.	SRV's and TBV's	72 mBtu/Hr
B.	SRV's and RCIC	68 mBtu/Hr
C.	SRV's and TBV's	68 mBtu/Hr
D.	SRV's and RCIC	72 mBtu/Hr

Proposed Answer: B

Explanation: For the first part of the answer, with both EHC pumps unavailable the main turbine bypass valves will not be able to be used for the event. Per OP-NM-101-111-1001, "Transient Mitigation Guidelines (TMG)" Attachment 2, Unit 2 Specific Transient Mitigation Guidelines (TMG), section 2.3.7:

Standardize Pressure Control Bands for conditions other than nominal:

800 – 1000 psig	SRVs are cycling, condenser & turbine pressure control available.
800 – 1000 psig	Using SRVs when condenser and / or turbine pressure control system is NOT available
500 – 1000 psig	Sustained opening of SRVs required due to loss of nitrogen to the DW. This does not apply when in EOP-C3 due to the need to conserve reactor water inventory.
500 – 600 psig	Reduce and stabilize reactor pressure to allow booster pump injection on loss of HP feed or High Pressure Injection

For the second part of the answer, using the "Decay Heat Load Vs. Time After Shutdown - RFO14" graph provided, the Full Core decay Heat curve would be used. The question asked for decay heat load 24 hours after shutdown. Using "1" on the x-axis and following it up to the full core decay heat load curve, the plotted decay heat load would be 68 mBtu/Hr.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed.

A. Plausible – Both parts of the distracter are incorrect, but plausible. For the first part of the distracter the candidate could not recognize that the failure of both EHC pumps would prevent the use of the main turbine bypass valves. For the second part of the distracter the candidate could mistakenly use the Full Core Decay Heat + Spent Fuel Pool Decay Heat line to determine decay heat load.

C. Plausible – The first part of the distracter is incorrect, but plausible the second part of the distracter is correct. For the first part of the distracter, the candidate could not recognize that the failure of both EHC pumps would prevent the use of the main turbine bypass valves.

D. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter the candidate could mistakenly use the Full Core Decay Heat + Spent Fuel Pool Decay Heat line to determine decay heat load.

Technical Reference(s): OP-NM-101-111-1001, "Transient Mitigation Guidelines (TMG)" Attachment 2, Unit 2 Specific Transient Mitigation Guidelines (TMG), section 2.3.7., N2-OP-115 Attachment 4

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01, Objective # N2-EOP-RPV-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295003 EA2.03
	Importance Rating	4.2

High Secondary Containment Area Radiation Levels / 9

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Cause of high area radiation

Proposed Question: #83

The plant is operating at rated power with the following:

- 2RMS-RE102, "Equipment drains sumps and pumps E" initial reading is 3.29 E^{-1}
- 2RMS-RE104, "Equipment drains sumps and pumps W" initial reading is 1.08 E^{0}
- A plant event occurs that results in the following:
  - 2RMS-RE102, "Equipment drains sumps and pumps E" area radiation monitor goes into red alarm on DRMS and is now reading 4.86 E^{+2}
  - 2RMS-RE104, "Equipment drains sumps and pumps W" area radiation monitor reading rises to 1.84 E^{0}
- N2-EOP-SC has been entered

> Note:
> - WCS Pump casing leaks drain to 2DER-TK2A
> - WCS NRHX flange break drains to 2DFR-TK2A

Which one of the following describes (1) a potential source of the rising area radiation level and (2) identifies the required implementation of procedures for this event if the source cannot be isolated?

A. (1) WCS pump casing leak
(2) Remain in N2-EOP-SC and enter N2-EOP-RPV to perform a manual reactor scram

B. (1) WCS pump casing leak
(2) Transition out of N2-EOP-SC and utilize N2-OP-101D & C to commence a normal reactor shutdown

C. (1) Flange break on the inlet side of the WCS NRHX
(2) Remain in N2-EOP-SC and enter N2-EOP-RPV to perform a manual reactor scram

D. (1) Flange break on the inlet side of the WCS NRHX
(2) Transition out of N2-EOP-SC and utilize N2-OP-101D & C to commence a normal reactor shutdown

Proposed Answer: A

Explanation: For the first part of the answer, this would be considered a primary system discharging into the secondary containment. Per N2-EOP-SC flowchart step SC-5, If a primary system is discharging into the secondary containment and the discharge cannot be isolated then it is required to enter N2-EOP-RPV control to perform a reactor scram.

A WCS pump casing leak would drain to equipment drain 2DER-ED3401 (PID-037B to PID-63A) and eventually 2DER-TK2A and per EM-002A, 2DER-TK2A is located in reactor building on elevation 175' at approximately reactor building 180° (180° is due east, true magnetic north is at reactor building 90°). Also per EM-002A, 2RMS-RE102 is at approximately reactor building 200° on elevation 175' in the reactor building. Therefore, with the radiation levels given above, the leak must originate from a source that is draining into 2DER-TK2A. 2RMS-RE102, "Equipment drains sumps and pumps E" value has risen more than 1000 times its original value where 2RMS-RE104, "Equipment drains sumps and pumps W" has remained relatively unchanged.

EM-002H, Machine location Reactor Building Plan 306'6' shows the NRHX on the 270 degree azimuth of the reactor building. Using EB10Q Floor & Equip Drainage Plan Elevation 306'6", the floor drain discussed in the question would be DNF 7204. DNF 7204 drains to EB-10P (E-5), which then drains to EB-10M (E-8), which then drains to EB-10K (E-8), which then drains to EB-10G (E-8), which then drains to EB-10W (E-8), which then drains to EB-10C (E-8), which then drains to EB-10B to Floor Drain Sump 2DFR-TK2A at 0 degree azimuth which is on the west side of the reactor building.

<u>Note:</u> This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

B. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the candidate could incorrectly proceed down the non-primary system discharging leg and think that a controlled reactor shutdown is required per N2-OP-101D & C.
C. Plausible – The first part of the distracter is incorrect, but plausible the second part of the distracter is correct. For the first part of the distracter the candidate could think The candidate could incorrectly assess and determine that leak is draining into a sump that is on the west side (reactor building 270°) which is near 2RMS-RE104 (reactor building 340°) and the second part of the distracter is correct.
D. Plausible – Both parts of the distracter are incorrect, but plausible. In the first part of the distracter, the candidate could incorrectly assess and determine that leak is draining into a sump that is on the west side (reactor building 270°) which is near 2RMS-RE104 (reactor building 340°). For the second part of the distracter, the candidate could incorrectly proceed down the non-primary system discharging leg and think that a controlled reactor shutdown is required per N2-OP-101D & C.

Technical Reference(s): N2-EOP-SC, N2-EOP-RPV, EM-002A

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-291000C01, RBO-03

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295036 2.4.41
	Importance Rating	4.6

Secondary Containment High Sump/Area Water Level / 5

Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #84

The plant is shutdown in preparation for a refueling outage with the following:

- Spent Fuel Pool gates are still installed
- Vessel head closure bolts are still tensioned
- Reactor water temperature is 185°F
- RPV floodup has not yet been commenced
- Shutdown cooling is in service
- RPV level is being maintained 227-243 inches on the shutdown range level indicator using CNM-HIC137 in automatic and WCS reject to the main condenser
- A loss of RPV level indication occurs for a unknown reason
- Reactor Building floor drain sump levels rise and reach the High-High level
- Annunciator 851453, "Reactor Bldg Floor Drain System Trouble" alarms
- This condition has been occurring for 15 minutes
- Efforts are in progress to restore RPV level indication

The declaration of an Alert emergency condition is based upon which one of the following?

A. Loss or potential loss of the RCS barrier only

B. Loss or potential loss of the Containment barrier only

C. Loss or potential loss the RCS and Fuel Clad barrier only

D. Loss of the Fuel Clad & RCS barrier and potential loss of the Containment barrier

Proposed Answer: A

Explanation: Per EP-AA-1013 Addendum 4 section 5.2.C.1, "The following criteria are the bases for event classification related to fission product barrier loss or potential loss:

Unusual Event:
Any loss or any potential loss of Containment

Alert:
Any loss or any potential loss of either Fuel Clad or RCS

Site Area Emergency:
Loss or potential loss of any two barriers

General Emergency:
Loss of any two barriers and loss or potential loss of third barrier"

Per EAL CA3.1,
"RPV water level < 17.8 in.
OR
RPV water level cannot be monitored for $\geq$15 min. with ANY UNPLANNED RPV leakage indication, Table C-2 (Note 4)"

Table C-2 RPV Leakage Indications

- Drywell equipment drain sump level rise
- Drywell floor drain sump level rise
- Reactor building equipment sump level rise
- Reactor Building floor drain sump level rise
- Suppression Pool level rise
- UNPLANNED rise in RPV make-up rate
- Observation of UNISOLABLE RCS leakage

Per EAL CA3.1 bases, page 136 of 263, "The inability to restore and maintain level after reaching this setpoint would be indicative of a failure of the RCS barrier."

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

B. Plausible –The candidate could think that this is one barrier degradation and think that since the plant is in Mode 4 that the containment barrier is applicable barrier for the determination of the Alert classification.
C. Plausible –The candidate could incorrectly think that since the distracter said "potential loss" that this would be the appropriate barrier. Also the candidate could think that both barriers are affected during this event.
D. Plausible –The candidate could incorrectly think that since the distracter said potential loss for the containment barrier that it could apply. Also, the candidate could think that both the RCS and Fuel Clad barriers would be impacted, and since the plant is in Mode 4 that the different EAL chart and associated requirements would be applicable.

