Facility:	Oyster Creek ILT 09-1 NRC Exam Outline				Date of Exam:				May 17, 2010							
Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Plant Evolutions	1	3	4	3				3	3			4	20	3	4	7
	2	1	1	1				1	1			2	7	1	2	3
	Tier Totals	4	5	4				4	4			6	27	4	6	10
2. Plant Systems	1	2	3	2	3	2	3	2	3	2	3	2	26	2	3	5
	2	1	1	1	1	1	1	1	2	1	1	1	12	0 | 2	1	3
	Tier Totals	3	4	3	4	3	4	3	4	3	4	3	38	4	4	8
3. Generic Knowledge & Abilities Categories				1		2		3		4			10	1 | 2	3 | 4	7
				2		2		3		3				1 | 2	2 | 2	

Note:
1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only 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

ILT 09-1 NRC Exam Outline
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
295028 High Drywell Temperature / 5					X		EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Reactor pressure	3.9	1
295019 Partial or Total Loss of Inst. Air / 8					X		AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Instrument air system pressure	3.6	2
295018 Partial or Total Loss of CCW / 8					X		AA2.03 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Cause for partial or complete loss	3.5	3
295006 SCRAM / 1						X	2.4.18 - Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.	4.0	4
295038 High Offsite Release Rate						X	2.4.11 - Emergency Procedures / Plan: Knowledge of abnormal condition procedures.	4.2	5
295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1						X	2.4.20 - Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes.	4.3	6
700000 Generator Voltage and Electric Grid Disturbances						X	2.2.40 - Equipment Control: Ability to apply technical specifications for a system.	4.7	7
295025 High Reactor Pressure / 3	X						EK1.05 - Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : Exceeding safety limits	4.4	39
600000 Plant Fire On-site / 8	X						AK1.02 - Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire Fighting	2.9	40
295004 Partial or Total Loss of DC Pwr / 6	X						AK1.05 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Loss of breaker protection	3.3	41
295030 Low Suppression Pool Water Level / 5		X					EK2.07 - Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: Downcomer/ horizontal vent submergence	3.5	42
295018 Partial or Total Loss of CCW / 8		X					AK2.01 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: System loads	3.3	43
295021 Loss of Shutdown Cooling / 4		X					AK2.03 - Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: RHR/shutdown cooling	3.6	44
295031 Reactor Low Water Level / 2			X				EK3.04 - Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL : Steam cooling	4.0	45
295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown / 1			X				EK3.01 - Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Recirculation pump trip/runback: Plant-Specific	4.1	46

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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#
295023 Refueling Acc Cooling Mode / 8			X				AK3.02 - Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS : Interlocks associated with fuel handling equipment	3.4	47
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4				X			AA1.01 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Recirculation System	3.5	48
295038 High Off-site Release Rate / 9				X			EA1.01 - Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Stack-gas monitoring	3.9	49
295016 Control Room Abandonment / 7				X			AA1.04 - Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : A.C. electrical distribution	3.1	50
295019 Partial or Total Loss of Inst. Air / 8					X		AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Instrument air system pressure	3.5	51
295006 SCRAM / 1					X		AA2.03 - Ability to determine and/or interpret the following as they apply to SCRAM : Reactor water level	4.0	52
295003 Partial or Complete Loss of AC / 6					X		AA2.04 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : System lineups	3.5	53
295026 Suppression Pool High Water Temp. / 5						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	54
295005 Main Turbine Generator Trip / 3						X	2.2.42 - Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	3.9	55
295024 High Drywell Pressure / 5						X	2.2.37 - Ability to determine operability and/or availability of safety related equipment.	3.6	56
700000 Generator Voltage and Electric Grid Disturbances						X	2.4.4 - Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.5	57
295028 High Drywell Temperature / 5		X					EK2.02 - Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Components internal to the drywell	3.2	58
K/A Category Totals:	3	4	3	3	3/3	4/4	Group Point Total:		20/7

ILT 09-1 NRC Exam Outline
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#
295009 Low Reactor Water Level / 2					X		AA2.02 - Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL : Steam flow/feed flow mismatch	3.7	8
295008 High Reactor Water Level / 2						X	2.1.20 - Conduct of Operations: Ability to interpret and execute procedure steps.	4.6	9
295013 High Suppression Pool Temperature / 5						X	2.1.23 - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.4	10
295017 High Off-site Release Rate / 9	X						AK1.03 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE : Protection of the general public	3.8	59
295035 Secondary Containment High Differential Pressure / 5		X					EK2.01 - Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following: Secondary containment ventilation	3.6	60
295033 High Secondary Containment Area Radiation Levels / 9			X				EK3.01 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Emergency depressurization	3.3	61
295034 Secondary Containment Ventilation High Radiation / 9				X			EA1.04 - Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : SBGT/FRVS: Plant-Specific	4.1	62
295002 Loss of Main Condenser Vac / 3					X		AA2.02 - Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM : Reactor power: Plant-Specific	3.2	63
295007 High Reactor Pressure / 3						X	2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.4	64
295022 Loss of CRD Pumps / 1						X	2.2.2 - Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	4.6	65
K/A Category Totals:	1	1	1	1	1/1	2/2	Group Point Total:		7/3

ILT 09-1 NRC Exam Outline
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				A2.07 - 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: A.C. electrical failure	2.8	11
215004 Source Range Monitor								X				A2.01 - Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply degraded	2.9	12
211000 SLC											X	2.4.4 - Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	4.7	13
263000 DC Electrical Distribution											X	2.1.30 - Conduct of Operations: Ability to locate and operate components, including local controls.	4.0	14
400000 Component Cooling Water											X	2.4.16 - Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.	4.4	15
212000 RPS	X											K1.01 - Knowledge of the physical connections and/or cause- effect relationships between REACTOR PROTECTION SYSTEM and the following: Nuclear instrumentation	3.7	1
223002 PCIS/Nuclear Steam Supply Shutoff	X											K1.02 - Knowledge of the physical connections and/or cause- effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the Reactor water cleanup	3.3	2
262001 AC Electrical Distribution		X										K2.01 - Knowledge of electrical power supplies to the following: Off-site sources of power	3.3	3
263000 DC Electrical Distribution		X										K2.01 - Knowledge of electrical power supplies to the following: Major D.C. loads	3.1	4

ILT 09-1 NRC Exam Outline
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G		Imp.	Q#
215005 APRM / LPRM			X									K3.08 - Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: Core thermal calculations	3.0	5
259002 Reactor Water Level Control			X									K3.03 - Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following: Rod worth minimizer: Plant-Specific	2.7	6
239002 SRVs				X								K4.08 - Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Opening of the SRV from either an electrical or mechanical signal	3.6	7
400000 Component Cooling Water				X								K4.01 - Knowledge of CCWS design feature(s) and or interlocks which provide for the following: Automatic start of standby pump	3.4	8
205000 Shutdown Cooling					X							K5.02 - Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : Valve operation	2.8	9
264000 EDGs					X							K5.05 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET): Paralleling A.C. power sources	3.4	10
261000 SGTS						X						K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM: Process radiation monitoring	2.9	11
262002 UPS (AC/DC)						X						K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : AC electrical power	2.7	12
209001 LPCS							X					A1.07 - Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Emergency generator loading	3.0	13

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Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
218000 ADS							X					A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: ADS valve tail pipe temperatures	3.4	14
215003 IRM								X				A2.04 - Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Up scale or down scale trips	3.7	15
300000 Instrument Air								X				A3.02 - Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: Air temperature	2.9	16
207000 Isolation (Emergency) Condenser									X			A3.03 - Ability to monitor automatic operations of the ISOLATION (EMERGENCY) CONDENSER including: Reactor water level: BWR-2,3	3.5	17
211000 SLC									X			A2.02 - 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: Failure of explosive valve to fire	3.6	18
215004 Source Range Monitor										X		A4.06 - Ability to manually operate and/or monitor in the control room: Alarms and lights	3.2	19
259002 Reactor Water Level Control										X		A4.03 - Ability to manually operate and/or monitor in the control room: All individual component controllers when transferring from manual to automatic modes	3.8	20
215005 APRM / LPRM											X	2.4.6 - Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.	3.7	21
264000 EDGs											X	2.4.3 - Emergency Procedures / Plan: Ability to identify post-accident instrumentation.	3.7	22
215004 Source Range Monitor		X										K2.01 - Knowledge of electrical power supplies to the following: SRM channels/detectors	2.6	23

ILT 09-1 NRC Exam Outline
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G		Imp.	Q#
239002 SRVs						X						K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES : Nuclear boiler instrument system (pressure indication)	3.2	24
212000 RPS										X		A4.17 - Ability to manually operate and/or monitor in the control room: Perform alternate reactivity/ shutdown operations	4.1	25
300000 Instrument Air				X								K4.01 - Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Manual/automatic transfers of control	2.8	26
K/A Category Totals:	2	3	2	3	2	3	2	2/2	2	3	2/3	Group Point Total:	26/5	

ILT 09-1 NRC Exam Outline
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G		Imp.	Q #
272000 Radiation Monitoring								X				A2.03 - Ability to 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: A.C. electrical failure	3.1	16
215001 Traversing In-core Probe											X	2.2.22 - Equipment Control: Knowledge of limiting conditions for operations and safety limits.	4.7	17
259001 Reactor Feedwater								X				A2.08 - Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of DC electrical power	2.6	18
241000 Reactor/Turbine Pressure Regulator	X											K1.14 - Knowledge of the physical connections and/or cause- effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: A.C. electrical power	2.8	27
226001 RHR/LPCI: CTMT Spray Mode		X										K2.02 - Knowledge of electrical power supplies to the following: Pumps	2.9	28
201003 Control Rod and Drive Mechanism			X									K3.01 - Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Reactor power	3.2	29
201002 RMCS				X								K4.06 - Knowledge of REACTOR MANUAL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Emergency In rod insertion	3.5	30
223001 Primary CTMT and Aux.					X							K5.01 - Knowledge of the operational implications of the following concepts as they apply to PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES : Vacuum breaker/relief operation	3.1	31
201006 RWM						X						K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) : Power Supply	2.8	32
233000 Fuel Pool Cooling/Cleanup							X					A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: Surge tank level	2.6	33

ILT 09-1 NRC Exam Outline
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				A2.07 - Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to retract during accident conditions: Mark-I&II(Not-BWR1)	3.4	34
204000 RWCU									X			A3.05 - Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: Reactor water temperature	2.8	35
201001 CRD Hydraulic										X		A4.01 - Ability to manually operate and/or monitor in the control room: CRD pumps	3.1	36
259001 Reactor Feedwater											X	2.4.34 - Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	4.2	37
290002 Reactor Vessel Internals								X				A2.03 - Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Control rod drop accident	3.6	38
K/A Category Totals:	1	1	1	1	1	1	1	2/2	1	1	1/1	Group Point Total:		12/3

Facility:	Oyster Creek ILT 09-1 NRC Exam Outline	Date: May 17, 2010				
Category	K/A #	Topic	RO		SRO-Only	
			IR	Q#	IR	Q#
1. Conduct of Operations	2.1.23	Ability to perform specific system and integrated plant procedures during all modes of plant operation.			4.4	19
	2.1.18	Ability to make accurate, clear and concise logs, records, status boards, and reports.	3.6	66		
	2.1.38	Knowledge of the station's requirements for verbal communications when implementing procedures.	3.7	67		
	Subtotal			2		1
2. Equipment Control	2.2.17	Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, coordination with the transmission system operator.			3.8	20
	2.2.36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.			4.2	24
	2.2.36	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	3.1	68		
	2.2.14	Knowledge of the process for controlling equipment configuration or status.	3.9	69		
	Subtotal			2		2
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.			3.7	21
	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.			3.1	23
	2.3.14	Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.4	70		

	2.3.5	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	2.9	71		
	2.3.11	Ability to control radiation releases.	3.8	74		
	Subtotal			3		2
4. Emergency Procedures / Plan	2.4.50	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.			4.0	22
	2.4.40	Knowledge of SRO responsibilities in emergency plan implementation.			4.5	25
	2.4.25	Knowledge of fire protection procedures.	3.3	72		
	2.4.46	Ability to verify that the alarms are consistent with the plant conditions.	4.2	73		
	2.4.2	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	4.5	75		
	Subtotal			3		2
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
RO 2/1	215005 K3.08	215005 K3.01 was very close in content to another KA in the outline (212000 K1.01: Relation between APRMs and RPS) and has been replaced. KA 215005 K3.08 (3.0/3.4) was randomly selected as a replacement.
RO 2/1	262002 K6.01	262002 K6.03 UPS at Oyster Creek (Vital AC) does not utilize static inverters. KA 262002 K6.01 (2.7/2.9) was randomly selected as a replacement.
RO 2/1	300000 A3.02	300000 K2.01 This KA is on the candidates' audit examination. KA 300000 A3.02 (2.9/2.7) was randomly selected as a replacement.
RO 2/1	211000 A2.02	211000 A3.01 This KA is related to the automatic operation of the SLC System. The SLC System at Oyster Creek has no automatic operation. 211000 A2.02 (3.6/3.9) was randomly selected as a replacement.
RO 2/2	223001 K5.01	223001 K5.10 An operationally relevant question could not be developed. KA 223001 K5.01 (3.1/3.3) was randomly selected as a replacement.
RO 2/2	201006 K6.01	201006 K6.04 The RWM does not lose functionality upon the loss of the Process Computer. KA 201006 K6.01 (2.8/3.2) was randomly selected as a replacement.
RO 2/2	204000 A3.05	204000 A3.02 There is no automatic response of the RWCU System related to reactor water quality. KA 204000 A3.05 (2.8/2.8) was randomly selected as a replacement.
RO 2/2	259001 2.4.34	271000 2.4.34 An operationally relevant question could not be developed. KA 259001 G2.4.34 (4.2/4.1) was randomly selected as a replacement.
RO 2/2	290002 A2.05	290002 A2.03 An operationally relevant question could not be developed. KA 290002 A2.05 (3.7/4.2) was randomly selected as a replacement.
RO 1/1	600000 AK1.02	600000 AK1.01 An operationally relevant question could not be developed. KA 600000 AK1.02 (2.9/3.1) was randomly selected as a replacement.
RO 1/1	295021 AK2.03	295021 AK2.06 An operationally relevant question could not be developed. KA 295021 AK2.03 (3.6/3.6) was randomly selected as a replacement.
RO 1/1	295001 AA1.01	295001 AA1.03 This KA is very close to question on the candidates' audit exam. 295001 AA1.01 (3.5/3.6) was randomly selected as a replacement.

RO 1/1	295038 EA1.07	295038 EA1.05 The PASS System is neither monitored nor operated by Licensed Operators. KA 295038 EA1.01 (3.9/4.2) was randomly selected as a replacement.
RO 1/1	295005 G2.2.37	295005 G2.2.4 This KA applies to a dual-unit facility. Oyster Creek is a single-unit facility. KA 259005 G2.2.37 (3.6/4.6) was randomly selected as a replacement.
RO 1/2	295007 G2.1.7	295007 G2.1.27 An operationally relevant question could not be developed. KA 295007 G2.1.7 (4.4/4.7) was randomly selected as a replacement.
RO 1/2	295022 G2.2.2	295022 G2.2.25 There is no safety limit associated with loss of CRD Pumps and LCO bases for loss of CRD Pumps is not appropriate at the RO level. KA 295022 G2.2.2 (4.6/4.1) was randomly selected as a replacement.
RO 3	G2.2.36	G2.2.39 This KA is not appropriate at the RO level. KA G2.2.36 (3.1/4.2) was randomly selected as a replacement.
RO 3	G2.2.14	G2.2.17 An operationally relevant question could not be developed. G2.2.14 (3.9/4.3) was randomly selected as a replacement.
RO 3	G2.4.2	G2.4.22 An operationally relevant question could not be developed. G2.4.2 (4.5/4.6) was randomly selected as a replacement.
SRO 1/1	295038 G2.4.11	295030 G2.4.11 Oyster Creek has no abnormal condition procedure (ABN) related to low Suppression Pool water level. KA 295038 G2.4.11 (4.0/4.2) was randomly selected as a replacement.
SRO 2/1	211000 G2.4.4	223002 G2.4.34 An operationally relevant question could not be developed. KA 211000 G2.4.4 (4.5/4.7) was randomly selected as a replacement.
SRO 2/1	263000 G2.1.30	300000 G2.1.30 An operationally relevant question could not be developed. KA 213000 G2.1.30 (4.4/4.0) was randomly selected as a replacement.
SRO 2/1	400000 G2.4.16	400000 G2.4.3 An operationally relevant question could not be developed. KA 400000 G2.4.16 (3.5/4.4) was randomly selected as a replacement.
SRO 2/2	259001 A2.08	239001 A2.06 An operationally relevant question could not be developed. KA 259001 A2.08 (2.5/2.6) was randomly selected as a replacement.
SRO 3	G2.2.36	G2.2.39 An operationally relevant question could not be developed. KA G2.2.36 (3.1/4.2) was randomly selected as a replacement.
SRO 3	G2.4.40	G2.4.27 An operationally relevant question could not be developed. KA G2.4.40 (2.7/4.5) was randomly

		selected as a replacement.
RO 1/2	295017 AK1.02	295017 AK1.03 An operationally relevant question could not be developed. KA 295017 AK1.02 (3.8/4.3) was randomly selected as a replacement.

The written outline used for the Oyster Creek ILT 09-1 NRC initial license exam was developed by commercially available software. More specifically, the Boiling Water Reactor Outline Generation Software, version 2.02, developed by Western Technical Services, was used to develop the outline.

Facility: Oyster Creek
Date of Examination: May 17, 2010
Examination Level: RO ☒ SRO ☐
Operating Test Number: OC 2010

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations	N, R	Determine Thermal Limit Restrictions with the EPR Out of Service IAW 202.1; G2.1.7 (4.4)
Conduct of Operations	D, S	Perform Week 4 of 680.4.007, Safety Related Equipment Verification; G2.1.29 (4.1)
Equipment Control	D, R, P	Perform Manual Core Heat Balance Calculation IAW 1001.6; G2.2.12 (3.7)
Radiation Control		
Emergency Procedures/Plan	M, S	Perform Actions of Shift Communicator During an Emergency; G2.4.39 (3.9)

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

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

JPM	RO Admin JPM Description
RO1	The candidate will be provided plant data, including that the EPR has just been declared inoperable. The candidate will be asked to calculate the new thermal limit restrictions (for MFLPD, MFLCPR and MAPRAT) IAW procedure 202.1, Attachment 7, and then determine if any actions are required. One actual thermal limit will be greater than allowed by the new thermal limit restrictions and certain actions are required by the procedure. This JPM can be performed as a group in the classroom.
RO2	The candidate will be directed to perform Week 4 of procedure 680.4.007, Safety Related Equipment Verification. With the procedure, the candidate will confirm the correct EOP jumper placement in the simulator for rated power. Several jumpers will be misplaced in several EOP Jumper boxes. This JPM will be performed individually in the simulator.
RO3	The candidate will be provided plant data and told to perform a reactor core heat balance calculation IAW procedure 1001.6. This JPM can be performed as a group in the classroom.
RO4	The candidate will be provided a completed State/Local Notification Form, used during declared emergencies. The candidate will review the form and discover any discrepancies. The candidate will then initiate the notification process. This JPM will be performed individually in the Shutdown Panel Room in the back if the simulator.

Facility: Oyster Creek	Date of Examination: May 17, 2010
Examination Level: RO ☐ SRO ☒	Operating Test Number: OC 2010

Administrative Topic (See Note)	Type Code*	Describe activity to be performed
Conduct of Operations	N,R	Approve Reactivation of License Logs; G2.1.4 (3.8)
Conduct of Operations	M, R	Apply Work Hour Rules; G2.1.5 (3.9)
Equipment Control	N, R	Review Completed Surveillance Test 619.3.016, High Drywell Pressure Scram Test And Calibration; G2.2.12 (4.1)
Radiation Control	N, R	Determine Recommendation for KI Issuance for Off-site Emergency Workers And On-site Personnel During An Emergency; G2.3.14 (3.8)
Emergency Procedures/Plan	M, R	Classify An Emergency And Initiate A State/Local Notification Form; G2.4.40 (4.5)

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

JPM	SRO Admin JPM Description
SRO1	The candidate will be directed to review and approve several Reactivation of License Logs. These logs are used to document actions required to reactivate an inactive licensed operator to allow performance of on-shift duties. The candidate will determine that several logs are deficient and one log is complete for approval. This JPM can be performed as a group in the classroom.
SRO2	The candidate will be provided the schedule for several operators and will identify any work hour restrictions IAW LS-AA-119-1003, Calculating Work Hours. The candidate will identify that two operators will exceed the work hour rules and one operator will meet all work hour requirements. This JPM can be performed as a group in the classroom.
SRO3	The candidate will review a completed surveillance test, 619.3.016, High Drywell Pressure Scram Test and Calibration. The candidate will determine that one pressure switch is inoperable, and will apply this to the Scram and Primary Containment Isolation sections of Tech Spec Table 3.1.1. This JPM can be performed as a group in the classroom.
SRO4	Given emergency conditions, the candidate will determine the need to administer potassium iodine (KI) to offsite field monitoring teams and to onsite emergency workers IAW EP-AA-113, Personnel Protective Actions. This JPM can be performed as a group in the classroom.
SRO5	Given emergency conditions, the candidate will reclassify the emergency. Then, the candidate will be directed to complete a State/Local Notification Form. This JPM can be performed as a group in the classroom.

Facility: Oyster Creek	Date of Examination: May 17, 2010
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐	Operating Test Number: OC 2010

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

System / JPM Title	Type Code*	Safety Function
a. Perform Recirculation Pumps Trip Circuitry Test, 603.4.001 w/ Multiple Recirculation Pump Trips (Alternate Path); 202001 A2.04 (3.7/3.8)	P, D, A	1
b. Shutdown Second RWCU Pump; 204000 A4.01 (3.1/3.0)	D	2
c. Transfer to the MPR and Raise RPV Pressure; 241000 A4.01 (3.8/3.8)	M	3
d. Shutdown Core Spray with Actuating Signals Present (Alternate Path); 209001 A4.01 (3.8/3.6)	D, A, EN	4
e. Purge the Primary Containment (Alternate Path); 223001 A4.07 (4.2/4.1)	N, A	5
f. De-energize 1A1 Transformer by Cross-tieing USS 1A1 to USS 1B1; 262001 A1.05 (3.2/3.5)	N, L	6
g. Delete a Substitute Control Rod Position in the RWM and Initiate the Power Ops Mode; 201006 A4.02 (2.9/2.9)	M	7
h. Restart RB Ventilation System with Fan Failure (Alternate Path); 288000 A4.01 (3.1/2.9)	N, A	9

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

System / JPM Title	Type Code*	Safety Function
i. Vent the Control Rod Drive Over Piston Volume; 201003 A2.01 (3.4/3.6)	D, R, E	1
j. Supply Alternate Air Supply for Isolation Condenser Makeup Valves; 207000 A1.09 (3.7/3.7)	D, R, E	4
k. Swap Static Chargers from C1 to C2; 263000 K1.02 (3.2/3.3)	D, R	6

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

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

Facility: Oyster Creek	Date of Examination: May 17, 2010
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐	Operating Test Number: OC 2010

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

System / JPM Title	Type Code*	Safety Function
a. Perform Recirculation Pumps Trip Circuitry Test, 603.4.001 w/ Multiple Recirculation Pump Trips (Alternate Path); 202001 A2.04 (3.7/3.8)	P, D, A	1
b.		
c. Transfer to the MPR and Raise RPV Pressure; 241000 A4.01 (3.8/3.8)	M	3
d. Shutdown Core Spray with Actuating Signals Present (Alternate Path); 209001 A4.01 (3.8/3.6)	D, A, EN	4
e. Purge the Primary Containment (Alternate Path); 223001 A4.07 (4.2/4.1)	N, A	5
f. De-energize 1A1 Transformer by Cross-tieing USS 1A1 to USS 1B1; 262001 A1.05 (3.2/3.5)	N, L	6
g. Delete a Substitute Control Rod Position in the RWM and Initiate the Power Ops Mode; 201006 A4.02 (2.9/2.9)	M	7
h. Restart RB Ventilation System with Fan Failure (Alternate Path); 288000 A4.01 (3.1/2.9)	N, A	9

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

System / JPM Title	Type Code*	Safety Function
i. Vent the Control Rod Drive Over Piston Volume; 201003 A2.01 (3.4/3.6)	D, R, E	1
j. Supply Alternate Air Supply for Isolation Condenser Makeup Valves; 207000 A1.09 (3.7/3.7)	D, R, E	4
k. Swap Static Chargers from C1 to C2; 263000 K1.02 (3.2/3.3)	D, R	6

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

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

Facility: Oyster Creek Date of Examination: May 17, 2010

Exam Level: RO ☐ SRO-I ☐ SRO-U ☒ Operating Test Number: OC 2010

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

System / JPM Title	Type Code*	Safety Function
a.		
b.		
c.		
d. Shutdown Core Spray with Actuating Signals Present (Alternate Path); 209001 A4.01 (3.8/3.6)	D, A, EN	4
e. Purge the Primary Containment (Alternate Path); 223001 A4.07 (4.2/4.1)	N, A	5
f. De-energize 1A1 Transformer by Cross-tieing USS 1A1 to USS 1B1; 262001 A1.05 (3.2/3.5)	N, L	6
g.		
h.		

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

i. Vent the Control Rod Drive Over Piston Volume; 201003 A2.01 (3.4/3.6)	D, R, E	1
j. Supply Alternate Air Supply ~~for~~ Isolation Condenser Makeup Valves; 207000 A1.09 (3.7/3.7)	D, R, E	4
k. INJ CST → C.S.		2

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

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

JPM	Control Room/In-Plant System JPMs Description
Sim1	The candidate will start the performance of surveillance test Recirculation Pumps Trip Circuitry Test, 603.4.001. After several manipulations, several recirculation pumps will trip. The candidate will then enter ABN-2, Recirculation System Failures, and scram the reactor IAW ABN-1. (Alternate Path)
Sim2	The candidate will remove the second RWCU Pump from service IAW procedure 303, Reactor Cleanup Demineralizer System. The candidate will throttle closed the pump discharge valve, control system pressure with the FCV, and stop the pump.
Sim3	The candidate will be directed place the MPR in control IAW ABN-9, Electronic Pressure Regulator Malfunction. The candidate will then be directed to restore RPV pressure to the normal value with the MPR.
Sim4	The candidate will be directed to secure the Core Spray System IAW procedure 308, Emergency Core Cooling System Operation, due to an inadvertent initiation. With the initiation signals still present, the candidate will press the Override pushbuttons, the Actuated pushbuttons, verify Parallel Isolation Valves closed, and secure a Core Spray Booster Pump, which will not trip. The candidate will notify an EO to trip the pump locally. Following the local trip of the Booster Pump, the candidate will then trip the Core Spray Pump. (Alternate Path)
Sim5	The candidate will be directed to purge the Torus IAW procedure 312.9, Primary Containment Control. The candidate will verify valve positions, place a switch in Bypass, open the Torus valves and monitor stack gas activity. When Torus pressure approaches 0 psig, the valves Torus valves are closed and then the Drywell vent valves are opened. When stack activity reaches 1000 cps, the candidate will close the Drywell valves. (Alternate Path)
Sim6	The candidate will be directed to crosstie USS 1A1 to USS 1B1 IAW procedure 337, 4160 Volt Electrical System. The candidate will manipulate several breakers including a synch scope.
Sim7	The candidate will be directed to delete a previously substituted control rod position in the RWM IAW procedure 409, Operation of the Rod Worth Minimizer. The candidate will press several keys on the RWM panel to delete the substitute control rod position, and will be cued to then initiate a full core scan, which is also performed with keys on the RWM Panel. Following this, the candidate is directed to

	initiate the Power Ops Mode of the RWM.
Sim8	The candidate will be directed to restart the RB HVAC System IAW Support Procedure 50, Reactor Building Ventilation Restart. When restarting the supply fans, one fan will not start and the candidate will start a different supply fan. (Alternate Path)
Plant1	The candidate will vent the CRD over-piston volume IAW Support Procedure 21, Alternate Insertion of Control Rods. The candidate will make several simulated valve manipulations in the Reactor Building.
Plant2	The candidate will be directed to provide an alternate air supply to the Isolation Condenser makeup valves IAW procedure 307, Isolation Condenser System. The candidate will make several valve manipulations and connect the air hose.
Plant3	The candidate will be directed to swap DC chargers from C1 to C2 IAW procedure 340.3, 125 Volt DC Distribution System C. The candidate will simulate opening and closing several breakers.

Scenario Outline

Facility: Oyster Creek **Scenario No.:** 1 **Op Test No.:** OC 2010

Examiners: ______ **Operators:** ______

Initial Conditions:
- The plant is at 97% power
- RWCU Pump B is tagged out of service

Turnover:
- Surveillance test Standby Gas Treatment System 10-Hour Run – System 1, 651.4.002, is in-progress

Event No.	Malf. No.	Event Type*		Event Description
1	NA	N	BOP	Perform Automatic Scram Contactor Test, 619.4.025
2	NA	R	ATC	Withdraw control rods IAW the ReMA
3	MAL-CRD005_1835	C TS	ATC BOP SRO	Responds to a continuously outward drifting control rod
4	VLV-RCU001 VLV-RCU004 BKR-RCU001 MAL-RCU007	C TS	BOP SRO	Responds to RWCU System high pressure and failure of system to automatically isolate
5	MAL-SCN003A	TS	SRO	Respond to trip of SGTS Fan 1-8
6	MAL-CRD006	C	ATC	Respond to multiple drifting control rods
7	MAL-NSS005	M	All	Respond to an RPV coolant leak in the Primary Containment
8	MAL-OED001B MAL-FWC003A	C	All	Respond to the loss of Startup Transformer B and MFRV A Closure

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor Transient, (TS) Tech Specs

Simulator Summary

Event	Event Summary
1	The BOP will complete Automatic Scram Contactor Test, 619.4.025. The BOP will place the Subchannel Test 2A switch to Trip, verify proper plant response, place the switch back to Normal, and reset the ½ scram. The same will be performed with the 2B switch. [Normal Evolution: BOP]
2	The ATC will withdraw several control rods IAW the Reactivity Management Approval (ReMA) form from OP-AB-300-1003 and procedure 302.2. The ATC will turn Rod Power on, select the control rods and withdraw to the desired positions. [Reactivity Manipulation: ATC]
3	The ATC will respond to a control rod which drifts out of the core. The ATC will respond to the ROD DRIFT annunciator and ABN-6, Control Rod Malfunctions. The ATC will identify the drifting control rod, select the control rod and insert to its original position and release the drive switch. The ATC will identify that the control rod is still drifting outward. The ATC will insert the control rod to position 00, while the BOP scrams the single control room from a back panel. With the control rod successfully at position 00, it will be isolated IAW 302.1, Control Rod Hydraulic System. The SRO will declare the control rod inoperable and will apply Tech Spec 3.2.B.4. [Component Failure: ATC; Component Failure: BOP; Tech Spec: SRO]
4	The BOP will respond to an annunciator (D7b) for a high pressure condition in the Reactor Water Cleanup System (RWCU). This condition should have isolated the RWCU System but did not. The BOP will trip the RWCU Pump and isolate the RWCU System manually. The SRO will apply Tech Spec 3.5.A.3 for isolation valve failures. [Component Malfunction: BOP; Tech Spec: SRO]
5	The Standby Gas Treatment System (SGTS) Fan which was running for the surveillance test will trip. The SRO will declare the SGTS Fan inoperable and will apply Tech Specs 3.5.B.6.a(1). [Tech Specs: SRO]
6	The ATC will identify/report multiple drifting control rods and IAW the ROD DRIFT annunciator response, will insert a manual scram IAW ABN-1, Reactor Scram. [Component Failure: ATC]
7	The crew will respond to an RPV water leak in the Primary Containment and the SRO will direct entering the RPV Control – No ATWS EOP and the Primary Containment Control EOP. The SRO will

direct spraying the Drywell, which will be effective. [Major Evolution]

8	Startup Transformer B will experience a fault and MFRV A will fail closed. This will result in only Condensate Pump A and Feedwater Pump A having electrical power, and the injection path will only be through the low flow regulating valve. RPV water level will lower due to the leak. [Component failures after EOP entry]
Critical Task 1	When multiple drifting control rods are recognized, then reactor is scrammed IAW ABN-1, Reactor Scram
Critical Task 2	When Drywell or Torus exceeds 12 psig, or before Drywell bulk temperature reaches 281 °F, spray the Drywell IAW Support Procedure 29, Initiation of the Containment Spray System for Drywell Sprays
Critical Task 3	Reduce RPV pressure to allow low pressure systems to inject into the RPV or Emergency Depressurize the RPV when RPV water level reaches 0" with at least one injection source running

Scenario Outline

Facility: Oyster Creek **Scenario No.:** 2 **Op Test No.:** OC 2010

Examiners: ______ **Operators:** ______

Initial Conditions:
- **The plant is at 90% power**
- **The RWM is inoperable and bypassed**
- **Service Water Pump 1-2 is OOS**

Turnover:
- **Perform Anticipatory Scram Turbine Stop Valve Closure Test (>45% Load), 619.4.002**
- **Raise reactor power to rated**

Event No.	Malf. No.	Event Type*		Event Description
1	NA	N	BOP	Perform Anticipatory Scram Turbine Stop Valve Closure Test (>45% Load), 619.4.002
2	NA	R	ATC	Raise reactor power to 100% with Recirculation Flow
3	ICH-NSS118A RLY-RPS044B RPS043B RPS048B RPS047B	C TS	ATC SRO	Responds to RE05B RPV water level instrument failure (low) without the expected ½ scram response on RPS 1
4	MAL-MSS005A	C	BOP	Responds to trip of Steam Packing Exhauster 1
5	CNH-FWH001B CNH-FWH004B CNH-FWH007B	C	ATC	Responds to partial loss of feedwater heating
6	MAL-NSS026C	I TS	BOP SRO	Responds to EMRV acoustic monitor failure (NR108C)
7	MAL-RSX001	M	All	Responds to rising main steam and offgas radiation monitors due to fuel failures
8	VLV-ICS005 VLV-ICS006 MAL-ICS003A	C	All	Responds to unisolable Isolation Condenser steam leak with fuel failures leading to Emergency Depressurization

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor Transient, (TS) Tech Specs

Simulator Summary

Event	Event Summary
1	The BOP will complete the last portion of turbine stop valve testing surveillance IAW Anticipatory Scram Turbine Stop Valve Closure Test (>45% Load), 619.4.002. The BOP will manipulate the test pushbutton for Stop Valve # 3, verify expected plant response, then return the Test switch to the Off position, and again verify proper plant response. [Normal Evolution: BOP]
2	The ATC will raise power to rated with recirculation flow IAW a Reactivity Management Approval (ReMA) Form and procedure 202.1, Power Operation. [Reactivity Manipulation: ATC]
3	The ATC will respond to a downscale failure of an RPS water level instrument (RE05B) and the expected ½ scram will not occur. The SRO will direct entry into ABN-39, RPS Failures. The ATC will insert a ½ scram on RPS1 and the SRO will apply Tech Specs Table 3.1.1.A. [Component Failure: ATC; Tech Specs: SRO]
4	The BOP will respond to the failure of the in-service steam packing exhauster. The BOP will start the standby Exhauster and throttle open associated discharge valve to maintain the correct vacuum. [Component Failure: BOP]
5	The ATC will respond to the loss of the Feedwater Heating sting A (HP, IP and LP). IAW ABN-17, the ATC will reduce reactor power to 20% below to pre-trip power level. [Component Failure: ATC]
6	The BOP will respond to indications of an open EMRV. The BOP will diagnose the event as a failed acoustic monitor and will bypass the alarms IAW procedure 413. The SRO will apply Tech Specs 3.13. [Instrument Failure: BOP; Tech Spec: SRO]
7	The Crew will respond to rising indications in main steam and offgas radiation monitors. The Crew will enter ABN-26, High Main Steam/Offgas/Stack Effluent Activity, and will reduce power, initiate a shutdown, then manually scram the reactor, and close the MSIVs, the Isolation Condenser vents, and reactor sample valves. [Major]
8	A steam leak will occur in the Isolation Condenser System, which will be unisolable. The steam leak combined with the fuel failures will result in exceeding the Max Safe radiation levels and/or temperature levels in two areas and the SRO will direct an Emergency Depressurization of the RPV. [Component Failure after EOP]

Critical Task 1	IAW in ABN-26, scram the reactor and close the MSIVs when the OFFGAS HI-HI alarm comes in and does not clear within 15 minutes.
Critical Task 2	With a primary system discharging into the Reactor Building, and radiation levels in two or more areas exceed the MAX SAFE values, or temperature levels in two or more areas exceed the MAX SAFE values, then Emergency Depressurized the RPV.