Technical Reference(s): EP-AA-1013, addendum 4, section CA3.1, EP-AA-1013 Addendum 4 section 5.2.C.1.

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EPL001C01, Objective # N2-EAL-UP-CE-1.09

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	500000 EA2.02
	Importance Rating	3.5

High CTMT Hydrogen Conc. / 5

Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Oxygen monitoring system availability

Proposed Question: #85

The plant is operating at rated power with the following:

- A LOCA occurs
- Suppression Pool level is 200.5 ft
- During the LOCA response actions the LOCA isolation signal for the Division 1 and 2 Containment Monitoring System (CMS) is defeated and CMS is unisolated
- 2CMS*P2A & B, H2/O2 ANALYZER PMP are both running.
- After the 30 minute warm up for CMS has been completed it is determined that Drywell Hydrogen concentrations are reading 5.5% and Drywell oxygen concentration indications are not functioning.
- Suppression chamber hydrogen concentration is determined to be 1%
- The offsite release rate is expected to stay below the ODCM limit
- Containment Temperature is 200°F
- Containment Pressure is 16 psig

With respect to N2-EOP-PCH Drywell table only, which one of the following actions should be directed?

<u>**Note:**</u> A portion of N2-EOP-PCH is attached on the following page.

A. Operate recombiners with suction from the drywell (OP-62, Section E.2.0)

B. Trip all recirc. pumps and drywell unit coolers and operate drywell sprays (EOP-6.22)

C. Purge the drywell through the suppression pool with nitrogen at the maximum rate (EOP-6.25)

D. Purge the drywell through the suppression pool with air or nitrogen at the maximum rate (EOP-6.25)

DRYWELL

		Drywell O_2: < 5%	≥ 5% or unknown — Suppression Chamber H_2: None	< 6%	≥ 6% or unknown
Drywell H_2	None	No action	No action	32	33
	< 6%	31			
	≥ 6% or unknown				

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- IF....... drywell and suppression chamber hydrogen concentrations are below 5%
  OR
  drywell and suppression chamber oxygen concentrations are below 5%,
  AND.... drywell parameters are within the limits of recombiner operation (Table T),
  THEN...operate recombiners with suction from the drywell (OP-62, Section E.2.0).
  - Stop recombiners taking a suction on the drywell when:
    - Drywell parameters reach the limits of recombiner operation (Table T),
      OR
    - Drywell or suppression chamber hydrogen is above 5%
      AND
      drywell or suppression chamber oxygen is above 5%.
- IF....... the offsite release rate is expected to stay below the ODCM limit,
  AND.... the drywell can be vented,
  THEN...purge the drywell with nitrogen (EOP-6.25).
  - ☛ OK to defeat isolations except high radiation.
  - ☛ If suppression pool water level is below El. 217 ft, purge through the suppression pool if possible.
  - Stop the purge when:
    - Hydrogen is no longer detected in the drywell,
      OR
    - The offsite release rate reaches the ODCM limit.

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32

- IF....... drywell and suppression chamber hydrogen concentrations are below 5%
  OR
  drywell and suppression chamber oxygen concentrations are below 5%,
  AND.... drywell parameters are within the limits of recombiner operation (Table T),
  THEN...operate recombiners with suction from the drywell (OP-62, Section E.2.0).
  - Stop recombiners taking a suction on the drywell when:
    - Drywell parameters reach the limits of recombiner operation (Table T),
      OR
    - Drywell or suppression chamber hydrogen is above 5%
      AND
      drywell or suppression chamber oxygen is above 5%.
- IF....... the offsite release rate is expected to stay below the General Emergency level
  OR
  you cannot assure adequate core cooling,
  AND.... the drywell can be vented,
  THEN...purge the drywell with nitrogen at the maximum rate (EOP-6.25).
  - ☛ OK to defeat all isolations.
  - ☛ If suppression pool water level is below El. 217 ft, purge through the suppression pool if possible.
  - Stop the purge when:
    - The offsite release rate reaches the General Emergency level and adequate core cooling is assured,
      OR
    - Hydrogen is no longer detected in the drywell
      AND
      drywell oxygen is less than 5% or hydrogen is no longer detected in the suppression chamber

6

33

1. Stop recombiners taking suction on the drywell (OP-62, Section G.1.0).
2. **BLOW DOWN**:
   1. Enter EOP-RPV while continuing here. ↓
   2. Enter EOP-C2 while continuing here. ↓
3. IF....... the drywell can be vented,
   THEN...purge the drywell with air or nitrogen at the maximum rate (EOP-6.25).
   - ☛ **Use whichever method will reduce hydrogen below 6% or oxygen below 5% faster.**
   - ☛ **OK to defeat all isolations.**
   - ☛ **OK to exceed release rate limits.**
   - ☛ **If suppression pool water level is below El. 217 ft, purge through the suppression pool if possible.**
4. IF....... drywell temperature is below the Drywell Spray Initiation Limit (Fig K),
   AND.... suppression pool water level is below El. 217 ft,
   THEN...spray the drywell while continuing here: ↓
   1. Trip all recirc pumps.
   2. Trip all drywell unit coolers.
   3. Operate drywell sprays (EOP-6.22).
      - ⚠ **Operating RHS with suppression pool water level below El. 195 ft may cause system damage.**
      - ☛ Operate sprays even if core cooling will be lost.
      - ☛ OK to use alternate spray sources but only if you can restore and maintain suppression chamber pressure below the Primary Containment Pressure Limit (Fig D):
        - Service Water (EOP-6.5)
        - Fire Water (EOP-6.6)
      - ☛ OK to defeat spray interlocks (EOP-6.22).
      - ☛ Reducing primary containment pressure affects margin to NPSH limits (EOP-6.29).

Proposed Answer: C

Explanation: Per N2-EOP-PC it is required to enter N2-EOP-PCH when hydrogen is detected in the drywell or suppression chamber. Per N2-EOP-PCH Drywell table:

DRYWELL

<table>
<tr><td colspan="2" rowspan="3"></td><td colspan="4">Drywell O_2</td></tr>
<tr><td rowspan="2">< 5%</td><td colspan="3">≥ 5% or unknown
Suppression Chamber H_2</td></tr>
<tr><td>None</td><td>< 6%</td><td>≥ 6% or unknown</td></tr>
<tr><td rowspan="3">Drywell H_2</td><td>None</td><td>No action</td><td>No action</td><td rowspan="2">32</td><td rowspan="3">33</td></tr>
<tr><td>< 6%</td><td rowspan="2">31</td><td></td></tr>
<tr><td>≥ 6% or unknown</td><td colspan="2"></td></tr>
</table>

4

With Drywell hydrogen concentration indication given to be 5.5%, Drywell oxygen concentration given to be unknown and suppression chamber hydrogen concentration given to be at 1%, action number 32 would be required.

32

- IF........drywell and suppression chamber hydrogen concentrations are below 5%
  OR
  drywell and suppression chamber oxygen concentrations are below 5%,
  AND.... drywell parameters are within the limits of recombiner operation (Table T),
  THEN...operate recombiners with suction from the drywell (OP-62, Section E.2.0).
  - Stop recombiners taking a suction on the drywell when:
    - Drywell parameters reach the limits of recombiner operation (Table T),
      OR
    - Drywell or suppression chamber hydrogen is above 5%
      AND
      drywell or suppression chamber oxygen is above 5%.
- IF........the offsite release rate is expected to stay below the General Emergency level
  OR
  you cannot assure adequate core cooling,
  AND.... the drywell can be vented,
  THEN...purge the drywell with nitrogen at the maximum rate (EOP-6.25).
  - OK to defeat all isolations.
  - If suppression pool water level is below El. 217 ft, purge through the suppression pool if possible.
  - Stop the purge when:
    - The offsite release rate reaches the General Emergency level and adequate core cooling is assured,
      OR
    - Hydrogen is no longer detected in the drywell
      AND
      drywell oxygen is less than 5% or hydrogen is no longer detected in the suppression chamber

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed.

A. Plausible – This is plausible if the candidate does not evaluate N2-EOP-PCH Table "T". Per table "T" hydrogen concentration limit is < 5%. In the given it was provided to be 5.5%, therefore use of the recombiners would not be permitted.

B. Plausible – This is plausible if the candidate incorrectly determines the actions per the Drywell table to be action #33.

D. Plausible – This is plausible if the candidate incorrectly determines the actions per the Drywell table to be action #33.

Technical Reference(s): N2-EOP-PCH, EOP Bases NER-2M-039 Rev 0800 PCH pages 3-126 through 3-128

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCHC01, Objective # N2-EOP-PCH-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	203000 A2.14
	Importance Rating	3.9

RHR/LPCI: Injection Mode

Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Initiating logic failure

Proposed Question: #86

The plant is in Mode "4" with the following:

- RHS A is in shutdown cooling
- N2-OSP-RHS-R001, "Division 2 ECCS Functional Test" is in progress
- When LPCI B & C manual initiation pushbutton is armed and depressed the initiation fails to function
- Annunciator 601652, "RHR B Manual Initiation Switch Armed" alarms
- Annunciator 601651, "RHR B System Actuated" does NOT alarm
- Further troubleshooting reveals that the problem has been isolated to the arm and depress pushbutton on 2CEC*PNL601

Which one of the following describes the impacts to LPCI availability and the associated mitigating actions?

	LPCI impacts	Mitigating actions
A.	RHS B & C are available for injection	Restore channel to OPERABLE status within 24 hrs
B.	RHS B & C are NOT available for injection	Restore channel to OPERABLE status within 24 hrs
C.	RHS B & C are NOT available for injection	Verify at least two ECCS injection/spray subsystems are OPERABLE
D.	RHS B & C are available for injection	Verify at least two ECCS injection/spray subsystems are OPERABLE

Proposed Answer: D

Explanation: In the given information, the question stated that troubleshooting revealed that the problem was isolated to the arm and depress pushbutton on 2CEC*PNL601. This does not prevent the Division 1 ECCS pumps from starting from LOCA signals or from being manually started and manually lined up for injection. So in this case RHS B & C are available for injection. The mitigating actions are determined by using T.S. since the failure during the surveillance was isolated to the arm and depress pushbutton on 2CEC*PNL601, T.S. 3.3.5.1 would be evaluated. Using T.S. table 3.3.5.1-1, function 2k is applicable. The plant is in mode 4 so function 2k is only applicable when associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2. LCO 3.5.2 states that "Two ECCS injection/spray subsystems shall be OPERABLE" which in this case is satisfied since LPCS and RHS A are operable even with RHS A in SDC. (per SR3.5.2.4 note, "One low pressure coolant injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable."). Therefore, there are no required actions from LCO 3.5.2 or 3.3.5.1. The only mitigating action that should be performed is to verify at least two ECCS injection/spray subsystems are OPERABLE.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the question involving the Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1) of TS.

A. Plausible –The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. T.S. table 3.3.5.1-1 requires that the required actions of condition C be evaluated. There are two required actions contained in Condition C (C.1 & C.2). Required action C.1 note 1 states, "Only applicable in MODES 1, 2, and 3 and note 2 states, "Only applicable for Functions 1.e, 1.f, 1.g, 1.h, 1.i, 1.j, 2.e, 2.f, 2.g, 2.h, and 2.i." So, required action C.1 is not applicable. Required action C.2 states to "Restore channel to OPERABLE status" within 24 hours, however this action is not required either since the applicable mode or other specified condition is 4(a) and per the given information, this would not be applicable. If the candidate does not recognize note (a) at the bottom of T.S. table 3.3.5.1-1, then the candidate could pick this as an answer choice.

B. Plausible – Both parts of the distracter are incorrect, but plausible. In the first part of the distracter, the only failure for the Division 1 ECCS systems is the failure of the manual initiation pushbutton, therefore the systems will still initiate automatically with a LOCA signal and can also be started and aligned manually for injection. The candidate could believe that this malfunction would prevent the use of these RHR system for LPCI. The second part of the distracter, T.S. table 3.3.5.1-1 requires that the required actions of condition C be evaluated. There are two required actions contained in Condition C (C.1 & C.2). Required action C.1 note 1 states, "Only applicable in MODES 1, 2, and 3 and note 2 states, "Only applicable for Functions 1.e, 1.f, 1.g, 1.h, 1.i, 1.j, 2.e, 2.f, 2.g, 2.h, and 2.i." So, required action C.1 is not applicable. Required action C.2 states to "Restore channel to OPERABLE status" within 24 hours, however this action is not required either since the applicable mode or other specified condition is 4(a) and per the given information, this would not be applicable. If the candidate does not recognize note (a) at the bottom of T.S. table 3.3.5.1-1, then the candidate could pick this as an answer choice.

C. Plausible –The first part of the distracter is incorrect, but plausible the second part of the distracter is correct. In the first part of the distracter, the only failure for the Division 1 ECCS systems is the failure of the manual initiation pushbutton, therefore the systems will still initiate automatically with a LOCA signal and can also be started and aligned manually for injection. The candidate could believe that this malfunction would prevent the use of these RHR systems for LPCI. In

the second part of the distracter, the mitigating actions are determined by using T.S. since the failure during the surveillance was isolated to the arm and depress pushbutton on 2CEC*PNL601, T.S. 3.3.5.1 would be evaluated. Using T.S. table 3.3.5.1-1, function 2k is applicable. The plant is in mode 4 so function 2k is only applicable when associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2. LCO 3.5.2 states that "Two ECCS injection/spray subsystems shall be OPERABLE" which in this case is satisfied since LPCS and RHS A are operable even with RHS A in SDC. (per SR3.5.2.4 note, "One low pressure coolant injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable."). Therefore, there are no required actions from LCO 3.5.2 or 3.3.5.1. The only mitigating action that should be performed is to verify at least two ECCS injection/spray subsystems are OPERABLE.

Technical Reference(s): T.S. 3.3.5.1, T.S. Table 3.3.5.1-1 (2K section), T.S. 3.5.2

Proposed references to be provided to applicants during examination: T.S. 3.3.5.1 and table 3.3.5.1-1, T.S. 3.5.2 no bases

Learning Objective: 2101-205000C01, RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)2

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	211000 A2.04
	Importance Rating	3.4

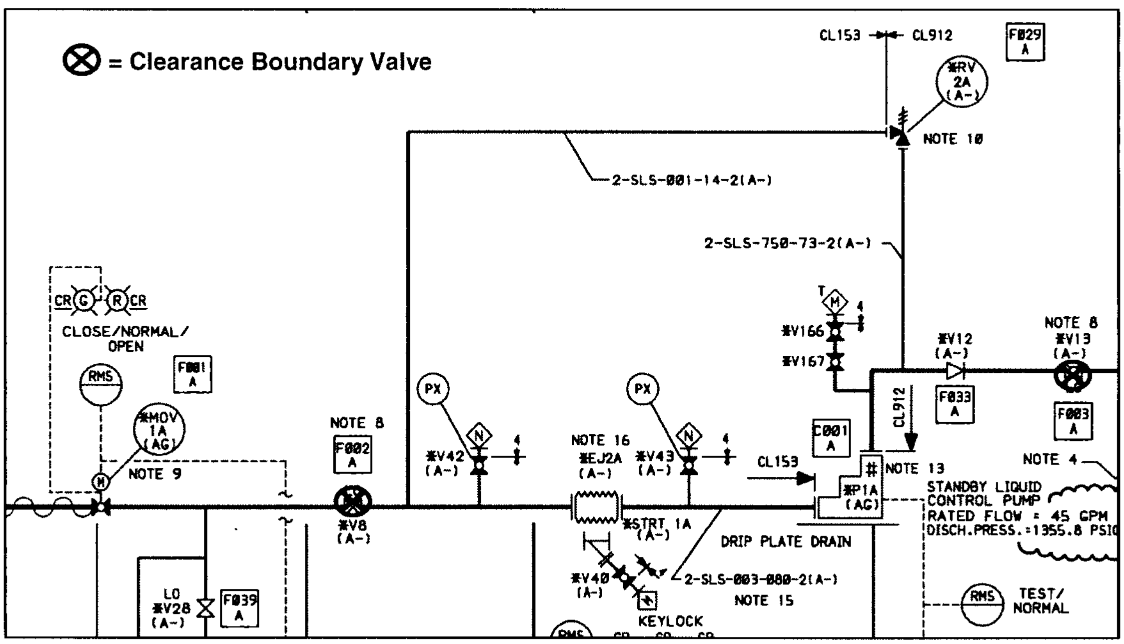
SLC

Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadequate system flow

Proposed Question: #87

The plant is operating at rated power with the following:

- 2SLS*P1A is tagged out of service using the following clearance boundary valves (see drawing below) due to a cracked pump housing:
  - 2SLS*V8, "2SLS*P1A Suction Isolation"
  - 2SLS*V13, "2SLS*P1A Discharge Valve"
- A subsequent ATWS occurs that requires standby liquid control injection
- When the operator initiates 2SLS*P1B, relief valve 2SLS*RV2B immediately lifts due to discharge check valve 2SLS*V14 being stuck closed.



Which one of the following describes the required procedural direction and the amount of boron required to be injected per N2-EOP-C5?

	IAW N2-EOP-6.15, "SLS HYDRO PUMP INJECTION" inject SLS using...	Inject Boron Until...
A.	2SLS*P1A piping only	Cold Shutdown Boron amount has been injected (SLS tank level below 1450 gallons)

B.	2SLS*P1A piping only	the entire contents of the SLS tank has been injected
C.	2SLS*P1B piping only	Cold Shutdown Boron amount has been injected (SLS tank level below 1450 gallons)
D.	2SLS*P1B piping only	the entire contents of the SLS tank has been injected

Proposed Answer: D

Explanation: The failure of 2SLS*P1B discharge check valve does not prevent the use of the SLS 'B' piping for hydro pump injection since the hydro pump is connected to the suction pressure test connection which is upstream of the SLS pump. For the second part of the answer, per NER-2M-039, Rev 08.00 page 3-336, section for EOP-C5 step (Q-10 and Q-11), "Once initiated, boron injection is continued until the entire contents of the SLS tank have been injected or alternate boron injection is completed in accordance with the appropriate EOP-6 attachment. RPV depressurization may be initiated in accordance with the Pressure branch, once Cold Shutdown Boron amount has been injected (SLS tank level below 1450 gallons). Continuing injection until the entire contents of the SLS tank has been injected provides the design basis concentration margin."

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed.

K/A Justification: In this question the candidate would have to predict the impact (status) of the plant in order to make a determination of which procedure to direct. The K/A says to predict the impact and then determine correct procedure guidance to mitigate the event. Predicting the impact and determining a procedure section is inherently contained in the first part of the answer choice.