Scenario Outline

Facility: Oyster Creek **Scenario No.:** 3 **Op Test No.:** OC 2010

Examiners: ____________ **Operators:** ____________

____________ ____________

____________ ____________

Initial Conditions:
- **The plant is at 100% power**
- **Air Compressor 3 is tagged out of service**

Turnover:
- **Reduce power IAW the ReMA**
- **Perform 323.6, Backwashing Condensers**

Event No.	Malf. No.	Event Type*		Event Description
1	NA	R	ATC	Reduces reactor power with recirculation flow to 97%
2	NA	N	BOP	Performs Condenser A North Condenser Backwash procedure
3	MAL-SLC003A	TS	SRO	Respond to Standby Liquid Control System 1 loss of squib continuity
4	BKR-RFC001 MAL-RFC002A	C	BOP	Respond to abnormalities on Recirculation Pump A
5	MAL-EDS003B	C TS	All	Respond to the loss of 480 VAC USS 1A2
6	BKR-CRD001 MAL-CRD010 MAL-CRD007	C	ATC	Respond to CRD Pump NC08B trip leads to a manual scram; Four control rods remain at position 48
7	MAL-NSS005C MAL-PCN008	M C	All	Respond to primary coolant leak in the Drywell
8	VLV-CNS005	C	All	Respond to the loss of Drywell Sprays

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor Transient, (TS) Tech Specs

Simulator Summary

Event	Event Summary
1	The ATC will lower reactor power with recirculation flow IAW the Reactivity Management Approval Form (ReMA) too allow condenser backwashing in the next event [Reactivity Manipulation: ATC]
2	The BOP will perform a backwash of the condenser half A North IAW procedure 323.6, Backwashing Condensers. The BOP will verify the correct lineup, place the Backwash Control switch in Backwash, place the Cond A North switch to Close, verify the lineup, place the Cond A North switch to Open, place the Backwash Control switch to Close, and then verify the proper lineup. [Normal Evolution: BOP]
3	The BOP will respond to a loss of Standby Liquid Control System 1 squib valve continuity. The SRO will declare Standby Liquid Control System 1 inoperable and will apply Tech Specs 3.2.C.3.a. [Tech Specs: SRO]
4	The BOP will respond to the trip of the Recirculation MG Set A field breaker. The drive motor breaker will fail to auto trip. The BOP will trip the drive breaker, enter ABN-2, Recirculation System Failures, and close the pump discharge valve. [Component Failure: BOP]
5	The Crew w ill respond to the loss of 480 VAC USS 1A2. The ATC will start CRD Pump NC08B and reset the RPS 1 ½ scram and ½ isolation when RPS power is restored. The BOP will start RBCCW Pump 1-2, secure Reactor Building ventilation, initiate Standby Gas Treatment System 2, restore power to RPS 1, and reset alarms. The SRO will review and apply Tech Specs 3.7.B. [Component Failure: ATC & BOP; Tech Specs: SRO]
6	The ATC will respond to CRD Pump NC08B trip. HCU accumulator trouble alarms will then be received. The Alarm Response for CRD Pump trip directs a manual scram given no CRD Pumps and multiple HCU accumulator alarms. Four control rods will remain at position 48 following the manual scram. [Component Failure: ATC]
7/8	The Crew will respond to a primary coolant leak in the Primary Containment. At the same time, a pre-existing leak in a Drywell downcomer will allow communication between the Drywell air space and the Torus air space. The SRO will direct Drywell Sprays but sprays will not operate. The SRO will direct Emergency Depressurization when it has been determined that Torus pressure cannot be maintained below the Primary System Pressure (PSP) Curve. [Major Event; Component Failure after EOP]

Critical Task 1	With RPV pressure > 850 psig and CRD charging water pressure cannot be immediately re-established, and two or more accumulator alarms are received, then scram the reactor
Critical Task 2	When it has been determined that Torus pressure cannot be maintained below the Primary System Pressure (PSP) Curve, then Emergency Depressurize the RPV

Scenario Outline

Facility: Oyster Creek **Scenario No.:** 4 **Op Test No.:** OC 2010

Examiners: ____________ **Operators:** ____________

Initial Conditions:
- **14% power during a startup (IC 152)**
- **The RWM is inoperable and Bypassed**
- **Control Room HVC System A is inoperable**

Turnover:
- **Startup in progress**

Event No.	Malf. No.	Event Type*		Event Description
1	NA	N	BOP	Swaps Service Water Pumps
2	NA	R	ATC	Withdraws control rods to raise reactor power
3	MAL-CRD007	C	ATC	Respond to indications of a stuck control rod
4	MAL-EDS004B	C TS	BOP SRO	Respond to the loss of Vital Bus 1B2
5	LOA-RCP003 MAL-RCP003C MAL-RCP004C	C TS	BOP SRO	Responds to Recirculation Pump C inner seal failure, then outer seal failure
6	MAL-NSS025C	C	ATC	Responds to an open EMRV leading to a manual scram
7	CAEP ATWS	M	All	Responds to an electric ATWS
8	PMP-SLC001A PMP-SLC002A	C	RO	Respond to Standby Liquid Control Pump shaft break

* **(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor Transient, (TS) Tech Specs**

Simulator Summary

Event	**Event Summary**
1	The BOP will swap Service Water Pumps to equalize run times. The BOP will start the standby pump, stop the running pump, and then verify expected conditions locally with the EO. [Normal Evolution: BOP]
2	The TC will withdraw control rods to raise reactor power IAW the pull sheet and 302.2. [Reactivity Manipulation: ATC]
3	The ATC will respond to indications of a stuck control rod. The ATC will raise drive pressure and attempt to move the control rod. The control rod will then move. The ATC will then return drive pressure to normal and continue withdrawing control rods. [Component Failure: ATC]
4	The BOP will respond to and diagnose the loss of Vital Bus 1B2 and will perform actions IAW ABN-51, Loss of VMCC 1B2. The BOP will place RPS 2 on an alternate power supply, reset alarms and place a diesel fire pump in the manual mode. The SRO will apply Tech Specs 3.7 for the bus loss. [Component Failure: BOP; Tech Specs: SRO]
5	The BOP will respond to a leak in Recirculation Pump C inner seal, followed by a leak in the outer seal. The SRO will direct entry into ABN-2 to trip and isolate the pump. The SRO will refer to Tech Specs 3.3.D for unidentified RCS leakage. [Component Failure: BOP; Tech Spec: SRO]
6	The BOP will respond to indications of an open EMRV. The SRO will direct entry into ABN-40. The ATC will place feedwater level control in manual, while the BOP will attempt to close the valve. When determined the valve will not close, the ATC will balance the feedwater level controller and place back in auto. The ATC will then scram the reactor IAW ABN-1. [Component Failure: ATC]
7	The Crew will diagnose an electric ATWS and the SRO will direct entry into RPV Control – With ATWS. The ATC will perform actions to insert control rods and the BOP will perform actions to control Torus water temperature and RPV water level. [Major Evolution]
8	Due to the Torus water heatup, Standby Liquid Control injection will be directed. The first SLC Pump started will have a broken shaft and the Candidate will start the second pump. [Component Failure after EOP]

Critical Task 1	Shutdown the reactor IAW Support Procedure 21, Alternate Insertion of Control Rods
Critical Task 2	With reactor power > 2% during an ATWS, terminate and prevent injection into the RPV except CRD and Boron injection
Critical Task 3	With the reactor at power and an open EMRV which cannot be closed, then manually scram the reactor

ADAMS MASTER EXAM FILE PACKAGE

ADAMS DOCUMENT COVER SHEET

DOCUMENT TITLE: Oyster Creek - Draft Written Exam

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SITE:	Oyster Creek
Exam DATES:	MAY 17, 2010 to MAY 21, 2010
Chief Examiner:	J. D'Antonio
TAC NO:	U01771

ADAMS Profiled & entered into DPC on ________ by ______

EXAMINATION ANSWER KEY

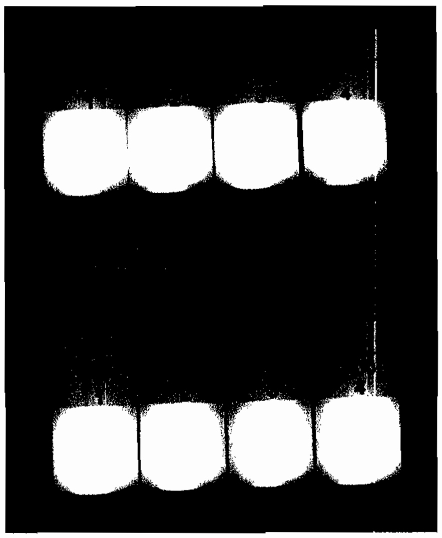
ILT 09-1 NRC RO Exam

1 | **ID: 09-1 NRO1** | **Points: 1.00**

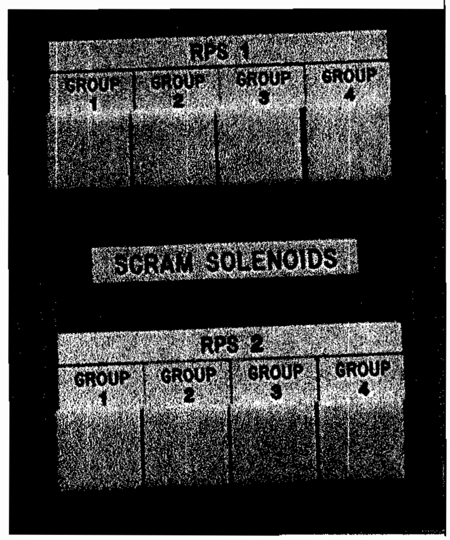
The plant was at rated power. Recirculation flow was at maximum.

Which of the following combination of **simultaneous** events will result in the indication below changing from **CONDITION 1** to **CONDITION 2**?

Condition 1



Condition 2



	Event 1	Event 2
A.	IRM 18 indicates upscale	APRM 8 indicates downscale
B.	APRM 4 indicates 114%	APRM 6 indicates 114%
C.	APRM 1 indicates INOP	APRM 6 indicates 120%
D.	APRM 3 indicates downscale	APRM 7 indicates INOP

Answer: C

Answer Explanation:

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

QID: 09-1 NRO1		
Question # / Answer	**1**	**Developer/Date: NTP 11/11/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
212000 RPS K1.01 Knowledge of the physical connections and/or cause- effect relationships between REACTOR PROTECTION SYSTEM and the following: Nuclear instrumentation						**3.7**	**3.9**
Level	**RO**	**Tier**	**2**	**Group**	**1**		

General References	**237E566**	**RAP-G1f RAP-G2f**	**RAP-G2f RAP-H7a**
Explanation	The plant is at rated power, and the question asks what events will cause all group scram solenoids to change state from currently energized to de-energized, as pictured. A 1/2 scram from RPS 1 APRMs (APRMs 1-4 HI-HI (>118%) or INOP) plus a 1/2 scram from RPS 2 APRMs (APRMs 5-8 HI-HI (>118%) or INOP) will cause all group scram solenoids to be de-energized. Correct answer C shows a 1/2 scram on RPS 1 from an inoperable RPS 1 APRM, plus a 1/2 scram on RPS 2 from an RPS APRM HI-HI. Answer A will result in a rodblock only, which does not impact the scram solenoids, and is incorrect. Answer B shows an RPS 1 and RPS 2 APRM above the rodblock setpoint of 113%, but below the scram setpoint of the HI-HI. Again, RPS in not impacted, and answer B is incorrect. Answer D shows an RPS 1 APRM downscale, which only provides a rod block, plus an RPS 2 APRM INOP resulting in a 1/2 scram on RPS 2 only. Thus, only RPS 2 group scram solenoids are affected but not all group scram solenoids are affected. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0029 LO 215-10446		

<table>
<tr><td colspan="3">Question Source (New, Modified, Bank)</td><td colspan="2">New</td></tr>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X
3:SPK</td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td>5</td><td>55.43</td><td></td></tr>
<tr><td colspan="4">(SRO Only)</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
</table>

Group Heading **REACTOR NEUTRON MONITORS**	**G-1-f**

[redacted]

CONFIRMATORY ACTIONS:

❑	IF	half scram signal exists,	
	THEN	**DETERMINE** cause of trip by checking APRM recorders at Panel 4F **and** APRM cabinets at Panels 3R and 5R.	[ ]

AUTOMATIC ACTIONS:

Reactor scram coincident with Channel II trip.

MANUAL CORRECTIVE ACTIONS:

❑	IF	APRM is high,	
	THEN	**REDUCE** reactor power by inserting control rods in accordance with the rod sequence.	[ ]
❑	IF	unit is inoperative,	
	THEN	**CHECK** the affected APRM cabinet for the following:	
		• improper mode switch position	[ ]
		• module removed	[ ]
		• more than three LPRM inputs bypassed.	[ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 2)

[redacted]

MANUAL CORRECTIVE ACTIONS: (continued from Page 1 of 2)

❑	**REFER** to Procedure 403, LPRM-APRM System Operations for bypassing APRM.		[ ]
❑	IF	a full scram condition exists,	
	THEN	**REFER** to ABN-1, Reactor Scram.	[ ]
❑	IF	failure of APRM channels results in conditions less conservative than those permitted by Technical Specifications,	
	THEN	**SHUTDOWN** the reactor.	[ ]
❑	IF	all APRM indication is lost,	
	THEN	manually **SCRAM** the reactor IAW ABN-1, Reactor Scram.	[ ]
❑	**USE** IRMs and SRMs to monitor reactor power.		[ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Core power exceeding predetermined level for the existing recirculation flow condition as described by 0.90 x 10^{-6}w + 65.1, or module inoperable, indicating mode switch on APRM drawer not in operate position, module removed, or more than three LPRM inputs bypassed. These are trip signal inputs to Reactor Protection System Channel I.	0.90 x 10^{-6}w + 65.1 (Maximum setpoint of 118% power) or Module Inoperable.	RJ19A and RJ19B Reference Drawings: GE 237E566 Sh. 1A & 1B GE 706E812 Sh. 19 & 22 GU 3E-611-17-009 Sh. 1

[redacted]

CONFIRMATORY ACTIONS:

❑ IF half scram signal exists,

THEN **DETERMINE** cause of trip by checking APRM recorders at Panel 4F **and** APRM cabinets at Panels 3R and 5R. [ ]

AUTOMATIC ACTIONS:

Reactor scram coincident with Channel I trip.

MANUAL CORRECTIVE ACTIONS:

❑ IF APRM is high,

THEN **REDUCE** reactor power by inserting control rods in accordance with the rod sequence. [ ]

❑ IF unit is inoperative,

THEN **CHECK** the affected APRM cabinet for the following:

- improper mode switch position [ ]
- module removed [ ]
- more than three LPRM inputs bypassed. [ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 2)

(RED)

APRM
HI-HI/INOP II

MANUAL CORRECTIVE ACTIONS:

- ❑ **REFER** to Procedure 403, LPRM-APRM System Operations. [ ]
- ❑ IF a full scram condition exists,
  THEN **REFER** to ABN-1, Reactor Scram. [ ]
- ❑ IF failure of APRM channels results in conditions less conservative than those permitted by Technical Specifications,
  THEN **SHUTDOWN** the reactor. [ ]
- ❑ IF all APRM indication is lost,
  THEN manually **SCRAM** the reactor IAW ABN-1, Reactor Scram. [ ]
- ❑ **USE** IRMs and SRMs to monitor reactor power. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Core power exceeding predetermined level for the existing recirculation flow condition as described by $0.90 \times 10^{-6}w + 65.1$, or module inoperable, indicating mode switch on APRM drawer not in operate position, module removed, or more than three LPRM inputs bypassed. These are trip signal inputs to Reactor Protection System Channel II.	$0.90 \times 10^{-6}w + 65.1$ (Maximum setpoint of 118% power) or Module Inoperable.	RJ19C and RJ19D Reference Drawings: GE 237E566 Sh. 1A & 1B GE 706E812 Sh. 26 & 29 GU 3E-611-17-009 Sh. 1

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

2 **ID: 09-1 NRO2** **Points: 1.00**

The plant was at rated power when an event occurred. The Panel Operator reports the following observations:

- ISOL COND - LOGIC TRAIN ACTUATED I annunciator is in alarm
- ISOL COND - LOGIC TRAIN ACTUATED II annunciator is in alarm
- EDG 1 UNIT IDLING light is energized
- EDG 1 UNIT START light is de-energized
- EDG 2 UNIT IDLING light is energized
- EDG 2 UNIT START light is de-energized
- RPV pressure indicates 425 psig and lowering slowly

Which of the following is correct?

A. **All** 4 RWCU Isolation Valves indicate **closed**.

B. **All** 4 Core Spray Parallel Isolation Valves indicate **open**.

C. **Both** EDG 1 and EDG 2 Output Breakers indicate **closed**.

D. **All** 10 RB HVAC MAIN SUPPLY HEADER VALVES indicate **open**.

Answer: A

Answer Explanation:

QID: 09-1 NRO2		
Question # / Answer	**2**	**Developer/Date: NTP 11/11/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
223002 PCIS/Nuclear Steam Supply Shutoff K1.02 - Knowledge of the physical connections and/or cause- effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the Reactor water cleanup				**3.3**	**3.5**
Level	**RO**	**Tier**	**2**	**Group**	**1**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	**EMG-SP1**	**341** **RAP-C1a**	**330** **2621.828.0.0013**
Explanation	The plant was at rated power when an event occurred which resulted in the following (from the indications provided): both Isolation Condensers are in service and both EDGs have idle started. The ICs will auto initiate from either an RPV water level lo-lo (90") or RPV high pressure (1051 psig). The EDGs will auto idle start from either an RPV water level lo-lo (86"), a high Drywell pressure (3 psig) or from Low Lube Oil Temperature. The only single event that will both initiate the ICs and idle start the EDGs is an RPV water level lo-lo. On an RPV water level lo-lo, all 4 RWCU isolation valves will close if open (not all 4 valves close on an RWCU system upset condition). Answer A is correct. Under the conditions provided, the Core Spray System has auto initiated. But since RPV pressure is 425 psig, the Core Spray Parallel Isolation Valves are still closed and will auto open when RPV pressure lowers to 305 psig. Answer B is incorrect. The conditions provided show only that the EDGs have idle started and are not supplying their respective emergency busses. Therefore, the EDG output breakers are open. Answer C is incorrect. The conditions provided show that the Standby Gas treatment System has auto started and normal RB HVAC has secured. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0039 LO 204-10445		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X** **3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

AmerGen An Exelon Company	OYSTER CREEK GENERATING STATION PROCEDURE	Number **EMG-SP1**
SUPPORT PROCEDURE 1 Title **CONFIRMATION OF AUTOMATIC INITIATIONS AND ISOLATIONS**		Revision No. 0

SYSTEM		OPERATING DETAILS
Cleanup System Isolation	IF	Any of the following conditions exist: • RPV water level at or below **86 in.** and **not** bypassed • Drywell pressure at or above **3.0 psig** and **not** bypassed • RWCU HELB Alarms
	THEN	**CONFIRM** closed the following Cleanup Isolation valves: (Panel 3F/11F) V-16-1 [ ] V-16-14 [ ] V-16-2 [ ] V-16-61 [ ]
Shutdown Cooling System Isolation	IF	Any of the following conditions exist: • RPV water level at or below **86 in.** • Drywell pressure at or above **3.0 psig**
	THEN	**CONFIRM** closed the following SDC Isolation Valves: (Panel 11F) V-17-54 [ ] V-17-19 [ ]
Isolation Condenser Initiation	IF	Any of the following conditions exist or have occurred: • RPV water level at or below **86 in.** • Reactor pressure at or above **1050 psig.**
	THEN	**CONFIRM** that both Isolation Condensers did initiate. (ICs may have been removed from service by Pressure Control Leg.) [ ]

OVER

bypassed (the starting resistors are automatically short-circuited) so that the engine starts and accelerates to rated speed as quickly as possible.

d) If the engine does not start after 15 seconds of cranking, an SEQ Fault shuts down the engine.

e) Following engine ignition, the diesel accelerates to full speed in about 15 seconds.

f) The generator field is flashed automatically. The governor and voltage regulator establish 4160 VAC, 60 Hz EDG output power.

g) The EDG output breaker automatically closes to energize the emergency bus.

b. EDG Start Signals

1) Idle-Start Signals

The EDGs automatically start after a 10-second time delay and accelerate to 400 rpm upon receipt of any of the following signals:

a) RPV Lo-Lo Water Level (86" TAF) from Core Spray system initiation logic.

b) Drywell High Pressure (3.0 psig) from Core Spray system initiation logic.

c) Engine Low Lube-Oil Temperature (LOTS set at 85 °F) - Lube oil is warmed by engine heat

The EDG continues to run at idle speed until the start signal has been reset or a manual stop has been initiated. Following a 15-minute time delay the engine shuts-down.

2) Fast-Start Signals

The EDGs automatically start and accelerate to 900 RPM, and assume emergency bus loads in 15 seconds upon receipt of any of the following signals:

a) Loss of Power:

LOGIC TRAIN I ACTUATED

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Sustained high Rx pressure	1051 psig	6K9 From: RE15A or 16K110A or 6K57
OR		OR
Lo-Lo Rx water level	90" above TAF	6K10 From: RE15C or 16K110C or 6K57
OR		
V-14-34 manually opened	6K57 Energized	

Reference Drawings:

JC 19529 Sh. 1
BR 3029 Sh. 2
GE 157B6397 Sh. 15
GU 3E-611-17-005 Sh. 1

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

3 **ID: 09-1 NRO3** **Points: 1.00**

Which of the following correctly states the **normal** power supply to USS 1E1?

A. R144 Line

B. Transformer SA

C. Transformer SB

D. North Yard Distribution J69361

Answer: A

Answer Explanation:

QID: 09-1 NRO1		
Question # / Answer	1	**Developer/Date: NTP**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
262001 AC Electrical Distribution K2.01 - Knowledge of electrical power supplies to the following: Off-site sources of power					3.3	3.6
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**BR 3000**		**BR 3002 sh. 4**			
Explanation	The R144 line is the normal offsite power supply to USS 1E1, and the J69361 Line is the alternate offsite power supply. Answer A is correct. All other answers are off-site power supplies, just not to USS 1E1.					
References to be provided during exam:	**None**					
Learning Objective	2621.828.0.016B LO 262-10435					

Question Source (New, Modified, Bank)	**New**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

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1:F</td><td>Comprehension or Analysis</td><td></td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Fact</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td>7</td><td>55.43</td><td></td></tr>
<tr><td colspan="4">(SRO Only)</td></tr>
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BR 3000, Rev 011, ELECTRICAL POWER SYSTEM KEY ONE LINE DIAGRAM (tif) - Imagenation

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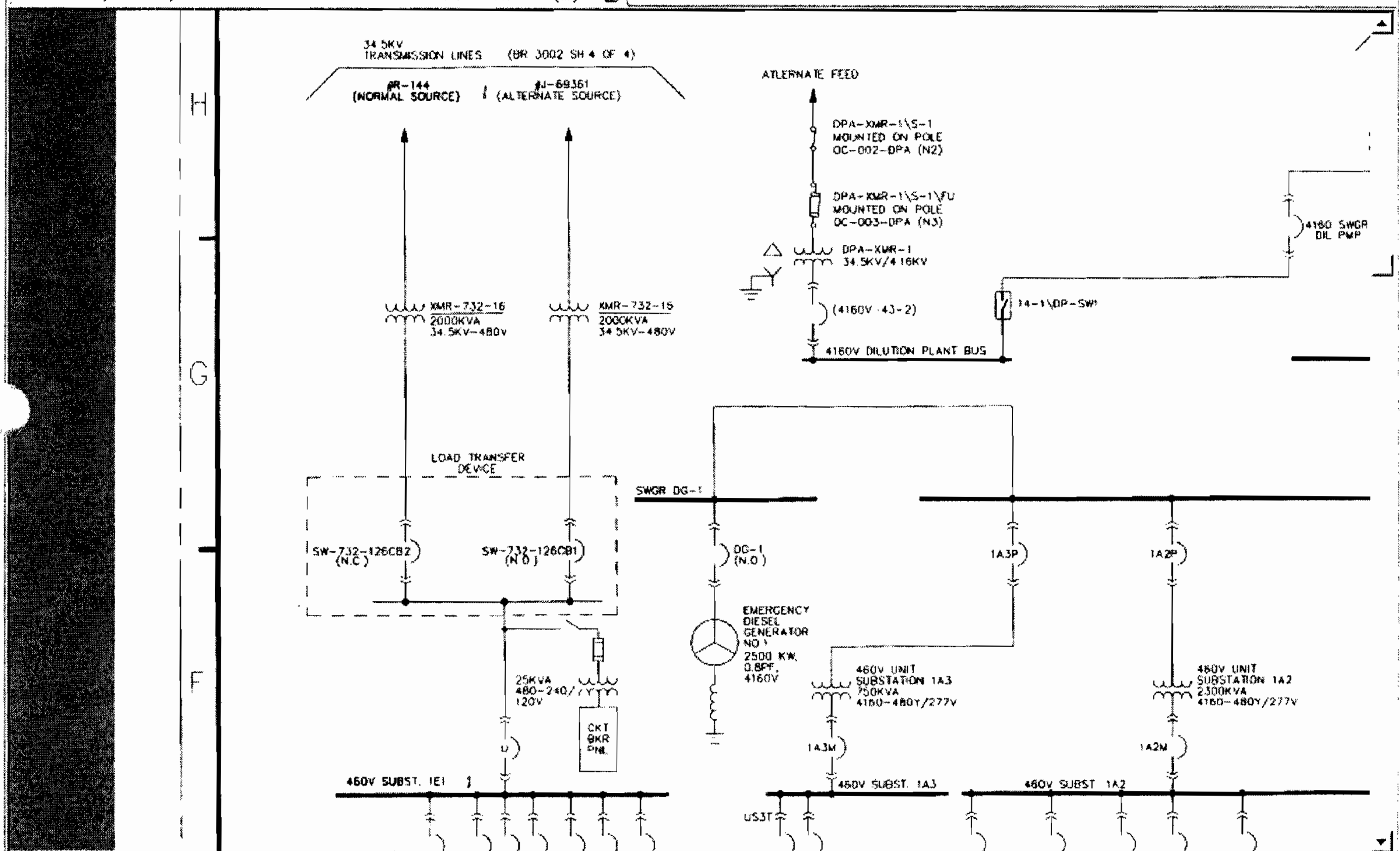
BR 3000, Rev 011, ELECTRICAL POWER SYSTEM KEY ONE LINE DIAGRAM (tif)



Coord: (3984,2714) viewing

4 **ID: 09-1 NRO4** **Points: 1.00**

Which of the following receives its normal DC power from 125 VDC Bus C?

A. 125 VDC DC-D

B. Emergency Seal Oil Pump

C. 4160V Bus 1A control power

D. 480V USS 1B1 control power

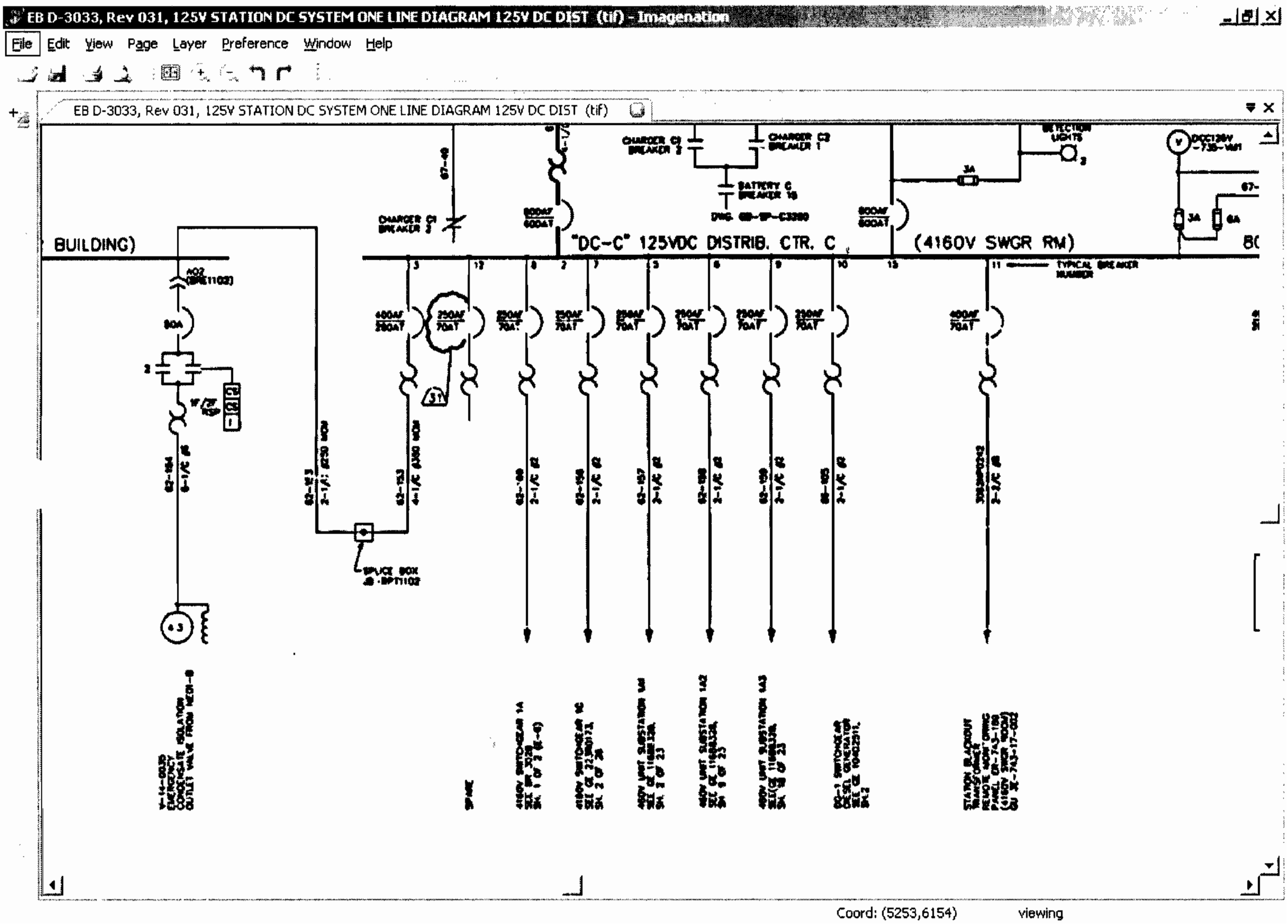
Answer: C

Answer Explanation:

QID: 09-1 NRO4		
Question # / Answer	4	Developer/Date: NTP 11/12/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
263000 DC Electrical Distribution K2.01 - Knowledge of electrical power supplies to the following: Major D.C. loads					3.1	3.4
Level	RO	Tier	2	Group	1	
General References	3028	3033				
Explanation	The 125 VDC Bus C supplies the normal power for breaker control to loads on 4160V Bus 1A. Answer C is correct. 125 VDC DC-D receives its normal DC power from DC B. Answer A is incorrect. The Emergency Seal Oil Pump receives its normal breaker control power from Bus A. Answer B is incorrect USS 1B1 receives its normal breaker control power from DC A. Answer A is incorrect. All loads listed are DC power and are plausible..					
References to be provided during exam:	None					
Learning Objective	2621.828.0.0012 LO 1106					

Question Source (New, Modified, Bank)			Modified	
Cognitive Level	Memory or Fundamental Knowledge	X 1:F	Comprehension or Analysis	
	NUREG 1021 Appendix B: Fact			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				


EB D-3033, Rev 031, 125V STATION DC SYSTEM ONE LINE DIAGRAM 125V DC DIST (tif) - Imagenation
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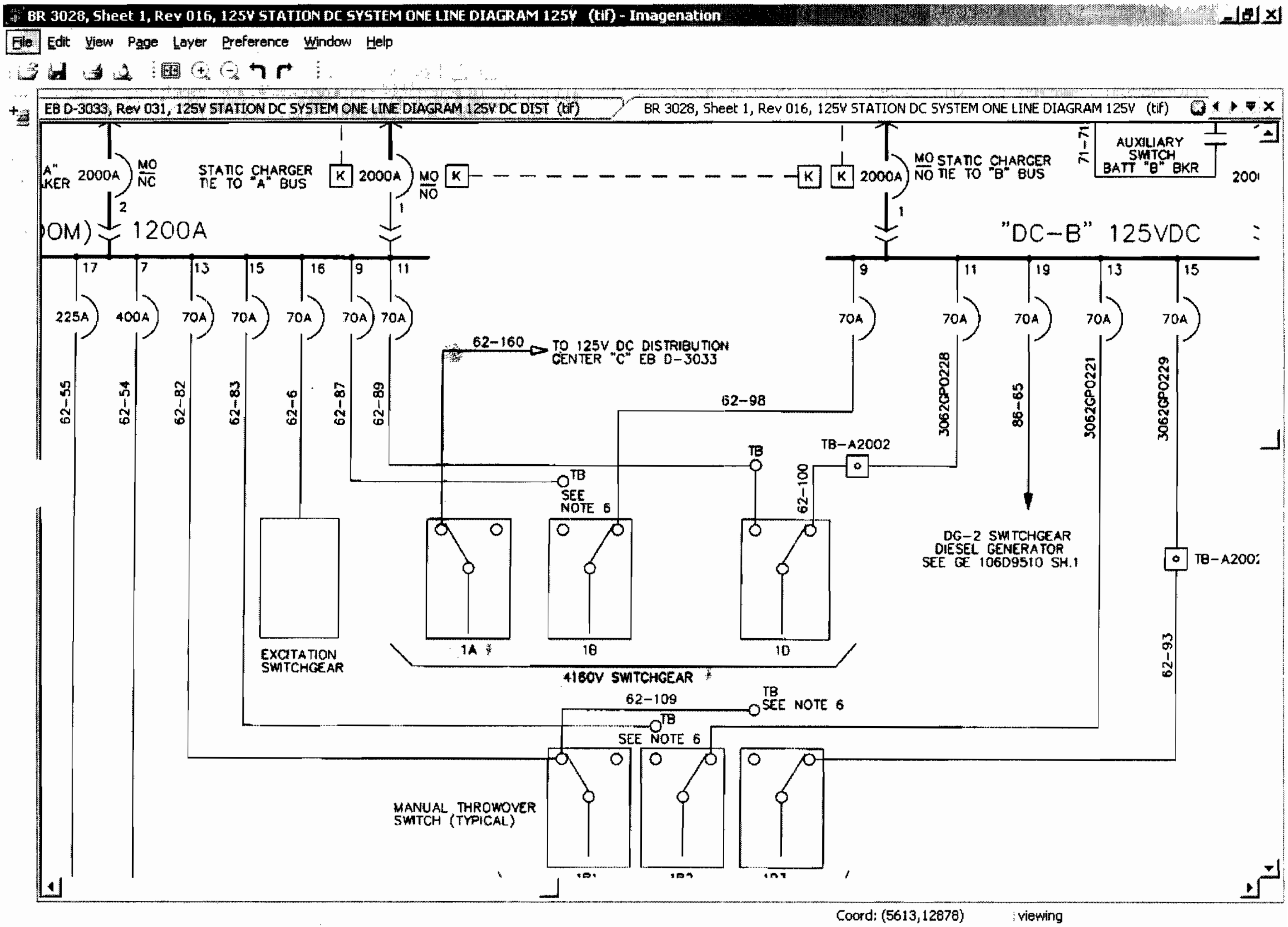
EB D-3033, Rev 031, 125V STATION DC SYSTEM ONE LINE DIAGRAM 125V DC DIST (tif)
BUILDING)
"DC-C" 125VDC DISTRIB. CTR. C
(4160V SWGR RM)
CHARGER C1 BREAKER 2
CHARGER C2 BREAKER 1
BATTERY C BREAKER 15
TYPICAL BREAKER NUMBER
SPLICE BOX
SPARE
Coord: (5253,6154)
viewing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File Edit View Page Layer Preference Window Help
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BR 3028, Sheet 1, Rev 016, 125V STATION DC SYSTEM ONE LINE DIAGRAM 125V (tif)
2000A
MO NC
STATIC CHARGER TIE TO "A" BUS
K
2000A
MO NO
K
K
K
2000A
MO NO
STATIC CHARGER TIE TO "B" BUS
71-71
AUXILIARY SWITCH BATT "B" BKR
1200A
"DC-B" 125VDC
17 7 13 15 16 9 11
9 11 19 13 15
225A 400A 70A 70A 70A 70A 70A
70A 70A 70A 70A 70A
62-55 62-54 62-82 62-83 62-6 62-87 62-89
3062GP0228 86-65 3062GP0221 3062GP0229
62-160
TO 125V DC DISTRIBUTION CENTER "C" EB D-3033
62-98
TB
TB-A2002
TB
SEE NOTE 6
62-100
DG-2 SWITCHGEAR DIESEL GENERATOR SEE GE 106D9510 SH.1
EXCITATION SWITCHGEAR
1A
1B
1D
4160V SWITCHGEAR
62-93
62-109
TB SEE NOTE 6
TB
SEE NOTE 6
MANUAL THROWOVER SWITCH (TYPICAL)
Coord: (5613,12878)
viewing

5 **ID: 09-1 NRO5** **Points: 1.00**

The plant was at rated power when the applied voltage to LPRM 20-49D was lost. (LPRM 20-49D inputs into APRM 6)

Which of the following states the impact on APRM 6 indicated reactor power and on reactor power indication provided by heat balance?

	APRM 6 Power Indication	Heat Balance Power Indication
A.	Indicates lower	Indicates lower
B.	Indicates lower	No impact
C.	No impact	Indicates lower
D.	No impact	No impact

Answer: B

Answer Explanation:

QID: 09-1 NRO5		
Question # / Answer	**5**	**Developer/Date: NTP 11/12/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
215005 APRM/LPRM K3.08 - Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: Core thermal calculations				**3.0**	**3.4**
Level	**RO**	**Tier**	**2**	**Group**	**1**
General References	**GFES**	**NF-AB-770**	**LP 2621.828.0.0029**		

Explanation	When the applied voltage is lost to the LPRM detector, it can no longer collect all the generated ion pairs and the LPRM output will go down. As this single LPRM output lowers, APRM 6 indication will also lower since the LPRM is in its normal state and not bypassed from the APRM. The heat balance on the other hand, is not affected by the number of neutron counts and will remain the same since there is no change in reactor power. Answer B is correct and answer A is incorrect. The other answers are plausible if the candidate does not understand neutron detector operation, how the APRM is affected by LPRM inputs or heat balance calculations. The APRM would show no impact if the LPRM were bypassed. Answers C & D are incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0029 215-10453		

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

ATTACHMENT 1
Typical Heat Balance Equation and Terms
Page 1 of 1

$$CTP = (Q_S + Q_{RAD} + Q_{Cuin}) - (Q_{CUout} + Q_{FW} + Q_{CRD} + Q_P)$$

CTP = core thermal power

Q_S = energy rate of steam = $\dot{m}_S \times h_S \times c$

Q_{RAD} = radiative losses (constant) may include contributions from unmonitored sources

Q_{CUin} = energy rate of RWCU inlet (out of reactor) = $\dot{m}_{CU} \times h_{CUin} \times c$

Q_{CUout} = energy rate of RWCU outlet (into the reactor) = $\dot{m}_{CU} \times h_{CUout} \times c$

Q_{FW} = energy rate of FW = $\dot{m}_{FW} \times h_{FW} \times c$

Q_{CRD} = energy rate of CRD water = $\dot{m}_{CRD} \times h_{CRD} \times c$

Q_P = recirculation pump work = $\eta_P \times \sum_{i=1}^{n} RPP_i$

$\dot{m}$ = mass flow rate (typically in units of Mlb/hr)

$\dot{m}_S = \dot{m}_{FW} + \dot{m}_{CRD}$

h = enthalpy of fluid (typically in units of BTU/lb)

c = conversion factor (typically 0.293 $MW_{th} \bullet hr \bullet lb/BTU \bullet Mlb$)

η_P = recirculation pump efficiency (typically in units of MW_{th}/MW_e)

RPP_i = recirculation pump power (typically in units of MW_e)

6 **ID: 09-1 NRO6** **Points: 1.00**

The plant is at 5% power during a startup. The steam flow signal from the Digital Feedwater Control System suddenly drifts up to 50% steam flow.

Which of the following states the impact on the Rod Worth Minimizer (RWM)?

A. The RWM will no longer enforce compliance with the control rod sequence.

B. The RWM will insert both an insert block and a withdraw block due to the RWM Fault.

C. The RWM will continue to enforce compliance with the control rod sequence until the Low Power Setpoint is reached.

D. The RWM will automatically shift to the Power Operations Mode to enforce compliance with the control rod sequence.

Answer: A

Answer Explanation:

QID: 09-1 NRO6		
Question # / Answer	**6**	**Developer/Date: NTP 11/12/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
259002 Reactor Water Level Control **K3.03 - Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following:** **Rod worth Minimizer**				**2.7**	**2.9**
Level	**RO**	**Tier**	**2**	**Group**	**1**
General References	**409**	**2621.828.0.0041**			

Explanation	The plant is starting up at 5% power. The RWM is in service in the low power mode, enforcing the control rod sequence. The RWM determines reactor power level by the amount of steam flow, which comes from the Digital Feedwater Control System. In the current low power mode, the RWM will enforce the control rod sequence up to the low power setpoint (LPSP), which is set at 35% power. When the steam flow to the RWM rises to 50%, the RWM recognizes that reactor power is above the LPSP and that it will no longer enforce compliance to the control rod sequence. Answer A is correct. The RWM does have a high power operations mode, which when activated manually, will also enforce compliance to the control rod sequence. There is no automatic swap between the RWM modes. A failure of control rod position information can result in both an insert and withdraw block. The other answers are plausible but incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0041 LO 217-10444

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

ATTACHMENT 409-2 (continued)

Low Power Mode (LPM)
Also called Startup/Shutdown Mode. Previously the only mode available. This is the default mode the RWM boots up in. In this mode, if the defined low power sequence is loaded, it is enforced below the LPSP, and monitored up to the LPAP. The LPM is disabled whenever the Power Operations Mode (POM) is active (POM started).

LPAP
System Low Power Alarm Point.

LPSP
System Low Power Set Point.

Menu
A list of program options presented to the user on the terminal.

Mispositioned Control Rod
Per Plant Procedures, a mispositioned rod is:

1. A correctly selected control rod was moved more than one notch beyond its intended position.
2. A correctly selected control rod was inadvertently moved one notch beyond its intended position and unknowingly left in this position (i.e. next rod selected, control rod evolution completed, etc.).
3. An incorrectly selected control rod was moved.

MMI
Man-machine Interface

MUX
Acronym for multiplexer device. A multiplexer is a piece of computer hardware which allows access to multiple input/ouput points from a single hardware device.

Notch
Unit by which a control rod is moved. Control rods are withdrawn 1 notch at a time, corresponding to a change in rod position by 2. For example, if a rod in position 08 is withdrawn 4 notches, its target position is 16. (A notch position is always an even number)

Off-Line
If applied to a program - a non real-time execution of a program or process; if applied to data - storage of data external to the computer memory, such as on tape or disk.

ATTACHMENT 409-2 (continued)

V. System Interactions

A. Vital Power System

1. RWM System receives power from plant computer's uninterruptible power supply (UPS).

2. If power is lost, RWM System must be bypassed to move rods when below LPSP.

B. Rod Position Indication System

1. Provides rod position data to RWM System

C. Reactor Manual Control System

1. Supplies rod selection inputs to the RWM.

2. Receives rod block output signals.

3. RWM System failure below LPSP gives error lockout. RWM must be bypassed or repaired to continue moving rods.

D. Main Steam System

1. Supplies total steam flow signal to RWM System.

2. Used to determine LPAP and LPSP, which are adjustable.

3. Loss of signal causes RWM to become active regardless of power level.

VI. System Operation

A. Normal Operation - Startup

1. During startup up to 35% power (LPSP), RWM monitors rod selection and movement.

2. Enforces adherence to operating sequence by generating error and block signals.

3. From 35% power to 40% power (LPAP), RWM monitors rod selection and movement.

4. Error signals generated, but NO blocks.

5. Above 40% power, RWM does not monitor rod selection or movement.

B. Normal Operation - Shutdown

1. When power reaches 40% (LPAP), RWM begins to monitor rod selection and movement.

2. Error signals generated, but NO blocks.

3. When power reaches 35% (LPSP), RWM still monitors rod selection and movement, and gives error and block signals to enforce adherence to the sequence.