A. Plausible – Both parts of the distracter are incorrect, but plausible. For the first part of the distracter, the clearance boundary valve for the applied clearance encompasses the suction pressure test connection that is used when aligning SLS hydro pump to SLS 'A'. Therefore, this path would not be permissible. The candidate might think that the hydro pump hooks up on a valve outside of the clearance boundary. For the second part of the distracter, per NER-2M-039, Rev 08.00 page 3-336, section for EOP-C5 step (Q-10 and Q-11), "Once initiated, boron injection is continued until the entire contents of the SLS tank have been injected or alternate boron injection is completed in accordance with the appropriate EOP-6 attachment. RPV depressurization may be initiated in accordance with the Pressure branch, once Cold Shutdown Boron amount has been injected (SLS tank level below 1450 gallons). Continuing injection until the entire contents of the SLS tank has been injected provides the design basis concentration margin." The candidate may confuse the two requirements and think that only injecting the Cold Shutdown Boron amount is correct.
B. Plausible – The first part of the distracter is incorrect, but plausible the second part of the distracter is correct. For the first part of the distracter, the clearance boundary valve for the applied clearance encompasses the suction pressure test connection that is used when aligning SLS hydro pump to SLS 'A'. Therefore, this path would not be permissible. The candidate might think that the hydro pump hooks up on a valve outside of the clearance boundary.
C. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter, per NER-2M-039, Rev 08.00 page 3-336, section for EOP-C5 step (Q-10 and Q-11), "Once initiated, boron injection is continued until the entire contents of the SLS tank have been injected or alternate boron injection is completed in accordance with the appropriate EOP-6 attachment. RPV depressurization may be initiated in accordance with the Pressure branch, once Cold Shutdown Boron amount has been injected (SLS tank level below 1450 gallons). Continuing injection until the entire contents of the SLS tank has been injected provides the design basis concentration margin." The candidate may confuse the two requirements and think that only injecting the Cold Shutdown Boron amount is correct.

Technical Reference(s): N2-EOP-6.15, section 6.2 and figure 1, EOP bases C5 step Q10, EOP-C5 flowchart

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-211000C01, RBO-12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	217000 2.1.23
	Importance Rating	4.4

RCIC

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: #88

The plant is operating at rated power with the following:

- The plant is a month away from its next scheduled refueling outage
- A spurious reactor scram occurs due to an MSIV isolation
- All control rods insert
- N2-EOP-RPV control has been entered
- Reactor pressure is being controlled using the SRVs

With regards to RCIC and reactor water level control, which one of the following describes (1) the correct Unit Supervisor (US) procedure direction and (2) the associated level indicator that should be directed for reactor water level control & monitoring after the level 8 interlocks have been defeated, per OP-NM-101-111-1001, "TRANSIENT MITIGATION GUIDELINES (TMG)"?

	Direct crew to perform N2-EOP-6.20, "DEFEATING RPV WATER LEVEL INTERLOCKS" and defeat the RCIC Level 8 Interlocks by…	Directed Level Indicator
A.	removing trip units B22 N693A & B, "RPV HI WTR LVL 8" (located on 2CEC*PNL629)	Narrow Range
B.	placing keylock switch E31A-S4A, RHR/RCIC ISOLATION BYPASS, to BYPASS (located on 2CEC*PNL632)	Fuel Zone
C.	placing keylock switch E31A-S4A, RHR/RCIC ISOLATION BYPASS, to BYPASS (located on 2CEC*PNL632)	Narrow Range
D.	removing trip units B22 N693A & B, "RPV HI WTR LVL 8" (located on 2CEC*PNL629)	Fuel Zone

Proposed Answer: A

Explanation: Per OP-NM-101-111-1001, "TRANSIENT MITIGATION GUIDELINES (TMG)", Attachment 2, Unit 2 Specific Transient Mitigation Guidelines (TMG), section 2.2.3:

2.2.3 - EOP-RPV, step L-3 allows for bypassing of the high RPV water level interlock for RCIC and CSH when the system is being used to control RPV level, particularly when the RPV is being depressurized. It is critical that the operator closely monitors RPV water level whenever above the high level trip point with the trip bypassed, and take immediate action to stop injection when the level trend indicates that water level may rise to the steam lines.

- When below 600 psig, water level calibration errors may result in CSH and RCIC injection termination on high level when the actual level is below the high end of the specified control band.
- Overriding the high RPV water level interlock to maintain RCIC and / or CSH injection systems in service is appropriate when SRV's are being utilized for pressure control or RPV depressurization. Operating experience has shown that RPV level swells when opening an SRV can cause these systems to trip due to level calibration errors associated with the wide level instrumentation. Therefore when high level (L8) trips for the RCIC and/or HPCS systems are defeated, it is prudent then to control level using the narrow range level instrumentation since this instrument represents actual level for these plant conditions.

In this condition (MSIV's closed with high decay heat load) the SRV's will be cycled frequently to avoid RPV level 3 scrams during the initial phases of the event. The use of RCIC is strategic in that RCIC will use some of the steam produced by decay heat. This will remove energy from the steam and reduce the impacts of adding heat to the primary containment. RCIC will also provide for reactor water level control. Per N2-EOP-6.20, the method used to defeat the RCIC high water level interlocks is by removing the tamper bar and then by removing trip units B22 N693A & B, "RPV HI WTR LVL 8" (located on 2CEC*PNL629).

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed.

B. Plausible – The first and second part of the distracter are both incorrect, but plausible. For the first part of the distracter it is required per N2-EOP-6.20 to remove the trip unit tamper bar and then remove trip units B22 N693A & B, "RPV HI WTR LVL 8", however there are bypass switches associated with bypassing the RCIC high temperature isolations that are located 2CEC*PNL632. The candidate could mistake these switches for the method that is required to be used by N2-EOP-6.20. For the second part of the distracter the candidate could think that since the reactor has been shutdown, the calibration conditions for the Fuel Zone level indicator would be the most accurate.
C. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part of the distracter it is required per N2-EOP-6.20 to remove the trip unit tamper bar and then remove trip units B22 N693A & B, "RPV HI WTR LVL 8", however there are bypass switches associated with bypassing the RCIC high temperature isolations that are located 2CEC*PNL632. The candidate could mistake these switches for the method that is required to be used by N2-EOP-6.20. The second part of the distracter is correct.
D. Plausible – The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. For the second part of the distracter the candidate could think that since the reactor has been shutdown, the calibration conditions for the Fuel Zone level indicator would be the most accurate.

Technical Reference(s):	OP-NM-101-111-1001, "TRANSIENT MITIGATION GUIDELINES (TMG)", Attachment 2, Unit 2 Specific Transient Mitigation Guidelines (TMG), section 2.2.3 and N2-EOP-6.20, "DEFEATING RPV WATER LEVEL INTERLOCKS"

Proposed references to be provided to applicants during examination: None

Learning Objective:	2101-EOPRPVC01, Objective #N2-EOP-RPV-CE-02
Question Source:	New
Question History:	
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	55.43(b)5
Comments:	

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	262002 2.2.40
	Importance Rating	4.7

UPS (AC/DC)

Equipment Control: Ability to apply Technical Specifications for a system.

Proposed Question: #89

The plant is operating at rated power. Conditions are as follows:

- 2VBA*UPS2B, Division II UPS is in operation
- The DC supply breaker to 2VBA*UPS2B is inadvertently tripped open and cannot be shut.

Which one of the following is allowed per Technical Specifications?

A. 2VBA*UPS2D must be placed in operation within 8 hours or the reactor must be in Mode 3 within 20 hours.

B. 2VBA*UPS2D must be placed in operation within 24 hours or the reactor must be in Mode 3 within 36 hours.

C. The reactor must be in Mode 3 within 20 hours. 2VBA*UPS2D cannot be placed in operation to satisfy the requirements of Technical Specifications.

D. The reactor must be in Mode 3 within 36 hours. 2VBA*UPS2D cannot be placed in operation to satisfy the requirements of Technical Specifications.

Proposed Answer: B

Explanation: Without a DC power supply, UPS2B is inoperable. Per TS Bases for 3.8.7, either UPS2B or 2D need to be operable and in operation in order to meet LCO 3.8.7. Per TS 3.8.7 you have 24 hours to restore an operable UPS to service. If you don't restore an UPS to service, you must be in mode 3 within 12 hours. So 24 + 12 hours is 36 hours.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(2) based on the question involving the application of TS Required Actions (Section 3) in accordance with rules of application requirements (Section 1). Per NUREG1021, section ES-401, attachment 2 Figure 1.

A. Plausible – Although UPS2D can be placed in operation to satisfy technical specifications, you have 24 hours per TS 3.8.7 to place in it in operation, not 8 hours. Plausible in that if the candidate thinks that having UPS2B inoperable makes the Div I 120 VAC uninterruptable electrical power subsystem inoperable, then you would have 9 hours per TS 3.8.7, condition B.
C. Plausible – TS bases allows UPS2D to be placed in service to satisfy Tech Specs.
D. Plausible – TS bases allows UPS2D to be placed in service to satisfy Tech Specs.

Technical Reference(s): T.S. 3.8.7 and bases

Proposed references to be provided to applicants during examination: T.S. 3.8.7 and 3.8.8 no bases.

Learning Objective: 2101-262002C01, RBO-14

Question Source: Bank

Question History: 2011 Audit 88

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)2

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	300000 2.1.20
	Importance Rating	4.6

Instrument Air

Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: #90

The Plant is operating at rated power with the following:

- An Instrument air leak occurs
- N2-SOP-19 has been entered by the control room crew
- Current air header pressure is indicating 105 psig on 2IAS-PI101 (P851)
- Air header pressure is still lowering very slowly and is being evaluated by the control room crew

Which one of the following describes when a manual reactor scram is required be directed per N2-SOP-19 and OP-NM-101-111-1001, "Transient Mitigation Guidelines (TMG)" AND who the Shift Manger is required to notify per OP-AA-106-101, "Significant Event Reporting"?

	When a manual reactor scram is to be directed	Required Notification
A.	2IAS-PI194 (RB el. 261'), Inst air rcvr 2IAS-TK3 pressure, lowers to < 74 psig,	Duty Station Manager
B.	2IAS-PI101 (P851), Instrument air pressure, lowers to < 85 psig	Nuclear Duty Officer
C.	2IAS-PI194 (RB el. 261'), Inst air rcvr 2IAS-TK3 pressure, lowers to < 74 psig,	Nuclear Duty Officer
D.	2IAS-PI101 (P851), Instrument air pressure, lowers to < 85 psig	Duty Station Manager

Proposed Answer: A

Explanation: Per N2-SOP-19, "Loss of Instrument Air" a reactor scram is required when:

- 2IAS-PI194 (RB el. 261'), Inst air rcvr 2IAS-TK3 pressure, lowers to < 74 psig
- 2RDS-PI133 (RB el. 261'), scram air header pressure, lowers to < 60 psig
- 2IAS-PI101 (P851), Instrument air pressure, lowers to < 70 psig

OP-AA-106-1001, "Significant Event Reporting" step 4.2.2 states that the Shift Manager will notify the Duty Station Manager for any of the events listed in attachment 1. "Any unexpected significant plant transient"

B. Plausible – The first and second part of the distracter are both incorrect, but plausible. For the first part, the candidate could not recall the scram setpoint and think that since instrument air pressure is lowering and that IAS-SOV171 auto closes at 85 psig and that Annunciator 851208, INST AIR RCVR TK 2 PRESS LOW alarms at 85 psig that a scram would be required. For the second part of the distracter the candidate could think that it would be required by the Shift Manger to notify each of the duty officers both Duty and Nuclear. The Shift Manger is required to make notifications, but should also ensure oversight of crew activities.

C. Plausible – The first part of the distracter is correct and the second part of the distracter is incorrect, but plausible. For the second part of the distracter the candidate could think that it would be required by the Shift Manger to notify each of the duty officers both Duty and Nuclear. The Shift Manger is required to make notifications, but should also ensure oversight of crew activities.

D. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part, the candidate could not recall the scram setpoint and think that since instrument air pressure is lowering and that IAS-SOV171 auto closes at 85 psig and that Annunciator 851208, INST AIR RCVR TK 2 PRESS LOW alarms at 85 psig that a scram would be required.

Technical Reference(s): N2-SOP-19, OP-AA-106-101, "Significant Event Reporting" step 4.2.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101- SOP19C01, Objective N2-SOP-19-CE-02

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	241000 A2.23
	Importance Rating	2.6

Reactor/Turbine Pressure Regulator

Ability to (a) predict the impacts of the following on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Turbine high eccentricity

Proposed Question: #91

The plant is operating at 19% power with the following:

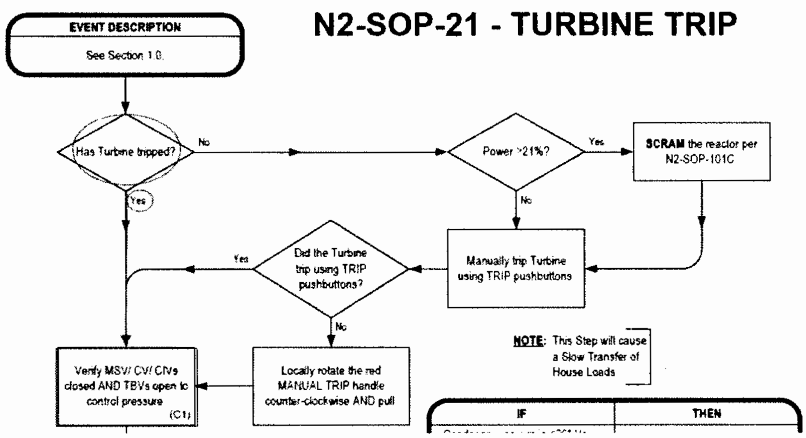
- A failure of the main turbine Electrohydraulic Control (EHC) speed control loop results in main turbine speed increasing
- As main turbine speed increases the mechanical overspeed trip device (unbalanced ring) reaches a high eccentricity.

Which one of the following describes the expected plant response and the required procedure direction to mitigate the consequences?

	Plant Response	Required Procedure Direction
A.	Overspeed trip finger repositions to operate the mechanical trip valve	Direct entry into N2-SOP-101C, "Reactor Scram" and direct reactor recirculation flow reduced to 55 mlbm/hr
B.	Overspeed trip finger repositions to operate the mechanical trip solenoid	Direct entry into N2-SOP-21, "Turbine Trip" and verify the main turbine is tripped and that Turbine Bypass Valves are controlling pressure
C.	Overspeed trip finger repositions to operate the mechanical trip valve	Direct entry into N2-SOP-21, "Turbine Trip" and verify the main turbine is tripped and that Turbine Bypass Valves are controlling pressure
D.	Overspeed trip finger repositions to operate the mechanical trip solenoid	Direct entry into N2-SOP-101C, "Reactor Scram" and direct reactor recirculation flow reduced to 55 mlbm/hr

Proposed Answer: C

Explanation: Per USAR section 10.2.2.2 (page 10.2-7), "The mechanical overspeed trip device is an unbalanced ring that is held concentric with the shaft by an adjustable spring. When the speed reaches the trip speed, the centrifugal force on the ring overcomes the force on the spring and the ring snaps to an eccentric position. In so doing, it strikes the trip finger to operate the MTV(Mechanical Trip Valve), which closes all turbine steam admission valves. This trips the main turbine because all turbine steam admission valves close on low ETS oil pressure. The mechanical overspeed trip device is located in the front standard and acts to prevent damage in case the turbine overspeeds due to failure of the EHC speed control loop. In this case, reactor power was given to be 19%. In accordance with N2-SOP-21, a reactor scram is not required, but verifying that the MSV / CV / CIVs are closed AND TBVs are open to control pressure is required.



Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to assess plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. This question is testing the concepts of whether or not a reactor scram is required at 19% reactor power and whether or not the mechanical overspeed trip device will cause a turbine trip. The candidates could confuse the concept that high turbine vibration does not cause a trip with the mechanical overspeed trip device.

A. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter the candidate might not recognize that reactor power is at 19% or forget that 19% reactor power does not require a reactor scram and think that directing a reactor scram is the required action.

B. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. The candidate may believe that the mechanical trip solenoid is actuated by the mechanical repositioning of the trip finger. In fact it is actuated electrically by the EHC alarm and Trip circuitry. Also, the mechanical trip solenoid does not actually block and vent hydraulic supply to the ETS fluid, it repositions the mechanical tip mechanism which results in the Mechanical trip valve blocking and venting hydraulic supply to the ETS fluid.

D. Plausible – Both the first and second part of the distracter are incorrect, but plausible. In the first part of the distracter, the candidate may believe that the mechanical trip solenoid is actuated by the mechanical repositioning of the trip finger. In fact it is actuated electrically by the EHC alarm and Trip circuitry. Also, the mechanical trip solenoid does not actually block and vent hydraulic supply to the ETS fluid, it repositions the mechanical tip mechanism which results in the Mechanical trip valve blocking and venting hydraulic supply to the ETS fluid. For the second part of the distracter, the candidate might not recognize that reactor power is at 19%

or forget that 19% reactor power does not require a reactor scram and think that directing a reactor scram is the required action.

Technical Reference(s): USAR section 10.2.2.2 (page 10.2-7), 2101-248000C01 RBO-2 content, N2-SOP-21 flowchart

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-248000C01, RBO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	259001 2.4.30
	Importance Rating	4.1

Reactor Feedwater

Emergency Procedures / Plan; 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.

Proposed Question: #92

The plant is operating at rated power with the following:

- Both operating feedwater pumps trip and cannot be restarted
- RPS trips resulting in full automatic reactor scram
- HPCS automatically starts
- CSH*MOV107 (CSH Injection Valve) is closed and cannot be opened
- RCIC is injecting 600 gpm following automatic actuation
- RPV water level lowered to 90 inches and is now slowly recovering
- All other plant systems responded as designed

Which one of the following identifies the **most restrictive offsite notification requirement** to the NRC Operations Center due to this transient?

A. Within 15 minutes because an Emergency Action Level (EAL) is met

B. Within 1 hour because an Emergency Action Level (EAL) is met

C. Within 4 hours because of the scram and ECCS actuations

D. Within 8 hours because of the RCIC injection and isolation actuations

Proposed Answer: C

Explanation: Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. 10CFR50.72(b)(2)(iv)(B) NRC Operations Center ENS within 4 hours. Written report required by 10CFR50.73.

Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. 10CFR50.72(b)(2)(iv)(A) NRC Operations Center ENS within 4 hours. Written report required by 10CFR50.73.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on testing the ability to assess a plant condition and select the correct procedure section (LS-AA-1020 for 10CFR50.72 Notification Requirements) to determine correct NRC notification requirements.