C. Power Operations Mode

1. General Description

   a. Enforces all rod movements during power operation (>10%) as defined by Reactor Engineering.

   b. Monitors control rod movement vs. Maneuver Request Sheet sequence.

   c. Blocks rod movement whenever a rod is beyond its sequence limits.

   d. Greatly reduces the probability and consequences of control rod mispositioning events.

   e. Rods blocks are not applied until a rod is one notch beyond limits.

   f. During continuous rod withdrawal, rod will travel at least one additional notch after block is initiated.

7 **ID: 09-1 NRO7** **Points: 1.00**

The plant was at rated power when an MSIV closure event occurred. RPV pressure **peaked** at 1215 psig.

Which of the following is correct regarding the operation of the EMRVs and the RPV Safety Relief Valves (SRV) at the peak pressure?

	EMRV Operation	SRV Operation
A.	**All** EMRVs opened through actuation of a pressure switch	5 SRVs opened through actuation of a pressure switch
B.	**All** EMRVs opened from RPV pressure overcoming the valve spring pressure	4 SRVs opened through actuation of a pressure switch
C.	**All** EMRVs opened from RPV pressure overcoming the valve spring pressure	5 SRVs opened from RPV pressure overcoming the valve spring pressure
D.	**All** EMRVs opened through actuation of a pressure switch	4 SRVs opened from RPV pressure overcoming the valve spring pressure

Answer: D

Answer Explanation:

QID: 09-1 NRO7		
Question # / Answer	**7**	**Developer/Date: NTP 11/12/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

<table>
<tr><td colspan="6">239002 SRVs
K4.08 - Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Opening of the SRV from either an electrical or mechanical signal</td><td>3.6</td><td>3.7</td></tr>
<tr><td>Level</td><td>RO</td><td>Tier</td><td>2</td><td>Group</td><td colspan="3">1</td></tr>
<tr><td>General References</td><td colspan="2">420</td><td colspan="2">UFSAR 5.2.2</td><td colspan="3">729E182</td></tr>
<tr><td>Explanation</td><td colspan="7">The plant was at rated power when the MSIVs closed and RPV pressure peaked at 1215 psig.
2 EMRVs open at 1085 psig (TS value) and 3 open at 1105 psig. 4 SRVs open at 1212 psig and 5 open at 1221 psig. Therefore at an RPV pressure of 1215 psig, all EMRVs and 4 SRVs have opened.
The EMRVs open for ADS (Automatic Depressurization System) and in the Pressure Relief Mode. In the pressure relief mode, the EMRVs open to protect the RPV from an over-pressure condition. But, regardless if EMRV is opened manually, for the ADS function, or in the pressure relief mode, the EMRV solenoid must be energized to open the valve. In the pressure relief mode, a pressure switch will actuate to energize the solenoid. If there were no power available to the EMRV solenoids, the EMRVs will not open, regardless of the RPV pressure.
The SRVs on the other hand, are purely mechanical. When RPV pressure overcomes the SRV spring pressure, the SRV will open, and can open with no electrical power required.
Therefore, all EMRVs have open from actuation of the pressure switch, and 4 SRVs have opened since RPV pressure has overcome the valve spring pressure.
Answer D is correct.
The other answers are plausible if the candidate does not understand the operating mechanism of the valves or confuses the valve open setpoints. All other answers are incorrect.</td></tr>
<tr><td colspan="3">References to be provided during exam:</td><td colspan="3">None</td><td colspan="2"></td></tr>
<tr><td>Learning Objective</td><td colspan="7">2621.828.0.0005 LO 368</td></tr>
</table>

Question Source (New, Modified, Bank)	**New**

Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

FUNCTION	DEVICE	ACTION	TECH. SPEC. LIMIT	CORRECTED TECH. SPEC. LIMIT (See Note 1)	INSTRUMENT SETPOINT (See Note 2)	CORRECTED INSTRUMENT SETPOINT (See Note 3)
NOTE 1: These values are obtained by adding to the Tech. Spec. limit the associated head correction factors for each instrument and represented maximum allowable trip values. During instrument calibration and surveillance testing, observed operation outside these limits should be reported in an Incident Report.						
NOTE 2: These values represent the magnitude of the process variable at which the instruments trip. The difference between each value and its associated Tech. Spec. Limit accounts for any instrument drift or added conservatism included in the instrument setting. The Instrument Setpoint value is normally the source used for RAP Procedures.						
NOTE 3: These values are obtained by adding to the instrument setpoint the associated head correction for each instrument, as applicable and specifies the value at which the instrument is to be set during calibration.						
1. High Reactor Pressure	Safety Valves	Valves Open	4 @ 1212 psig $\pm$36 psi 5 @ 1221 psig $\pm$36 psi	4 @ 1212 psig $\pm$36 psi 5 @ 1221 psig $\pm$36 psi	4 @ 1212 psig 5 @ 1221 psig	4 @ 1212 psig 5 @ 1221 psig
	Relief NR108 Valves A B C D E	Relief Valves Open	$\leq$ 1085 psig $\leq$ 1105 psig $\leq$ 1105 psig $\leq$ 1085 psig $\leq$ 1105 psig	$\leq$ 1094.15 psig $\leq$ 1119.5 psig $\leq$ 1111.8 psig $\leq$ 1097.2 psig $\leq$ 1117.2 psig	1065 psig 1085 psig 1085 psig 1065 psig 1085 psig	1074 $\pm$ 2.5 psig 1099 $\pm$ 2.5 psig 1091 $\pm$ 2.5 psig 1077 $\pm$ 2.5 psig 1097 $\pm$ 2.5 psig
	Relief NR108 Valves A B C D E	Relief Valves Closed	NONE	NONE	1010 psig 1052 psig 1052 psig 1010 psig 1052 psig	1019 $\pm$ 2.5 psig 1066 $\pm$ 2.5 psig 1059 $\pm$ 2.5 psig 1022 $\pm$ 2.5 psig 1064 $\pm$ 2.5 psig
	RE 03 A & B C & D	Scram	$\leq$ 1060 psig	$\leq$ 1068.2 psig $\leq$ 1065.9 psig	1045 psig 1045 psig	1053 $\pm$ 3 psig 1051 $\pm$ 3 psig
	RE 15 A & B C & D	Isolation Condenser Initiation - and - Recirc Pump Trip (No Time Delay)	$\leq$ 1060 psig, with time delay $\leq$ 3 sec.	$\leq$ 1068.3 psig $\leq$ 3 sec. $\leq$ 1066.1 psig $\leq$ 3 sec.	1051 psig 1.5 $\pm$ 1 sec. 1051 psig 1.5 $\pm$ 1 sec.	1059 $\pm$ 3 psig 1.5 $\pm$ 1 sec 1057 $\pm$ 3 psig 1.5 $\pm$ 1 sec
	ID 77	Alarm	NONE	NONE	1030 psig	1030 psig
	PT-622-1018 PT-622-1019	Alternate Rod Injection	NONE	NONE	1090 $\pm$ 5 psig	1115 $\pm$ 5 psig

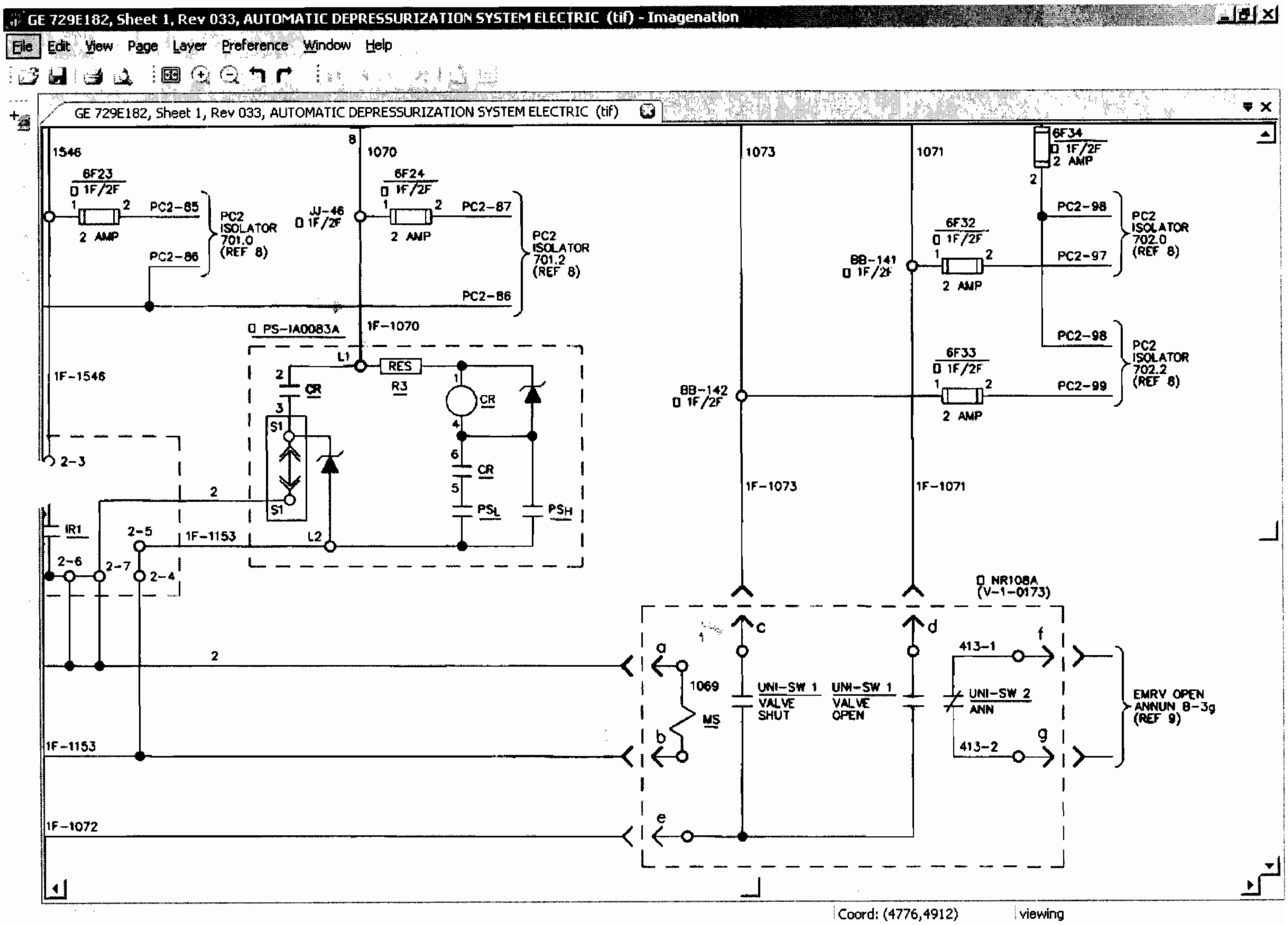
The design temperature for various system components varies according to the specific operating condition. The design temperature for the reactor vessel is based on the saturation temperature corresponding to the design pressure. Therefore, no specific system temperatures are designated as safety or operating limits.

In addition to the calculations required by the ASME Code, the vessel specification required additional stress analyses including stresses resulting from internal pressure, external and internal loadings, the effects of steady and fluctuating temperatures and loads. These analyses were conducted for regions involving changes of shape, structural discontinuities and points of concentrated loadings. The allowable stresses were those stated in the ASME Code. (See Section 5.3.)

Calculations were also conducted for cyclic conditions including normal startup and shutdown (**240**), daily reduction to 75 percent power (10,000) and weekly reduction to 50 percent power (2000) power cycling, control rod worth test (50,000), loss of feedwater heaters (80), scram (200), turbine trips (40), overpressure to 1250 psig (1), overpressure to 1375 psig (1), relief or bypass valve fails open (1). **These values of cyclic conditions were originally documented in Reference 1 of FSAR section 5.3.4 (FDSAR Amendment 16). The number of heatup and cooldown cycles were reanalyzed for reactor vessel studs and reactor vessel seal skirt due to their high fatigue usage factors and documented in Reference 10 to allow for higher number of cycles than expected in the original analysis. A review of the original analysis showed that the components with the highest fatigue usage factors are reactor vessel studs (usage factor of 0.796) and RV basin seal skirt (usage factor is 0.665). These components have the potential to exceed the allowable fatigue usage factor if the number of thermal cycles (e.g., heatup/cooldown) exceed the design assumptions. All other components have relatively low usage factors and are not expected to exceed the fatigue usage factor limit of 0.8 for the design life of 40 years. For these remaining components, the maximum usage factor is 0.1 for feedwater nozzles, providing a margin to the allowable usage factor of 0.8 by a factor of 8. Since the increase in the number of heatup and cooldown cycles is by a factor of 2 (from 120 to 240), there is sufficient margin available for the remaining components.** Table 5.2-2 compares design cycles with Bureau of Ships fatigue curves. This table highlights the wide margins of safety considered in the design of the Reactor Coolant Pressure Boundary (RCPB). To further assure the integrity of the RCPS, a comprehensive Inservice Inspection and Testing Program has been instituted.

5.2.2.2 <u>Design Evaluation</u>

Overpressure control and protection of the reactor vessel and main steam piping is provided by the 40 percent of rated steam flow Turbine Bypass System capacity, a 40 percent capacity set of the Electromatic Relief Valves (EMRVs), a set of spring loaded safety valves and the Isolation Condenser System. The bypass system is generally sufficient to relieve pressure transients in normal operating situations including full capacity turbine load rejection. The EMRVs will


GE 729E182, Sheet 1, Rev 033, AUTOMATIC DEPRESSURIZATION SYSTEM ELECTRIC (tif) - Imagenation
File Edit View Page Layer Preference Window Help
GE 729E182, Sheet 1, Rev 033, AUTOMATIC DEPRESSURIZATION SYSTEM ELECTRIC (tif)
1546
6F23
1F/2F
2 AMP
PC2-85
PC2-86
PC2 ISOLATOR 701.0 (REF 8)
8
1070
JJ-46
1F/2F
6F24
1F/2F
2 AMP
PC2-87
PC2-86
PC2 ISOLATOR 701.2 (REF 8)
1073
1071
6F34
1F/2F
2 AMP
PC2-98
PC2-97
PC2 ISOLATOR 702.0 (REF 8)
6F32
1F/2F
2 AMP
BB-141
1F/2F
PC2-98
PC2-99
PC2 ISOLATOR 702.2 (REF 8)
6F33
1F/2F
2 AMP
BB-142
1F/2F
PS-IA0083A
1F-1070
1F-1546
L1
RES
R3
CR
S1
PSL
PSH
L2
2-3
IR1
2-5
2-6
2-7
2-4
1F-1153
1F-1073
1F-1071
NR108A
(V-1-0173)
1069
MS
UNI-SW 1 VALVE SHUT
UNI-SW 1 VALVE OPEN
413-1
UNI-SW 2 ANN
413-2
EMRV OPEN ANNUN B-3g (REF 9)
1F-1153
1F-1072
Coord: (4776,4912)
viewing

Explanation	The plant is in a normal configuration at rated power. When the TBCCW System pressure drops to 79 psig and after a small time delay (10 seconds), any non-running TBCCW is signaled to start. Answer B is correct. All other pumps will start if voltage is lost to its respective 4160V emergency bus and loading by its respective EDG. RBCCW Pump 1 has no auto start on low system flow, but does if voltage is lost to USS 1A2 (fed from Bus 1C and EDG 1). When voltage is lost, there will be no RBCCW flow and EDG 1 will pickup USS 1A2 and the pump resumes flow. Answer A is incorrect but plausible. SW Pump 1-1 is independent on the breaker position of the other SW pump. SW Pump starts similar to RBCCW. Answer c is incorrect but plausible. CRD Pump B will auto start from a loss of volts to 4160V Bus 1D, not Bus 1C. Answer D is incorrect but plausible.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0048 LO 274-10443

Question Source (New, Modified, Bank)				**Modified**
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:F**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Fact			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading TB CCW	Q - 1 - f

DISCH PRESS LO	

CONFIRMATORY ACTIONS:

- **VERIFY** start of Standby TBCCW pump. [ ]

AUTOMATIC ACTIONS:

Starts standby TBCCW Water pump.

MANUAL CORRECTIVE ACTIONS:

- **VERIFY** trip of operating TBCCW pump or excessive demand on system. [ ]
- **CHECK** operation of running pumps. [ ]
- **RETURN** system to normal operation as necessary. [ ]
- IF **no** TBCCW pumps can be started,

  THEN **REFER** to Procedure ABN-20, TBCCW Failure Response. [ ]

9 **ID: 09-1 NRO9** **Points: 1.00**

The plant is shutdown and is cooling down using the Shutdown Cooling System (SDC). Present plant conditions include the following:

- Shutdown Cooling Loops A and B are in service
- RECIRC PUMP SUCTION TEMPS indicates 175 °F and lowering
- A, B, C and D Reactor Recirculation Loops are in an Idle configuration
- Recirculation Pump E is operating

Which of the following will result in the **greatest** impact on the RPV cooldown rate?

A. SDC Loop A DISCHARGE V-17-55 indicates 0%.

B. Shutdown Cooling Loop B senses 3 psig suction pressure.

C. Reactor Recirculation Pump E trips and the loop is placed in an Idle configuration.

D. Shutdown Cooling System Inlet Isolation Valve V-17-19 indicates red and green lights on.

Answer: D

Answer Explanation:

QID: 09-1 NRO9		
Question # / Answer	**9**	**Developer/Date: NTP 11/14/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
205000 Shutdown Cooling K5.02 - Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : Valve operation					**2.8**	**2.9**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**BR E1129 GE 148F711**		**GE 157B6350 sh. 157A**		**RAP-C2d, -C3d 305**	

Explanation	The plant is shutdown and is cooling down with 2 SDC loops. If the SDC inlet isolation valve V-17-19 comes off the full open position, all SDC pumps are interlocked to trip. Thus, with the red and green lights on, SDC pumps will trip resulting in a total loss of SDC and there will be no cooling of the RPV coolant and the cooldown rate will cease. Answer D is correct. When the SDC loop A discharge valve indicates 0%, the valve is closed. This SDC loop is no longer providing any cooling. But with the SDC loop B still in service, there is still about 50% of the initial amount of SDC cooling the RPV. Answer A is incorrect. When SDC loop B senses 3 psig suction pressure, the SDC B pump will trip. This results in 50% of the initial amount of SDC cooling the RPV. Answer B is incorrect. The SDC System takes a suction on Recirculation Pump E suction piping and discharges to the discharge piping of Recirculation Pump E. When the recirculation Pump trips, SDC could short cycle through the tripped pump instead of through the RPV, and would result in less cooling and a reduced RPV cooldown. But with the recirculation pump in an idle condition, its discharge valve is closed, which is the an allowable configuration and short cycling is not a concern. Answer C is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0045 LO 205-10445	

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

PUMP A TRIP

MANUAL CORRECTIVE ACTIONS: (continued from Page 2 of 3)

- ❑ IF there is a loss of control power,

  THEN **RESTORE** 125 VDC power in accordance with Procedure ABN-55, DC Bus C and Panel/MCC Failures. [ ]

CAUSES:		SETPOINTS:	ACTUATING DEVICES:
Breaker trip		Breaker tripped	Relay 30T
Trip Function:	Drive motor overload	340 amps	Solid State Trip Device
	Low suction pressure	4 psig, TD=1.5s	PSL-43A through TDR-214-001 relay
	Inlet water temperature high	350°F, TD=1.5s	TSH-42A through TDR-214-001 relay
	V-17-19 SDC Inlet Isolation Valve closed	Not fully open	SW 20IC through 6x16

Reference Drawings:

BR E1129
GE 148F711
GU 3E-611-17-005 Sh. 1

PUMP B TRIP

MANUAL CORRECTIVE ACTIONS: (continued from Page 2 of 3)

- ☐ IF there is a loss of control power,

  THEN **RESTORE** 125 VDC power in accordance with Procedure ABN-54, DC Bus B and Panel/MCC Failures. [ ]

CAUSES:		SETPOINTS:	ACTUATING DEVICES:
Breaker trip		Breaker tripped	Relay 30T
Trip Function:	Drive motor overload	340 amps	Solid State Trip Device
	Low suction pressure	4 psig, TD=1.5s	PSL-43B through TDR-214-002 relay
	Inlet water temperature high	350°F, TD=1.5s	TSH-42B through TDR-214-002 relay
	V-17-19 SDC Inlet Isolation Valve closed	Not fully open	SW 20IC through 6x16

Reference Drawings:

BR E1130
GE 148F711
GU 3E-611-17-005 Sh. 1

10 **ID: 09-1 NRO10** **Points: 1.00**

The plant is at rated power. The Operator is starting EDG 1 for Peaking Operation IAW procedure 341, Emergency Diesel Generator Operation, at the EDG Switchgear.

IAW procedure 341, which of the following actions and/or observations are required to parallel the EDG once it has been started?

	EDG 1 Output Voltage	EDG 1 Output Frequency	Synchroscope Position When EDG Output Breaker is Manually Closed by the Operator
A.	Slightly **higher** than line voltage	Slightly **higher** than line frequency	11 o'clock position
B.	Slightly **lower** than line voltage	Slightly **lower** than line frequency	12 o'clock position
C.	Slightly **higher** than line voltage	Slightly **lower** than line frequency	11 o'clock position
D.	Slightly **lower** than line voltage	Slightly **higher** than line frequency	12 o'clock position

Answer: A

Answer Explanation:

QID: 09-1 NRO10		
Question # / Answer	10	Developer/Date: NTP 11/14/09

Knowledge and Ability Reference Information		
K&A	Importance Rating	
	RO	SRO

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

264000 EDGs K5.05 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : Paralleling A.C. power sources					**3.4**	**3.4**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	341					
Explanation	The plant is at rated power with an operator starting EDG 1 at the local switchgear IAW procedure 341. To synchronize the EDG to the line, EDG output voltage must be slightly higher than line voltage, the synchroscope must be moving slowly in the fast direction. This means that the EDG output frequency is slightly higher than line frequency. The EDG output breaker will then be closed when the synchroscope hand reaches the 11 o'clock position. Answer A is correct. All other answers are plausible if the candidate does not know the relationships of voltage/frequency while paralleling two generator sources, but are incorrect in some fashion.					
References to be provided during exam:	**None**					
Learning Objective	2621.828.0.0013 LO 264-10446					

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

7.3.1.5

> **NOTE**
>
> After the engine has idled for 90 seconds from initial start, placing its MODE SELECTOR SWITCH in RUN will then increase engine speed to 900 RPM.

PLACE EDG-1 MODE SELECTOR SWITCH in the RUN position. [ ]

7.3.1.6 **VERIFY** engine speed increases. [ ]

7.3.1.7 **PLACE** EDG-1 MODE SELECTOR SWITCH in the EXC position. [ ]

7.3.1.8 **COMPARE** EDG-1 output voltage with line voltage using the KILOVOLT METER selecting any GEN or BUS position on the VOLTAGE/FREQUENCY SELECTOR SWITCH. [ ]

7.3.1.9

> **NOTE**
>
> EDG output voltage should be slightly higher than line voltage so that the machine will have lagging VARS when it is parallel with the system.

ADJUST EDG-1 output voltage to be slightly higher than line voltage using the VOLTAGE CONTROL SWITCH. [ ]

7.3.1.10 **SYNCHRONIZE** EDG-1 with the bus as follows:

1. **PLACE** EDG-1 SYNCHROSCOPE ON/OFF SWITCH in the ON position with the synchroscope key. [ ]
2. **OPERATE** EDG-1 GOVERNOR CONTROL SWITCH so that the synchroscope hand is moving slowly in the fast direction, and the synchronizing lights are pulsing slowly in unison. [ ]
3. **VERIFY** EDG-1 output voltage is slightly higher than line voltage. [ ]
4. **PLACE** EDG-1 VOLTAGE/FREQUENCY SELECTOR SWITCH in one of the following positions; GEN 1-2, GEN 2-3, or GEN 3-1. [ ]

5.

NOTE
The generator is synchronized to the system when the synchroscope hand is at the twelve o'clock position. Synchronizing lights should be out at this position.

CONFIRM the synchroscope hand is moving slowly in the fast direction. [ ]

6. WHEN EDG-1 synchroscope hand reaches the eleven o'clock position,

THEN **PLACE** EDG-1 BREAKER CONTROL SWITCH in the CLOSED position. [ ]

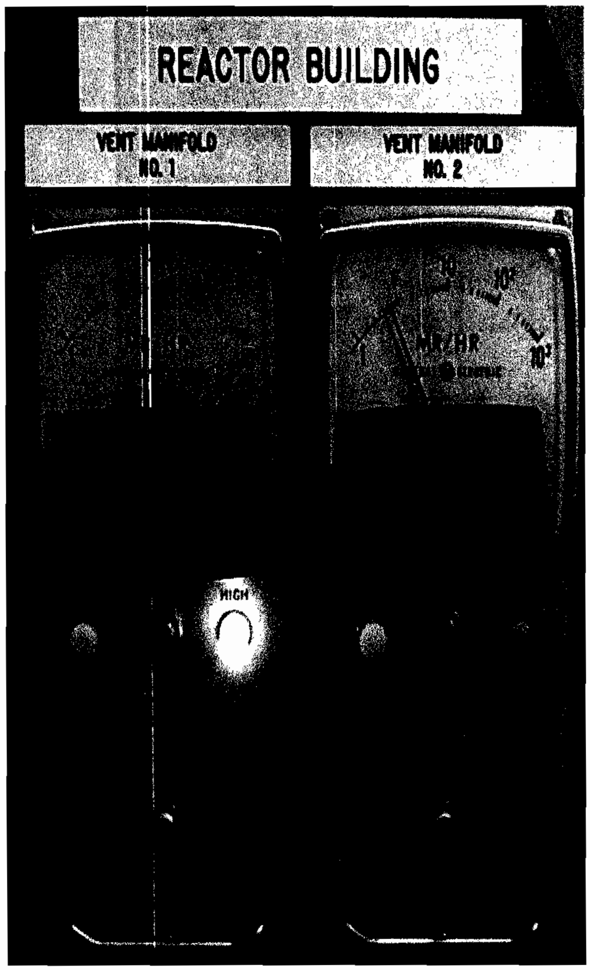
11 **ID: 09-1 NRO11** **Points: 1.00**

The plant is at rated power. The STANDBY GAS SELECT switch is in position SYS 2.

An event occurred which resulted in the following annunciator alarming:

- RADIATION MONITORS PROCESS RX BLDG - VENT HI

The Operator observed the following indications:

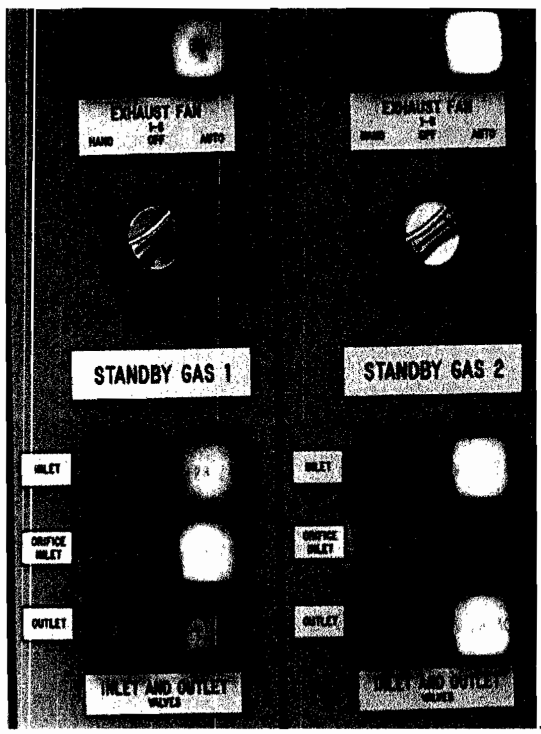


Which of the following shows the **first** response of the Standby Gas Treatment System (SGTS), if any?

C.

EXHAUST FAN
EXHAUST FAN
STANDBY GAS 1
STANDBY GAS 2
INLET
INLET
OUTLET
OUTLET
INLET AND OUTLET
INLET AND OUTLET

D.



EXHAUST FAN
EXHAUST FAN
STANDBY GAS 1
STANDBY GAS 2
INLET
INLET
OUTLET
OUTLET
INLET AND OUTLET
INLET AND OUTLET

Answer: A

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer Explanation:

QID: 09-1 NRO11		
Question # / Answer	**11**	**Developer/Date: NTP 11/14/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
261000 SGTS **K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM :** **Process radiation monitoring**					**2.9**	**3.1**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**330**	**RAP-10F1f**				
Explanation	The plant is at rated power when an event occurs. The provided indications shows that RB Vent Rad Monitor #1 has failed upscale. The logic to auto start SGTS is 1 rad monitor above the setpoint (9 mr/hr). Therefore, the SGTS has received an initiation signal. When the auto start signal is received, both SGTS fans start and SGTS valves for both fans re-align, as shown in Answer A. Answer A is correct. Since SGTS 1 is the selected system, then after flow has been established in system 1, SGTS System 2 will shutdown and re-align as shown in Answer C. This re-alignment occurs in about 2-3 minutes after the start signal is received. Since the question asks for the correct indications after only 1 minute, then the system alignment still looks as in Answer A. Answer C is incorrect. Answer B shows SGTS in its standby condition. If the candidate does not understand the logic to start SGTS, this could be selected. Answer B is incorrect. Answer D shows a start of both SGTS fans but System has a broken shaft but remains in service and the orifice valve for System 2 has closed since it is the only fan drawing a suction. Answer D is incorrect.					
References to be provided during exam:	**None**					

Learning Objective	2621.828.0.0042 LO 261-10445

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B: Predict an event of outcome.			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

5.3 Automatic Operation of the Standby Gas Treatment System

5.3.1

> **CAUTION**
>
> When normal Reactor Building ventilation system is **not** operating and the water evaporator is running, there is a potential for radioactive contamination due to water condensing out in the ductwork and dripping through the duct joints located in the CRD rebuild room.

IF SGTS has initiated,

THEN **CONFIRM** truck ventilation hose and water evaporator secured.

5.3.2 IF initiation is from an automatic startup signal,

THEN **VERIFY** the following events occur (Panel 11R):

5.3.2.1 **VERIFY** both SGTS Exhaust Fans EF-1-8 and EF-1-9 start. [ ]

5.3.2.2 **VERIFY** the following SGTS valves open:

- System I inlet V-28-23 [ ]
- System I orifice V-28-24 [ ]
- System I outlet V-28-26 [ ]
- System II inlet V-28-27 [ ]
- System II orifice V-28-28 [ ]
- System II outlet V-28-30 [ ]

5.3.2.3 **VERIFY** the following Reactor Building fans trip:

- Supply Fan SF-1-12 [ ]
- Supply Fan SF-1-13 [ ]
- Supply Fan SF-1-14 [ ]
- Exhaust Fan EF-1-5 [ ]
- Exhaust Fan EF-1-6 (only if lined up to the Reactor Building). [ ]

(Blue)

E

VENT
HI

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** high radiation level on redundant indicators. (Panel 2R) [ ]

AUTOMATIC ACTIONS:

Reactor Building isolation and initiation of the standby gas treatment system.

MANUAL CORRECTIVE ACTIONS:

- ❑ **CONFIRM** high radiation condition. [ ]
- ❑ IF confirmed,

  THEN **ENTER** EOP EMG-3200.11, Secondary Containment Control. [ ]

  - ❑ **NOTIFY** the Shift Manager. [ ]

NOTE
This alarm indicates that a parameter has exceeded or has the potential to exceed an Emergency Action Level (EAL).

- ❑ **REFER** to Procedure EP-AA-1010, Exelon Nuclear Radiological Emergency Plan Annex for Oyster Creek Station to determine EAL classification. [ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 2)

(Blue)

VENT
HI

E

MANUAL CORRECTIVE ACTIONS: (continued from Page 1 of 2)

- As directed, **INITIATE** the Reactor Building evacuation alarm. [ ]
- **ANNOUNCE** using the paging system the evacuation requirements of the building. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
NOTE A loss of power to or a failure of Power Supplies RN-37 will result in the Automatic action described on this procedure.	9 mr/hr	RN04A1 VIA RN07A1 VIA PART OF RN25 TRIP AUX UNIT RN04A2 VIA RN07A2 VIA PART OF RN25 TRIP AUX UNIT
Upscale trip of the Reactor Building ventilation radiation monitor.		Reference Drawings: GU 3E-611-17-003 GE 706E841

12 **ID: 09-1 NRO12** **Points: 1.00**

The plant was at rated power when the following annunciator alarmed:

- 1B2 MN BREAKER TRIP

Which of the following states the impact on the Vital AC System?

A. CIP-3 auto transfers to USS 1A2.

B. VACP-1 auto transfers to VMCC 1A2.

C. PAIPP 2 must be manually transferred to VMCC 1A2.

D. MCC 1AB2 must be manually transferred to VMCC 1A2.

Answer: B

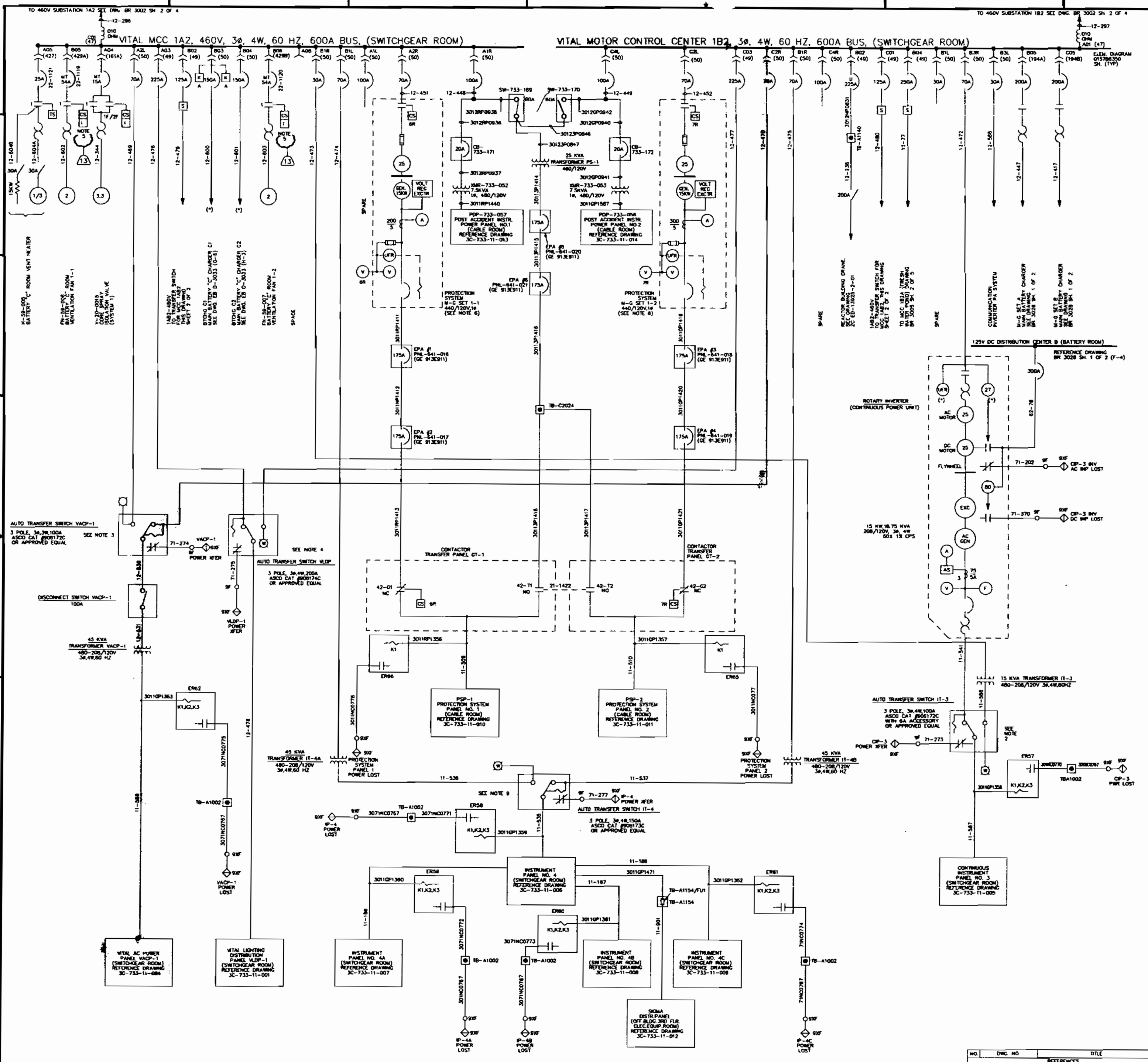
Answer Explanation:

QID: 09-1 NRO12		
Question # / Answer	12	Developer/Date: NTP 11/16/09

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
262002 UPS (AC/DC) K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : AC electrical power						2.7	2.9
Level	RO	Tier	2	Group	1		
General References	ABN-48		RAP-9XF3c		3013 sh. 1 3002 sh. 2		

Explanation	The plant was at rated power when indications of the loss of USS 1B2 appeared. USS 1B2 powers VMCC 1B2. VMCC normally powers MCC 1AB2, the rotary inverter which supplies CIP-3, and VACP-1. When power is lost to VACP-1, an auto transfer switch operates to power VACP-1 from VMCC 1A2. Answer B is correct. The inverter to CIP-3 will transfer to a DC power supply (and VMCC as a backup). Answer A is incorrect. USS 1B2 power PAIPP 2, and it power supply is not transferable to another supply. Answer c is incorrect. When power is lost to MCC 1AB2, an auto transfer switch operates to power MCC 1AB2 from VMCC 1A2. Answer D is correct.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0056 LO 262-10445

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				



REV	ZONE	REVISIONS DESCRIPTION
13		ECR OC 07-00542

NOTES:

1. SEE DWG GU 3E-700-10-001 FOR SYMBOLS.
2. AUTO TRANSFER SWITCH IT-3. FOR DETAIL ELEMENTARY DIAGRAM AND CONNECTION DIAGRAM INCLUDING INDICATING LIGHTS SEE JCP & L DWG 19642 SH 1.
3. AUTO TRANSFER SWITCH VACP-1 FOR DETAIL ELEMENTARY DIAGRAM AND CONNECTION DIAGRAM INCLUDING INDICATING LIGHTS SEE JCP & L DWG. 19643 SH. 1A
4. AUTO TRANSFER SWITCH VLDP & 1AB2 FOR DETAIL ELEMENTARY DIAGRAM AND CONNECTION DIAGRAM INCLUDING INDICATING LIGHTS SEE JCP & L DWG 19643 & 19643 SH. 1B
5. LOCATED AT HVAC CONTROL PANEL OUTSIDE BATTERY "C" ROOM.
6. FOR REACTOR PROTECTION SYSTEM M-G SET 1-1 & 1-2 CONTROL SEE DRAWING GE 913E911.
7. ELEVATIONS AND TABULATIONS FOR VITAL MCC 1A2 AND MCC 1B2 SEE SHEET 2 OF 2.
8. (*) WILL MARK ALL EQUIPMENT NOT VERIFIED DUE TO HIGH RADIATION OR PLANT CONDITIONS.
9. AUTO TRANSFER SWITCH IT-4, FOR DETAIL ELEMENTARY DIAGRAM AND CONNECTION DIAGRAM INCLUDING INDICATING LIGHTS SEE JCP & L DWG 19642 SH.2.
10. TED 136070ML-TEDUV1;REFER TO VM-OC-5089

NO.	DWG NO	TITLE
REFERENCES		

CAD FILE NAME: BR 3013 S1 R13

REVISION 0 OF THIS DRAWING WAS REDRAWN FROM BR 3013, REV 49.
REVISION 7 OF THIS DRAWING WAS TRANSLATED TO AUTOCAD FROM CADAM FILE V3E.733,11,0013,001-,0701

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B.VERGARA 07/16/07

AmerGen

AC VITAL POWER SYSTEM
ONE LINE DIAGRAM
VITAL MCC 1A2 AND 1B2

OYSTER CREEK | DWG. NO. BR 3013 | SH. 1 OF 2 | REV. 13

SCALE : NONE

WA. NO.

DWG NO. BR 3013 SH. 1 OF 2

STICK#1

4.11.1 **REVIEW** Attachment ABN-48-2, A Battery Load List, for loads on A Bus [ ]

AND **TAKE** appropriate compensatory measures (i.e. monitor annunciator panels and proper operation of V-2-17 and feed pump minimum flow valves.) [ ]

4.11.2 IF A Bus voltage drops to 110 VDC,

THEN **DIRECT** electricians to take individual cell voltages [ ]

AND **MONITOR** for cell reversal (ser 3-99). [ ]

4.11.3 IF A bus voltage drops to 105 VDC,

THEN **CONSIDER** removing A Battery from service. [ ]

4.12 **CONFIRM** VMCC 1AB2, VACP-1 and IP-4 have transferred to their alternate power supplies by verifying the following alarms are received:

- MCC-1AB2 PWR XFER (9XF-2-c) [ ]
- VACP-1 PWR XFER (9XF-3-c) [ ]
- IP-4 PWR XFER (9XF-7-c) [ ]

4.13

NOTE

Steps 4.13 through 4.32 may be performed in any order.