A. Plausible – The presented transient does not result in any EALs being met, therefore notifications per (a)(1)(i) are not required. If an EAL had been met and an Emergency declared, the local state and county notifications are required within 15 minutes.
B. Plausible – The presented transient does not result in any EALs being met, therefore notifications per (a)(1)(i) within 1 hour are not required.
D. Plausible – There are 8 hour notification requirements due to RCIC actuation and isolation logic/system actuations (b)(3)(iv)(A), but these are not the most restrictive, based on the transient conditions provided.

Technical Reference(s): LS-AA-1020 Rev. 22 Reportability Reference Manual Volume 1 - Table SAF page 5 of 125

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-ADMTPSC01, EO-1.4 & S101-ADMTPSC57

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	219000 2.4.47
	Importance Rating	4.2

RHR/LPCI: Torus/Pool Cooling Mode

Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: #93

N2-EOP-C5 execution is in progress following a failure to scram. The following conditions exist:

- Reactor pressure band is 800-1000 psig using SRV's
- Suppression pool water temperature is currently 125°F and rising 1 degree/minute
- A small drywell steam leak has occurred and has been isolated
- Drywell pressure reached 1.68 psig 12 minutes ago and is now 2.3 psig and stable
- RHR 'A' and 'B' have been placed in suppression pool cooling
- Service water pumps A, B, C, D are in service
- Suppression pool water level is now reported to be at 200.5 feet and steady

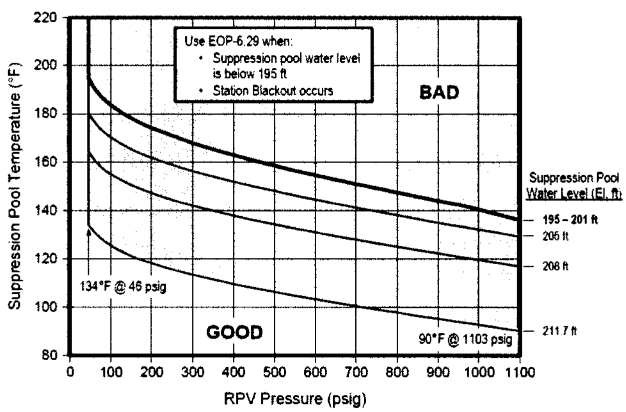
Based on the above conditions:

(1) When will the HCTL FIRST be exceeded if conditions remain as described above?

AND

(2) What action is required in an attempt maximize RHR suppression pool cooling?

M **Heat Capacity Temperature Limit**



	HCTL Exceeded in:	Action required:
A.	Approx. 15 minutes	Direct the start of a fifth service water pump
B.	Approx. 15 minutes	Direct turbine building non-essential service water headers isolated
C.	Approx. 23 minutes	Direct the start of a fifth service water pump
D.	Approx. 23 minutes	Direct turbine building non-essential service water headers isolated

Proposed Answer: A

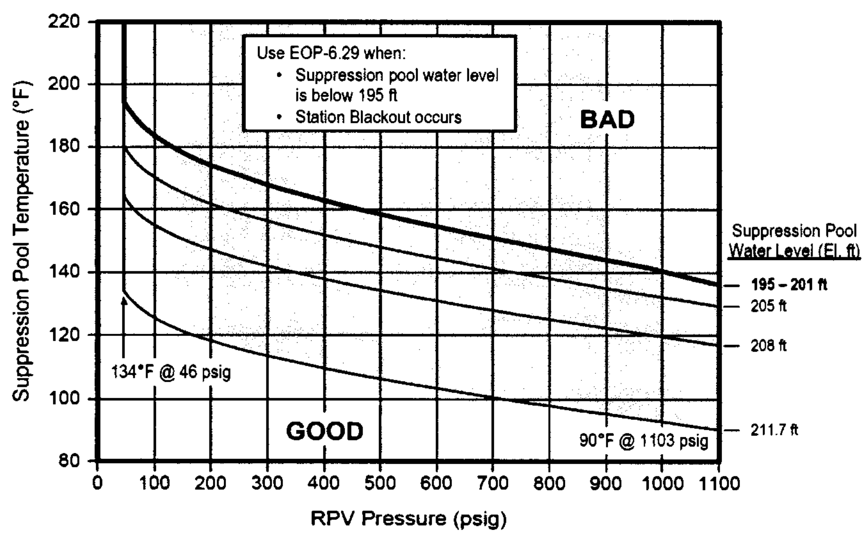
Explanation: With a suppression pool level of 200.5 feet, the 195 - 201 foot curve is the limit used. Since the pressure band is 800-1000 psig, the limiting pressure would be 1000 psig so using figure 'M', the 195 – 201 foot limit and a pressure of 1000 psig, the suppression pool temperature limit would be 140 degrees. Therefore, the suppression pool temperature limit will be reached in 15 minutes. As for the second part of the answer, with rising suppression pool temperature it would be required to maximize RHR suppression pool cooling. With only 4 service water pumps in service, service water flow through the RHR heat exchangers would be limited to 2000 gpm. In order to turn the trend of suppression pool temperature, rated service water flow (7400 gpm) through the RHS heat exchangers would be required. Per the note in N2-OP-31 section H.11.0 "Five SWP pumps are required to be in operation to provide rated flow to one or both RHS heat exchangers when non- essential headers are also being supplied by the Service Water system." Therefore, start of a fifth service water pump should be directed.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to both 1) assess plant conditions (normal, abnormal, or emergency) and then 2) select a procedure or section of a procedure to mitigate, recover, or with which to proceed. Per ES-401, Attachment 2, section II.E.

B. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the action to isolate turbine building loads by closing the turbine building non-essential header isolation valves to allow higher SWP flow through the RHS heat exchangers is a viable option, but only when more than 4 SWPs cannot be started. OP-11 P&L #26 states "Following a LOCA, if four or less SWP pumps are available, to supply greater than 2000 GPM SWP flow to one or both RHR Heat Exchangers, it may be necessary to isolate Turbine Building loads by closing 2SWP*MOV3A OR *MOV3B OR *MOV599 AND 2SWP-V8. This is only required if more than four SWP pumps can not be placed in service. The candidate could confuse this requirement since it is a viable option under certain circumstances.

C. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part of the distracter, with a pressure band of 800-1000 psig the candidate could mistakenly think that the limit would be based off the 800 psig RPV pressure and calculate the suppression pool temp limit to be 148°F.

M **Heat Capacity Temperature Limit**



From 125°F to 148°F at a rise in temp of one degree per minute it would take approximately 23 minutes.

D. Plausible – Both the first and second part of the distracter are incorrect, but plausible. For the first part of the distracter, with a pressure band of 800-1000 psig the candidate could mistakenly think that the limit would be based off the 800 psig RPV pressure and calculate the suppression pool temp limit to be 148°F. From 125°F to 148°F at a rise in temp of one degree per minute it would take approximately 23 minutes. For the second part of the distracter, the action to isolate turbine building loads by closing the turbine building non-essential header isolation valves to allow higher SWP flow through the RHS heat exchangers is a viable option, but only when more than 4 SWPs cannot be started. OP-11 P&L #26 states "Following a LOCA, if four or less SWP pumps are available, to supply greater than 2000 GPM SWP flow to one or both RHR Heat Exchangers, it may be necessary to isolate Turbine Building loads by closing 2SWP*MOV3A OR *MOV3B OR *MOV599 AND 2SWP-V8. This is only required if more than four SWP pumps can not be placed in service. The candidate could confuse this requirement since it is a viable option under certain circumstances.

Technical Reference(s): N2-OP-31, section H.11.0, N2-OP-11 P&L # 26.0

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01, Objective # N2-EOP-PC-CE-C02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.43
	Importance Rating	4.3

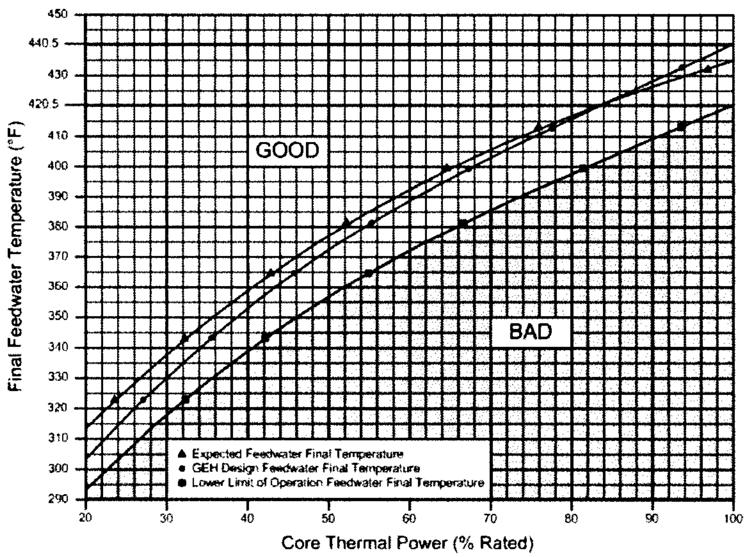
Ability to use procedures to determine the effects on reactivity of plant changes, such as RCS temperature, secondary plant, fuel depletion, etc.

Proposed Question: #94

The plant is operating at rated power with the following:

- A loss of FWS-E6A (6th Point Feedwater Heater) occurs
- An immediate rise in reactor power is indicated due to the transient
- The crew has entered N2-SOP-08, "Unplanned Power Changes"
- Core thermal power has been reduced to 3580 MWth
- Final feedwater temperature is 408°F

FIGURE 1: FEEDWATER TEMPERATURE/THERMAL POWER LIMIT



Which one of the following describes the effect of the loss of feedwater heating on the plant prior to any operator action AND the required direction?

	Effect	Required Direction
A.	Rise in reactor rod line	Direct a plant shutdown per N2-OP-101C
B.	Initial xenon concentration rise	Direct a plant shutdown per N2-OP-101C
C.	Initial xenon concentration rise	Direct reactor power reduced using N2-SOP-101D
D.	Rise in reactor rod line	Direct reactor power reduced using N2-SOP-101D

Proposed Answer: D

Explanation: Per N2-SOP-08, "Loss of Feedwater Heating" Note on page 8 of 31, "Loss of feedwater heating will cause rod line to rise." For the required direction portion of the question, N2-SOP-08 page 8 of 31 states to "Adjust Reactor power per N2-SOP-101D to maintain the following:

- FW temperature (NSSTA101, 102, 103, 104, OR FWS-TI64A, TI64B) in the GOOD area of Figure 1"

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to both 1) assess plant conditions (normal, abnormal, or emergency) and then 2) select a procedure or section of a procedure to mitigate, recover, or with which to proceed. Per ES-401, Attachment 2, section II.E.

A. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the candidate could think that you are not allowed to restore and maintain FW temperature within the limits of N2-SOP-08 and therefore think that since temperature is in the "bad" region that a plant shutdown is required.

B. Plausible – Both the first and second part of the distracter are incorrect, but plausible. For the first part of the distracter, a loss of feedwater heating will cause reactor power to rise, this will in turn cause the burnout term of the xenon equation to outweigh the production term initially and cause xenon concentration to lower initially. The candidate could incorrectly determine that due to the rise in power xenon production rate would increase. For the second part of the distracter, the candidate could think that you are not allowed to restore and maintain FW temperature within the limits of N2-SOP-08 and therefore think that since temperature is in the "bad" region that a plant shutdown is required.

C. Plausible – The first part of the distracter is incorrect, but plausible and the second part of the distracter is correct. For the first part of the distracter, a loss of feedwater heating will cause reactor power to rise, this will in turn cause the burnout term of the xenon equation to outweigh the production term initially and cause xenon concentration to lower initially. The candidate could incorrectly determine that due to the rise in power xenon production rate would increase.

Technical Reference(s): N2-SOP-08, "Loss of Feedwater Heating"

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-260000C01, RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.38
	Importance Rating	4.5

Knowledge of conditions and limitations in the facility license.

Proposed Question: #95

A plant startup is in progress with the following:

- The Mode Switch is in Startup/Hot Standby
- Suppression Pool average temperature is 93°F and lowering slowly due to the performance of N2-OSP-ICS-Q@002, RCIC Pump and Valve Operability Test and System Integrity Test and ASME XI Functional Test
- RCIC testing was completed 3 hours ago
- RCIC is operable and in a standby lineup

Which of the following describes a limitation in the facility license for these conditions?

The Mode Switch...

A. CANNOT be placed in RUN due to suppression pool average temperature unless evaluated by plant oversight review committee (PORC)

B. CANNOT be placed in RUN due to Suppression Pool average temperature unless a risk assessment is performed and risk management actions are established

C. CAN be placed in RUN as long as Suppression Pool average temperature remains less than or equal to 105°F

D. CAN be placed in RUN, but Suppression Pool average temperature must be less than or equal to 90°F within 24 hours

Proposed Answer: B

Explanation: Normally, the mode switch would be allowed to be placed in RUN without performing a risk assessment. In the stem, it states that SP temperature exceeds the limit per LCO 3.6.2.1 (90°F). LCO 3.0.4b permits a Mode change if an LCO is not met after a risk assessment is performed and risk management actions are established.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(1) requires the application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1) per ES-401, Attachment 2, Figure 1.

A. Plausible – During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events.
C. Plausible – The RCIC surveillance is no long in-progress, so the 105°F limit is no longer applicable. LCO 3.6.2.1 is not satisfied at 93°F. Plausible in that the candidate might not recognize this condition.
D. Plausible – The Mode Switch can be placed in RUN if LCO 3.0.4b is met, but temperature must be restored in 21 hours, not 24. The candidate may not recall this requirement.

Technical Reference(s): TS 3.6.2.1, TS Bases 3.6.2.1, LCO 3.0.4

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-223001C01, RBO-14

Question Source: Bank

Question History: 2011 Columbia 88

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)1

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.5
	Importance Rating	2.9

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #96

The plant is at 100% power. Conditions are as follows:

- A review of past surveillance tests reveals the following radiation monitor trip set-points:
  - 2HVR*CAB14A, Above Refuel Floor Radiation Monitor A: 2.30 x 10^{-3} µCi/cc
  - 2HVR*CAB14B, Above Refuel Floor Radiation Monitor B: 2.68 x 10^{-3} µCi/cc
  - 2HVR*CAB32A, Below Refuel Floor Radiation Monitor A: 2.47 x 10^{-3} µCi/cc
  - 2HVR*CAB32B, Below Refuel Floor Radiation Monitor B: 2.28 x 10^{-3} µCi/cc

Which one of the following describes the Technical Specification implications of these setpoints?

A. An adequate number of channels are operable, therefore no Technical Specifications actions are required.

B. The inoperable channel(s) must be placed in trip within 12 hours. All required functions currently maintain isolation capability.

C. The inoperable channel(s) must be placed in trip within 24 hours. All required functions currently maintain isolation capability.

D. Isolation capability for all required functions is NOT maintained and must be restored within 1 hour.

Proposed Answer: C

Explanation: All of these radiation monitors are required to have a trip setpoint of ≤ 2.46×10^{-3} μCi/cc. 2HVR*CAB14B and 2HVR*CAB32A are both inoperable because their trip setpoints exceed this value. However, with 2HVR*CAB14A and 2HVR*CAB32B still operable, Technical Specification Table 3.3.6.2-1 Functions 3 and 4 still maintain isolation capability. Therefore Technical Specification 3.3.6.2 Required Action A.1 requires placing the two inoperable channels in trip within 24 hours.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(2) requires the application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1) per ES-401, Attachment 2, Figure 1.

A. Plausible –An adequate number of channels are still operable to initiate the required function, but NOT enough channels are operable to meet the Technical Specification requirements.
B. Plausible –12 hours is the correct time in this Technical Specification for Function 2 (high Drywell pressure), but 24 hours is the correct time for Functions 3 and 4.
D. Plausible –Two radiation monitors are inoperable, but they are in separate Functions, therefore each individual Function still maintains isolation capability.

Technical Reference(s): T.S. 3.3.6.2 Condition A, T.S. Table 3.3.2.1-1

Proposed references to be provided to applicants during examination: Technical Specification 3.3.6.2 including Table 3.3.6.2-1 No Bases.

Learning Objective: N2-261000-RBO-14

Question Source: Bank

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.47
	Importance Rating	4.2

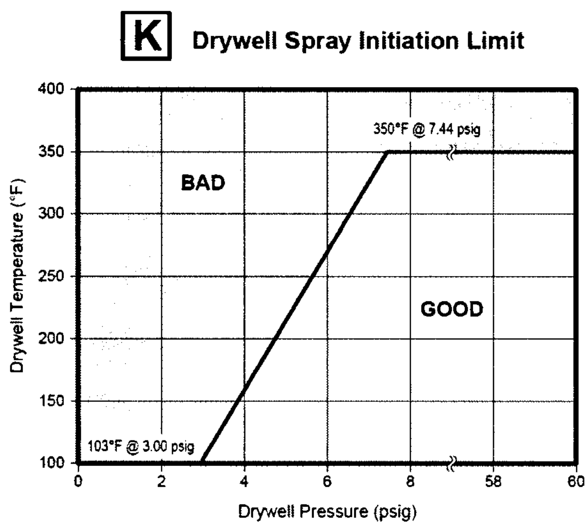
Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: #97

The plant is operating at rated power with the following:

- A steam leak into the drywell has occurred.
- Reactor water level is 80 inches and stable.
- Reactor pressure is 700 psig and slowly lowering due to the steam leak.
- Suppression Pool water level is 199.5 feet.
- Drywell pressure is currently 4 psig and is rising at 0.1 psig per minute.
- Drywell temperature is 110°F and is rising at 8°F per minute.

Assume all parameter trends remain constant.



Which one of the following describes:

(1) When or if the Drywell Spray Initiation Limit will be exceeded, and

(2) the action required to be directed if drywell temperature approaches and is projected to exceed 340°F?

	(1)	(2)
A.	Will not be exceeded	Operate drywell sprays per N2-EOP-6.22
B.	Approximately 20 minutes	Keep trying to lower drywell temperature below 150°F per N2-EOP-PC
C.	Will not be exceeded	Start all available drywell cooling per N2-EOP-6.24
D.	Approximately 20 minutes	Perform an RPV Blowdown per N2-EOP-C2

Proposed Answer: D

Explanation: Per Figure “K” on N2-EOP-PC, at the 20 minute point drywell pressure will be 6.0 psig and drywell temperature will be 270.0°F. This plots above or on the bad side of Figure “K”. The line formula for the limit is y = 55.63x - 63.89. As per the chart below the limit at 6.0 psig drywell pressure is 269.89.