NOTE

Due to loss of power to DWEDT controller, the DWEDT will overflow to the 1-8 sump and cause a rise in the unidentified leak rate.

MONITOR the following plant parameters using SPDS:

- Containment pressure [ ]
- Bulk Drywell temperature [ ]

Group Heading	
VITAL POWER AC XFERS	9XF-3-c

VACP-1 **PWR XFER**	

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** operation of the automatic transfer switch. [ ]

AUTOMATIC ACTIONS:

480 volt supply power to transformer VACP-1 transferred from Vital MCC 1B2 to Vital MCC 1A2.

MANUAL CORRECTIVE ACTIONS:

- ❑ **REFER** to ABN-51, Loss of VMCC 1B2 and ABN-58, Instrument Power Failures. [ ]
- ❑ **REFER** to Procedure 339, Vital Power System. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Automatic transfer switch for 120 Volt Vital A C Power Panel VACP-1 operation.	Dropout @ 70% v. Pickup @ 90% v.	Relays 1V, 2V, 3V Reference Drawings: GU 3E-611-17-022 BR 3013, Sh. 1 JC 19643

13 **ID: 09-1 NRO13** **Points: 1.00**

The plant was starting up after an outage. RECIRC PUMP SUCTION TEMPS indicates 350 °F.

The Operator notes a rising Drywell pressure and reports Drywell Unidentified leakage of 50 GPM and steady.

5 minutes later, the following annunciators alarmed **simultaneously**:

- LKOUT RELAY 86/S1A TRIP
- LKOUT RELAY 86/S1B TRIP
- DW PRESS HI-HI RV 46 A/B
- DW PRESS HI-HI RV 46 C/D

Which of the following states the EDG 1 loading sequence and the impact on RPV water level 60 seconds **after** EDG 1 output breaker closes? (**Note**: the pumps are listed in order of starting sequence)

	EDG 1 Loading Sequence	Impact on RPV Water Level
A.	Core Spray Main Pump C Core Spray Booster Pump C CRD Pump NC08B	Level is rising
B.	Core Spray Main Pump A Core Spray Booster Pump A CRD Pump NC08A Service Water Pump 1-1	Level is steady
C.	Core Spray Main Pump A Core Spray Booster Pump A CRD Pump NC08A	Level is rising
D.	CRD Pump NC08B Core Spray Main Pump C Core Spray Booster Pump C	Level is steady

Answer: C

Answer Explanation:

QID: 09-1 NRO13		
Question # / Answer	13	Developer/Date: NTP 11/16/09

Knowledge and Ability Reference Information						
K&A					Importance Rating RO	Importance Rating SRO
209001 LPCS A1.07 - Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Emergency generator loading					3.0	3.1
Level	RO	Tier	2	Group	1	
General References	341		201 Att. 7		RAP-S1b RAP-C1f RAP-B2e	
Explanation	The plant was starting up with coolant temperature at 350 °F (120 psig). Drywell pressure began to rise from a leak of 50 gpm (steady). 5 minutes later, Drywell pressure reached the LOCA signal setpoint, simultaneous with a loss of offsite power. Both EGDs start and close onto their respective emergency bus (in < 10 seconds). With a LOOP/LOCA, EDG1 loads as follows: lighting/instrument bus, Core spray main Pump NZ01A, Core Spray Booster Pump NZ03A, and CRD Pump A. For a non-LOOP LOCA, Service Water and RBCCW also start. Since RPV pressure is below the opening setpoint for the Core Spray Parallel Isolation Valves (305 psig), these valves open. With 2 sets of Core Spray injecting into the RPV (plus the already running Feed/Condensate System), RPV water level will rise. Answer C is correct. Answer A lists loads on EDG 2 and is incorrect. Answer B states a component on EDG 1 which would auto start from just a LOOP (and no LOCA). It also has the incorrect RPV water level impact. Answer B is incorrect. Answer D has loads which are powered from EDG 1, but sequence is incorrect, as the water level impact. Answer D is incorrect.					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

References to be provided during exam:	**Attachment 201-7**	
Learning Objective	2621.828.0.0013 LO 264-10444	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**x 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Saturation Temperature (degrees F) from Pressure (PSIG) in one pound increments
To read temperature at 108 psig go to row 10x and column 8.
Note: 0 psig = 14.7 psia

PSIG \ x	0	1	2	3	4	5	6	7	8	9
0x	212.0	215.4	218.5	221.5	224.4	227.2	229.8	232.3	234.8	237.1
1x	239.4	241.6	243.7	245.8	247.8	249.8	251.7	253.5	255.3	257.1
2x	258.8	260.5	262.1	263.7	265.3	266.8	268.3	269.8	271.2	272.6
3x	274.0	275.4	276.7	278.1	279.4	280.6	281.9	283.1	284.4	285.6
4x	286.7	287.9	289.0	290.2	291.3	292.4	293.5	294.5	295.6	296.6
5x	297.7	298.7	299.7	300.7	301.7	302.6	303.6	304.6	305.5	306.4
6x	307.3	308.2	309.1	310.0	310.9	311.8	312.6	313.5	314.3	315.2
7x	316.0	316.8	317.7	318.5	319.3	320.1	320.8	321.6	322.4	323.2
8x	323.9	324.7	325.4	326.1	326.9	327.6	328.3	329.0	329.8	330.5
9x	331.2	331.9	332.6	333.2	333.9	334.6	335.3	335.9	336.6	337.2
10x	337.9	338.5	339.2	339.8	340.5	341.1	341.7	342.3	342.9	343.6
11x	344.2	344.8	345.4	346.0	346.6	347.2	347.7	348.3	348.9	349.5
12x	350.1	350.6	351.2	351.8	352.3	352.9	353.4	354.0	354.5	355.1
13x	355.6	356.2	356.7	357.2	357.8	358.3	358.8	359.3	359.8	360.4
14x	360.9	361.4	361.9	362.4	362.9	363.4	363.9	364.4	364.9	365.4
15x	365.9	366.4	366.8	367.3	367.8	368.3	368.8	369.2	369.7	370.2
16x	370.6	371.1	371.6	372.0	372.5	372.9	373.4	373.9	374.3	374.8
17x	375.2	375.6	376.1	376.5	377.0	377.4	377.8	378.3	378.7	379.1
18x	379.6	380.0	380.4	380.8	381.3	381.7	382.1	382.5	382.9	383.3
19x	383.8	384.2	384.6	385.0	385.4	385.8	386.2	386.6	387.0	387.4
20x	387.8	388.2	388.6	389.0	389.4	389.8	390.2	390.5	390.9	391.3
21x	391.7	392.1	392.5	392.8	393.2	393.6	394.0	394.3	394.7	395.1
22x	395.5	395.8	396.2	396.6	396.9	397.3	397.7	398.0	398.4	398.7
23x	399.1	399.4	399.8	400.2	400.5	400.9	401.2	401.6	401.9	402.3
24x	402.6	403.0	403.3	403.7	404.0	404.3	404.7	405.0	405.4	405.7
25x	406.0	406.4	406.7	407.0	407.4	407.7	408.0	408.4	408.7	409.0
26x	409.4	409.7	410.0	410.3	410.7	411.0	411.3	411.6	411.9	412.3
27x	412.6	412.9	413.2	413.5	413.8	414.2	414.5	414.8	415.1	415.4
28x	415.7	416.0	416.3	416.6	417.0	417.3	417.6	417.9	418.2	418.5
29x	418.8	419.1	419.4	419.7	420.0	420.3	420.6	420.9	421.2	421.5
30x	421.8	422.1	422.3	422.6	422.9	423.2	423.5	423.8	424.1	424.4
31x	424.7	425.0	425.2	425.5	425.8	426.1	426.4	426.7	426.9	427.2
32x	427.5	427.8	428.1	428.4	428.6	428.9	429.2	429.5	429.7	430.0
33x	430.3	430.6	430.8	431.1	431.4	431.7	431.9	432.2	432.5	432.7
34x	433.0	433.3	433.5	433.8	434.1	434.3	434.6	434.9	435.1	435.4
35x	435.7	435.9	436.2	436.4	436.7	437.0	437.2	437.5	437.7	438.0
36x	438.3	438.5	438.8	439.0	439.3	439.5	439.8	440.0	440.3	440.6
37x	440.8	441.1	441.3	441.6	441.8	442.1	442.3	442.6	442.8	443.1
38x	443.3	443.5	443.8	444.0	444.3	444.5	444.8	445.0	445.3	445.5
39x	445.7	446.0	446.2	446.5	446.7	447.0	447.2	447.4	447.7	447.9
40x	448.2	448.4	448.6	448.9	449.1	449.3	449.6	449.8	450.0	450.3
41x	450.5	450.7	451.0	451.2	451.4	451.7	451.9	452.1	452.4	452.6
42x	452.8	453.1	453.3	453.5	453.7	454.0	454.2	454.4	454.6	454.9
43x	455.1	455.3	455.5	455.8	456.0	456.2	456.4	456.7	456.9	457.1
44x	457.3	457.6	457.8	458.0	458.2	458.4	458.7	458.9	459.1	459.3
45x	459.5	459.8	460.0	460.2	460.4	460.6	460.8	461.1	461.3	461.5
46x	461.7	461.9	462.1	462.3	462.6	462.8	463.0	463.2	463.4	463.6
47x	463.8	464.0	464.2	464.5	464.7	464.9	465.1	465.3	465.5	465.7
48x	465.9	466.1	466.3	466.5	466.7	467.0	467.2	467.4	467.6	467.8
49x	468.0	468.2	468.4	468.6	468.8	469.0	469.2	469.4	469.6	469.8
50x	470.0	470.2	470.4	470.6	470.8	471.0	471.2	471.4	471.6	471.8
51x	472.0	472.2	472.4	472.6	472.8	473.0	473.2	473.4	473.6	473.8
52x	474.0	474.2	474.4	474.6	474.8	475.0	475.2	475.3	475.5	475.7
53x	475.9	476.1	476.3	476.5	476.7	476.9	477.1	477.3	477.5	477.7
54x	477.8	478.0	478.2	478.4	478.6	478.8	479.0	479.2	479.4	479.5

ASME Properties of Saturated Steam from 1967 IPC Formulation for Industrial Use and other IAPWS releases

4.5.2

> NOTE
>
> Time indications are relative to Diesel Generator breaker closure.

Loss of Power Concurrent With a LOCA-BOTH Diesel Generators Available ,

TIME (SECONDS)	EDG 1 LOAD	EDG 2 LOAD	Load Rating (KW)
0	Lighting, Instrumentation and Controls Ventilation, Security, Battery Chargers, Miscellaneous small motors and transformers losses	Lighting, Instrumentation and Controls, Ventilation, Security, Battery Chargers, Miscellaneous small motors and transformer losses	13
	These loads start as needed	These loads start as needed	
	Core Spray Main Pump NZ01A	Core Spray Main Pump NZ01B	474
5	Core Spray Booster Pump NZ03A	Core Spray Booster Pump NZ03B	247
60	CRD Pump NC08A	CRD Pump NC08B	200

4.5.2.1

> CAUTION
>
> Manual equipment starts prior to the completion of automatic load sequencing may lead to an EDG trip.

WHEN the automatic loading sequence is complete and the diesel generator load has stabilized,

THEN **PERFORM** the following:

1. **ADD** Turbine Bldg. and Reactor Bldg. Loads manually to EDG as required. [ ]
2. **REFER** to Attachment 341-7, Emergency Diesel Generator Load List. [ ]

Group Heading STARTUP XFMRS S1A	S-1-b

LKOUT RELAY 86/S1A TRIP	

CONFIRMATORY ACTIONS:

❑ **VERIFY** trip of Startup Breaker S1A.	[ ]
❑ **VERIFY** trip of Bank 5 OCB.	[ ]
❑ **VERIFY** Dilution Plant power lost. If fed from S1A	[ ]

AUTOMATIC ACTIONS:

Trips 4160V Breaker S1A.

If Bus 1A was being supplied from the Startup Transformer, DG-1 will fast-start and pick up Bus 1C.

MANUAL CORRECTIVE ACTIONS:

❑ IF	DG-1 fast started and picked up Bus 1C,	
THEN	**MONITOR** Diesel Generator load to prevent overloading.	[ ]
❑ IF	Transformer S1A (Bk. 5) was supplying Bus 1A,	
THEN	**REFER** to the following procedures:	
	❑ ABN-1, Reactor Scram	[ ]
	❑ ABN-2, Recirculation System Failues	[ ]
	❑ ABN-17, Feedwater System Abnormal Conditions	[ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 2)

Group Heading TORUS/DRYWELL	C-1-f

DW PRESS HI-HI RV 46 A/B	

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** high drywell pressure. (Panel 1F/2F and 12XR) [ ]
- ❑ **VERIFY** start of core spray pumps and diesel generators. [ ]

AUTOMATIC ACTIONS:

Starts core spray pumps and diesel generators.

MANUAL CORRECTIVE ACTIONS:

- ❑ **ENTER** EMG-3200.01A, RPV Control - No ATWS [ ]

  OR

  EMG-3200.01B, RPV Control with ATWS [ ]

  AND

  EMG-3200.02 Primary Containment Control. [ ]

> NOTE
>
> This alarm indicates that a parameter has exceeded or has the potential to exceed an Emergency Action Level (EAL).

- ❑ **REFER** to Procedure EP-AA-1010, Exelon Nuclear Radiological Emergency Plan Annex for Oyster Creek Station to determine EAL classification. [ ]

**SYSTEM 1
FLOW
PERMISSIVE**

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
NOTE: This alarm will activate only if all three conditions are met indicating that core spray should be injecting into the depressurized Rx core.		
Booster pump differential pressure greater than 30.5/28.5 psid (RV40A/RV40C)	30.5 psid 28.5 psid	DPS RV40A or DPS RV40C
AND		AND
Core Spray pump discharge pressure greater than 100 psig	105 psig	PS RV29A or RV29C
AND		AND
Reactor pressure less than 305 psig	305 psig	RE17A or RE17B

Reference Drawings:

NU 5060E6003 Sh. 1 & 3
GU 3E-611-17-004 Sh. 1

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

14 **ID: 09-1 NRO14** **Points: 1.00**

The plant was at rated power when the following annunciator alarmed:

- SV/EMRV NOT CLOSED

The Operator reported the following indications:

- EMRV NR108A **and** NR108B indicate in the VALVE OPEN REGION
- **All** AUTO DEPRESS VALVE green lights are lit
- EMRV DISCH NR108A, B & E (Panel 1F/2F) indicates 328 °F
- MWe has dropped by 30 MWe

Which of the following states the status of the EMRVs?

A. **Both** EMRVs are full open, as evidenced by the magnitude of the drop in electrical output.

B. **Either** or **both** EMRVs are open, but must be confirmed at the EMRV Tailpipe Temperature Indicator (RB 23').

C. **Both** EMRVs are open as evidenced by the VALVE OPEN REGION and Panel 1F/2F EMRV DISCH temperature.

D. **Either** or **both** EMRVs are open, but must be confirmed by the red HI-ALARM lights in the Acoustic Monitoring Panel (15R).

Answer: B

Answer Explanation:

QID: 09-1 NRO14		
Question # / Answer	14	Developer/Date: NTP 11/16/09

Knowledge and Ability Reference Information		
K&A	Importance Rating	
	RO	SRO

<table>
<tr><td colspan="5">218000 ADS
A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: ADS valve tail pipe temperatures</td><td>3.4</td><td>3.6</td></tr>
<tr><td>Level</td><td>RO</td><td>Tier</td><td>2</td><td>Group</td><td colspan="2">1</td></tr>
<tr><td>General References</td><td colspan="2">RAP-B4g</td><td colspan="2">ABN-40</td><td colspan="2">UFSAR Table 5.1-1</td></tr>
<tr><td>Explanation</td><td colspan="6">The plant is at power when indications show that one or both EMRVs have opened (by other than solenoid energization). When a single EMRV opens, the VALVE OPEN REGION will be indicated for that EMRV and for another EMRV located physically close-by.
EMRVs A, B & E have a common monitored discharge tailpipe, which will show an elevated temperature for any of the 3 EMRVs being open. Thus, from the Panel 1F/2F tailpipe temperature indications, it will not show conclusively which valve is open. Individual EMRV tailpipe indicators on RB23' will show elevated temperatures for the individually open valves. Thus one or both EMRVs are open, but must be determined from the individual tailpipe indications locally. Answer B is correct.
The number of open EMRVs cannot be ascertained from the drop in electrical power. A single EMRV full open, passes about 600,000 lb/hr of steam at 1250 psig, which represents about 8% power, or about 50 MWe. If both valves were open 50%, this would provide about the same load drop for a single valve open 100%. Thus load drop alone cannot be used to tell how many EMRVs are open. Answer A is incorrect.
As stated, EMRVs A & B have a common tailpipe discharge with a common temperature indication and cannot be used to tell which is open. Also, as stated it is common for 1 open EMRV to influence the acoustic monitor of a near-by EMRV such that both indicate in the open region. Answer C is incorrect.
Under the conditions provided, the Hi-ALARM lights for EMRVs A & B will be lit, but cannot be used to determine which valves are open. Answer D is incorrect.</td></tr>
<tr><td colspan="2">References to be provided during exam:</td><td colspan="3">None</td><td colspan="2"></td></tr>
</table>

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Learning Objective	2621.828.0.0005 LO 7319

Question Source (New, Modified, Bank)				New
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describing or recognizing relationships			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

2.3 Other indications

- Red VALVE OPEN indication light is illuminated if the solenoid is energized
- Acoustic monitoring system indications
- EMRV discharge temperature indications
- Lowering RPV pressure
- Drop in generator load (MWe)
- Rising Torus temperature
- Indicated steam flow less than indicated feed flow
- EMRV Tailpipe Temperature Indicator (RB 23' elevation).

3.0 IMMEDIATE OPERATOR ACTIONS

None

4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 **VERIFY** the EMRV condition by observing the following, as practical, given plant conditions:

- IF the solenoid is energized,
  THEN Red VALVE OPEN indication light is illuminated [ ]
- Acoustic monitoring system indications [ ]
- Rising EMRV discharge temperature indications [ ]
- Lowering RPV pressure [ ]
- Drop in generator load (MWe) [ ]
- Rising Torus temperature [ ]
- Indicated steam flow less than indicated feed flow [ ]
- Rising EMRV tailpipe temperature indication (RB 23' elevation). [ ]

15 **ID: 09-1 NRO15** **Points: 1.00**

The plant was starting up after an outage. Present plant conditions include the following:

- REACTOR MODE SELECTOR switch is in STARTUP
- **No** ½ scrams or rodblocks currently exist
- **All** IRMs Recorders are on the 0-125% scale in Range 2
- Control rod withdrawals have been on-hold for the last 20 minutes

Several annunciators then alarmed. The Operator reported that **all** RPS Group Solenoids indicate energized.

The IRM indications 20 minutes ago, **and** the IRM indications now **after** the annunciators were received are shown below:

<u>20 minutes ago:</u>

IRM	Individual IRM Indicating Lights LIT (Panel 4F)	IRM Recorder Reading
11	None	8
12	None	9
13	HI-HI HIGH DN SCL OR INOP	90
14	None	10

<u>After the annunciators were received:</u>

IRM	Individual IRM Indicating Lights LIT (Panel 4F)	IRM Recorder Reading
11	DN SCL OR INOP	6
12	None	8
13	HI-HI HIGH DN SCL OR INOP	125
14	None	9

Which of the following states the plant response from this event and the required Operator action?

	Plant Response	Operator Action
A.	Rodblock from IRM 11 INOP	Bypass IRM 11
B.	Rodblock and 1/2 scram from IRM 13 INOP	Bypass IRM 13
C.	Rodblock and 1/2 scram from IRM 13 upscale	Range up on IRM 13
D.	Rodblock from IRM 11 downscale	Range down on IRM 11

Answer: D

Answer Explanation:

QID: 09-1 NRO15		
Question # / Answer	**15**	**Developer/Date: NTP 11/17/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
215003 IRM **A2.04 - Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Up scale or down scale trips**						**3.7**	**3.8**
Level	**RO**	**Tier**	**2**	**Group**	**1**		
General References	**RAP-G4e**	**RAP-H7a**	**RAP-H7a**				

Explanation	The plant is starting up after an outage and IRM 11-14 indications are provided both from 20 minutes ago and after the annunciator was received. A comparison of the 2 sets of indications shows that IRM11, 12 & 14 each went down slightly. IRM 11 picked up the downscale or inop light. IRM 14 went up to fill scale with no change in indicating lights. IRM 11 has the DN SCL or INOP light lit. A 7% on the 0-125% scale would account for this light being lit (setpoint for downscale is <7% on the 0-125% scale). No other indications are provided which show that IRM 11 is INOP. Also, an INOP IRM would produce a 1/2 scram and the RPS 1 group solenoids would be de-energized. Therefore, IRM 11 is reading below the downscale setpoint, and this would impose a control rod block (except when the IRM is on Range 1). Thus, IRM 11 is downscale causing a rodblock. A downscale IRM will cause the DN SCL OR INOP light to be lit. To correct this, the Operator must range down on the IRM. Answer D is correct. An INOP IRM 11 would cause a 1/2 scram which de-energizes the RPS 1 scram solenoids. Since RPS is normal, no 1/2 scram exists and thus IRM 11 cannot be INOP. The normal corrective action for an INOP neutron monitor is to bypass the monitor. Answer A is incorrect. The normal plant response from an upscale or inop IRM (in STARTUP mode) is both a rodblock and a 1/2 scram. But, it also shows that the HI-HI, HIGH, & DNSCL OR INOP lights are lit on IRM 13. The only way these can be lit is if IRM 13 were bypassed. With the IRM bypassed, it cannot generate any protective functions even though its count reading changes. Answers B & C are incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0029 LO 215-10444	

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**5**	**55.43**	

	(SRO Only)
	Time to Complete: 1-2 minutes

IRM DNSCL

CONFIRMATORY ACTIONS:

- **VERIFY** downscale level. (Panel 4F) [ ]

AUTOMATIC ACTIONS:

Rod withdrawal block when operating in the REFUEL or STARTUP modes except in Range 1.

MANUAL CORRECTIVE ACTIONS:

- **CHECK** that power level is consistent with IRM range switch position. [ ]
- **PLACE** associated range switch to the next lower range position to maintain IRM reading between 25% and 75% of full scale. [ ]
- **CHECK** that selector switch on the IRM cabinet is not in one of the zero positions. [ ]
- **CHECK** for loss of power or component failure. [ ]
- **CONFIRM** all IRM's fully inserted by observing ALL IN light is lit in the REFUEL or STARTUP modes [ ]
- IF the number of IRM channels or instruments per trip systems becomes less than Tech Spec 3.1 requirements, [ ]

  THEN **SHUTDOWN** Reactor in accordance with Procedure 203, Plant Shutdown. [ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 3)

IRM
DNSCL

MANUAL CORRECTIVE ACTIONS: (continued from Page 1 of 3)

❑	IF	more than one IRM channel per trip system becomes inoperable when the reactor mode switch is in STARTUP and the required conditions are met,	
	THEN	**PLACE** mode switch in RUN.	[ ]
❑	IF	the mode switch cannot be placed in RUN,	
	THEN	**PERFORM** the following:	
		1) **PLACE** the Administrative Switch in the Rod Block position. (Panel 4F)	[ ]
		2) **SHUTDOWN** the reactor in accordance with Procedure 203, Plant Shutdown.	[ ]
		3) **USE** SRMs and operable IRMs to monitor Reactor power.	[ ]
❑	IF	all IRM indication is lost in the STARTUP mode,	
	THEN	manually **SCRAM** the reactor IAW ABN-1, Reactor Scram.	[ ]
❑	IF	more than one IRM channel per trip system becomes inoperable when the reactor mode switch is in RUN,	
	THEN	**ATTEMPT** repairs as soon as possible.	[ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 3 of 3)

IRM DNSCL

MANUAL CORRECTIVE ACTIONS: (continued from Page 2 of 3)

- ❑ IF alarms are spurious

  OR

  alarm conditions clear,

  THEN **RESET** the sealed in alarms on the associated IRM Drawers as follows:

  - Momentarily **POSITION** Reset Switch to RESET and **RELEASE.** [ ]

- ❑ WHEN all sealed in alarms have been reset in the associated IRM Drawers,

  THEN **VERIFY** annunciator window clears. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Level less than 7% on the 125% scale or less than 2.2% on the 40% scale, channel component failure, or loss of power to channel.	7% on 125% scale; 2.2% on 40% scale	RH06A and RH06B
		Reference Drawings: GE 706E812, Sh. 9,10,11, 12,13,14,15, & 16 GU 3E-611-17-009 Sh. 1

ROD BLOCK

CAUSES:		SETPOINTS:	ACTUATING DEVICES:
IRM/APRM:	IRM level greater than 106/125 scale with APRM level less than 2% and mode switch in STARTUP or REFUEL.		
APRM Downscale:	APRM level less than 2/150 scale with mode switch in RUN.		
IRM Downscale:	IRM level less than 5/125 scale except in Range 1.		
SRM High:	SRM level greater than 1 x 10^5 and mode switch in STARTUP or REFUEL (below IRM Range 8).		
Timer Malfunction:	Failure of timer switch during rod out sequence.		
APRM High:	APRM level greater than (.9 x 10^{-6}) w + 60.1 with a Maximum Value of 115%		
IRM Detector Position:	IRM detector not full in with mode switch in STARTUP or REFUEL.		

(Continued on Page 14 of 14)

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

16 **ID: 09-1 NRO16** **Points: 1.00**

The plant is at rated power with the following air system indications:

- Air Compressor 1 is running in Lead
- Air Compressor 2 is in standby in Lag

Which of the following states the impact on Air Compressor 1 and Air Compressor 2 if **all** closed cooling water was lost to Air Compressor 1?

	Air Compressor 1	**Air Compressor 2**
A.	Will remain running	Remains in Standby
B.	Will auto trip from a high system air temperature signal	Will auto start on a low system air pressure signal
C.	Will auto trip from a high system air temperature signal	Will auto start immediately
D.	Will auto trip from a low cooling water pressure signal	Will auto start on a low system air pressure signal

Answer: B

Answer Explanation:

QID: 09-1 NRO16		
Question # / Answer	16	Developer/Date: NTP 11/17/09

Knowledge and Ability Reference Information		
K&A	Importance Rating	
	RO	SRO

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

300000 Instrument Air A3.02 - Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: Air temperature					**2.9**	**2.7**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**RAP-M5a**	**334**				
Explanation	The plant is at rated power with the lead air compressor running and the lag air compressor in a standby condition. Air system pressure is normal. A loss of closed cooling water occurs to air compressor 1. The lead compressor (#1) will auto trip from a high air temperature signal, and the lead compressor will auto start when system air pressure lowers to the auto start setpoint. Answer B is correct. Answer A is incorrect since air compressor 1 trips and air compressor will auto start. Answer C is incorrect since the lag compressor will not start immediately since air pressure is above the auto start setpoint (95 psig). Answer D is incorrect since there is no compressor trip signal from low cooling water pressure.					
References to be provided during exam:	**None**					
Learning Objective	2621.828.0.0043 LO 279-10444					

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solving a problem with knowledge and its meaning			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

COMPR 1 TRIP	

SETPOINTS

ALARMS (Trips)	Parameter Setpoint
Inlet Restriction	1st Stage inlet pressure <13.3 psi vacuum unloaded, or >psig loaded.
High I/C Press	> 39 psi **AND** 1ST Stage Disch Temp is > 410°F **OR** Unit is unloaded **AND** 2nd Stage Inlet Press is > 5 psi.
High 2nd Stage Press	2nd Stage Disch Press >140 psi.
High Line Air Press	Package Disch Press >140 psi.
Low Brg Oil Press	Bearing Oil Press <34 psig for 2 seconds and the unit is running.
High 1st Stage Temp	1st Stage Disch Temp >440 deg F.
High I/C Air Temp	2nd Stage Inlet Temperature >140 deg F.
High 2nd Stage Temp	2nd Stage Disch Temp >486 deg. F.
High Brg Oil Temp	Bearing Oil Temp >170 deg. F.
Starter Fault 1SL, Starter Fault 2SI	Starter coils energized and aux. contacts fail to close **OR** Starter coil is deenergized and aux. contact fails to open.
Main Motor Overload	Motor Overload Relay Contacts open.
Fan Motor Overload	Fan Motor Overload Relay contacts open.
Remote Stop Failure	The Remote Stop Button remains open and either start button (switch) is pressed.
Remote Start Failure	If unit is started from the remote start switch and the start contacts stay closed for 7 seconds after the unit starts.

Setpoints (contined on Page 3 of 3)

7.3 Procedure - Normal Operation

NOTE

1. Lead air compressor operation is as follows
   - Lead air compressor unloads at about 120 psig.
   - Lead air compressor loads at about 105 psig.
2. If #1 or #2 Air Compressor are lined up for Lag operation, then operation is as follows:
   - Lag air compressor unloads at 110 psig.
   - Lag air compressor auto starts at 95 psig, runs unloaded for 10 seconds and then loads.
   - If Lag air compressor runs for 10 minutes, then the Lag air compressor will shutdown and return to "Stopped in Auto Restart" mode status.
3. If #3 Air Compressor is the Lag air compressor, then operation is as follows:
   - Compressor unloads at 105 psig.
   - Compressor starts at 90 psig.
   - Compressor loads at 85 psig.
4. Air System Alarms are as follows:
   - At 95 psig the RCVR 1 PRESS LO (M-3-a)
   - At 85 psig RCVR 2/INST AIR PRESS LO (M-3-b) and RCVR 3 PRESS LO (M-3-c)
   - At 80 psig the RCVR 2/INST AIR PRESS LO (M-3-b)

7.3.1 NOTE

Hard Card (Attachment 334-11) applies to this section.

BLOWDOWN air system components as follows:

7.3.1.1 NOTE

Step 7.3.1.2 does **not** apply to V-6S-322 since water flow cannot be observed. The valve should be cycled and remain open for 3 to 5 seconds prior to reclosing.

17 **ID: 09-1 NRO17** **Points: 1.00**

The plant was at rated power when an event occurred. The Operator reports that RPV water level is 80" and rising slowly.

One minute later, the following annunciator alarmed:

- COND B FLOW HI POSSIBLE RUPTURE

With **no** Operator action, which of the following states the final positions of the Isolation Condenser System Valves (Steam and Condensate Return) and associated Vent Valves?

	IC A	IC B
A.	System A Valves open Vent Valves open	System B Valves closed Vent Valves closed
B.	System A Valves open Vent Valves closed	System B Valves closed Vent Valves closed
C.	System A Valves open Vent Valves open	System B Valves open Vent Valves closed
D.	System A Valves open Vent Valves closed	System B Valves closed Vent Valves open

Answer: B

Answer Explanation:

QID: 09-1 NRO17		
Question # / Answer	17	**Developer/Date: NTP 12/11/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

207000 Isolation (Emergency) Condenser A3.03 - Ability to monitor automatic operations of the ISOLATION (EMERGENCY) CONDENSER including: Reactor water level					**3.5**	**3.7**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**EMG-SP1**		**307**		**RAP-C3b RAP-C1a**	
Explanation	The plant was at rated power when an event resulted in an RPV water level of 80". As water level lowers from its normal 155", both Isolation Condensers will auto initiate at 90" (1 condensate valve in each loop goes open to initiate the condenses, and the normally open vent valves go closed). So far, all System valves are open and all vent valves go closed. When the Possible Rupture B comes in, this will close all System valves in System B only, and the vent valves remain closed. Thus, System A valves are open with their associated vent valves closed. System B valves are closed with the vent valves closed. Answer B is correct. The other answers provide various positions. If the candidate does not know the initiation signals and their impact on the system/vent valves, or does not know how the Possible Rupture annunciator impacts the Isolation Condensers, than all other answers are plausible.					
References to be provided during exam:		**None**				
Learning Objective	2621.828.0.0023 LO 2030					

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

AmerGen An Exelon Company	OYSTER CREEK GENERATING STATION PROCEDURE	Number **EMG-SP1**
Title	**SUPPORT PROCEDURE 1** **CONFIRMATION OF AUTOMATIC INITIATIONS AND ISOLATIONS**	Revision No. 0

SYSTEM		OPERATING DETAILS
Cleanup System Isolation	IF	Any of the following conditions exist: • RPV water level at or below **86 in.** and **not** bypassed • Drywell pressure at or above **3.0 psig** and **not** bypassed • RWCU HELB Alarms
	THEN	**CONFIRM** closed the following Cleanup Isolation valves: (Panel 3F/11F) V-16-1 [ ] V-16-14 [ ] V-16-2 [ ] V-16-61 [ ]
Shutdown Cooling System Isolation	IF	Any of the following conditions exist: • RPV water level at or below **86 in.** • Drywell pressure at or above **3.0 psig**
	THEN	**CONFIRM** closed the following SDC Isolation Valves: (Panel 11F) V-17-54 [ ] V-17-19 [ ]
Isolation Condenser Initiation	IF	Any of the following conditions exist or have occurred: • RPV water level at or below **86 in.** • Reactor pressure at or above **1050 psig.**
	THEN	**CONFIRM** that both Isolation Condensers did initiate. (ICs may have been removed from service by Pressure Control Leg.) [ ]

OVER

5.3.3

CAUTION

If reactor water level exceeds 180" TAF, initiation of an Isolation Condenser can cause water hammer to occur.

IF Reactor water level is greater than 180 in. TAF,

THEN **OPERATE** Isolation Condensers IAW Section 6.0 of this procedure until water level is below 180 in. TAF or until Reactor is completely depressurized. [ ]

5.3.4 **CONFIRM** V-14-34, Emergency Condenser NE01A Condensate Return Valve and V-14-35 ISO Cond 'B' Condensate Return Valve open automatically by observing valve indicating lights. (Panel 1F/2F) [ ]

5.3.5 **CONFIRM** the following Emergency Condenser NE01A(B) High Point Vent Valves close automatically:

- V-14-1 [ ]
- V-14-19 [ ]
- V-14-5 [ ]
- V-14-20 [ ]

5.3.6 **RESPOND** to multiple Recirc Pump trip IAW ABN-2, Recirculation System Failures. [ ]

5.3.7

NOTE

The Rad Pro Guidelines for performing Isolation Condenser surveys after initiation include:

- 75' and 95' EL Reactor Building surveys
- Smears outside on flat surfaces
- Water and soil samples under the discharge nozzles
- Air samples (including tritium) downwind

NOTIFY Rad Pro that an Isolation Condenser was placed in service so that the program for air, dose rate and soil sampling can be implemented. [ ]

COND B FLOW HI POSSIBLE RUPTURE

CONFIRMATORY ACTIONS:

- **VERIFY** Closed System B Isolation Valves. [ ]

Check for indication of pipe break:

- Annunciator C-8-b, COND AREA TEMP HI alarmed. [ ]
- Rise in area temperatures.
  (Panel 10R) [ ]
- **CHECK** for level changes.
  (Panel 2F) [ ]
- **CHECK** for shell temperature rise on TR IG02.
  (Panel 2F) [ ]

Check for indication of tube leak:

- **CHECK** for level changes.
  (Panel 2F) [ ]
- **CHECK** for shell temperature rise on TR IG02.
  (Panel 2F) [ ]

COND B FLOW HI POSSIBLE RUPTURE

AUTOMATIC ACTIONS:

Closes Isolation Condenser System B Valves:

- V-14-32, Iso Cond 'B' Steam Inlet Valve
- V-14-33, Steam Inlet Valve to 'B' Emergency Condenser
- V-14-35, Iso Cond 'B' Condensate Return Valve
- V-14-37, Isolation Valve Emergency Condenser NE01B

MANUAL CORRECTIVE ACTIONS:

NOTE

This alarm indicates that a parameter has exceeded or has the potential to exceed an Emergency Action Level (EAL).

❑ **REFER** to Procedure EP-AA-1010, Exelon Nuclear Radiological Emergency Plan Annex to determine EAL classification. [ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 3 of 4)

18 **ID: 09-1 NRO18** **Points: 1.00**

The plant was at rated power when the Operator reported the following observation:

- The Standby Liquid Control (SLC) System 1 continuity meter indicates 0 amps

Investigation revealed that the SLC Pumps were **not** impacted and that the squib valves remained physically intact.

An event then occurred which required SLC to be injected. The Operator placed the STANDBY LIQUID CONTROL keylock switch to the FIRE SYS 1 position.

Which of the following states the correct current panel indications and the next required Operator action?

	Indications	Required Action
A.	• SLC System 1 PUMP ON light is **lit** • Reactor power indicates lowering	• Verify PUMP DISCH PRESS greater than RPV pressure
B.	• STDBY LIC CNTRL - FLOW ON annunciator is **on** • Reactor power indicates lowering	• Verify RWCU isolated
C.	• SLC System 1 PUMP ON light is **lit** • Reactor power indicates stable	• Initiate SLC System 2
D.	• STDBY LIC CNTRL - FLOW ON annunciator is **on** • Reactor power indicates stable	• Initiate SLC System 2

Answer: C

Answer Explanation:

QID: 09-1 NRO18		
Question # / Answer	18	Developer/Date: NTP 12/12/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
211000 SLC A2.02 - 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: Failure of explosive valve to fire					3.6	3.9
Level	RO	Tier	2	Group	1	
General References	157B6350 sh. 188	RAP-G2b		EMG-SP22		

Explanation	The plant was at rated power when the SLC System 1 continuity meter indicated 0 amps. This indicates a loss of power to SLC System 1 explosive valve. When the SLC keylock is placed in FIRE SYS 1, this would normally start SLC Pump 1 and fire the associated explosive valve. In the case in the question, the pump will still start but the explosive valve will not fire and open. The SLC system explosive valve remains in its normal standby state when the keylock is placed in position 1. Because the pump starts, its indicated pressure will indicate it normal pressure of being > RPV pressure. Since the explosive valve will not open and the System 2 explosive valve remains closed, there will be no change in reactor power from this attempted SLC injection. Also, the FLOW ON annunciator will not be on since there is no SLC flow. IAW SP-22, if proper SLC indications are not received, then initiate SLC System 2. Answer C is correct. Answer A & B are incorrect since reactor power is not lowering. Answer D is incorrect since the Flow On annunciator is not lit.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0046 LO 211-10445

Question Source (New, Modified, Bank)				**Modified**
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: predict an event or outcome			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

SQUIB VALVE OPEN

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** Standby Liquid Control PUMP ON and SQUIBS lights Lit. (Panel 4F). [ ]

> NOTE
>
> Continuity meters normally read 60% - 100%.
> (3-5 ma)

- ❑ **VERIFY** SQUIB continuity meters 14MR1/14MR2 indicate loss of continuity/control circuit power supply. (Behind Panel 4F) [ ]

- ❑ **CHECK** for pump breaker trip. [ ]

LOCKED IN ALARM COMPENSATORY ACTION:

❑ IF this alarm is locked in due to maintenance,

THEN **PERFORM** the following compensatory actions:

- **VERIFY** that associated system lights (for operable system) (SQUIBS and PUMP ON) are **not** Lit. (Panel 4F) [ ]

- **VERIFY** that the respective ammeter 14MR1 or 14MR2 (for operable system) indicates 60-100% (3-5 ma). (Rear of Panel 4F) [ ]

SQUIB VALVE OPEN

AUTOMATIC ACTIONS:

> **NOTE**
>
> Reactor Cleanup System will trip and isolate if Liquid Poison flow is >15 gpm.

NONE

MANUAL CORRECTIVE ACTIONS:

- ☐ In the event of inadvertent injection,
  - **SCRAM** the reactor in accordance with ABN-1, Reactor Scram. [ ]
  - **FOLLOW** actions defined in ABN-5, Inadvertent SLC Initiation. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Actuation of either Standby Liquid Control Squib Valve, NP05A or NP05B.	Valve NP05A or NP05B open	14MR1 or 14MR2
OR		
Loss of continuity/control circuit power supply in the system.	2 mA	14MR1 or 14MR2

Reference Drawings:

GE 157B6350 Sh. 188
GU 3E-611-17-009 Sh. 1

GE 157B6350 ...et 188, Rev 015, LIQUID POISION SYSTEM ELECTRIC ELEMENTA (tif) ...genation

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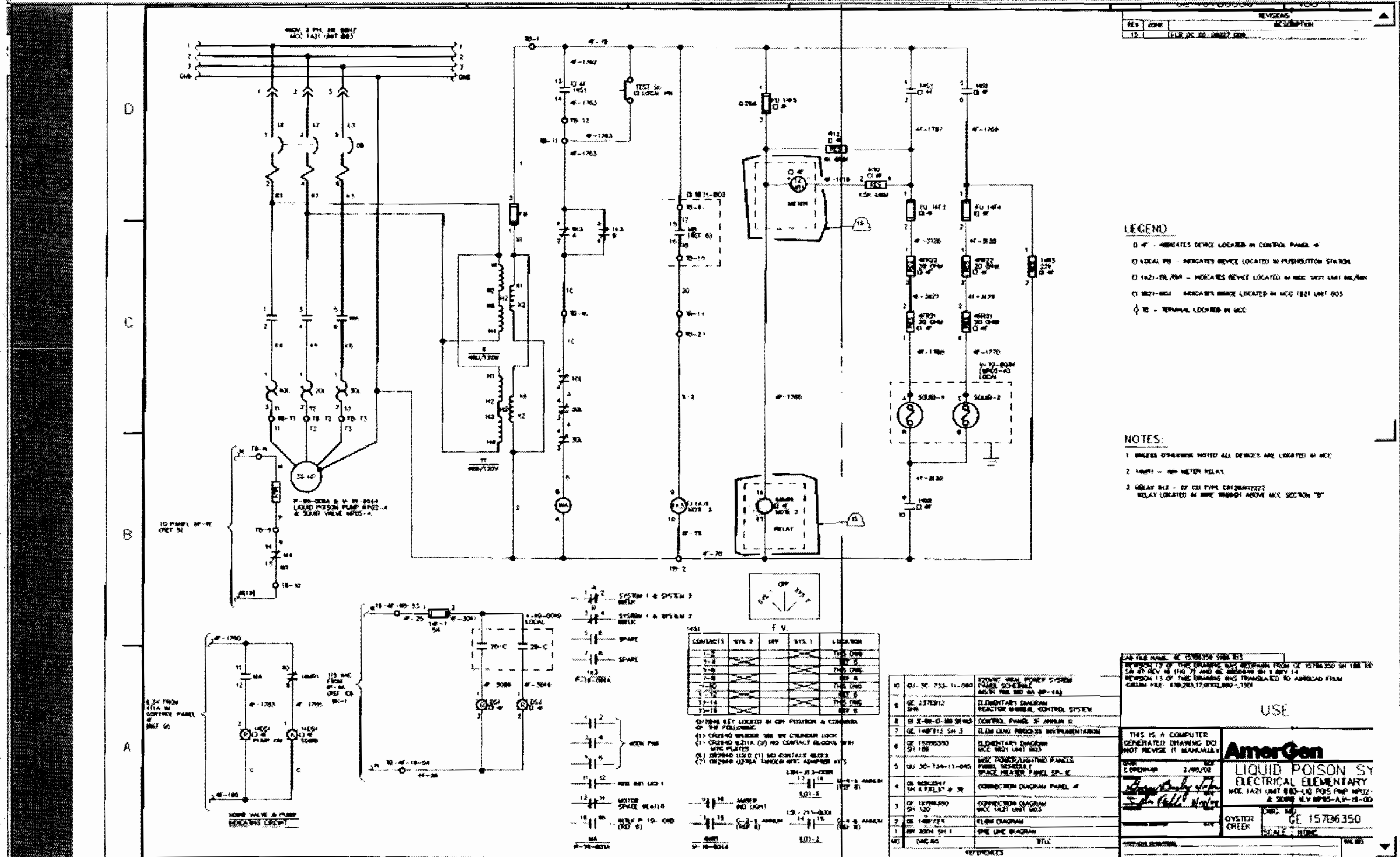
GE 157B6350, Sheet 188, Rev 015, LIQUID POISION SYSTEM ELECTRIC ELEMENTA (tif)



Am 626

Coord: (6116,2098) viewing

3.3 IF the above expected indications do **not** occur,

THEN **PERFORM** the following:

1. **PLACE** the STANDBY LIQUID CONTROL Keylock in the opposite position to use the other system. (Panel 4F) [ ]

2. **VERIFY** proper operation using step 3.2. [ ]

3.4 WHEN the SLC Tank is empty (Large Discharge Pressure Fluctuations or SLC Tank Level Indicates 150 gallons)

OR

injection with the Liquid Poison System is **no** longer required,

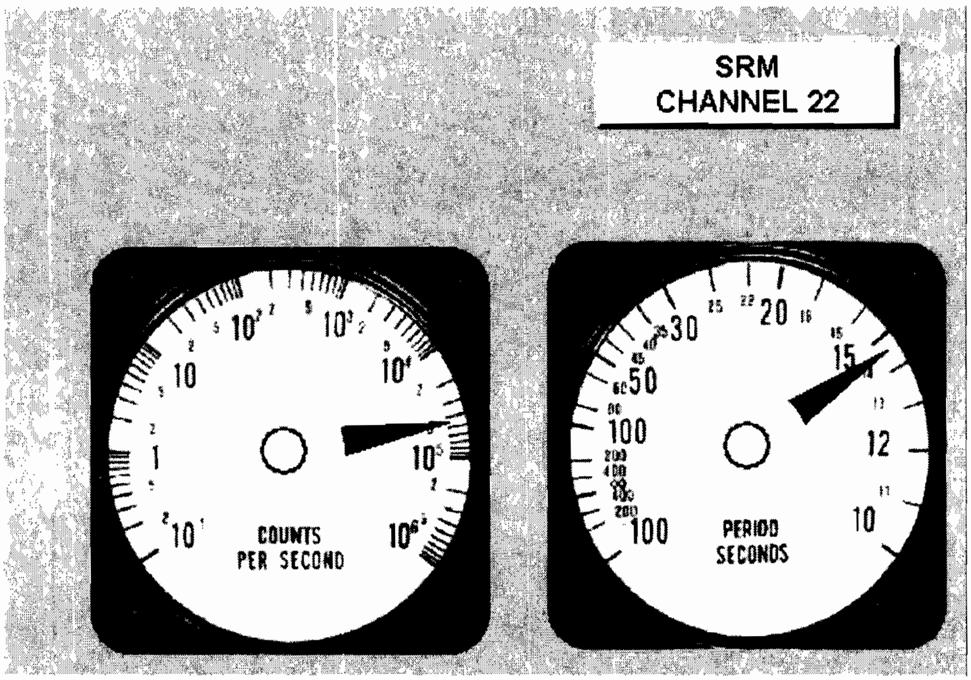
THEN **SECURE** any running SLC Pump. [ ]

19 **ID: 09-1 NRO19** **Points: 1.00**

The plant is starting up after an outage and control rod withdrawals are in progress. As the Panel Operator is withdrawing a control rod in close proximity to SRM 21, another Operators reports the following observations:

- SRM Drawer 21 indications as shown
- All other SRMs Drawers show no changes

Which of the following states the correct RAPs that the Panel Operator will refer to now?



A. • SRM PERIOD SHORT **only**

B. • SRM PERIOD SHORT **and**
• ROD BLOCK **only**

C. • SRM HI/INOP **and**
• ROD BLOCK **and**
SCRAM CONTACTOR OPEN

D. • ROD BLOCK **and**
• SRM HI/INOP **only**

Answer: A

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer Explanation:

QID: 09-1 NRO19		
Question # / Answer	**19**	**Developer/Date: NTP 12/12/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
215004 Source Range Monitor A4.06 - Ability to manually operate and/or monitor in the control room: Alarms and lights						**3.2**	**3.1**
Level	**RO**	**Tier**	**2**	**Group**	**1**		

General References			
Explanation	The plant is starting up and withdrawing control rods. During a control rod withdrawal, an operators provides SRM 21 drawer indications, which include of note: SRM counts is on-scale, period indicates about 13 seconds and the amber PERIOD light is on. With these last 2 indications, then the SRM PERIOD SHORT annunciator will be in alarm (setpoint is 30 seconds). A fast period does not cause a rodblock, and current SRM counts does not show upscale or downscale. Therefore, the only annunciator that is in the alarm state is SRM PERIOD SHORT. Answer A is correct. Answer B is incorrect since a fast period does not give a rodblock signal, although SRM counts will. But the current SRM 21 counts is below that setpoint. Answer C is also incorrect since even an upscale SRM will not result in a 1/2 scram which gives the scram contactor open annunciator. Answer D is incorrect since there are no indications to support either a high SRM counts or inoperable condition.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0029 LO 215-10444		

Question Source (New, Modified, Bank)	**New**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading REACTOR NEUTRON MONITORS	G-7-d

SRM PERIOD SHORT	

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** short period. (Panel 4F) [ ]

AUTOMATIC ACTIONS:

NONE.

MANUAL CORRECTIVE ACTIONS:

- ❑ IF persistent (i.e.; not prompt jump),

  THEN **PERFORM** the following:

  - ❑ **STOP** control rod withdrawal [ ]
  - ❑ **INSERT** control rod(s) to lengthen period above alarm point. [ ]
  - ❑ **CHECK** for movement of detector into high flux area. [ ]

- ❑ **REFER** to ABN-7, Unexplained Reactivity Change. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
NOTE During low power physics testing, periods of 20 seconds are permitted. A reactor period of less than 30 seconds.	Less than or equal to 30 sec.	RH06A and RH06B Reference Drawings: GE 706E812, Sh. 5, 6, 7, & 8 GU 3E-611-17-009 Sh. 2

20 **ID: 09-1 NRO20** **Points: 1.00**

The plant is at 30% power on a startup. Current plant conditions include the following:

- The MASTER FEEDWATER LEVEL CONTROLLER is in AUTO
- Feedwater Pumps A and C are in service
- A MFRV FLOW CONTROLLER is in AUTO
- C MFRV FLOW CONTROLLER is in MAN, to be placed in AUTO

IAW 317, Feedwater System, which of the following displays on the C MFRV FLOW CONTROLLER should be about equal to each other, in order to place the C MFRV FLOW CONTROLLER in AUTO? .

A. P-display and the Y-display.

B. S-display and the V-display.

C. Y-display and the S-display.

D. V-display and the P-display.

Answer: B

Answer Explanation:

QID: 09-1 NRO20		
Question # / Answer	**20**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
259002 Reactor Water Level Control A4.03 - Ability to manually operate and/or monitor in the control room: All individual component controllers when transferring from manual to automatic modes					**3.8**	**3.6**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**317**					

Explanation	The plant is starting with 2 Feedwater Pumps in service, one in AUTO and the other in MAN. The Operator is ready to place the second controller in AUTO. To do this, the S-display and V-display must be made about equal to each other to prevent any changes in feedwater flow when the individual controller is placed on the master controller. When S and V are approximately matched, then the individual controller is placed in AUTO. Answer D is correct. The MFRV controller has displays for S, V, P and Y and the distracters represent the incorrect combinations of the displays.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0018 LO 259-10446	

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

13.4 Transferring from Manual Control on the Individual Controller to the Master Feedwater Level Controller

13.4.1 IF none of the individual controllers are in MASTER MANUAL or MASTER AUTO control,

- A Feed Pump Controller [ ]
- B Feed Pump Controller [ ]
- C Feed Pump Controller [ ]

THEN **PLACE** the first controller in MASTER MANUAL control as follows:

13.4.1.1 **CONFIRM** the MASTER FEEDWATER LEVEL CONTROLLER in MAN. [ ]

13.4.1.2 **PLACE** the Master Feedwater Level Controller display to V-display. [ ]

13.4.1.3 **MATCH** the S display with the V display on the selecting string MFRV FLOW CONTROLLER by rotating the manual adjustment knob on the MASTER FEEDWATER LEVEL CONTROLLER.

- V-ID11A, MFRV Flow Controller Valve [ ]
- V-ID11B, MFRV Flow Controller Valve [ ]
- V-ID11C, MFRV Flow Controller Valve [ ]

13.4.1.4

> NOTE
>
> The Y display on the MFRV FLOW CONTROLLER provides an indication of deviation (S-V).

WHEN the S and V displays are approximately equal (zero deviation on Y display)

THEN **PLACE** the individual controller in AUTO. [ ]

13.4.1.5 **MONITOR** the following for any changes:

- Reactor level [ ]
- Feedwater flow [ ]

21 **ID: 09-1 NRO21** **Points: 1.00**

The plant was at rated power when a loss of USS 1A2 occurred. While investigating the loss of the bus, the operating CRD pump tripped on overload.

The Operator manually scrammed the reactor and then reported the following indications:

- RPS Group 1 Scram lights on Panel 4F and 6R are **energized**
- Many LPRMs indicate > 2%
- Turbine Bypass Valves are controlling RPV pressure

IAW SP-21, Alternate Insertion of Control Rods, which of the following actions is required?

A. De-energize the Scram Solenoids by placing both 100 amp Main RPS Breakers in OFF.

B. Place the REACTOR MODE SELECTOR switch in REFUEL and manually insert control rods.

C. De-energize the Scram Solenoids by placing the Sub Channel Test Keylocks in the TRIP position.

D. Raise the CRD Cooling Water differential pressure by opening the CRD Cooling Water PCV NC40.

Answer: C

Answer Explanation:

QID: 09-1 NRO21		
Question # / Answer	**21**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
215005 APRM / LPRM 2.4.6 - Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.					**3.7**	**4.7**
Level	**RO**	**Tier**	**2**	**Group**	**1**	

General References	EMG-SP21		
Explanation	The plant was at rated power when USS 1A2 was lost. This will remove power to CRD Pump A. When the second CRD Pump is lost on overload, all CRD pumps are lost and the operator scrams the plant. Indications show that several RPS Group solenoid lights are still energized and power is > 2% on several LPRMs. This shows that the plant is in an electrical ATWS. Also, with the turbine bypass valves controlling RPV pressure, the MSIVs must be open. IAW SP-21, for an electrical ATWS, a method to insert control rods is to place the sub-channel keylocks in TEST. Answer C is correct. The method in answer A can be used in an electrical ATWS but only when the MSIVs are closed. Answer A is incorrect. Because both CRD pumps are lost, the operator is unable to manually insert control rods. Answer B is incorrect. The method in answer D can be used in an electrical ATWS but with no CRD pumps, this will have no impact. Answer D is incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.845.0.0053 LO 200-10445A		

Question Source (New, Modified, Bank)				New
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

4.3.3 WHEN control rods are no longer moving in,

THEN **PERFORM** the following:

1. **CLOSE** Scram Air Header drain valve V-6-409. (RB 23 SE) [ ]

2. **OPEN** Scram Air Header isolation valve V-6-175 (RB 23 SE) [ ]

4.4 De-energize the Scram Solenoids

4.4.1 IF MSIVs are OPEN,

THEN **PERFORM** the following:

1. **PLACE** the following Sub channel Test Keylocks in the TRIP position. (Panels 6R/7R)

   - RPS Sub Channel 1A Keylock (Panel 6R) [ ]
   - RPS Sub Channel 1B Keylock (Panel 6R) [ ]
   - RPS Sub Channel 2A Keylock (Panel 7R) [ ]
   - RPS Sub Channel 2B Keylock (Panel 7R) [ ]

2. WHEN the control rods are no longer moving in,

   THEN **PLACE** the RPS Channel I and II Sub channel Test Keylocks in the NORMAL position. [ ]

OVER

22 **ID: 09-1 NRO22** **Points: 1.00**

The plant was at rated power when an event occurred and the plant scrammed. The following observations and annunciators in alarm include:

- LKOUT RELAY 86/S1A TRIP
- LKOUT RELAY 86/S1B TRIP
- EDG 2 - LKOUT RELAY TRIP
- ATWOS RX RECIRC PUMP TRIP - ACTUATE A I, ACTUATE B I, ACTUATE C II **and** ACTUATE D II
- The RPV has been rapidly depressurized to 450 psig and lowering, due to an RPV leak in the Turbine Building

RPV water level continues to lower. Which of the following RPV water level instruments can be used to determine RPV water level?

A. GEMAC A

B. YARWAY B

C. Fuel Zone A

D. Fuel Zone B

Answer: C

Answer Explanation:

QID: 09-1 NRO22		
Question # / Answer	**22**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
264000 EDGs **2.4.3 - Emergency Procedures / Plan: Ability to identify post-accident instrumentation.**					**3.7**	**3.9**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**EOP Users Guide**		**ABN-59** **RAP-E1a**		**3013 sh. 1** **RAP-E2a**	

Explanation	The plant was at rated power when an event occurred, including the following: the plant scrammed with a loss of offsite power, and the loss of EDG 2 to power Bus 1D. EDG 1 has started and is supplying Bus 1C, and the busses downstream: USS 1A2 & 1A3. USS 1A2 supplies VMCC 1A2, which supplies PAIPP-1. PAIPP-1 supplies power to Fuel Zone A. Answer C is correct. The question also shows that the recirculation pumps have tripped (ATWS annunciators) on RPV water level lo-lo of 90" and level is still lowering. The GEMAC lowest reading is 90" and is unable to provide indication less than 90". Answer A is incorrect. Because the RPV has rapidly depressurized to below 500 psig, and IAW the EOP Users Guide, the YARWAYS are not to be used to determine RPV water level. Answer B is incorrect. Answer D is incorrect since Fuel Zone 2 is powered from PAIPPS-2 (powered from VMCC 1B2, powered from USS 1B2, powered from Bus 1D) and has no power.

References to be provided during exam:	**None**	

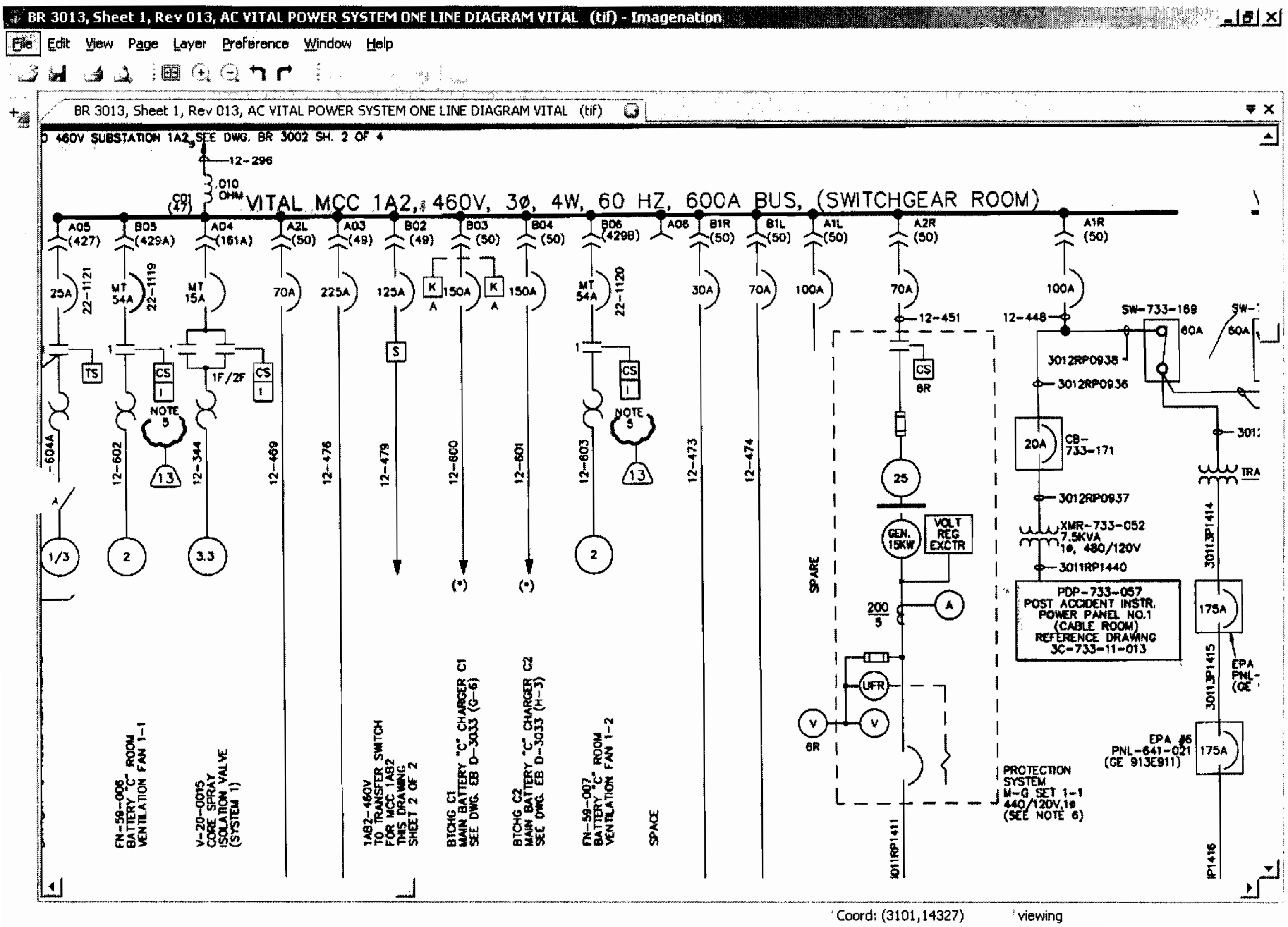
Learning Objective	2621.845.0.0052 LO 200-10445

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	**55.41**	**6**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

ATTACHMENT ABN-59-2

RPV LEVEL INSTRUMENT POWER SUPPLY REFERENCE

INSTRUMENT	**POWER SUPPLY**	**BREAKER**	**ANNUNCIATOR**
NARROW RANGE GE/MAC	CIP-3	1	9XF-4-b
WIDE RANGE GE/MAC	IP-4	15	9XF-5-b
RX PRESS/LEVEL A RECORDER UR-622-0024A	PAIPP-1	6	C-8-f
RX. PRESS/LEVEL B RECORDER UR-622-0024B	PAIPP-2	6	C-8-f
FUEL ZONE C & D (RSP) PRIMARY POWER SUPPLY ALTERNATE POWER SUPPLY	IP-4 DC-B	19 20	9XF-5-b n/a
CORE REGION (FUEL ZONE CH. A)	PAIPP-1	5	C-8-f
CORE REGION (FUEL ZONE CH. B)	PAIPP-2	4	C-8-g



BR 3013, Sheet 1, Rev 013, AC VITAL POWER SYSTEM ONE LINE DIAGRAM VITAL (tif) - Imagenation
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BR 3013, Sheet 1, Rev 013, AC VITAL POWER SYSTEM ONE LINE DIAGRAM VITAL (tif)
460V SUBSTATION 1A2, SEE DWG. BR 3002 SH. 2 OF 4
12-296
.010 OHM
C01 (47)
VITAL MCC 1A2, 460V, 3Ø, 4W, 60 HZ, 600A BUS, (SWITCHGEAR ROOM)
A05 (427)
B05 (429A)
A04 (161A)
A2L (50)
A03 (49)
B02 (49)
B03 (50)
B04 (50)
B06 (429B)
A06
B1R (50)
B1L (50)
A1L (50)
A2R (50)
A1R (50)
25A
22-1121
MT 54A
22-1119
MT 15A
70A
225A
125A
150A
150A
MT 54A
22-1120
30A
70A
100A
70A
100A
12-451
12-448
SW-733-169
60A
60A
3012RP0938
3012RP0936
TS
CS
1F/2F
CS
S
K
K
CS
6R
NOTE 5
NOTE 5
20A
CB-733-171
3012RP0937
XMR-733-052
7.5KVA
1Ø, 480/120V
3011RP1440
PDP-733-057
POST ACCIDENT INSTR. POWER PANEL NO.1 (CABLE ROOM)
REFERENCE DRAWING 3C-733-11-013
12-604A
12-602
12-344
12-469
12-476
12-479
12-600
12-601
12-603
12-473
12-474
13
13
1/3
2
3.3
2
25
GEN. 15KW
VOLT REG EXCTR
SPARE
200/5
A
UFR
V
V
6R
175A
175A
3011RP1414
3011RP1415
EPA PNL-
EPA #6 PNL-641-021 (GE 913E911)
PROTECTION SYSTEM M-G SET 1-1 440/120V,1Ø (SEE NOTE 6)
FN-59-006 BATTERY "C" ROOM VENTILATION FAN 1-1
V-20-0015 CORE SPRAY ISOLATION VALVE (SYSTEM 1)
1AB2-460V TO TRANSFER SWITCH FOR MCC 1AB2 THIS DRAWING SHEET 2 OF 2
BTCHG C1 MAIN BATTERY "C" CHARGER C1 SEE DWG. EB D-3033 (G-6)
BTCHG C2 MAIN BATTERY "C" CHARGER C2 SEE DWG. EB D-3033 (H-3)
FN-59-007 BATTERY "C" ROOM VENTILATION FAN 1-2
SPACE
3011RP1411
Coord: (3101,14327)
viewing

RPV WATER LEVEL CONTROL

WATER LEVEL

IF	THEN
RPV WATER LEVEL REACHES **180 IN.**	CLOSE THE ISOLATION CONDENSER DC VALVES
RAPID RPV DEPRESSURIZATION BELOW **500 PSIG** OCCURS	YARWAY LEVEL INSTRUMENT SHALL NOT BE USED TO DETERMINE RPV WATER LEVEL
PRIMARY CONTAINMENT WATER LEVEL AND TORUS PRESSURE CANNOT BE MAINTAINED BELOW FIG.A MPCWLL **AND** ADEQUATE CORE COOLING WILL STILL BE ASSURED AFTER EXTERNAL WATER SOURCES ARE SECURED	TERMINATE INJECTION TO THE RPV FROM SOURCES EXTERNAL TO PRIMARY CONTAINMENT
RPV WATER LEVEL IS ABOVE **61 IN.**	CONFIRM THE ADS TIMER SWITCHES IN AUTO

CONFIRM AUTOMATIC INITIATIONS AND ISOLATIONS PER

SUPPORT PROC - 1

DISCUSSION

Rapid depressurization of the RPV may cause flashing and possible loss of liquid inventory from the water level instrument reference legs resulting in erratic RPV water level indications substantially higher than actual. This effect applies only to RPV water level instruments with heated reference legs (YARWAY level instruments.) Since heated reference leg temperatures seldom exceed 450°F (saturation temperature for 500 psig), this phenomenon occurs only during rapid depressurization below 500 psig.

23 **ID: 09-1 NRO23** **Points: 1.00**

Which of the following states the correct power supply to the listed Nuclear Instrument?

	Nuclear Instrument	Power Supply
A.	SRM 23	24 VDC Panel A
B.	SRM 24	24 VDC Panel B
C.	IRM 14	24 VDC Panel B
D.	IRM 15	24 VDC Panel A

Answer: B

Answer Explanation:

QID: 09-1 NRO23		
Question # / Answer	23	Developer/Date: NTP 12/14/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
215004 Source Range Monitor K2.01 - Knowledge of electrical power supplies to the following: SRM channels/detectors					2.6	2.8
Level	RO	Tier	2	Group	1	
General References	706E812 sh. 4	401.1				
Explanation	IAW the references, 24 VDC Panel B supplies SRMs 21 & 22, and IRMs 11-14. 24 VDC Panel B supplies SRMs 23 & 24, and IRMs 15-18. Answer B is correct. The other answers are plausible but incorrect.					

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0012 LO 263-10445	

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:F**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Fact			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

GE 706E812, Sheet 4, Rev 012, NEUTRON MONITORING SYSTEM ELECTRICAL ELEME (tif) - Imagenation

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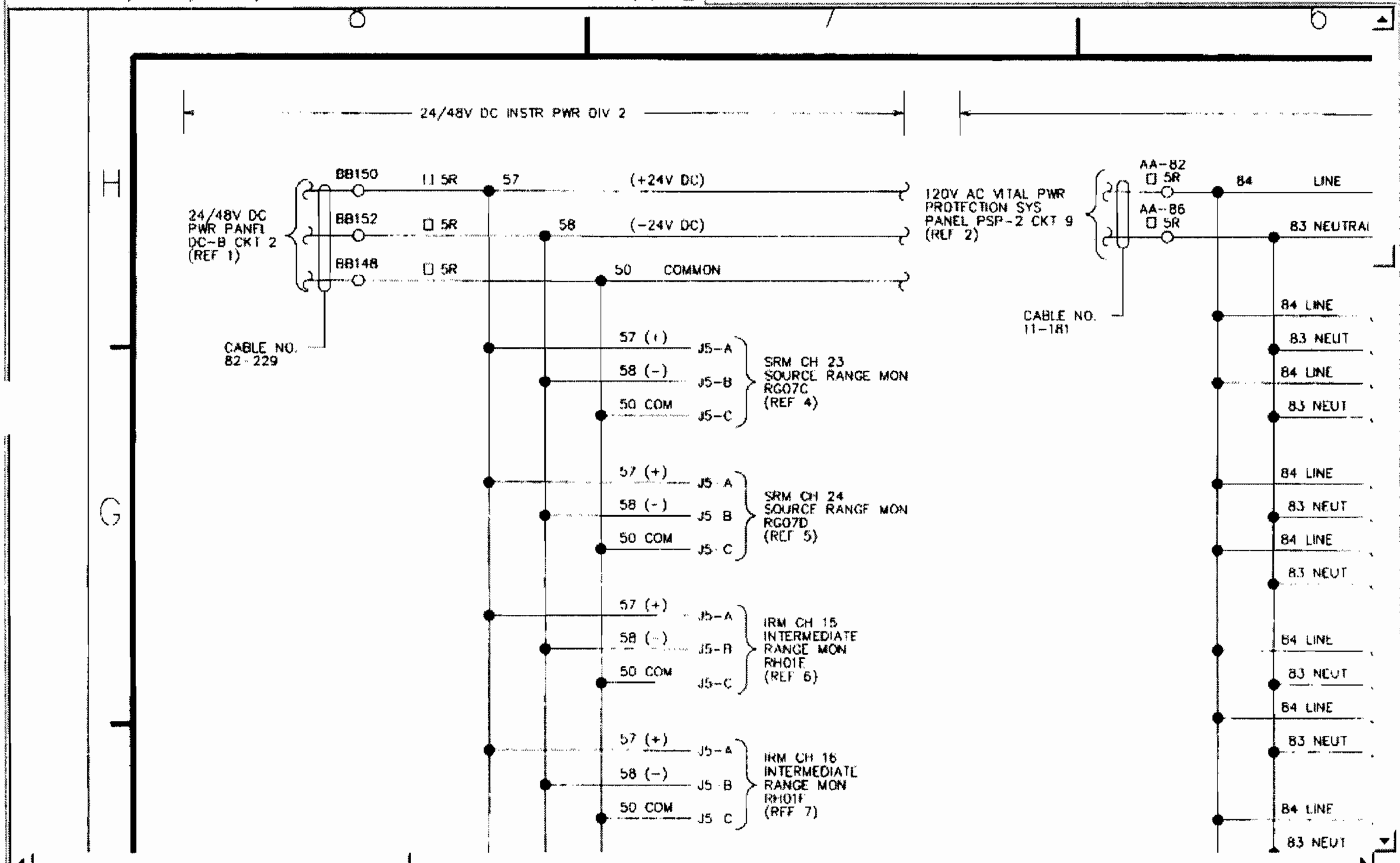
GE 706E812, Sheet 4, Rev 012, NEUTRON MONITORING SYSTEM ELECTRICAL ELEME (tif)



Coord: (3249,1204) viewing

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

24 **ID: 09-1 NRO24** **Points: 1.00**

The plant was at rated power when pressure switch PS-1A0083A, which inputs the RPV pressure signal into the open circuit for EMRV NR108A, failed **high**.

Which of the following states the required action regarding EMRV NR108A, and **following this action**, state the affect on EMRV NR108A to function for ADS?

	Required Action	Affect on ADS
A.	Place the AUTO DEPRESS VALVE NR108A switch in OFF	EMRV NR108A will **not** function for ADS
B.	Place the AUTO DEPRESS VALVE NR108A switch in OFF	ADS is unaffected
C.	Place the EMRV NR108A keylock switch in DISABLE	ADS is unaffected
D.	Place the EMRV NR108A keylock switch in DISABLE	EMRV NR108A will **not** function for ADS

Answer: B

Answer Explanation:

QID: 09-1 NRO24		
Question # / Answer	**24**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

239002 SRVs **K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES : Nuclear boiler instrument system (pressure indication)**					**3.2**	**3.4**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**729E182 sh 1**		**ABN-40**			
Explanation	The plant is at power when the RPV pressure sensing pressure switch fails high to EMRV NR108A. This will result in EMRV NR108A only, opening. The crew will enter ABN-40, Stock Open EMRV. The first manipulation of EMRV controls is placing the control switch for the EMRV to OFF. This removes any RPV pressure input to the EMRV and it will close. With the switch in OFF, the ADS function is not impacted and the valve will open as designed for ADS. Answer B is correct. Answer A is incorrect since the ADS function is still operable. IAW the ABN, if the valve will not close using its control switch, then the DISABLE switch is used. This action will disable both the ADS function and the RPV pressure relief function. But, since the valve did close by placing its control switch in OFF, the DISABLE switch is not manipulated. Answers C & D are incorrect.					
References to be provided during exam:		**None**				
Learning Objective	2621.828.0.0005 LO 379					

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X** **3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

4.2 **PERFORM** the following to close the EMRV:

4.2.1 IF a feedwater transient is in progress,

THEN **ALLOW** the transient to stabilize prior to performing step 4.2.2. [ ]

4.2.2

> NOTE
>
> Shifting from automatic to manual mode is a bumpless transfer and should have **no** effect on RPV level.

PLACE Feedwater Level Control in manual by depressing the AUTO/MAN pushbutton on the MASTER FEEDWATER CONTROLLER [ ]

4.2.2.1 **VERIFY** the red manual LED is illuminated. [ ]

4.2.2.2 **ADJUST** MASTER FEEDWATER CONTROLLER as required to control RPV water level within the normal band of 155-165” TAF. [ ]

4.2.3 **PLACE** the AUTO DEPRESS VALVE switch in OFF position for the open EMRV. [ ]

4.2.4 **DETERMINE** if the EMRV closed using any of the following indications:

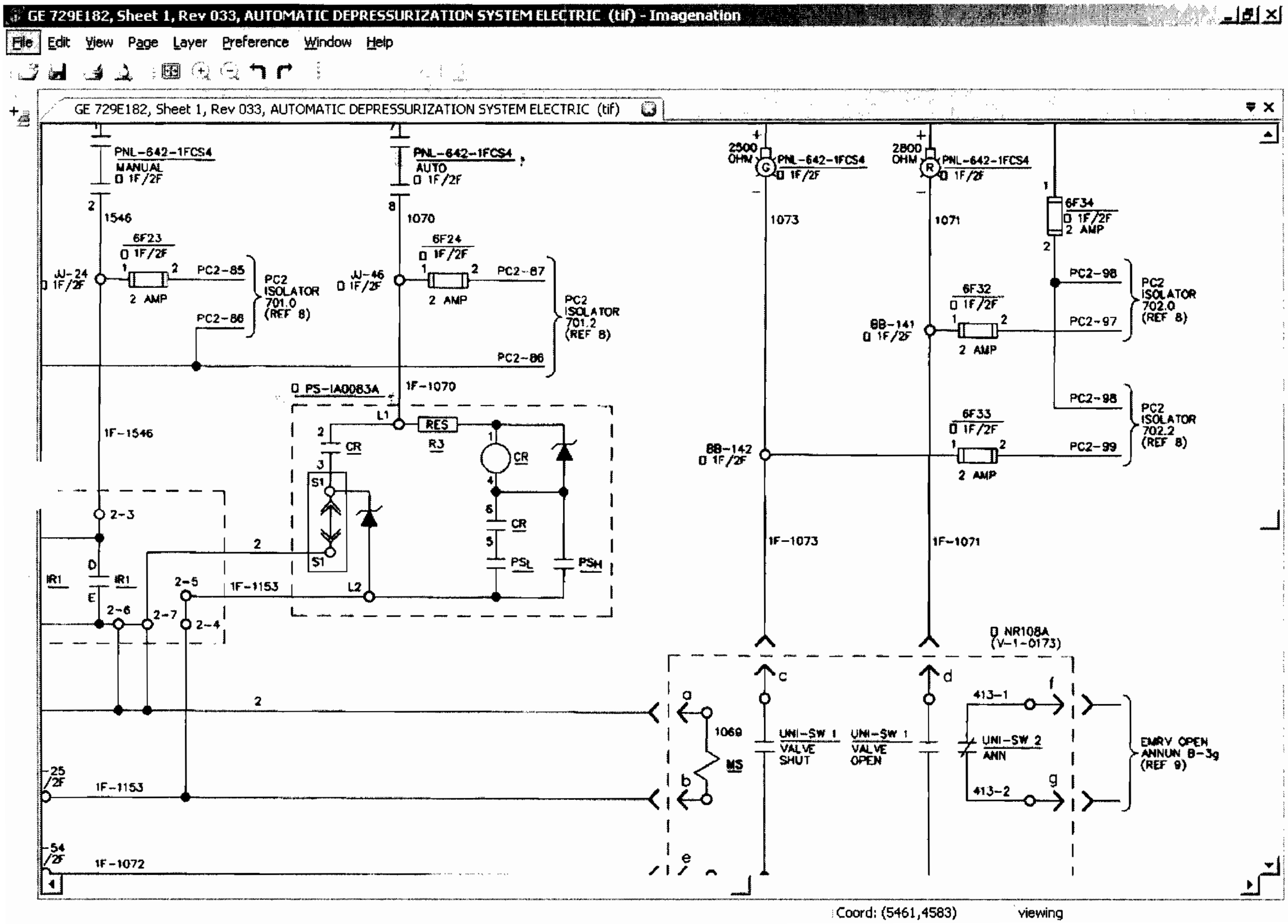
- Acoustic Monitor (15R) [ ]
- 180 Meter (1F/2F) [ ]
- Valve Solenoid Light (1F/2F). [ ]
- Generator Output (8F/9F) [ ]

4.2.5 IF the EMRV is still open,

THEN **CYCLE** the respective AUTO DEPRESS VALVE switch from OFF to MAN to OFF. [ ]

4.2.6 IF the EMRV is **not** closed,

THEN **REPEAT** steps 4.2.5 and 4.2.6 another three to five times in an attempt to close the EMRV. [ ]


GE 729E182, Sheet 1, Rev 033, AUTOMATIC DEPRESSURIZATION SYSTEM ELECTRIC (tif) - Imagenation
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GE 729E182, Sheet 1, Rev 033, AUTOMATIC DEPRESSURIZATION SYSTEM ELECTRIC (tif)
PNL-642-1FCS4
MANUAL
1F/2F
1546
6F23
1F/2F
JJ-24
1F/2F
2 AMP
PC2-85
PC2-86
PC2
ISOLATOR
701.0
(REF 8)
PNL-642-1FCS4
AUTO
1F/2F
1070
6F24
1F/2F
JJ-46
1F/2F
2 AMP
PC2-87
PC2-86
PC2
ISOLATOR
701.2
(REF 8)
PS-IA0083A
1F-1070
1F-1546
RES
R3
CR
PSL
PSH
1F-1153
2500
OHM
PNL-642-1FCS4
1F/2F
1073
2800
OHM
PNL-642-1FCS4
1F/2F
1071
6F34
1F/2F
2 AMP
PC2-98
PC2
ISOLATOR
702.0
(REF 8)
6F32
1F/2F
8B-141
1F/2F
2 AMP
PC2-97
PC2-98
6F33
1F/2F
8B-142
1F/2F
2 AMP
PC2-99
PC2
ISOLATOR
702.2
(REF 8)
1F-1073
1F-1071
NR108A
(V-1-0173)
1069
MS
UNI-SW 1
VALVE
SHUT
UNI-SW 1
VALVE
OPEN
413-1
UNI-SW 2
ANN
413-2
EMRY OPEN
ANNUN B-3g
(REF 9)
1F-1153
1F-1072
Coord: (5461,4583)
viewing

25 **ID: 09-1 NRO25** **Points: 1.00**

The plant was at rated power when an ATWS occurred. The Operator placed the individual scram switch for control rod 02-27 to the scram position, and the control rod fully inserted.

Which of the following states the correct indications for control rod 02-27 with its scram switch in the scram position?

A. The closest LPRM 's amber light is energized and the control rod position displays 00 with green back-lighting.

B. The red scram light on Panel 4F is illuminated and the control rod position displays a blank with green back-lighting.

C. The SCRAM CONTACTOR OPEN annunciator is energized and the control rod position displays a blank with red back-lighting.

D. One of the RPS GROUP SCRAM lights is de-energized and the control rod position displays a blank with green back-lighting.

Answer: B

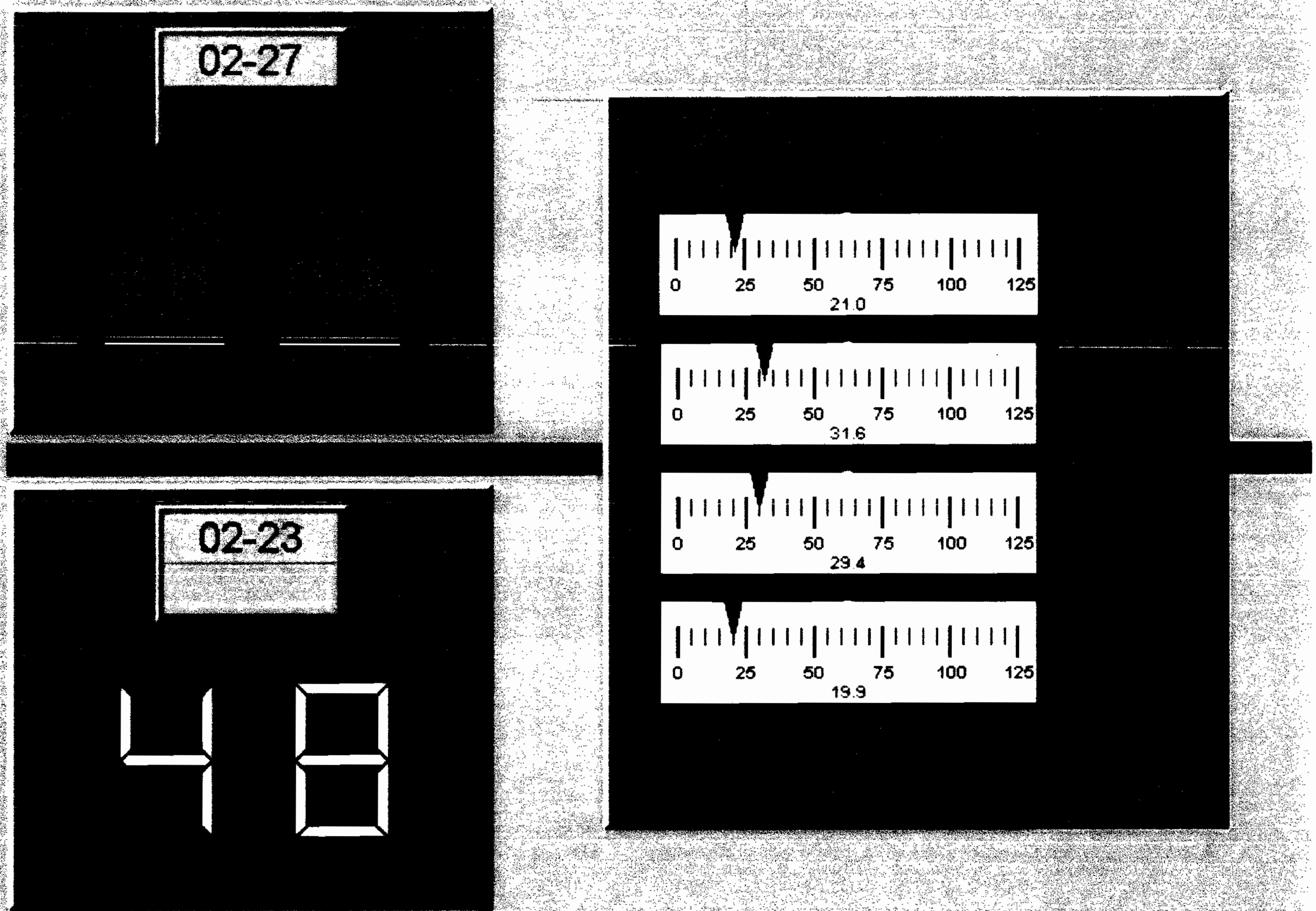
Answer Explanation:

QID: 09-1 NRO25		
Question # / Answer	25	Developer/Date: NTP 12/15/09

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
212000 RPS A4.17 - Ability to manually operate and/or monitor in the control room: Perform alternate reactivity/ shutdown operations						4.1	4.1
Level	RO	Tier	2	Group	1		
General References	Simulator						

Explanation	The plant was at power when an ATWS occurred and the operator inserts control rods using the Rod Scram Test Panel (Panel 6XR). When a single switch is placed in the scram position, this will de-energize the scram solenoids for the single selected control rod and the rod will scram, as it would with a normal scram. The red scram light will energize, showing that the scram valves have opened, and with the scram signal still present, the control rod will indicate over-travel in the inward direction, which is a blank green-backlight (00 not indicated). The 00 position will show after the scram signal is removed. Answer B is correct. The closest LPRM will indicate lower as in answer A, but the rod will not show 00. Answer A is incorrect. Since the scram switch only de-energizes a single control rod scram solenoids, it does not affect a group scram solenoid light. Answer D is incorrect. When the control rod moves to the over-in position with the scram switch in the scram position, the rod position will not display the red backlight. Also, the scram contactor open annunciator does not alarm using the individual scram switches. Answer C is incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0011 LO 79

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system status			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				



Single rod scram for control rod 02-27

26 **ID: 09-1 NRO26** **Points: 1.00**

The plant was at rated power with the following air compressor indications:

- Air Compressor 1 is running in Lead
- Air Compressor 2 is in Standby
- Air Compressor 3 is tagged out of service

The following annunciator then alarmed:

- COMPR 1 TRIP

Investigation revealed that the Air Compressor spuriously tripped on high bearing oil temperature as it had several times in the last few weeks. Bearing temperature was verified as normal. The SRO has directed a re-start of the compressor.

IAW 334, Instrument and Service Air, which of the following is correct to place Air Compressor 1 back in LEAD from the Control Room?

A. Reset the breaker locally. Then place the COMPRESSOR 1 control switch to START.

B. Reset the breaker locally. Then hold the COMPRESSOR 1 control switch in START for 3-5 seconds.

C. The local reset at the compressor must be pressed **once** and then wait for the start logic to be satisfied. Then place the COMPRESSOR 1 control switch to START.

D. The local reset at the compressor must be pressed **twice** and then wait for the start logic to be satisfied. Then hold the COMPRESSOR 1 control switch in START for 3-5 seconds.

Answer: D

Answer Explanation:

QID: 09-1 NRO26		
Question # / Answer	**26**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information

K&A						Importance Rating	
						RO	SRO
300000 Instrument A K4.01 - Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Manual/automatic transfers of control						2.8	2.9
Level	RO	Tier	2	Group	1		

General References	334	RAP-M4a	RAP-M5a
Explanation	The plant is at power with air compressor 1 is lead, air compressor 2 is standby, and air compressor 3 is tagged out of service. An annunciator alarms which describes a trip of compressor 1. The trip is spurious is and the SRO directs a restart of the compressor. IAW the procedure, to place air compressor 1 back in lead, the local reset at the compressor must be pressed twice and then wait for the start logic to be satisfied. Then hold the COMPRESSOR 1 control switch in START for 3-5 seconds. Answer D is correct. There is another annunciator, COMPR 1 BREAKER TRIP, which opens the air compressor breaker on overload only. This will require actions at the breaker to restart the air compressor. But this is not the annunciator given in the question. Answers A & B are incorrect. Answer C is incorrect since the air compressor switch must be held for 3-5 seconds, not just placed in start, and the reset must be pressed twice.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0043 LO 279-10447		

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge	X 1:I	Comprehension or Analysis	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

COMPR 1 TRIP

CONFIRMATORY ACTIONS:

- **CONFIRM** #1 Air Comp, green indicating light on panel 7F is lit. Locally on #1 Air Comp, any "Alarm Messages" indicate that a parameter had exceeded a trip setpoint. [ ]

AUTOMATIC ACTIONS:

Standby Air Compressor start.

MANUAL CORRECTIVE ACTIONS:

- **SWITCH** to standby Air Compressor 1-2 or 1-3 as operating unit. [ ]
- **REVIEW** the alarm history on the programmable controller to diagnose the cause of the trip. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Any #1 Compressor parameter reaching the setpoint for a trip.	**See pages 2-3**	#1 Air Compressor Intellis Controller
		Reference Drawings:

COMPR 1 TRIP

SETPOINTS

ALARMS (Trips)	Parameter Setpoint
Inlet Restriction	1st Stage inlet pressure <13.3 psi vacuum unloaded, or >psig loaded.
High I/C Press	> 39 psi **AND** 1ST Stage Disch Temp is > 410°F **OR** Unit is unloaded **AND** 2nd Stage Inlet Press is > 5 psi.
High 2nd Stage Press	2nd Stage Disch Press >140 psi.
High Line Air Press	Package Disch Press >140 psi.
Low Brg Oil Press	Bearing Oil Press <34 psig for 2 seconds and the unit is running.
High 1st Stage Temp	1st Stage Disch Temp >440 deg F.
High I/C Air Temp	2nd Stage Inlet Temperature >140 deg F.
High 2nd Stage Temp	2nd Stage Disch Temp >486 deg. F.
High Brg Oil Temp	Bearing Oil Temp >170 deg. F.
Starter Fault 1SL, Starter Fault 2SI	Starter coils energized and aux. contacts fail to close **OR** Starter coil is deenergized and aux. contact fails to open.
Main Motor Overload	Motor Overload Relay Contacts open.
Fan Motor Overload	Fan Motor Overload Relay contacts open.
Remote Stop Failure	The Remote Stop Button remains open and either start button (switch) is pressed.
Remote Start Failure	If unit is started from the remote start switch and the start contacts stay closed for 7 seconds after the unit starts.

Setpoints (contined on Page 3 of 3)

7.2.8 If #1 or #2 Air Compressors trip, manual start **cannot** take place until the trip is reset by pressing reset twice, and waiting about 7 seconds for the start logic to be satisfied.

7.2.9 The Air Compressors are limited to 6 starts per hour.

7.2.10 The #1 and #2 Air Compressors if running loaded, will automatically unload, depressurize, and continue to run for about 10 seconds after being given a stop signal. If #1 or #2 Air Compressors are running unloaded, they will immediately stop when given a stop signal.

7.2.11 When starting #1 or #2 Air Compressors from the Control Room, the control switch on Panel 7F must be held in the START position for about 3 to 5 seconds to satisfy the start logic. There is **no** delay when starting the compressors locally.

7.2.12 The enclosure panels of #1 and #2 Air Compressors should remain installed and latched during compressor operation, to ensure proper air cooling of internal components. It is acceptable to remove one panel briefly to check compressor parameters, (i.e. sump oil level).

7.2.13 Do **not** block the cooling air inlets at either ends of the enclosure of #1 or #2 Air Compressor, or the air outlet on the top of the enclosure, to prevent over heating.

7.2.14 If #1 or #2 Air Compressors are **not** running with "Stopped in Auto Restart" displayed on the local compressor display, that compressor can not be manually started locally or remotely. I order to perform a Manual start, the Panel 7F control switch must be placed in stop and allowed to spring return to normal after stop. This will remove the compressor from the "LAG" mode, and "Ready for Start, Local or Remote" will be displayed. At which time a manual start, either locally or remotely is possible.

7.2.15 A set of air desiccant dryer towers is considered operable if the sustained dew point is $\leq$ 22°F.

27 **ID: 09-1 NRO27** **Points: 1.00**

The plant is at rated power.

Complete the following statements:

(1) The loss of _____1_____ will result in the loss of the EPR.

(2) RPV pressure will stabilize at a _____2_____ value due to this power loss.

	(1)	(2)
A.	VACP-1	Lower
B.	VACP-1	Higher
C.	CIP-3	Lower
D.	CIP-3	Higher

Answer: D

Answer Explanation:

QID: 09-1 NRO27		
Question # / Answer	**27**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
241000 Reactor/Turbine Pressure Regulator K1.14 - Knowledge of the physical connections and/or cause- effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: A.C. electrical power	**2.8**	**2.9**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	2	Group	2
General References	RAP-Q6a	ABN-58	315.4		
Explanation	The plant is at rated power. In this condition, the EPR is in control, and the MPR relay position is 8-10% below that of the EPR. This means that the MPR setpoint is slightly above that of the EPR. When AC power (CIP-3) is lost to the EPR, the MPR takes control. Because the MPR setpoint was above the original EPR setpoint, it will call for a slightly lower turbine control valve opening which results in a slightly greater (than initial) RPV pressure. Therefore, if CIP-3 is lost, the MPR will control RPV pressure at a slightly higher pressure. Answer D is correct. Answers A & B are incorrect since they list an incorrect vital AC power supply. Answer C is incorrect since the event will result in a greater RPV pressure.				
References to be provided during exam:	None				
Learning Objective	2621.828.0.0051 LO 249-10446				

Question Source (New, Modified, Bank)				Modified
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

4.0 TRANSFERRING FROM MPR TO EPR AFTER STARTUP OR EXTENDED OPERATION

4.1 Prerequisites

NONE

4.2 Precautions and Limitations

4.2.1 **Do not** drive the EPR setpoint excessively beyond actual reactor pressure, as reactor pressure and level transients may result.

4.3 Instructions

4.3.1 CONFIRM the EPR power switch is in the ON position (Panel 7F). [ ]

4.3.2

> CAUTION
>
> Driving EPR setpoint excessively beyond actual reactor pressure may result in reactor pressure and level transients.

Slowly **LOWER** the EPR setpoint by placing the EPR Relay Position Control Switch in the Lower "↑%" position for approximately one-second periods. [ ]

4.3.3 **REPEAT** as necessary while watching the relay position indicator (Panel 7F). [ ]

4.3.4 WHEN the EPR relay position indicator starts to move upscale,

THEN **RAISE** the EPR setpoint just enough so that relay position indicator will slow down but continue to move very slowly in the upscale direction. [ ]

4.3.4

CAUTION

During the transfer, watch reactor steam pressure closely in case a burned out indicating light **cannot** alert the operator to the takeover.

WHEN the EPR servo reaches the approximate vicinity of the MPR relay position,

THEN **VERIFY** transfer to the EPR using the red indicating light. [ ]

4.3.5

NOTE

Monitor Mwe (Panel 9F) Control Valve position indication (Panel 14R) and Reactor Pressure (Panel 4F) for signs of oscillations.

IF oscillations occur during transfer,

THEN **PERFORM** the following:

a. **PLACE** the EPR power switch in OFF. [ ]

b. **VERIFY** the MPR in control. [ ]

4.3.6 **RAISE** the MPR setpoint so the relay position indication is 8-10% below the EPR relay position or as directed by the Operations Supervisor (**not** to exceed 12.5%). [ ]

5.0 ATTACHMENTS

None

3.0 PROCEDURE

3.1

> **NOTE**
>
> The US may prioritize the order of actions in the following sections.

Instrument Panel CIP-3

3.1.1 Indications

1. Annunciators

Engraving	Location
CIP-3 PWR LOST	9XF-4-b
DISCH PRESS LOW	D-2-b (3F)
CU RCP A OL/TRIP	D-3-b (3F)
CU FLOW LO	D-5-c (3F)
FILTER FLOW LOW	D-7-c (3F)
CAPGRMS TROUBLE	C-2-g (2F)
DW SUMP VLV CLOSED	C-4-h (2F)
TORUS VAC HI	C-6-e (2F)
RB/TORUS VAC BKR OPN	C-8-e (2F)
TRAIN A FLTRS P HI/HTR CKT FAIL	L-1-b (5F/6F}
TRAIN B FLTRS P HI/HTR CKT FAIL	L-4-b (5F/6F}
TIP PURGE PRESS HI/LO	G-8-e (3F)
APRM HI	G-3-f (3F)
MN STM VLVS OFF NORMAL	J-8-b (3F)
FCS/RFCS TROUBLE	J-8-c (5F/6F)
DEMIN EFFL CONDUCT HI	K-2-a (5F/6F)

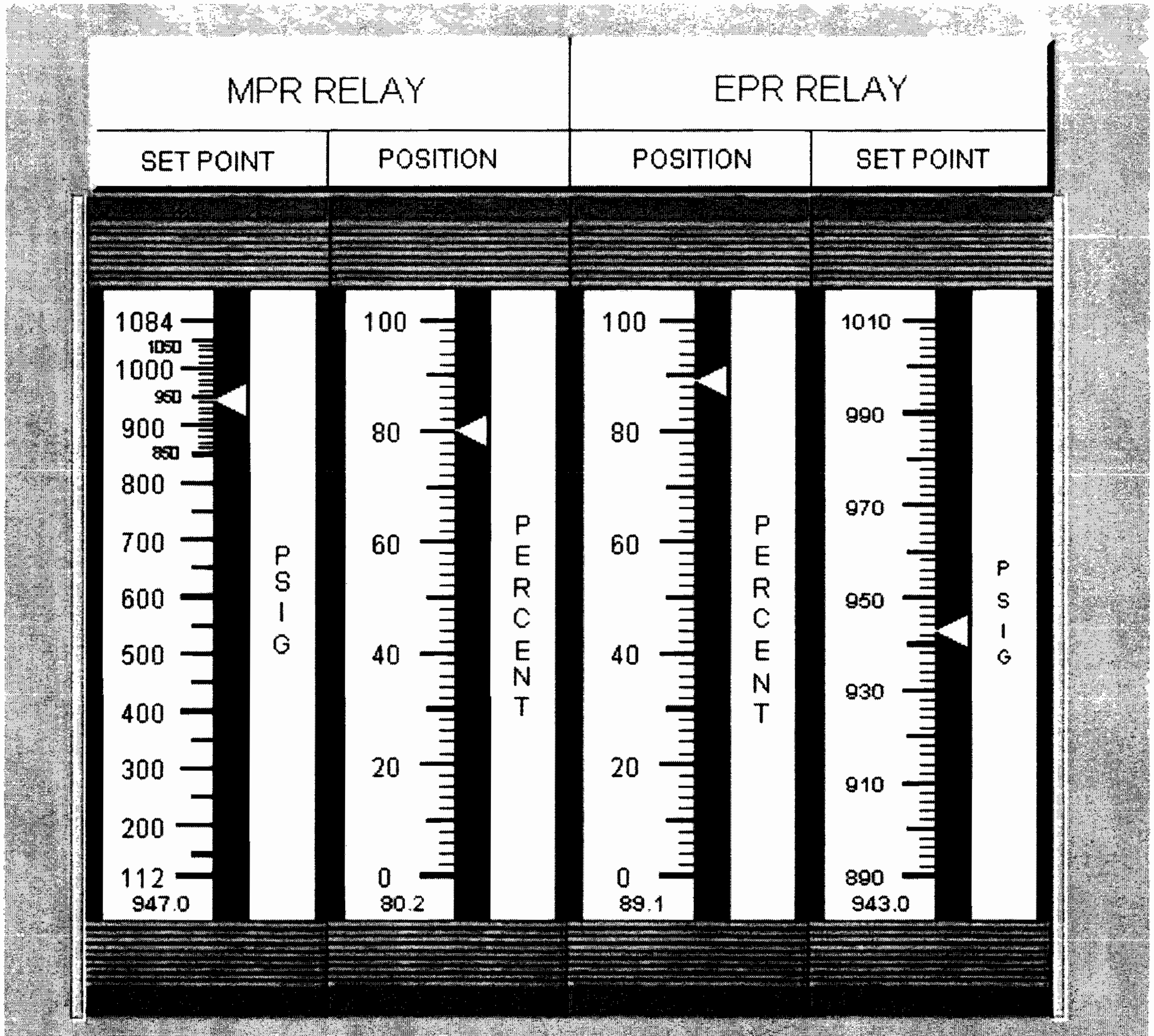
2. Plant Parameters (continued) - recorders fail as is or de-energized

Parameter	Location
Drywell Unidentified Leak Rate Recorder	3F
RWCU System Indicating Controllers	3F
Recirculation Pump Suction Temperature Indicators	3F
STEAM PRESSURE/STEAM FLOW (No Recorder Digital Display)	5F/6F
FEEDWATER FLOW/REACTOR LEVEL (No Recorder Digital Display)	5F/6F
REACTOR PRES/TURBINE PRES (No Recorder Digital Display)	5F/6F
REACTOR LEVEL and REACTOR PRESS (No Recorder Digital Display)	4F
SRMs	4F
IRMs/APRMs	4F
Stack Effluent/Off Gas Radiation	10F
Steam Line Radiation (No Recorder Digital Display)	10F
Off Gas Line/Sample Flow	10F

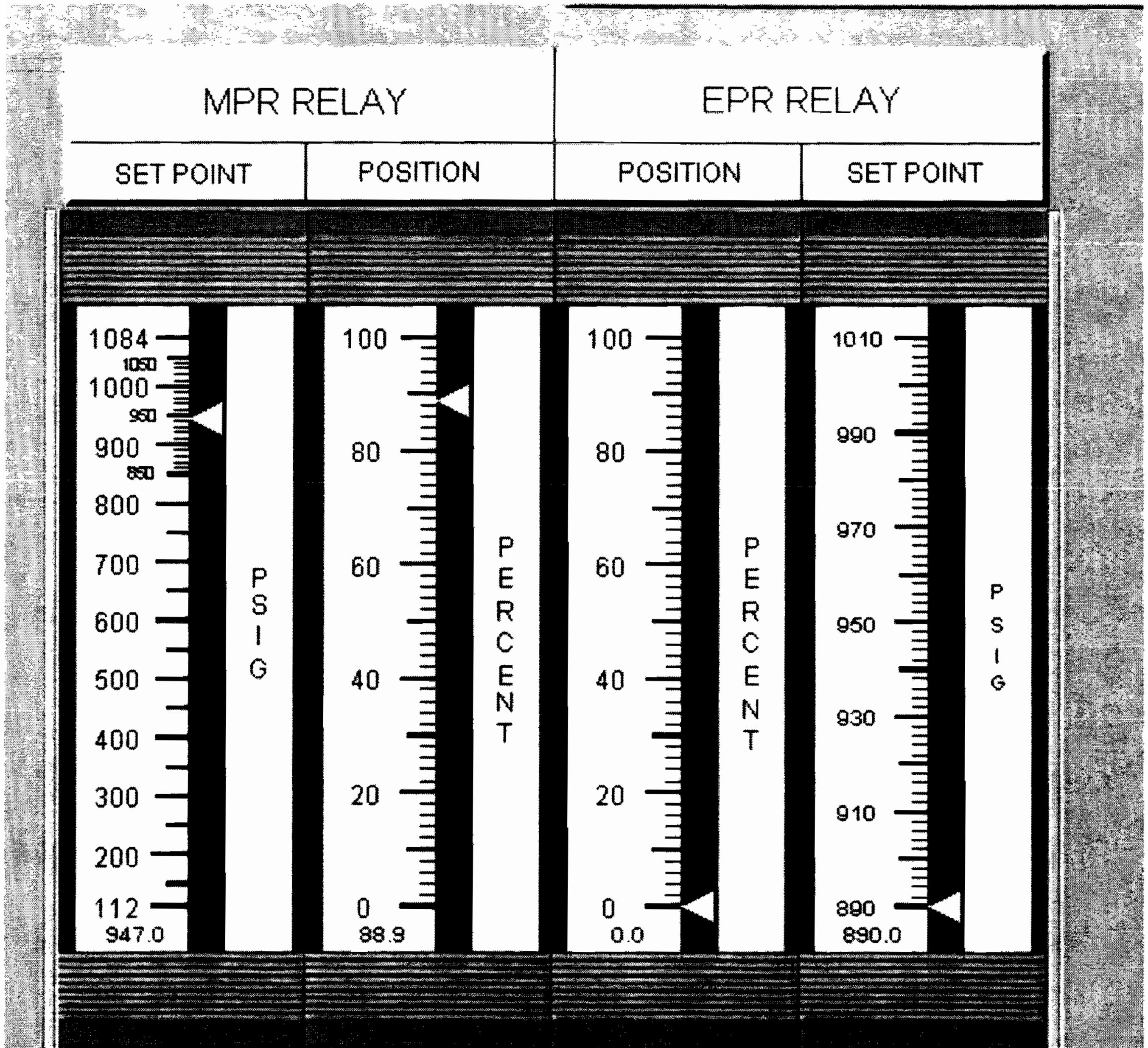
3. The following systems and components become inoperable:

System/Component	Location
Reactor Manual Controls Inoperable	4F
Control Rod Position Indication	4F
Individual Rod Scram Lights	4F
SLC Indicating Lights	4F
Main Turbine EPR	7F
Computer DCC-X (DCS)	9R

4. The following actuations take place



Rated power: No failures



Rated power with loss of CIP-3

28 **ID: 09-1 NRO28** **Points: 1.00**

The plant was at rated power when a loss of offsite power event combined with a loss of coolant accident occurred.

The Operator then reports the following annunciators in alarm:

- 1A2 MN BRKR OL TRIP
- 1A2 MN BRKR TRIP

Which of the following Containment Spray Loops remain available for the Containment Spray function?

A. Loops A and B

B. Loops C and D

C. Loops A and C

D. Loops B and D

Answer: B

Answer Explanation:

QID: 09-1 NRO28		
Question # / Answer	**28**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
226001 RHR/LPCI: CTMT Spray Mode K2.02 - Knowledge of electrical power supplies to the following: Pumps					**2.9**	**2.9**
Level	**RO**	**Tier**	**2**	**Group**	**2**	
General References	**310**		**RAP-U4a**			

Explanation	The plant was at power when a LOOP/LOCA occurred. Then, USS 1A2 was lost. Containment Spray pumps 51C and 51D are powered from 1B2 (which is currently being supplied from EDG 2). The Containment Spray System 2 valves needed to change state for the containment spray function are powered from 1B21B, which is fed from USS 1B2, which is currently being fed from EDG 2. Therefore, Pumps C & D are still available to perform the containment spray function. Answer B is correct. The other distracters list at least one pump which is not powered under the given circumstances.
References to be provided during exam:	**None**
Learning Objective	2624.828.0.0009 LO 226-10453

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading 460V STATION POWER 1A	U-4-a

1A2 MN BRKR OL TRIP	

CONFIRMATORY ACTIONS:

NONE

AUTOMATIC ACTIONS:

Loads on Bus 1A2 will shed on loss of power.

MANUAL CORRECTIVE ACTIONS:

❑ **ENTER** Procedure ABN-45, Loss of US 1A2.	[ ]
❑ **CHECK** Bus loads have shed.	[ ]
❑ **DETERMINE** cause of Breaker trip.	[ ]
❑ IF required, THEN **CORRECT** cause of Breaker trip.	[ ]
❑ **RESET** overload trip alarm at 1A2M.	[ ]
❑ **RESTORE** power to the bus by closing Breaker 1A2M IAW Procedure 338, 480 Volt Electrical System.	[ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 2)

ATTACHMENT 310–2

Electrical Check Off List for Containment Spray System 1

Item	Power Supply	Breaker Location	Position	Initial Perform/IV*
Containment Spray Pump Suction Valve V–21–7	MCC–1A21B	R.B. 23'6" COL RB-R3	Closed	_____/______
Containment Spray Pump Suction Valve V–21–9	MCC-1A21B	R.B. 23'6" COL RB-R3	Closed	_____/______
Containment Spray Valve V–21–11	MCC-1A21B	R.B. 23'6" COL RB-R3	Closed	_____/______
Containment Spray Valve V–21–17	MCC-1A21B	R.B. 23'6" COL RB-R3	Closed	_____/______
Press. Suppression Chamber Spray Valve V–21–18	MCC-1A21B	R.B. 23'6" COL RB-R3	Closed	_____/______
Heat Exchanger Outlet Valve V–3–88	MCC-1A21B	R.B. 23'6" COL RB-R3	Closed	_____/______
Containment Spray Pump (51A)	460V USS 1A2	R.B. SWGR Room	Racked In, Open, and Charged	_____/______
Containment Spray Pump (51B)	460V USS 1A2	R.B. SWGR Room	Racked In, Open, and Charged	_____/______
Emergency Service Water Pump 1-1 (52A)	4160V SWGR 1C	T.B. S.W. Corner Elev. 23'6"	Racked In, Open, and Charged	_____/______
Emergency Service Water Pump 1-2 (52B)	4160V SWGR 1C	T.B. S.W. Corner Elev. 23'6"	Racked In, Open, And Charged	_____/______

Completed by: ______________________ Signature ______________ Date __________ Time

Verified by: ______________________ Signature ______________ Date __________ Time

Reviewed and Approved by: ______________________ US Signature ______________ Date __________ Time

*Independent Verification

ATTACHMENT 310–2
(continued)

Electrical Check Off List for Containment Spray System 2

Item	Power Supply	Breaker Location	Position	Initial Perform/IV
Containment Spray Pump Suction Valve V–21–1	MCC-1B21B	R.B. 23'6" COL RB-R3	Closed	_____/_____
Containment Spray Pump Suction Valve V–21–3	MCC-1B21B	R.B. 23'6" COL RB-R3	Closed	_____/_____
Containment Spray Valve V–21–5	MCC-1B21B	R.B. 23'6" COL RB-R3	Closed	_____/_____
Containment Spray Valve V–21–13	MCC-1B21B	R.B. 23'6" COL RB-R3	Closed	_____/_____
Press. Suppression Chamber Spray Valve V–21–15	MCC-1B21B	R.B. 23'6" COL RB-R3	Closed	_____/_____
Heat Exchanger Outlet Valve V–3–87	MCC-1B21B	R.B. 23'6" COL RB-R3	Closed	_____/_____
Containment Spray Pump (51C)	460V USS 1B2	R.B. SWGR Room	Racked In, Open, and Charged	_____/_____
Containment Spray Pump (51D)	460V USS 1B2	R.B. SWGR Room	Racked In, Open, and Charged	_____/_____
Emergency Service Water Pump 1-3 (52C)	4160V SWGR 1D	T.B. S.W. Corner Elev. 23'6"	Racked In, Open, and Charged	_____/_____
Emergency Service Water Pump 1-4 (52D)	4160V SWGR 1D	T.B. S.W. Corner Elev. 23'6"	Racked In, Open, And Charged	_____/_____

Completed by: ____________________ ____________ ________
Signature Date Time

Verified by: ____________________ ____________ ________
Signature Date Time

Reviewed and Approved by: ____________________ ____________ ________
US Signature Date Time

29 **ID: 09-1 NRO29** **Points: 1.00**

The plant was at 50% power with control rod withdrawals in progress. Control rod 38-23 is the next control rod to be withdrawn. Control rod 38-23 is currently at position 00 and is to be withdrawn to its final position of 12 in the current RWM Step.

Consider the following sequence of events:

- Control rod 38-23 became uncoupled and stuck at position 00
- The drive for control rod 38-23 was withdrawn to the correct position
- The next control rod was selected and was being withdrawn

If control rod 38-23 then became completely unstuck, which of the following states the **first** plant indication that control rod 38-23 is now unstuck?

A. A rise in LPRMs near the control rod.

B. The ROD DRIFT annunciator will alarm.

C. The ROD OVERTRAVEL annunciator will alarm.

D. A rapid change in indicated control rod 38-23 position.

Answer: A

Answer Explanation:

QID: 09-1 NRO29		
Question # / Answer	29	Developer/Date: NTP 12/15/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
201003 Control Rod and Drive Mechanism K3.01 - Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Reactor power					3.2	3.4
Level	RO	Tier	2	Group	2	
General References	UFSAR 15.4.9		205		2621.828.0.0029	

Explanation	The plant is at 50% power with control rod withdrawals in progress. Control rod 38-23 is about to be withdrawn from position 00 to its final position of 12 in the current RWM Step. The control rod becomes uncoupled and stuck. Later, the control rod becomes unstuck and falls (drops) out of the core. The first indication will be a rise in LPRMs/APRMs located next to the unstuck control rod. Answer A is correct. The Rod Drift alarm will not annunciate since there will be no change in rod position reed switches. Answer B is incorrect. An uncoupled rod can be diagnosed by the Overtravel annunciator but this is only performed at position 48 and the control rod only went to position 12. Answer C is incorrect. Answer D is incorrect since rod position is independent from the control rod blade itself (but is dependent on the drive piston, which has been withdrawn to position 12) and will show position 12 prior to the control rod drop. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0011 LO 201-10450		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

I. Introduction

A. The purpose of the Neutron Monitoring System is to provide the capability to monitor neutron flux in the Reactor Core from shutdown conditions to the neutron flux anticipated in the case of overpower conditions requiring Reactor Scram. The Neutron Monitoring System provides Control Room alarms and indications and automatic protective functions, such as Rod Block and Scrams. Three basic types of chambers and signal conditioning equipment are used. The core neutron flux is monitored and indicated over the entire range by the Source Range Monitoring (SRM) System, the Intermediate Range Monitoring (IRM) System, and the Local Power Range Monitoring (LPRM/APRM) System. The fission chambers used in the three nuclear instrumentation systems are essentially the same. However, the various systems operate at different voltages and utilize different gas (argon) pressures. The Traveling in Core Probe (Tip) system is used for calibration of the LPRM detectors and to determine axial neutron flux levels for power distribution. The Control Room Operators monitor SRM's while shutdown and during startups and during refueling activities. SRM's are used when the Reactor is taken critical during startups. IRM's are used during startups and will provide scram functions if necessary. APRM's are used during the Run Mode and scram functions are provided. Operators rely on Nuclear Instrumentation indications and functions for safe plant operation and are vital to Reactivity Management.

1. Present the Learning Objectives.
2. Discuss current NRC Maintenance Rule Status
   a. If A1, discuss plans to return it to an A2 status.

B. Importance

Operators rely on Nuclear Instrumentation indications and functions for safe plant operation and are vital to Reactivity Management.

30 **ID: 09-1 NRO30** **Points: 1.00**

IAW ABN-6, Control Rod Malfunctions, which of the following events requires the Operator to place the NOTCH OVERRIDE switch to the EMERG ROD IN position?

A. When there are simultaneous multiple drifting control rods.

B. An outward drifting control rod when there is an RMCS timer malfunction.

C. A control rod is to be inserted following a double-notching event on a withdrawal.

D. When a withdrawn control rod will not insert with the ROD CONTROL switch in the ROD IN position.

Answer: B

Answer Explanation:

QID: 09-1 NRO30		
Question # / Answer	**30**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
201002 RMCS **K4.06 - Knowledge of REACTOR MANUAL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Emergency In rod insertion**						**3.5**	**3.5**
Level	**RO**	**Tier**	**2**	**Group**	**2**		
General References	**ABN-6**						
Explanation	The reference states that if one control rod is moving out and a timer malfunction is indicated, then apply an EMERG ROD IN signal using the NOTCH OVERRIDE switch. Answer B is correct. All other distracters are incorrect but use of the EMERG ROD IN would function to insert a malfunctioning control rod and is plausible.						

References to be provided during exam:	None	
Learning Objective	2621.828.0.0011 LO 201-10450	

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge	X 1:P	Comprehension or Analysis	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

5.3 IF one control rod is moving out and timer malfunction is indicated,

THEN **PERFORM** the following:

5.3.1 IMMEDIATE OPERATOR ACTIONS

1. **CONFIRM** Rod Power Switch is ON. [ ]
2. **SELECT** the rod. [ ]
3. **APPLY** an EMERG ROD IN signal using the NOTCH OVERRIDE switch. [ ]
4. WHEN the rod is returned to it's programmed position,

   THEN **PLACE** the ROD POWER switch to OFF. [ ]
5. **REMOVE** the EMERG ROD IN signal. [ ]

5.3.2 SUBSEQUENT OPERATOR ACTIONS

5.3.2.1 IF outward rod motion continues,

THEN **PERFORM** the following:

1. **PLACE** the Rod Power Switch to ON. [ ]
2. **SELECT** the Rod. [ ]
3. **RE-APPLY** the EMERG ROD IN signal. [ ]
4. **SCRAM** the affected control rod in accordance with Procedure 302.2, Control Rod Drive Manual Control System. [ ]
5. **REMOVE** the EMERG ROD IN signal. [ ]
6. **TURN** Rod Control Power OFF. [ ]

31 **ID: 09-1 NRO31** **Points: 1.00**

Consider the plant at rated power in 2 different conditions:

Condition 1: **No** annunciators alarmed
Condition 2: The following annunciators alarmed:

- TORUS/DW 1 VAC BRKR OPEN
- TORUS/DW 2 VAC BRKR OPEN

Which of the following is correct given the indications above?

During a LOCA in the Primary Containment, ...

A. indicated Torus pressure will **rise more rapidly** in **Condition 2**.

B. indicated Torus water level will **rise more rapidly** in **Condition 2**.

C. indicated Drywell pressure will **rise more slowly** in **Condition 1**.

D. the amount of water in the Downcomers will be **higher** in **Condition 1**.

Answer: A

Answer Explanation:

QID: 09-1 NRO31		
Question # / Answer	**31**	**Developer/Date: NTP 12/16/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
223001 Primary CTMT and Aux. K5.01 - Knowledge of the operational implications of the following concepts as they apply to PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES : Vacuum breaker/relief operation	**3.1**	**3.3**

Level	RO	Tier	2	Group	2
General References	GU 3E-243-21-1000 sh. 1	UFSAR 6.2.1.1.1 RAP-C4f, -C5f	604.4.016		
Explanation	There are 7 sets of vacuum breakers (of 2 parallel valves in each set) between the Drywell and the Torus air space. The valves are normally closed and automatically open when Drywell pressure drops below Torus pressure by about 0.5 psig. This helps to prevent exceeding the Drywell design negative pressure. Condition 2 shows that at least one vacuum breaker is open. This allows direct communication between the Drywell and the Torus air space. When a LOCA occurs in the Drywell, steam is designed to flow down the downcomers to below the Torus water level to suppress the steam. With a vacuum breaker open, some steam can flow directly from the Drywell into the Torus air space and this steam is not condensed and its pressure is not suppressed. Thus during a LOCA, the indicated torus pressure will rise faster in Condition 2. Answer A is correct. With additional pressure above the water level in the Torus, this will push more water up into the downcomers. Thus water level will not be higher for the same size LOCA in Condition 2. Answer B is incorrect. Because the steam from the LOCA can expand into the Torus space in Condition 2, Drywell pressure will rise more slowly in Condition 2 - not Condition 1. Answer C is incorrect. In LOCA conditions, Torus pressure lags behind Drywell pressure as Drywell pressure rises. With the vacuum breaker open, Torus pressure will be closer to Drywell pressure. As a result, the Dp between the Drywell and Torus will be smaller in Condition 2. With a smaller Dp, there will be more water in the downcomer in Condition 2. Answer D is incorrect.				
References to be provided during exam:	None				
Learning Objective	2621.828.0.0032 LO 432				

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge	X 1:F	Comprehension or Analysis	

	NUREG 1021 Appendix B: Fact				
	10CRF55 Content	55.41	5	55.43	
		(SRO Only)			
	Time to Complete: 1-2 minutes				

Group Heading TORUS/DRYWELL	C-4-f

TORUS/DW 1 VAC BRKR OPN	

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Any one of fourteen Torus to Drywell Vacuum Breakers Open .08 inches off their seating surface for more than 15 seconds.	Any Vac. Bkr. Open	Relay CRAL - A (LS-V26-1A through LS V26-14A)
		Reference Drawings: SS NQZ-0001 GU 3E-611-17-005 Sh. 1

Group Heading	
T O R U S / D R Y W E L L	**C - 5 - f**

T O R U S / D W 2 **V A C B R K R O P N**	

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Any one of fourteen Torus to Drywell Vacuum Breakers Open .08 inches off their seating surface for more than 15 seconds.	Any Vac. Bkr. Open	Relay CRAL - B (LS-V26-1B through LS V26-14B) Reference Drawings: SS NQZ-0001A GU 3E-611-17-005 Sh. 2

32 **ID: 09-1 NRO32** **Points: 1.00**

The plant was at 50% power during a startup with control rod manipulations in progress. The ROD WORTH MINIMIZER switch is in NORMAL.

An event then occurred which resulted in the loss of power to the RWM and the Plant Computer System.

Which of the following states the impact of the power supply loss on the Rod Worth Minimizer?

A. The RWM **will insert** a control rod block due to the loss of power.

B. The RWM **will not** insert a rod block from the power loss **but** will **not** enforce the control rod pattern.

C. The RWM **will not** insert a control rod block since reactor power is above the Low Power Setpoint.

D. The RWM **will insert** a control rod block due to the loss of control rod position information from the PPC.

Answer: A

Answer Explanation:

QID: 09-1 NRO32		
Question # / Answer	**32**	**Developer/Date: NTP 12/16/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
201006 RWM K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC): Power supply					**2.8**	**3.2**
Level	**RO**	**Tier**	**2**	**Group**	**2**	
General References	**2621.828.0.0041**	**409**		**201**		

Explanation	The plant is starting up with control rod withdrawals in progress. At this point in the startup, the RWM is in the Power Operations Mode, which acts very similarly to the low power mode in that it will still insert control rod blocks if control rods manipulations deviate from the planned control rod manipulations. The PPC UPS also powers the RWM. With the RWM in service (ie, not bypassed, or above the low power setpoint while in the low power mode), a loss of power to the RWM will result in a control rod block. Answer A is incorrect. The RWM sends control rod information to the PPC. Since the power loss does result in a rod block, answer B is incorrect. It is true that the RWM is above the LPSP, but the RWM will be in the power operations mode at this time in the startup and the LPSP does not apply, and a rodblock will still be applied from the loss of power. Answer C is incorrect. The RWM supplies control rod information to the PPC - not the other way around. Thus, there is a rodblock but not because the PPC does not/cannot provide control rod position information to the RWM. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2624.828.0.0041 LO 217-10444	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**		**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

V. System Interactions

A. Vital Power System

1. RWM System receives power from plant computer's uninterruptible power supply (UPS).

2. If power is lost, RWM System must be bypassed to move rods when below LPSP.

B. Rod Position Indication System

1. Provides rod position data to RWM System

C. Reactor Manual Control System

1. Supplies rod selection inputs to the RWM.

2. Receives rod block output signals.

3. RWM System failure below LPSP gives error lockout. RWM must be bypassed or repaired to continue moving rods.

D. Main Steam System

1. Supplies total steam flow signal to RWM System.

2. Used to determine LPAP and LPSP, which are adjustable.

3. Loss of signal causes RWM to become active regardless of power level.

2. Power operations mode functions the same as the low power RWM except:

   a. Rod block on select error

   b. Insert block on a single insert error

   c. Available at all power levels

   d. Can define individual rods vs. groups

   e. Stores up to 10 predefined sequences

   f. Manual activation only

   g. Can only go "forward" in sequence

   h. Monitors only the current sequence step

   i. Does not include Groups 1 thru 4

3. Procedure

   a. CRO programs sequence using Power Operations Mode Sequence Editor

   b. CRO initializes Power Operations Mode

   c. CRO verifies POM Sequence vs. Maneuver Request Sheet

   d. CRO performs rod movement IAW Maneuver Request Sheet & Monitors RWM

   e. When steps are added or changed on Maneuver Request Sheet, Shift Engineer edits POM sequence

4. Function Keys

   a. Start/Stop POM

      1) function requires system to be in BYPASS, provides for clean transition from low power to Power Operation Mode.

      2) If in low power mode, displays START POM in red. (POM status block in green)

      3) If in Power Operation Mode display in green (POM status block in red)

33 **ID: 09-1 NRO33** **Points: 1.00**

The Spent Fuel Pool Cooling System was in service with one pump, filter and demineralizer. The flow controller is currently set at 70% open with flow established at 400 GPM when the following occurred:

- The flow controller failed to 100% open

Which of the following states the **initial** affect on water level in the Skimmer Surge Tanks **and** in the Fuel Pool?

	Skimmer Surge Tanks Level	Fuel Pool Level
A.	Lower	Lower
B.	Higher	Higher
C.	Lower	Higher
D.	Higher	Lower

Answer: C

Answer Explanation:

QID: 09-1 NRO33		
Question # / Answer	33	Developer/Date: NTP 12/17/09

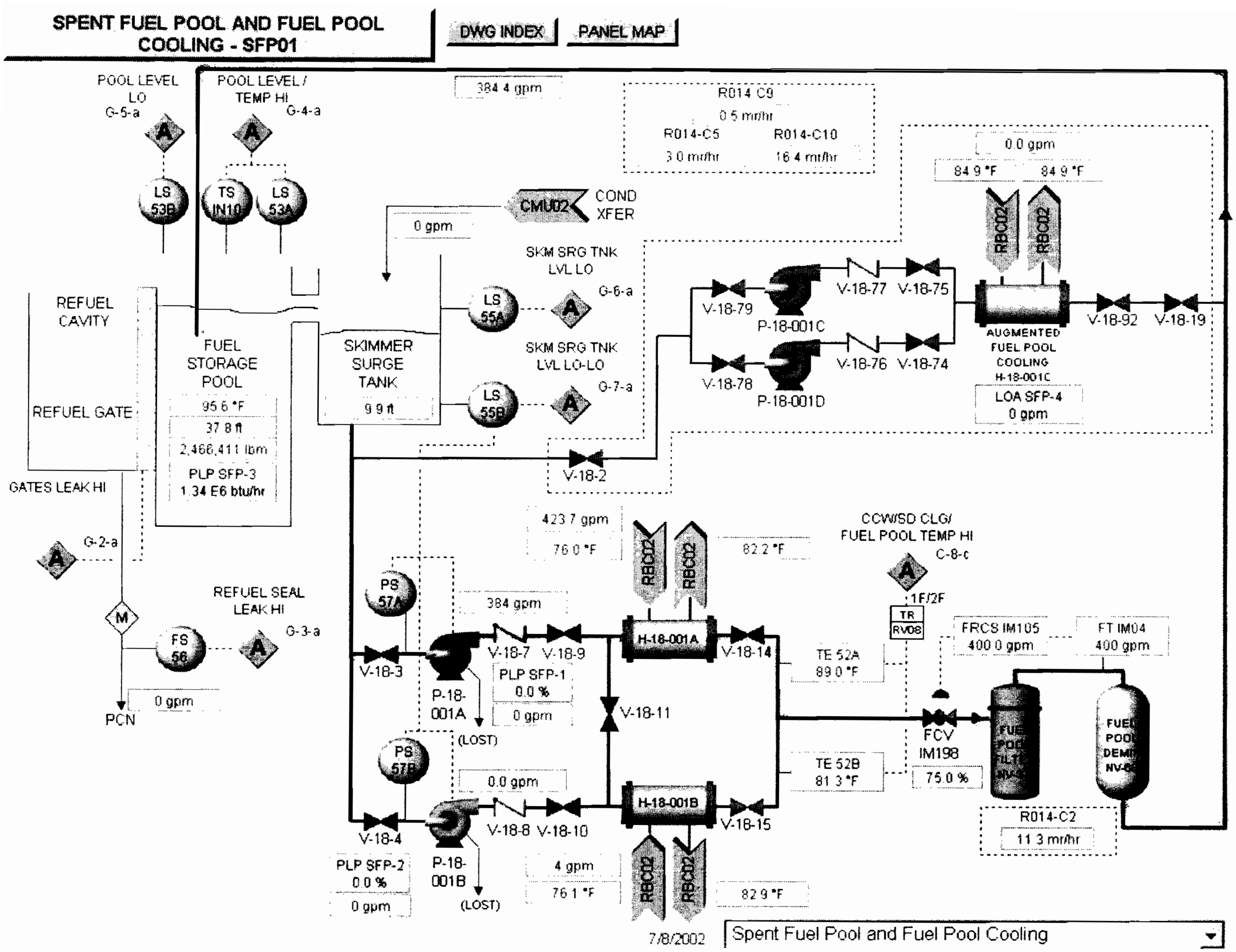
Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
233000 Fuel Pool Cooling/Cleanup A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: Surge tank level					2.6	2.9
Level	RO	Tier		Group		

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	237E756		
Explanation	The fuel pool overflows into the skimmer surge tanks. The fuel pool cooling pumps take a suction on the skimmer surge tanks and discharge directly back into the fuel pool. The fuel pool water level and skimmer surge tanks water level is in equilibrium and steady at a system flow rate of 400 gpm. To maintain steady levels, 400 gpm must flow from the fuel pool into the skimmer surge tanks. When the flow controller is opened further, the skimmer surge tank level will drop to accommodate more flow. As the flow rate rises into the fuel pool, water level there will initially rise, until again equilibrium is achieved. Answer C is correct. The other answers are plausible if the candidate does not understand the system and flow path.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0020 LO 231-10445		

Question Source (New, Modified, Bank)				Bank
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B:			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				


SPENT FUEL POOL AND FUEL POOL COOLING - SFP01
DWG INDEX
PANEL MAP
POOL LEVEL LO
G-5-a
POOL LEVEL / TEMP HI
G-4-a
LS 53B
TS IN10
LS 53A
384 4 gpm
R014 C9
0 5 mr/hr
R014-C5
3 0 mr/hr
R014-C10
16 4 mr/hr
0 0 gpm
84 9 °F
84 9 °F
CMU02
COND XFER
0 gpm
RBC02
RBC02
SKM SRG TNK LVL LO
G-6-a
LS 55A
SKM SRG TNK LVL LO-LO
G-7-a
LS 55B
REFUEL CAVITY
FUEL STORAGE POOL
SKIMMER SURGE TANK
9 9 ft
REFUEL GATE
95 6 °F
37 8 ft
2,466,411 lbm
PLP SFP-3
1 34 E6 btu/hr
GATES LEAK HI
G-2-a
V-18-79
V-18-77
V-18-75
P-18-001C
V-18-78
V-18-76
V-18-74
P-18-001D
AUGMENTED FUEL POOL COOLING H-18-001C
LOA SFP-4
0 gpm
V-18-92
V-18-19
V-18-2
423 7 gpm
76 0 °F
82 2 °F
CCW/SD CLG/ FUEL POOL TEMP HI
C-8-c
1F/2F
TR RV08
M
REFUEL SEAL LEAK HI
G-3-a
FS 56
0 gpm
PCN
PS 57A
384 gpm
V-18-3
P-18-001A
V-18-7
V-18-9
H-18-001A
V-18-14
PLP SFP-1
0.0 %
0 gpm
(LOST)
V-18-11
TE 52A
89 0 °F
FRCS IM105
400 0 gpm
FT IM04
400 gpm
FCV IM198
75.0 %
PS 57B
0.0 gpm
V-18-4
P-18-001B
V-18-8
V-18-10
H-18-001B
V-18-15
TE 52B
81 3 °F
R014-C2
11 3 mr/hr
PLP SFP-2
0 0 %
0 gpm
4 gpm
76 1 °F
82 9 °F
(LOST)
7/8/2002
Spent Fuel Pool and Fuel Pool Cooling

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

34 **ID: 09-1 NRO34** **Points: 1.00**

The plant was at rated power. The Operator had just placed TIP 3 and 4 at the core top location, when the following annunciators alarmed:

- CORE SPRAY - SYSTEM 1 AUTOSTART
- CORE SPRAY - SYSTEM 2 AUTOSTART

10 minutes later, the Operator reports the following observations:

- TIP CHANNEL 3
  - IN SHIELD white light is energized
  - DETECTOR POSITION displays 02
- TIP CHANNEL 4
  - IN SHIELD white light is de-energized
  - DETECTOR POSITION displays 255
- The TIP red light (Panel 11F) is energized
- **No** TIPs can be moved

Which of the following states the status of the TIPs 3 & 4, and the required actions IAW 405.2, Operation of the TIP System?

	TIP Status	Required Action
A.	TIP 3 has isolated TIP 4 has **not** isolated	Fire the shear valve for TIP 4
B.	TIP 4 has isolated TIP 3 has **not** isolated	Fire the shear valve for TIP 3
C.	TIP 3 has isolated TIP 4 has **not** isolated	Manually retract TIP 4 locally
D.	TIP 4 has isolated TIP 3 has **not** isolated	Manually retract TIP 3 locally

Answer: A

Answer Explanation:

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

QID: 09-1 NRO34		
Question # / Answer	34	Developer/Date: NTP 12/17/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
215001 Traversing In-core Probe A2.07 - Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to retract during accident conditions: Mark-I&II(Not-BWR1)					3.4	3.7
Level	RO	Tier	2	Group	2	
General References	405.2		RAP-B1e RAP-B1f		EMG-SP1	

Explanation	The plant is at power with TIPs 3 & 4 at the core top. The provided annunciators show that a LOCA signal has been generated (3 psig Drywell pressure or RPV water level at or below 86"). These signals also isolate the Primary Containment and RPV, including the TIPs. On an isolation, the TIPs automatically retract and the ball valves close. Conditions show that with the Panel 11F TIP red light on, then at least one TIP ball valve is open. It also shows that the in shield light for TIP 4 is de-energized, which means that the TIP 4 has not retracted to the in shield position and the ball valve will be open. The ball valve normally auto closes when the TIP is retracted into the shield. The TIP 4 detector position (lowest is in shield and counts up as the detector moves out of the shield) shows that it is not in shield. IAW the 405.2, with a ball valve open and cannot be closed, then it directs that the shear valve be fired for the applicable TIP. Answer A is correct. Answer B is incorrect since it lists the incorrect TIP as not being in-shield. Answer C is incorrect since it lists the correct TIP but the incorrect action, although TIPs can be manually cranked locally . Answer C is incorrect. Answer D is incorrect since although TIPs can be manually cranked locally, this is not the procedure direction. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0029 LO 215-10445	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading CORE SPRAY 1	B-1-e

SYSTEM 1 AUTOSTART	

MANUAL CORRECTIVE ACTIONS: (continued from Page 1 of 2)

NOTE

This alarm indicates that a parameter has exceeded or has the potential to exceed an Emergency Action Level (EAL).

- ❑ **REFER** to Procedure EP-AA-1010, Exelon Nuclear Radiological Emergency Plan Annex for Oyster Creek Station to determine EAL classification. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Low low reactor water level	90" above TAF	RE02AY5 RE02BY5 RE02CY5 RE02DY5 (Panel 18R & 19R Relay Modules) P.S. RV46 A, B, C, D
OR		
High drywell pressure	Drywell press. 2.9 psig	
		Reference Drawings: NU 5060E6003 GU 3E-611-17-004 Sh. 1

SYSTEM	OPERATING DETAILS
Core Spray System Start	IF Any of the following conditions exist: • RPV water level at or below **86 in.** and **not** bypassed. • Drywell pressure at or above **3.0 psig** and **not** bypassed. AND Core Spray is **not** defeated per EOPs, THEN **CONFIRM** the following: (Panel 1F/2F) • Start of one Main Pump in each system. [ ] • At least one Booster Pump running. [ ]
Primary Containment Isolation	IF Any of the following conditions exist: • RPV water level at or below **86 in.** and **not** bypassed. • Drywell pressure at or above **3.0 psig** and **not** bypassed. THEN **CONFIRM** closed the following valves that are **not** required to be open by the Emergency Operating Procedures: System — Valve No. DW Vent/Purge (Panel 11F) — V-27-1 [ ] V-27-2 [ ] V-27-3 [ ] V-27-4 [ ] Torus Vent (Panel 11F) — V-28-17 [ ] V-28-18 [ ] (continued)

SYSTEM	OPERATING DETAILS		
Primary Containment Isolation (continued)	Torus 2" Vent Bypass (Panel 11F)	V-28-47	[ ]
	DWEDT (Panel 11F)	V-22-1	[ ]
		V-22-2	[ ]
	DW Floor Sump (Panel 11F)	V-22-28	[ ]
		V-22-29	[ ]
	Torus/Rx Bldg. (Panel 11F)	V-26-16	[ ]
	Vacuum Breakers (Panel 11F)	V-26-18	[ ]
	TIP Valves (Panel 11F)	Common Ind.	[ ]
	DW 2" Vent Bypass (Panel 12XR)	V-23-21	[ ]
		V-23-22	[ ]
	N_2 Purge/Hardened Vent (Panel 12XR)	V-23-13	[ ]
		V-23-14	[ ]
		V-23-15	[ ]
		V-23-16	[ ]
	N_2 Makeup (Panel 12XR)	V-23-17	[ ]
		V-23-18	[ ]
		V-23-19	[ ]
		V-23-20	[ ]

5.1.2.30 **PERFORM** Attachment 405.2-2 Final Panel/Switch Checks.

1 [ ] 2 [ ] 3 [ ] 4 [ ]

5.2 Emergency Operation of TIP System.

5.2.1

NOTE

Shear valve is a safety device designed to cut guide tube and drive cable, and to seal tube if a loss of coolant accident occurs and drive cable is unable to be withdrawn from guide tube or if ball valve does **not** seal properly. Operation of shear valve will restore Reactor containment integrity.

IF Primary Containment Isolation occurs, and TIPs are in Service,

THEN **PERFORM** the following:

5.2.1.1 **VERIFY** TIPs return to In Shield.

1 [ ] 2 [ ] 3 [ ] 4 [ ]

5.2.1.2 **VERIFY** Ball Valve lamp on DCU is dimly illuminated.

1 [ ] 2 [ ] 3 [ ] 4 [ ]

5.2.1.3 **VERIFY** Ball Valve Close lamp on Valve Control Monitor is illuminated.

1 [ ] 2 [ ] 3 [ ] 4 [ ]

5.2.2

NOTE

Condition required for Ball Valve to Close is TIP Detector shall be IN-SHIELD.

IF Ball Valve fails to close,

THEN **CONFIRM** Man. Valve Control Switch in CLOSED.

1 [ ] 2 [ ] 3 [ ] 4 [ ]

5.2.3 IF Ball Valve fails to close and TIP detector is retracted to less than 30 inches,

THEN **PERFORM** the following:

5.2.3.1 **POSITION** Mode Switch on DCU to OFF.

1 [ ] 2 [ ] 3 [ ] 4 [ ]

5.2.3.2 **VERIFY** Ball Valve Closed Lamp on Valve Control Monitor Illuminated.

1 [ ] 2 [ ] 3 [ ] 4 [ ]

5.2.4

CAUTION

Placing Keylock Switch for a squib firing circuit in fire position for an extended period has potential to damage circuit components.

IF TIP detector fails to retract into shield or Ball Valve fails to close

AND

Containment integrity shall be ensured,

THEN **PERFORM** the following:

5.2.4.1 **ACTUATE** respective shear valve by rotating Key Locked Switch to FIRE.
(Panel 4R)

1 [ ] 2 [ ] 3 [ ] 4 [ ]

5.2.4.2 **RETURN** Switch to MONITOR within three seconds.

1 [ ] 2 [ ] 3 [ ] 4 [ ]

6.0 ATTACHMENTS

- 405.2-1, Troubleshooting TIP's
- 405.2-2, Final Panel/Switch Checks

5.1.1.4 **VERIFY** following at selected DCU: (Panel 4R)

- IN-SHIELD Lamp Illuminated (Indicating TIP in Shield)

  1 [ ] 2 [ ] 3 [ ] 4 [ ]

- Valve Lamp .... Illuminated Dim (Indicating Ball Valve is Closed)

  1 [ ] 2 [ ] 3 [ ] 4 [ ]

- Core Top...................... Extinguished

  1 [ ] 2 [ ] 3 [ ] 4 [ ]

- In-Core......................... Extinguished

  1 [ ] 2 [ ] 3 [ ] 4 [ ]

- Scan .... Extinguished

  1 [ ] 2 [ ] 3 [ ] 4 [ ]

- Reverse .... Extinguished

  1 [ ] 2 [ ] 3 [ ] 4 [ ]

- Fwd.............................. Extinguished

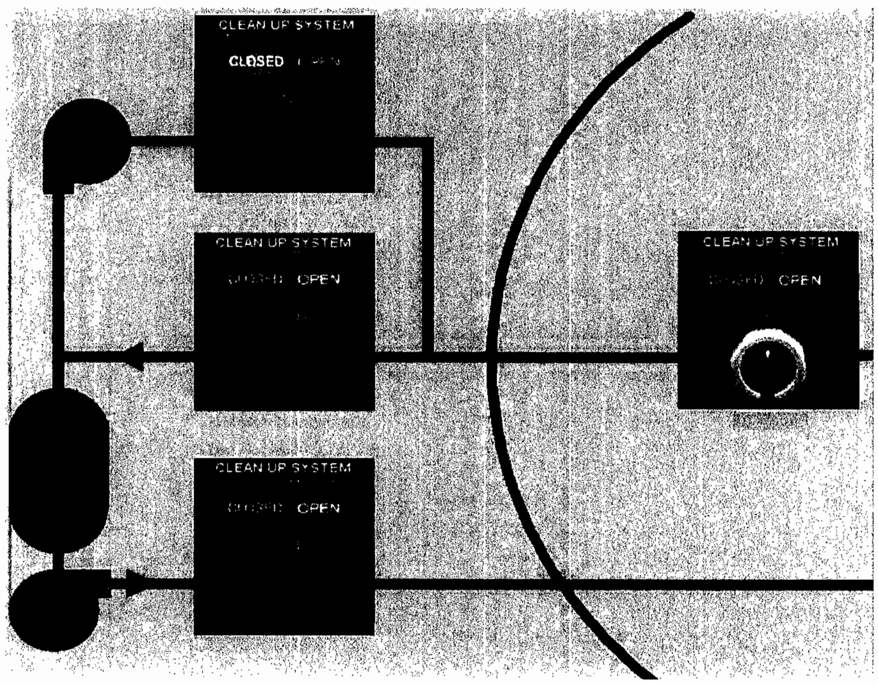
  1 [ ] 2 [ ] 3 [ ] 4 [ ]

35 **ID: 09-1 NRO35** **Points: 1.00**

The plant was at rated power when an event occurred, which resulted in the following annunciator:

- NRHX OUTLET TEMP HI

Which of the following states **all** RWCU valves that receive an isolation signal?



A. V-16-1
V-16-14
V-16-61

B. V-16-1
V-16-2
V-16-14

C. V-16-1
V-16-2
V-16-61

D. V-16-1
V-16-2
V-16-14
V-16-61

Answer: B

Answer Explanation:

QID: 09-1 NRO35		
Question # / Answer	**35**	**Developer/Date: NTP 12/17/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
204000 RWCU **A3.05 - Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: Reactor water temperature**						**2.8**	**2.8**
Level	**RO**		**Tier**	**2**	**Group**	**2**	
General References	**RAP-D8b**		**EMG-SP1**		**148F444, sh. 1**		
Explanation	The plant was at power when an event occurred which resulted in the alarm provided. When this alarms activates, it signals that V-16-1, V-16-2 and V-126-14 will auto close. Answer B is correct. All 4 valves will auto close on a Drywell high pressure, a RPV lo-lo water level, or a sensed RWCY line break. All other answers are plausible but incorrect.						
References to be provided during exam:		**None**					
Learning Objective	2621.828.0.0039 LO 204-10444						

Question Source (New, Modified, Bank)				**Bank**
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**		**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading CLEANUP SYSTEM	D-8-b

NRHX OUTLET TEMP HI	

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** high system temperature. (RK05, TIS-IJ33) [ ]

AUTOMATIC ACTIONS:

Isolation of:

V-16-1, CU Inlet Isolation Valve From Reactor Vessel
V-16-2, Inlet Isolation Valve To Cleanup Auxiliary Pump
V-16-14, Clean-Up Inlet Isolation Valve

AND

Trip of Cleanup Recirc. Pump

OR

Trip of Cleanup Aux Pump

MANUAL CORRECTIVE ACTIONS:

- ❑ **REDUCE** letdown flow. [ ]
- ❑ **CHECK** RBCCW flow and temperature to the Non-Regen Heat Exchanger. [ ]
- ❑ **CORRECT** problem. [ ]
- ❑ **RETURN** system to normal operation in accordance with Procedure 303, Reactor Cleanup Demineralizer System. [ ]
- ❑ **CHECK** condition of Cleanup Filter in service. [ ]
- ❑ **DIRECT** Chemistry to sample Cleanup Demin effluent. [ ]

36 **ID: 09-1 NRO36** **Points: 1.00**

The plant was at rated power with CRD Pump A in service.

The Operator then makes the following report:

- CRD Pump A red and green lights are de-energized
- DRIVE WATER FLOW indicates about 0.25 GPM
- CLG WTR REACTOR ΔP indicates 20 psid

Which of the following states the cause of the given indications?

A. CRD Pump A has tripped.

B. The Drive Water PCV has failed closed.

C. The Cooling Water PCV has failed open.

D. CRD Pump A has lost breaker control power.

Answer: D

Answer Explanation:

QID: 09-1 NRO36		
Question # / Answer	36	Developer/Date: NTP 12/17/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
201001 CRD Hydraulic A4.01 - Ability to manually operate and/or monitor in the control room: CRD pumps					3.1	3.1
Level	RO	Tier	2	Group	2	
General References	BR E1132	GE 116B8328 sh. 11A		302.2		

Explanation	The plant is at power with the indications provided. The breaker power for CRD Pump A is DC, which also provides power to the red/green lights. The loss of DC does not impact the status of the breaker (remains closed). Answer D is correct. If the pump had tripped, then cooling water ΔP would lower. The value given is a normal value Answer A is incorrect. If the candidate thought that cooling water ΔP was larger (or confused with drive water ΔP of 250 psid), then both answers B and C could account for the perceived lower cooling water pressure. But this is a normal value, and it would not account for loss of breaker indication of the pump. Answers B & C are incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2624.828.0.0012 LO 263-10453	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

BR E1132, Re 5, ELEMENTARY DIAGRAM 1A2-U033C CONTROL ROD DRIVE (CRD) (t magenation

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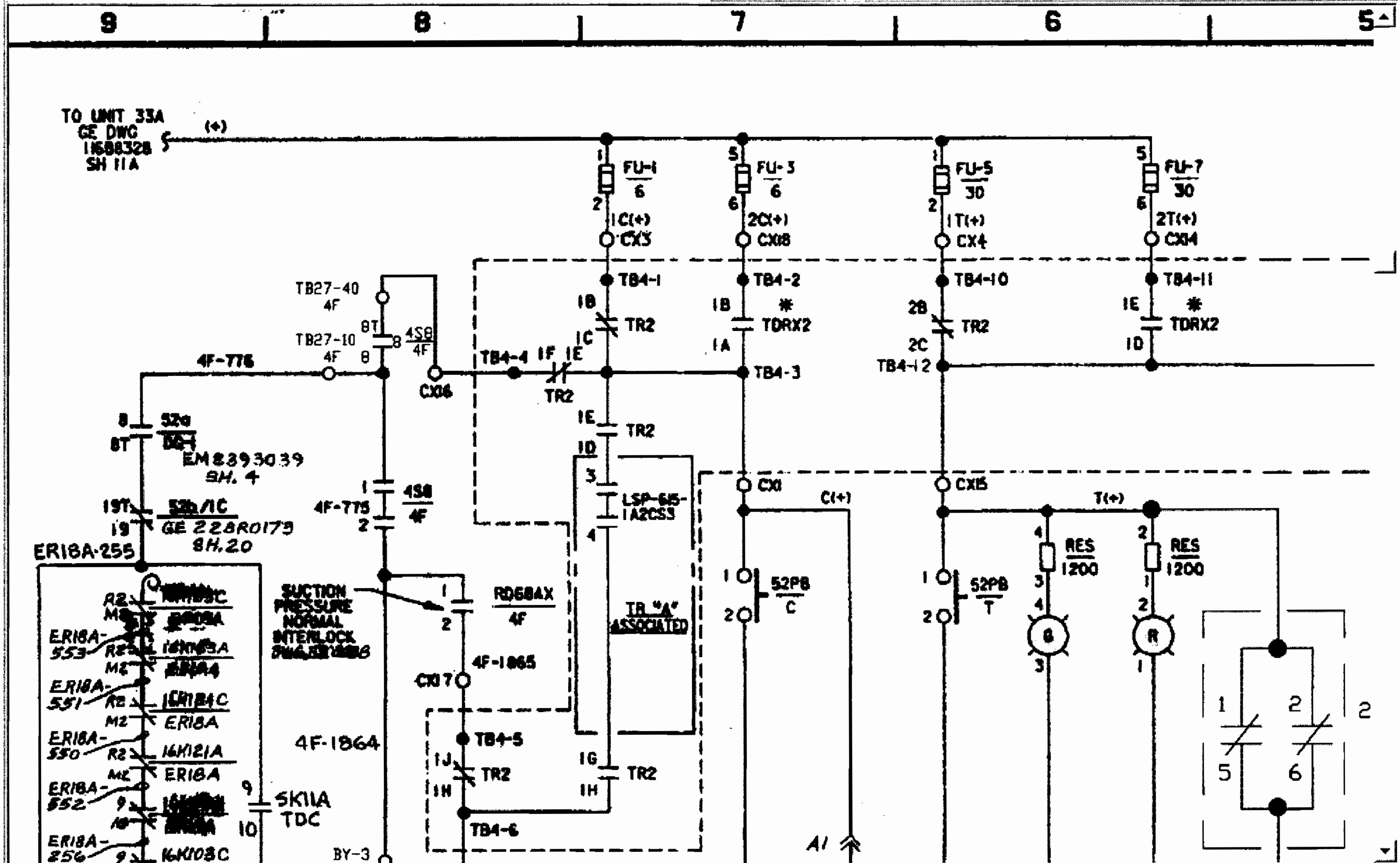
BR E1132, Rev 015, ELEMENTARY DIAGRAM 1A2-U033C CONTROL ROD DRIVE (CRD) (tif)



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GE 116B8328, Sheet 11A, Rev 018, CONTAINMENT SPRAY SYSTEM ELECTRICAL ELE (tif) agenation

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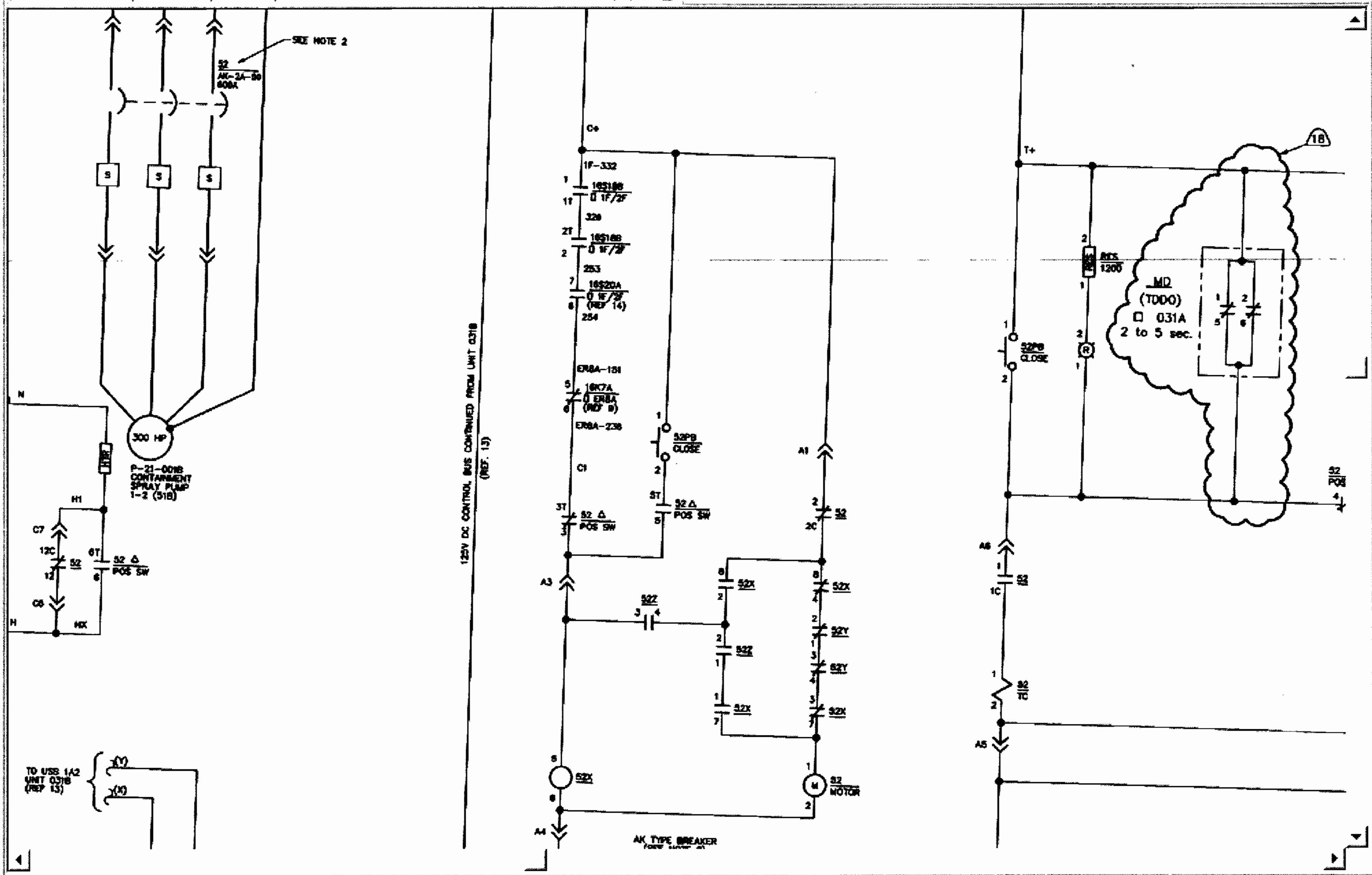
GE 116B8328, Sheet 11A, Rev 018, CONTAINMENT SPRAY SYSTEM ELECTRICAL ELE (tif)



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37 **ID: 09-1 NRO37** **Points: 1.00**

The plant was at rated power when the Shift Manager declared that a Control Room evacuation was required, and ABN-30, Control Room Evacuation, was entered.

The Operator only had enough time to scram the reactor and verify all control rods inserted, when **all** operators left the Control Room.

IAW ABN-30, which of the following states the required action outside of the Control Room and the associated plant impact of the action?

	Required Action	Plant Impact
A.	Close the MSIVs by isolating the air supply locally	The use of Isolation Condenser B for RPV pressure control at the RSP may be required
B.	Trip and lockout 4160 VAC 1D Main Breaker	Will require starting EDG 2 at the LSP DG2
C.	Trip all Condensate Transfer Pump breakers locally	May require controlling Condensate Transfer Pump at LSP 1B32
D.	Trip and lockout all Feedwater Pump breakers	CRD injection with CRD Pump B at the RSP may be required

Answer: D

Answer Explanation:

QID: 09-1 NRO37		
Question # / Answer	**37**	**Developer/Date: NTP 12/18/09**

Knowledge and Ability Reference Information	
K&A	**Importance Rating**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

		RO	SRO
	259001 Reactor Feedwater 2.4.34 - Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	4.2	4.1

Level	**RO**	**Tier**	**2**	**Group**	**2**

General References	**ABN-30**		
Explanation	The plant was at rated power when a control room evacuation was required and ABN-30 was entered. The following actions are attempted, if possible, prior to leaving the control room: Critical steps include: scram, trip recirculation pumps, close MSIVs, and trip all feedwater pumps. There are many other non-critical steps as well. If these critical steps cannot be performed prior to leaving the control room, Attachment ABN-30-1 provides direction on how to perform these actions in the plant. From what is given, the reactor was scrammed prior to leaving the control. The ABN does require tripping the feedwater pumps at the breakers. With a loss of high pressure injection now gone, using CRD Pump B from the RSP may be required. Answer D is correct. Closing the MSIVs is a critical step in the ABN, but the listed method to close the MSIVs is incorrect. The plant impact is also correct. Answer A is incorrect. Trip and lockout of breaker 1D is mentioned in the ABN, but is only required if no offsite power is available. The conditions stipulate that a successful scram was performed and can be assumed that offsite power is available. Therefore, answer B is incorrect. The ABN does not require tripping of condensate transfer pumps although it does require tripping of condensate pumps (non-critical step). Controlling the condensate transfer pump at the LSP is correct. But, answer D is incorrect since it refers to condensate transfer pumps and not condensate.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0017 LO 259-10445		

Question Source (New, Modified, Bank)	**New**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

<table>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td>X
1:P</td><td>Comprehension or Analysis</td><td></td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Procedure steps and cautions</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td>10</td><td>55.43</td><td></td></tr>
<tr><td colspan="4">(SRO Only)</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
</table>

4.2

NOTE

Attachment ABN-30-1, Backup Methods for Critical Functions, provides instruction to accomplish these actions from outside the Control Room if they **cannot** be completed prior to evacuation.

BEFORE or immediately after evacuating the Control Room,,

THEN **EXECUTE** the following steps:

- **SCRAM** the Reactor and **CONFIRM** all control rods are inserted to or beyond position 04. [ ]

CAUTION

At least one recirculation loop discharge and its associated suction valve shall remain open to ensure adequate communication between the vessel annulus and core region.

- **TRIP** all operating Reactor Recirculation pumps (3F):

  P-37-1, 'A' Reactor Recirculation Pump (NG01A) [ ]

  P-37-2, 'B' Reactor Recirculation Pump (NG01B) [ ]

  P-37-3, 'C' Reactor Recirculation Pump (NG01C) [ ]

  P-37-4, 'D' Reactor Recirculation Pump (NG01D) [ ]

  P-37-5, 'E' Reactor Recirculation Pump (NG01E) [ ]

- **CLOSE** the Main Steam Isolation Valves (MSIVs) (11F)

  V-1-7, 'A' Main Steam Line Outlet Isolation Valve (NS03A) [ ]

  V-1-8, 'B' Main Steam Line Outlet Isolation Valve (NS03B) [ ]

  V-1-9, 'A' Main Steam Line Outlet Isolation Valve (NS04A) [ ]

  V-1-10, 'B' Main Steam Line Outlet Isolation Valve (NS04B) [ ]

- **TRIP** all operating Reactor Feedwater pumps (5F/6F)

  P-2-2A, 'A' Reactor Feedwater Pump [ ]

  P-2-2B, 'B' Reactor Feedwater Pump [ ]

  P-2-2C, 'C' Reactor Feedwater Pump [ ]

4.3 IF time and plant conditions permit,

THEN **EXECUTE** the following actions prior or concurrent to evacuation, as directed by US. [ ]

OTHERWISE

PROCEED directly to step 4.4. [ ]

4.3.1 **TRIP** the Main Turbine by depressing pushbuttons on 7F. [ ]

4.3.2 IF the Main Generator did **not** trip,

THEN **OPEN** the following generator output breakers on Panel 12F-1:

- GC1 [ ]
- GD1 [ ]

4.3.3 **CONFIRM** Closed V-567-5, HWC Hydrogen Isolation Valve (5F/6F), indicating feedwater hydrogen injection is isolated. [ ]

4.3.4 **CONFIRM** plant electrical power transferred to startup transformers, as indicated by output breakers closed and transformers loaded:

- SA, Start-Up Transformer 'A' (8F/9F) [ ]
- SB, Start-Up Transformer 'B' (8F/9F). [ ]

4.3.10 **ISOLATE** the RWCU system by closing the following valves (1F/2F):

- V-16-1, CU Inlet Isolation Valve from Reactor Vessel [ ]
- V-16-2, Inlet Isolation Valve to Cleanup Auxiliary Pump [ ]
- V-16-14, Clean-Up Inlet Isolation Valve [ ]
- V-16-61. Regenerative Ht Exchanger Outlet to Reactor Vessel [ ]

4.4 <u>BEFORE</u> evacuating the Control Room,

<u>THEN</u> **OBTAIN** the following keys and radios from the SM office.

- OPS SPARE KEY RING (# 132) [ ]
- FIRE SAFE SHUTDOWN KEY RING # 1 (# 133) [ ]
- FIRE SAFE SHUTDOWN KEY RING # 2 (# 134) [ ]
- VITAL AREA ACCESS (2 keys) (# 121 and # 122) [ ]
- Portable Radios [ ]

4.5 **EVACUATE** the Control Room. The US will ensure all personnel have safely exited prior to leaving the Control Room. [ ]

4.5.1 **DIRECT** Shift Manager and STA to proceed to Technical Support Center (TSC), unless otherwise directed by US. [ ]

4.5.2 **DIRECT** remaining personnel to report to Remote Shutdown Panel (RSP), unless otherwise directed by US. [ ]

4.6 **PERFORM** any critical functions <u>**not**</u> completed in Step 4.2, in accordance with Attachment ABN-30-1, Backup Methods for Critical Functions. [ ]

4.7 **NOTIFY** Security (ext. 4957) of the Control Room evacuation. [ ]

4.8 **REQUEST** a Radiation Protection Technician be dispatched with appropriate monitoring equipment to the RSP in the B 480V Room. [ ]

ATTACHMENT ABN-30-1
(continued)

BACKUP METHODS FOR CRITICAL FUNCTIONS

	Intended Action	Backup method	
2.	Trip the Reactor Recirculation pumps	Manually **TRIP** and **LOCKOUT** (using the 69 Permissive Switch) the following Reactor Recirculation Pump Breakers:	
		• A Recirculation Pump, P-37-1 at 4160V 1A Bus, Unit A9	[ ]
		• B Recirculation Pump, P-37-2 at 4160V 1B Bus, Unit B4	[ ]
		• C Recirculation Pump, P-37-3 at 4160V 1A Bus, Unit A5	[ ]
		• D Recirculation Pump, P-37-4 at 4160V 1B Bus, Unit B8	[ ]
		• E Recirculation Pump, P-37-5 at 4160V 1A Bus, Unit A3	[ ]
3.	Trip the Reactor Feed Water pumps	Manually **TRIP** and **LOCKOUT** (using the 69 Permissive Switch) the following Feed Water pump breakers:	
		• A Feed Water Pump, P-2-2A at 4160V 1A Bus, Unit A8	[ ]
		• B Feed Water Pump, P-2-2B at 4160V 1B Bus, Unit B2	[ ]
		• C Feed Water Pump, P-2-2C at 4160V 1B Bus, Unit B10	[ ]

ATTACHMENT ABN-30-2

REMOTE AND LOCAL SHUTDOWN EQUIPMENT

> NOTE
>
> REFER TO OPERATING PROCEDURE 346 FOR DETAILS ON OPERATION.

Panel	Equipment/Instrumentation	Location
LSP-DG2	EDG No. 2 Control Transfer switches (3)	#2 EDG Vault
LSP-1D	Feeder Breaker to USS 1B2 (1B2P) Feeder Breaker to USS 1B3 (1B3P)	'D' 4160V Swgr. Room
RSP	IC 'B' Shell WTR LVL Indicator	'B' 480V Swgr. Room

Reactor Pressure indicators (2)

Fuel Zone Level Indicators (2)

IC 'B' Vent Valves V-14-1 and 19

IC 'B' DC Valves V-14-33 and 35

IC 'B' AC Valves V-14-32 and 37

IC 'B' Shell Water Makeup Valve V-11-34

Condensate Transfer AOV Valve for ICMU V-11-257

RBCCW pump 1-2

IF RBCCW Pump 1-2 breaker will **not** close from the RSP,
THEN manually **CLOSE** breaker at USS 1B2.

IF RBCCW Pump 1-2 trips spuriously,
THEN **REMOVE** trip fuses at USS 1B2 breaker cubicle and re-close breaker.

Shutdown Cooling Pump NU02B

CRD Pump NC08B

Main Breaker to USS 1B2 (1B2M)

EF1-20 (A/B battery room) exhaust fan
SF1-21 & EF1-21 'B' 480V SWGR room ventilation fans.

38 **ID: 09-1 NRO38** **Points: 1.00**

The plant was at 95% power. The Operator had just completed control rod withdrawals IAW the ReMA. The PPC Reactor Core State Parameters **prior to** and 15 minutes **after** the control rod manipulations are shown below:

Prior to control rod manipulations:

Thermal Limit	Fraction of Limit
MFLCPR	0.899
MFLPD	0.910
MAPRAT	0.917

After control rod manipulations:

Thermal Limit	Fraction of Limit
MFLCPR	0.995
MFLPD	1.002
MAPRAT	0.976

Which of the following states the potential impact on the nuclear fuel, and the **most restrictive** required action under the given conditions, IAW 202.1, Power Operations?

	Fuel Impact	**Required Action**
A.	Failures from transition boiling	Contact US, RE and monitor trend
B.	Failures from transition boiling	Follow TS 3.10, Notify RE Manager and Director Operations
C.	Failures from fuel pellet expansion	Follow TS 3.10, Notify RE Manager and Director Operations
D.	Failures from fuel pellet expansion	Contact US, RE, and monitor trend

Answer: C

Answer Explanation:

QID: 09-1 NRO38		
Question # / Answer	38	Developer/Date: NTP 12/18/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
290002 Reactor Vessel Internals A2.05 - Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Exceeding thermal limits					3.7	4.2
Level	RO	Tier	2	Group	2	
General References	202.1		GFES Thermo Chapter 9			

Explanation	The plant is at 95% power with control rod withdrawals just completed. The before/after reactor core state parameters data is provided. The before data shows that all thermal limits are within allowable limits. The after data shows that MFLPD is > 1.00. This would require the action listed in Column C of Attachment 202.1-5 and is the most restrictive action. Is can also be seen that MFLCPR is in violation of Column B of Attachment 202.1-5, but its action from the Attachment is less restrictive. IAW 202.1, Attachment 5, if MFLPD is > 0.98, then the action is to contact the US, RE and to monitor. When > 0.99, then restore to within the limits. When > 1.00, then TS 3.10 must be applied (which requires restoring back below the limit)rations. Operating the reactor with MFLPD > 1.00 can result in fuel failures due to fuel pellet expansion. Answer C is correct. Answers A is incorrect since these fuel failures can result from exceeding CPR limits, which are in violation of Column B in the Attachment, but the action in Column B is less restrictive than Column C in the correct answer. Answer B is incorrect since the transition boiling thermal limit MFLCPR does not violate the given action, which is the response when Column C is violated. Currently, MFLCPR only violates Column A & B. Answer D fuel impact may occur and shows that Column A limit for MAPRAT is exceeded , but the action is less limiting than the correct answer C. Answer D is incorrect.	
References to be provided during exam:	**202.1 Attachment 5**	
Learning Objective	2621.850.0.0090 LO 1520	

Question Source (New, Modified, Bank)				**New**
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPR**
	NUREG 1021 Appendix B: Solve a problem with references			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

ATTACHMENT 202.1-5 (cont'd)
CORE THERMAL LIMITS AND ACTIO5NS

MFLPD **LIMITS AND ACTIONS** **[With Reactor Power >25%]** **PLANT CONDITION:**	**(Column A)** **Contact US AND Reactor Engineer AND Monitor Thermal Limit Trend**[1]	**(Column B)** **Take Immediate Action To Restore Operation Within Limits (e.g., Inserting Control Rods OR Reducing Core Flow)**[2]	**(Column C)** **Follow Requirements of TS Section 3.10, As Applicable; Notify Mgr, Reactor Engineering AND Director, Operations**
Normal	> 0.98	> 0.99	> 1.00[3]

THERMAL LIMITS

PURPOSE	RADIOACTIVE RELEASE FROM THE PLANT WITHIN LIMITS		
FAILURE MECHANISM	FUEL CLADDING CRACKING DUE TO HIGH STRESS	GROSS CLADDING FAILURE DUE TO LACK OF COOLING	FUEL CLADDING CRACKING DUE TO LOSS OF COOLING
CAUSE OF FAILURE	FUEL PELLET EXPANSION	DECAY HEAT AND STORED HEAT FOLLOWING LOCA	LOSS OF NUCLEATE BOILING AROUND CLADDING
LIMITING CONDITION	1% PLASTIC STRAIN ON CLADDING	CLAD TEMPERATURE 2200°F	BOILING TRANSITION
ITEM MEASURED	LOCAL FUEL PIN POWER IN NODE	AVERAGE FUEL PIN POWER IN NODE	TOTAL FUEL BUNDLE POWER
LIMITING OPERATION	FULL POWER	LOCA	TRANIENT OPERATION
LIMITING PARAMETER	LHGR	APLHGR	CPR
CALCULATED PARAMETER	FLPD < 1.0	MAPRAT < 1.0	MFLCPR < 1.0

Fig 9-10

39 **ID: 09-1 NRO39** **Points: 1.00**

The plant was at rated power when an event occurred. The Operator reports the following observations:

- RPV pressure peaked at 1380 psig for 5 seconds, then quickly lowered to between 1000 - 1100 psig
- RPV water level lowered to 78" for 5 seconds then quickly restored to above 100"

Which of the following is correct regarding Safety Limits as defined in Tech Specs and the current plant status?

	Safety Limit	Plant Status
A.	• The Reactor Coolant System pressure safety Limit was exceeded	• The Isolation Condensers are in service • The Core Spray System has initiated
B.	• The Reactor Coolant System Pressure Safety Limit was exceeded	• The Isolation Condensers are in service • The Core Spray System has initiated and is injecting
C.	• The Fuel Cladding Integrity Safety Limit has been exceeded	• **All** EMRVs and SRVs have opened and are now closed
D.	• The Fuel Cladding Integrity Safety Limit has been exceeded	• The Core Spray System has initiated • Both EDGs have idle started

Answer: A

Answer Explanation:

QID: 09-1 NRO39

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Question # / Answer	39	Developer/Date: NTP 12/18/09

Knowledge and Ability Reference Information						
K&A					Importance Rating RO	Importance Rating SRO
295025 High Reactor Pressure EK1.05 - Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : Exceeding safety limits					4.4	4.7
Level	RO	Tier	1	Group	1	

General References	TS 2.1	TS 2.2	EMG-SP1
Explanation	The plant was at power when an event resulted in high RPV pressure and low RPV water level. The fuel cladding integrity safety limit for RPV water level is no less than 4'8" above the fuel, or 56". The RPV pressure safety limit is no greater than 1375 psig with fuel in the vessel. Therefore, the pressure safety limit has been exceeded. The isolation Condensers auto initiate on 1050 psig or RPV water level 90", and thus are in service. Core Spray auto initiates at 86" RPV water level or Drywell pressure of 3 psig. Thus, the Core Spray System is in service, although it will not inject until RPV pressure drops to 305 psig. Therefore, the RPV pressure safety limit has been exceeded and core spray has initiated and isolation condensers are in service. Answer A is correct. Because RPV pressure is greater than the core spray injection setpoint, core spray is not injecting. Answer B is incorrect. Both answers C & D are incorrect since the fuel clad integrity safety limit has not been exceeded. Both plant statuses are correct.		
References to be provided during exam:	None		
Learning Objective	2621.850.0.0090 LO 1658		

Question Source (New, Modified, Bank)	New

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK
	NUREG 1021 Appendix B: Solve a problem using			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

2.2 SAFETY LIMIT - REACTOR COOLANT SYSTEM PRESSURE

Applicability: Applies to the limit on reactor coolant system pressure.

Objective: Preserve the integrity of the reactor coolant system.

Specification: The reactor coolant system pressure shall not exceed 1375 psig whenever irradiated fuel is in the reactor vessel.

Bases:

The reactor coolant system(1) represents an important barrier in the prevention of the uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1375 psig was derived from the design pressures of the reactor pressure vessel, coolant piping, and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1200 psig at 570°F and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section I for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III for the isolation condenser and the ASA Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10% over

SYSTEM	OPERATING DETAILS
Cleanup System Isolation	<u>IF</u> Any of the following conditions exist: • RPV water level at or below **86 in.** and **<u>not</u>** bypassed • Drywell pressure at or above **3.0 psig** and **<u>not</u>** bypassed • RWCU HELB Alarms <u>THEN</u> **CONFIRM** closed the following Cleanup Isolation valves: (Panel 3F/11F) V-16-1 [ ] V-16-14 [ ] V-16-2 [ ] V-16-61 [ ]
Shutdown Cooling System Isolation	<u>IF</u> Any of the following conditions exist: • RPV water level at or below **86 in.** • Drywell pressure at or above **3.0 psig** <u>THEN</u> **CONFIRM** closed the following SDC Isolation Valves: (Panel 11F) V-17-54 [ ] V-17-19 [ ]
Isolation Condenser Initiation	<u>IF</u> Any of the following conditions exist or have occurred: • RPV water level at or below **86 in.** • Reactor pressure at or above **1050 psig.** <u>THEN</u> **CONFIRM** that both Isolation Condensers did initiate. (ICs may have been removed from service by Pressure Control Leg.) [ ]

OVER

SYSTEM	OPERATING DETAILS
Core Spray System Start	IF Any of the following conditions exist: • RPV water level at or below **86 in.** and **not** bypassed. • Drywell pressure at or above **3.0 psig** and **not** bypassed. AND Core Spray is **not** defeated per EOPs, THEN **CONFIRM** the following: (Panel 1F/2F) • Start of one Main Pump in each system. [ ] • At least one Booster Pump running. [ ]
Primary Containment Isolation	IF Any of the following conditions exist: • RPV water level at or below **86 in.** and **not** bypassed. • Drywell pressure at or above **3.0 psig** and **not** bypassed. THEN **CONFIRM** closed the following valves that are **not** required to be open by the Emergency Operating Procedures: System — Valve No. DW Vent/Purge (Panel 11F) — V-27-1 [ ] V-27-2 [ ] V-27-3 [ ] V-27-4 [ ] Torus Vent (Panel 11F) — V-28-17 [ ] V-28-18 [ ] (continued)

40 **ID: 09-1 NRO40** **Points: 1.00**

The plant was at rated power when an earthquake occurred. The Operator makes the following report:

- RPV water level is 80" and lowering slowly
- Breaker S1B indicates open
- A fire alarm has alarmed on Main Fire Panel A

A report is received of a fire in the RB NW Corner Room and the Fire Brigade is dispatched. The Fire Brigade Leader reports that the Fire Brigade actions will impact the operation of **all** pumps located in the NW corner room.

Which of the following states the RPV injection sources currently available and **not** impacted by the fire?

A.
- Feedwater Pumps B & C
- Core Spray Main Pumps A & B

B.
- Feedwater Pump A
- Core Spray Main Pumps B & D

C.
- Feedwater Pump A & B
- SLC Pumps A & B

D.
- Feedwater Pump A
- Core Spray Main Pumps B & C
- CRD Pumps A & B

Answer: B

Answer Explanation:

QID: 09-1 NRO40		
Question # / Answer	**40**	**Developer/Date: NTP 12/19/09**

Knowledge and Ability Reference Information	
K&A	**Importance Rating**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

	RO	SRO
600000 Plant Fire On-site **AK1.02 - Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire fighting**	**2.9**	**3.1**

Level	**RO**	**Tier**	**1**	**Group**	**1**

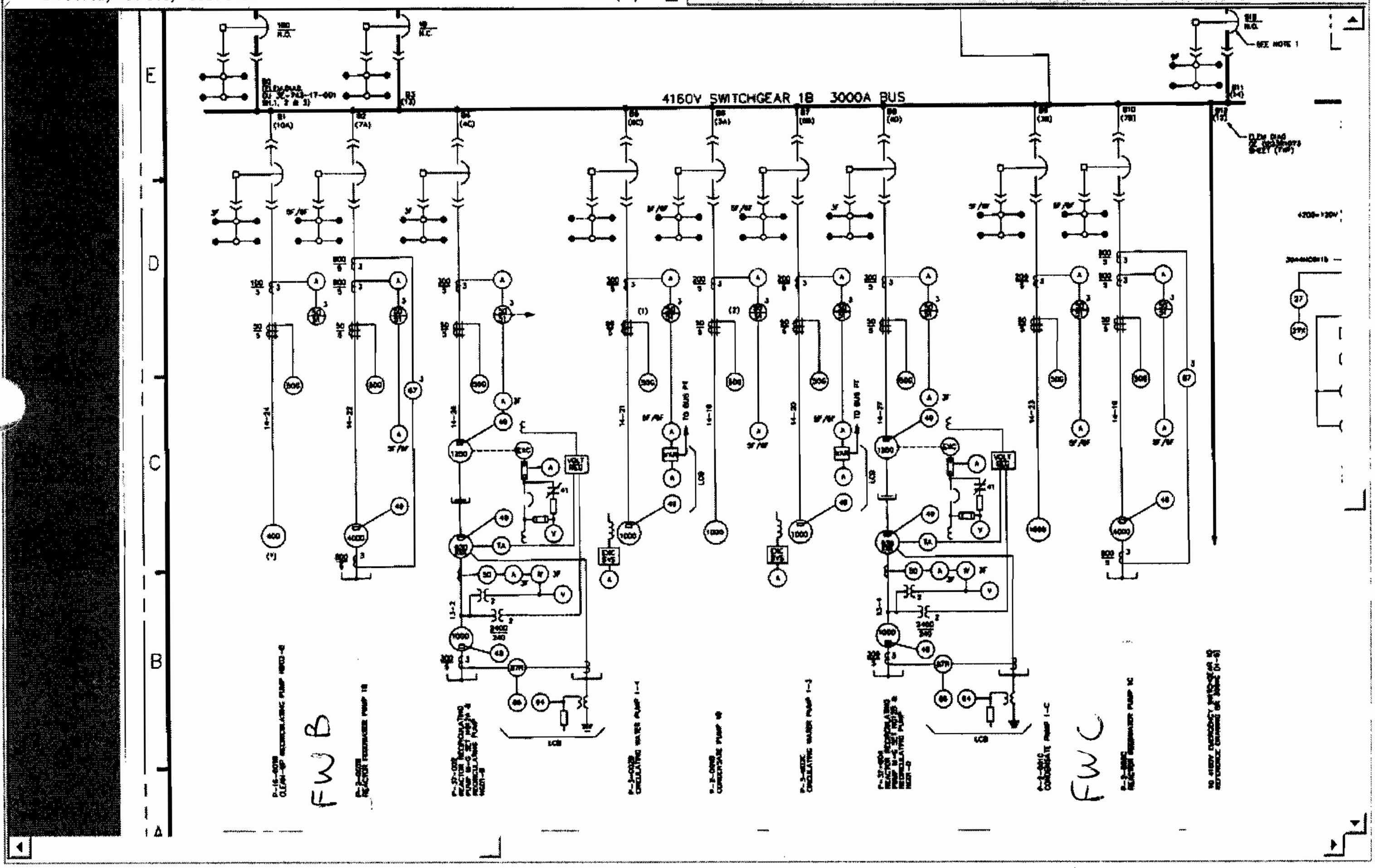
General References	**3E-153-02-001**	**BR 3001B**	
Explanation	The plant was at rated power when an earthquake occurred. This resulted in a low RPV water level, the loss of startup transformer 1B and a fire in the plant. The fire has been confirmed to impact all equipment in the RB NW corner room. This room contains both CRD pumps, and Core Spray main pumps A & C. Thus, this equipment is not available for RPV injection. Feedwater Pump A and Core Sprays B & D are available, both electrically and not impacted by the fire. Answer B is correct. Answer A is incorrect since Feedwater Pumps B & C are powered from Bus 1B, which has no power. Answer C is incorrect since it lists Feedwater Pump B. Answer D is incorrect since it lists the incorrect Core Spray pumps and it lists CRD pumps.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0010 LO 209-10445		

Question Source (New, Modified, Bank)		**New**		
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:S**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Structures and locations			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

BR 3001B, Rev 016, 4160V SYSTEM ONE LINE DIAGRAM 4160V SWITCHGEAR BUS 1B (tif) - Imagenation

File Edit View Page Layer Preference Window Help

BR 3001B, Rev 016, 4160V SYSTEM ONE LINE DIAGRAM 4160V SWITCHGEAR BUS 1B (tif)



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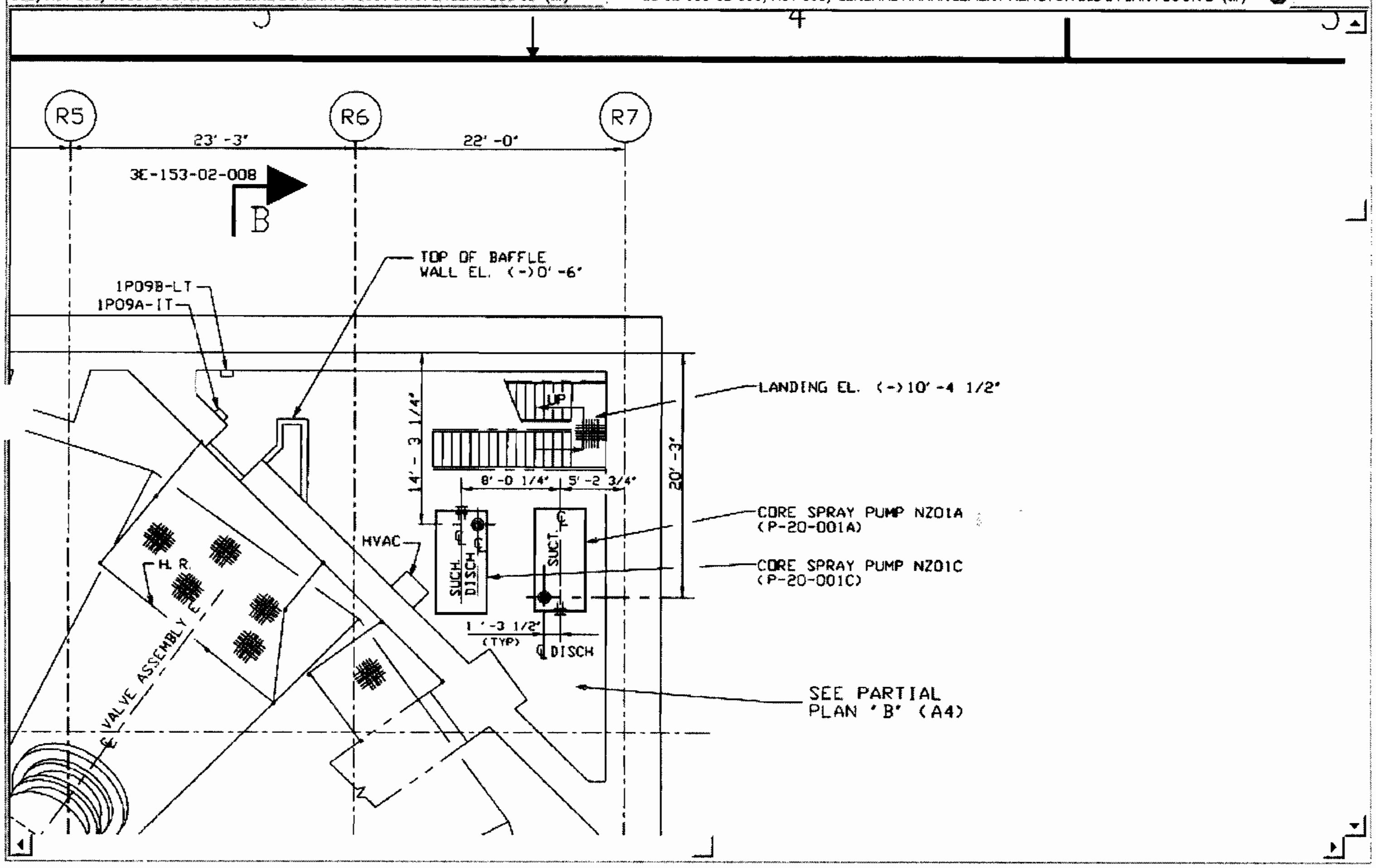
GU 3E-153-02-001, Rev 008, GENERAL ARRANGEMENT REACTOR BLDG PLAN FLOOR E (tif) - Imagenation

File Edit View Page Layer Preference Window Help

I01B, Rev 016, 4160V SYSTEM ONE LINE DIAGRAM 4160V SWITCHGEAR BUS 1B (tif) | GU 3E-153-02-001, Rev 008, GENERAL ARRANGEMENT REACTOR BLDG PLAN FLOOR E (tif)

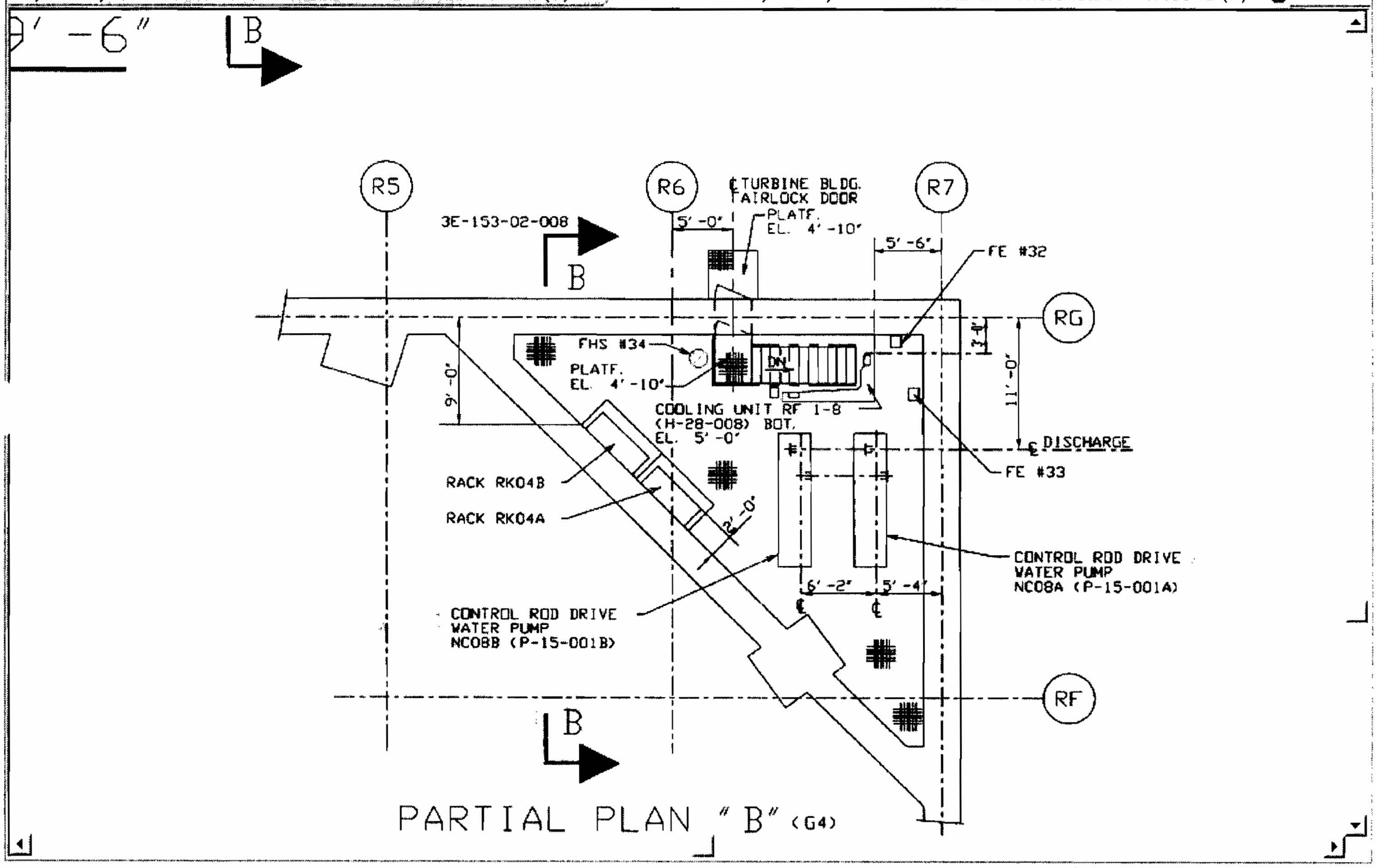


Coord: (11192,2010) viewing

GU 3E-153-02-001, Rev 008, GENERAL ARRANGEMENT REACTOR BLDG PLAN FLOOR E (tif) - Imagenation

File Edit View Page Layer Preference Window Help

I01B, Rev 016, 4160V SYSTEM ONE LINE DIAGRAM 4160V SWITCHGEAR BUS 1B (tif) | GU 3E-153-02-001, Rev 008, GENERAL ARRANGEMENT REACTOR BLDG PLAN FLOOR E (tif)



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41 **ID: 09-1 NRO41** **Points: 1.00**

The plant was shutdown and was cooling down with Shutdown Cooling (SDC) Pumps A and B, when the following annunciator alarmed:

- 1A2 DC LOST

Which of the following states the affect on Shutdown Cooling?

A. SDC Pump A is **only** able to be tripped from the LSP.

B. SDC Pump B is **unable** to trip on high suction temperature.

C. SDC Pump A is **unable** to trip on high suction temperature.

D. The RPV cooldown rate can **not** be adjusted from the Control Room.

Answer: C

Answer Explanation:

QID: 09-1 NRO41		
Question # / Answer	**41**	**Developer/Date: NTP 12/19/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
295004 Partial or Total Loss of DC Pwr AK1.05 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Loss of breaker protection				**3.3**	**3.4**
Level	**RO**	**Tier**	**1**	**Group**	**1**
General References	**BR E1129**	**RAP-U3d**			

Explanation	The plant was cooling down with SDC pumps A & B when DC power is indicated lost to USS Bus 1A2. USS 1A2 powers SDC Pump A. Therefore, DC control power is lost to SDC Pump A and not SDC Pump B. Since DC control power is lost, it cannot auto trip on high suction temperature. Answer C is correct. The same DC power is required for tripping when SDC Pump A is controlled from the LSP. Answer A is incorrect. SDC Pump B is powered from USS 1B2 which receives DC control Power from DC B and is not affected. Answer B is incorrect. The cooldown rate can still be adjusted from the control by several methods, one of which by tripping SDC Pump B. Answer D is incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0045 LO 205-10453	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:F**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Facts			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

1A2
DC LOST

CONFIRMATORY ACTIONS:

- ❑ **CHECK** Breaker position indication on USS 1A2 Supply and Load Breakers. [ ]
- ❑ **CHECK** loss of power to 125 VDC Distribution Center C. [ ]
- ❑ **CHECK** USS 1A2 DC Control Power Supply Breaker on 125 VDC Distribution Center C - Breaker #6. [ ]

AUTOMATIC ACTIONS:

Electrical tripping and closing functions for Breakers on USS 1A2 are defeated. Undervoltage protection is still available.

MANUAL CORRECTIVE ACTIONS:

- ❑ **INVESTIGATE** loss of DC control power. [ ]
- ❑ IF power has been lost to 125 VDC Distribution Center C,

  THEN **REFER** to ABN-55, DC Bus C and Panel/MCC Failures. [ ]
- ❑ **MAINTAIN** plant conditions constant. [ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 2)

42 **ID: 09-1 NRO42** **Points: 1.00**

The plant was at rated power when a leak developed in the Torus. A timeline of Torus water level is provided below (times are in minutes):

- T = 0 Torus water level indicated 145"
- T = 5 Torus water level indicated 140"
- T = 15 Torus water level indicated 130"

Which of the following represents the **soonest** that steam from a LOCA in the Primary Containment would **directly** pressurize the Torus air space?

A. T = 30 minutes

B. T = 36 minutes

C. T = 42 minutes

D. T = 56 minutes

Answer: B

Answer Explanation:

QID: 09-1 NRO42		
Question # / Answer	**42**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
295030 Low Suppression Pool Water Level EK2.07 - Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: Downcomer/ horizontal vent submergence				**3.5**	**3.8**
Level	**RO**	**Tier**	**1**	**Group**	**1**
General References	**EOP Users Guide**				

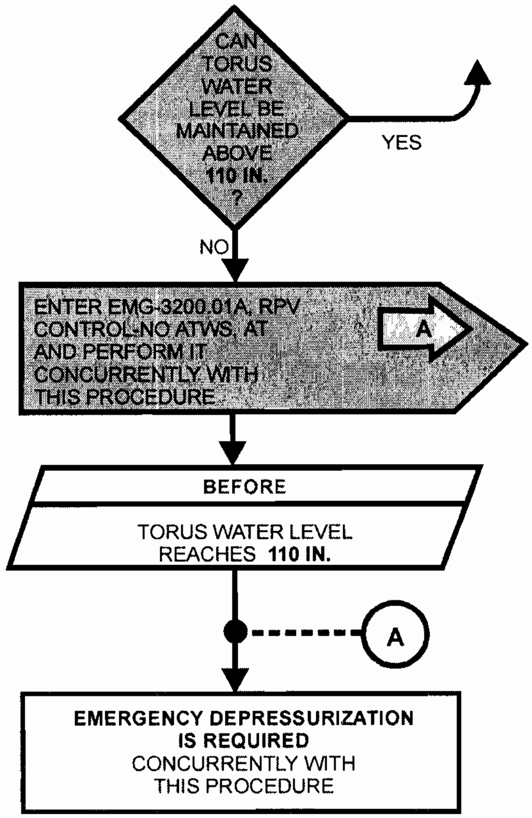
EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant was at rated power when a Torus water leak developed. The indications show that Torus water level is lowering at the rate of 1"/minute. IAW the reference, below 110", the Drywell vent header downcomer openings are uncovered and the pressure suppression function of the Primary Containment becomes inoperable. Steam discharged from a LOCA would exit the downcomers, bypass the water in the Torus and directly pressurize the Torus air space. Thus at an additional 20 minutes (or T = 35), Torus water level will be 110", and at T = 36 minutes, Torus water level will be less than 110". Answer b is correct. At T=30 minutes, the downcomers are still covered. Answer A is incorrect. At T=42 minutes, Torus water level is even further below 110", but it is not the soonest time that the downcomers become uncovered. Answer C is incorrect. At T = 56 minutes, Torus water level has past 90", which is the level that the EMRV discharge pipes become uncovered. Answer D is incorrect. All other answers are plausible but incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0032 LO 432

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

TORUS WATER LEVEL CONTROL



DISCUSSION

Below 110 in., the Drywell vent header downcomer openings are uncovered and the pressure suppression function of the Primary Containment becomes inoperable. Steam discharged from a LOCA would exit the downcomers, bypass the water in the Torus and directly pressurize the Torus airspace, a transient for which the Primary Containment is not designed. An Emergency RPV Depressurization is performed before 110 in. is reached, which transfers primary system energy to the Torus water to limit the consequences should a LOCA occur when Torus level drops below 110 in.

43 **ID: 09-1 NRO43** **Points: 1.00**

The plant was at rated power when a total loss of TBCCW occurred. The Operators performed **all** IMMEDIATE OPERATOR ACTIONs of ABN-1, Reactor Scram **and** ABN-20, TBCCW Failure Response.

One minute later, the Operator reports the following:

- All control rods indicate full-in
- RPV water level indicates 120" and lowering slowly
- TOTAL FEEDWATER FLOW indicates 6 x 10^6 LBS/HR
- RPV pressure indicates 920 psig and lowering slowly

Which of the following is the **next** annunciator to alarm under the given conditions that will require manual Operator actions IAW ABN-20?

A. GENERATOR - MN LEADS TEMP HI

B. RX RECIRC PUMP TRIP - MG BRG TEMP HI

C. CLEANUP SYSTEM - AUX PUMP CCW TEMP HI

D. FEED PUMPS - COND/FW PMP BRG TEMP HI

Answer: D

Answer Explanation:

QID: 09-1 NRO43		
Question # / Answer	**43**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295018 Partial or Total Loss of CCW AK2.01 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: System loads					**3.3**	**3.4**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**ABN-20**					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

<table>
<tr><td>Explanation</td><td colspan="2">The plant was at rated power when a total loss of TBCCW occurred. Immediate Operator Actions of ABN-1 include: scram the reactor, trip 2 feedwater pumps when RPV water level begins to rise, and to verify all control rods at ≤ position 04 and power is lowering. Immediate Operator Actions of ABN-20 include: scram the reactor IAW ABN-1, and to trip all recirculation pumps.
The stem states that RPV injection is 4×10^6 LBS/HR. Under these circumstances and those of ABN-1, 3 condensate pumps and 3 feedwater pumps are operating, with significant injection by feedwater. TBCCW cools the bearing of the condensate and feedwater pumps. With the feedwater/condensate pumps performing a lot of work with no cooling, their bearing temperatures will rise until the alarm point. ABN-20 will require further actions to prevent pump damage. Answer D is correct.
If the generator were on-line with no cooling provided by TBCCW, the generator leads temperature would rise. But since ABN-1 actions have been performed, the generator is no longer on-line and no further actions are required. Answer A is incorrect.
If TBCCW were lost to the recirculation MG sets, their temperatures would rise. But since ABN-20 actions have been performed, all recirculation MG sets are tripped when the recirculation pumps are tripped. There are no further actions. Answer B is incorrect.
At the given conditions, the aux. cleanup pump is not in service, and it is cooled by RBCCW. Answer C is incorrect.</td></tr>
<tr><td>References to be provided during exam:</td><td>None</td><td></td></tr>
<tr><td>Learning Objective</td><td colspan="2">2621.828.0.0048 LO 274-10437</td></tr>
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<tr><td colspan="4">Question Source (New, Modified, Bank)</td><td colspan="2">New</td></tr>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td colspan="2">X
3:PEO</td></tr>
<tr><td colspan="5">NUREG 1021 Appendix B: Predict an event or outcome</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td>7</td><td>55.43</td><td colspan="2"></td></tr>
<tr><td colspan="5">(SRO Only)</td></tr>
<tr><td colspan="6">Time to Complete: 1-2 minutes</td></tr>
</table>

4.9 **NOTE**

The indicated actions for the following systems, upon reaching their limits, may be performed in any order or concurrently.

- Feed and Condensate System — Step 4.9.1
- Stator Cooling Water — Step 4.9.2
- Turbine Lube Oil — Step 4.9.3
- Recirc MG Sets — Step 4.9.4

PERFORM the indicated actions for any of the following systems:

4.9.1 Feed and Condensate System

4.9.1.1 IF Condensate pump bearing temperature ≥ 185° F (J-8-f)

OR

C' Feed pump outer bearing temperature ≥ 195° F (J-8-f)

OR

Any other Feed pump bearing temperature ≥ 185° F (J-8-f),

THEN **MONITOR** bearing temperatures closely on Panel 12XR, Temperature Monitor 12XR-21. [ ]

4.9.1.2 IF All pump bearing temperatures ≥ 195° F, as indicated by Panel 12XR, Temperature Monitor 12XR-21 and therefore require all Feed and Condensate pumps to be shut down.

THEN **PERFORM** the following:

1. IF the reactor is in the STARTUP or RUN mode,

THEN **PERFORM** the following:

a. **CONFIRM** Feed pumps shutdown. [ ]

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

44 **ID: 09-1 NRO44** **Points: 1.00**

The plant was shutdown and was cooling down with the Shutdown Cooling System (SDC). Present plant conditions include the following:

- RPV water level indicates 160" and steady
- RECIRC PUMP SUCTION TEMPS indicates 340 °F
- Shutdown Cooling Pumps A and C are in service

Which of the following annunciators/indications will the **greatest** impact on the cooldown rate?

A. Annunciator 1A2 MN BRKR TRIP alarms.

B. Annunciator 1B2 MN BRKR TRIP alarms.

C. RECIRC PUMP SUCTION TEMPS rises to 351 °F.

D. SDC Pumps A and B suction pressure lowers to 10 psig.

Answer: C

Answer Explanation:

QID: 09-1 NRO44		
Question # / Answer	**44**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295021 Loss of Shutdown Cooling AK2.03 - Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: RHR/shutdown cooling					**3.6**	**3.6**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**305**					

Explanation	The plant is shutdown and cooling down with SDC pumps A & B. If recirculation loop temperatures exceed 350 °F, SDC isolation valves V-17-19 & V-17-54 auto close. When V-17-19 closes, then all SDC pumps trip. This results in a total loss of SDC and the greatest impact on the cooldown rate. Answer C is correct. SDC Pump A is powered from USS 1A2, and SDC pumps B & C are powered from USS 1B2. Thus a loss of either bus results in either the A or B & C pumps available, and also impacts the cooldown rate. Answers A & B are incorrect. The SDC pumps will trip on a low suction pressure of 4 psig, not 10 as stated in answer D. Answer D is incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0045 LO 205-10445	

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

4.2.11 To prevent SDC System flow from short-cycling the core, the E Recirc Loop Discharge Valve shall be CLOSED or the E Recirc Pump running.

4.2.12 If the Cleanup System is in service, the B Recirc Loop should **not** be the selected loop in those instances where one loop is required to be fully open.

4.2.13 Section 4.3 of this procedure is written to startup the SDC System in order to cooldown the Reactor. If system startup is to be done after cooldown, as when maintaining a temperature band during outages, those steps applicable only to startup for a cooldown may be omitted.

4.2.14 When initially placing the SDC System in service, monitor the RBCCW Pump suction temperature TI-541-9 (Pump 1-1) and TI-541-10 (Pump 1-2) closely to ensure the limits of Procedure 309.2 are **not** exceeded. RBCCW temperature out of the SDC System Heat Exchangers may initially be greater than 190°F, due to water already contained in the heat exchangers when the RBCCW outlet valve is opened, but shall be limited to less than 190°F once flow has stabilized.

4.2.15 The following trips are associated with SDC System operation:

- V-17-19 and V-17-54 will close if any Recirc Loop temperature exceeds 350°F.
- SDC Pumps will trip if individual pump suction pressure drops below 4 psig (with a 1.5 sec time delay).
- SDC Pumps will trip if the suction temperature for the pump rises above 350°F (with a 1.5 sec time delay).
- SDC Pumps will trip if V-17-19 closes.

4.2.16 The following valves are defined as Primary Containment Isolation Valves by Procedure 312.9 and FSAR Table 6.2-12 and are operated in this section:

- V-17-1
- V-17-2
- V-17-3
- V-17-55
- V-17-56
- V-17-57
- V-17-19
- V-17-54

4.2.17 **Do not** allow coolant temperature to go below 68 degrees F which is the lower limit of Shutdown Margin analysis.

45 **ID: 09-1 NRO45** **Points: 1.00**

The plant was at rated power when an event resulted in the following conditions:

- RPV water level indicates 0" and lowering slowly
- **No** RPV injection systems are available

The Steam Cooling EOP has been entered. Which of the following is correct?

IAW the EOP Users Guide, an RPV water level of ____(1)____, would still provide enough steam flow through the core to prevent exceeding ____(2)____ clad temperature.

	(1)	(2)
A.	-23"	1500 °F
B.	-28"	1500 °F
C.	-33"	1800 °F
D.	-38"	1800 °F

Answer: C

Answer Explanation:

QID: 09-1 NRO45		
Question # / Answer	45	Developer/Date: NTP 12/21/09

Knowledge and Ability Reference Information		
K&A	Importance Rating	
	RO	SRO
295031 Reactor Low Water Level EK3.04 - Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL : Steam cooling	4.0	4.3