	(x)	(Y)	
Minute #	DW Pressure	DW Temperature	Limit
0	4.00	110.00	158.63
1	4.10	118.00	164.193
2	4.20	126.00	169.756
3	4.30	134.00	175.319
4	4.40	142.00	180.882
5	4.50	150.00	186.445
6	4.60	158.00	192.008
7	4.70	166.00	197.571
8	4.80	174.00	203.134
9	4.90	182.00	208.697
10	5.00	190.00	214.26
11	5.10	198.00	219.823
12	5.20	206.00	225.386
13	5.30	214.00	230.949
14	5.40	222.00	236.512
15	5.50	230.00	242.075
16	5.60	238.00	247.638
17	5.70	246.00	253.201
18	5.80	254.00	258.764
19	5.90	262.00	264.327
20	6.00	[redacted]	269.89
21	6.10	[redacted]	275.453

Drywell temperature exceeds limit here

N2-EOP-PC requires an RPV Blowdown if you cannot restore and maintain drywell temperature below 340°F. Currently, there are no means available to reduce drywell temperature below 150°F. Per the EOP bases “If drywell temperature cannot be restored and maintained below 340°F (drywell design temperature and ADS qualification temperature), a blowdown is performed. This action minimizes any continuing direct energy release to the drywell through a primary system break and ensures that the SRVs are opened while still operable. Consistent with the definition of “restore,” a Blowdown is not required until it has been determined that drywell sprays are ineffective in reducing drywell temperature. It is not expected that either containment integrity or SRV operability will be immediately challenged when the temperature limit of 340°F is reached. If drywell temperature is already above 340°F when Element DWT-8 is reached, drywell sprays are still to be used, if available, in preference to a Blowdown. If sprays are effective in reducing drywell temperature, a Blowdown need not be performed. Extended operation above the specified temperature is not permitted, however. A determination that drywell temperature cannot be restored and maintained below 340°F may be made when, before, or after temperature actually reaches the value.”

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to both 1) assess plant conditions (normal, abnormal, or emergency) and then 2) select a procedure or section of a procedure to mitigate, recover, or with which to proceed. Per ES-401, Attachment 2, section II.E.

A. Plausible –Both the first and second part of the distracter are incorrect, but plausible. The candidate may incorrectly calculate the trending values and determine that the Drywell Initiation Spray Limit will not be exceeded and if this is the case, N2-EOP-PC would direct initiation of drywell sprays to reduce drywell temperature.
B. Plausible – The first part of the distracter is correct, the second part of the distracter is incorrect, but plausible. For the second part of the distracter, the candidate may think that since N2-EOP-PC directs this method as an option that this choice is correct, however no active methods remain available to reduce Drywell temperature and per the EOP-bases, “A determination that drywell temperature cannot be restored and maintained below 340°F may be made when, before, or after temperature actually reaches the value.”
C. Plausible – Both the first and second part of the distracter are incorrect, but plausible. The candidate may incorrectly calculate the trending values and determine that the Drywell Initiation Spray Limit will not be exceeded and if this is the case, starting all available drywell cooling per N2-EOP-6.24 is an option provided in N2-EOP-PC, however per the EOP bases “EOP-6.24 allows restoring cooling water to the drywell unit coolers only if the hottest drywell temperature did not exceed 250°F while cooling water was isolated.”

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01, Objective # N2-EOP-PC-CE-02

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

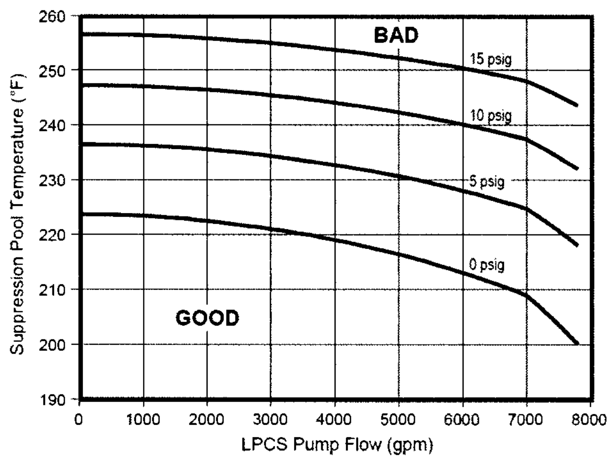
Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.25
	Importance Rating	4.2

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: #98

The plant is shutdown following a LOCA. Conditions are as follows:

- N2-EOP-RPV, RPV Control has been entered
- Current LPCS flow is 6000 gpm
- RPV water level is -14 inches and slowly rising
- The US determines that 4000 gpm flow from LPCS is needed to maintain adequate
- core cooling.
- Suppression Pool Water Temperature is 230°F (stable).
- Suppression Pool Water Level is 192.5 feet (stable).
- Suppression Pool Overpressure is 8 psig (stable).



Assuming no interpolation, which one of the following is the status of the LPCS pump and what actions, if any, are required to correct it?

The LPCS pump is...

A. operating within its NPSH limits. No actions are required.

B. NOT operating within its NPSH limits. Secure injection with LPCS and attempt maintain RPV level with other injection systems per N2-EOP-RPV.

C. NOT operating within its NPSH limits. Make the LPCS injection valve throttleable per N2-EOP-6.3. Maintain LPCS flow within the NPSH limits but above 4000 gpm.

D. NOT operating within its NPSH limits. Dispatch an operator to locally throttle the LPCS injection valve per N2-EOP-6.29. Maintain LPCS flow within the NPSH limits but above 4000 gpm.

Proposed Answer: C

Explanation: The LPCS pump is exceeding the NPSH limits. Using the 5 psig overpressure line with Suppression Pool Water Temperature at 230°F, the maximum amount of LPCS flow that is allowed would be approximately 5,200 gpm. Current LPCS pump flow is at 6,000 gpm. In this condition N2-EOP-6.29 directs the LPCS injection valve be made throttleable and flow lowered if it would not prevent meeting the flow chart objective. Since the US only needs 4000 gpm to meet the flowchart objective, then flow can be lowered provided it does not go below 4000 gpm.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on the need to both 1) assess plant conditions (normal, abnormal, or emergency) and then 2) select a procedure or section of a procedure to mitigate, recover, or with which to proceed.

A. Plausible – This is plausible if the candidate were to interpolate or use the 10 psig curve. With suppression chamber overpressure at 8 psig, the 5 psig curve is used, (not allowed to interpolate). With LPCS injecting at 6000 gpm, the LPCS pump is operating in the bad region of the NPSH curve.
B. Plausible – Although the LPCS is violating the NPSH limits, the US needs at least 4000 gpm from the pump and there is no procedural direction to secure the pump. The candidate may not think that injection valve throttling is allowed in N2-EOP-RPV control and decide that the only option is to trip the LPCS pump.
D. Plausible – There is no procedural direction to locally operate the LPCS injection valve in N2-EOP-6.29, however the candidate may think that this is the only way to throttle ECCS flow.

Technical Reference(s): N2-EOP-6.29, section 6.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOP00C02, Objective # N2-EOP-CL-CE-C01

Question Source: Bank

Question History: 2012 Audit 94

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(b)5

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.23
	Importance Rating	4.6

Ability to track Technical Specification limiting conditions for operations.

Proposed Question: #99

The plant is operating at rated power with the following:

- A Division III work week is in progress
- HPCS is scheduled to undergo planned maintenance

Which one of the following actions will require a control room narrative log entry per OP-AA-111-101, "Operating Narrative Logs and Records"?

A. Posting a "work in progress" area around the High Pressure Core Spray pump

B. Removal of a clearance associated with the High Pressure Core Spray System

C. Authorization of a material storage permit for the placement of scaffolding components

D. Placing protected equipment barrier tape around High Pressure Core Spray System components

Proposed Answer: B

Explanation: Per OP-AA-111-101, section 4.3.5 which states the following:
CAPTURE information in the control room narrative log in sufficient detail to accurately categorize system and component availability for systems and equipment subject to maintenance rule, NEI/NRC performance monitoring, etc.

1. Reports of system degradation which affect system/component operability or availability should be detailed enough to capture a complete understanding of the problem. For example, if oil level is reported low then what is the actual measurement? If a barrier seal is damaged, what is the number, how significant is the damage (1" tear, 1/4" hole, etc)?
2. For these systems and components, diligence is required to record ANY alteration or change in status which *affects operability or availability*, to include but not be limited to:
    A. Application of a clearance
    B. *Removal of a clearance*
    C. Removal of logic fuses
    D. Changes to Controller Positions
    E. Changes to logic configuration

Removing the clearance will affect operability /availability of the HPCS system.

Note: This question meets the SRO-only question guidelines for 10CFR55.43(b)(3) based on needing to have the knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose. Additional knowledge of the procedure's content is required to correctly answer the written test item. Per NUREG 1021, attachment 2, section II.E.

A. Plausible – The candidate may think that placing a "work in progress" area around components could affect operability of a component that would require a control room narrative log entry
C. Plausible – The candidate may think that erecting scaffolding could affect operability of a component and therefore think that it is required to log the activity.
D. Plausible – The candidate may think that placing protected barriers around components would require a control room narrative log entry, instead the requirements of OP-NM-108-117, "Protected Equipment at Nine Mile Point" apply which do not require a control room narrative log entry.

Technical Reference(s): OP-AA-111-101, section 4.3.5

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-ADMTPSC02, Objective # CNG-OP-1.01-CT-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(b)3

Comments: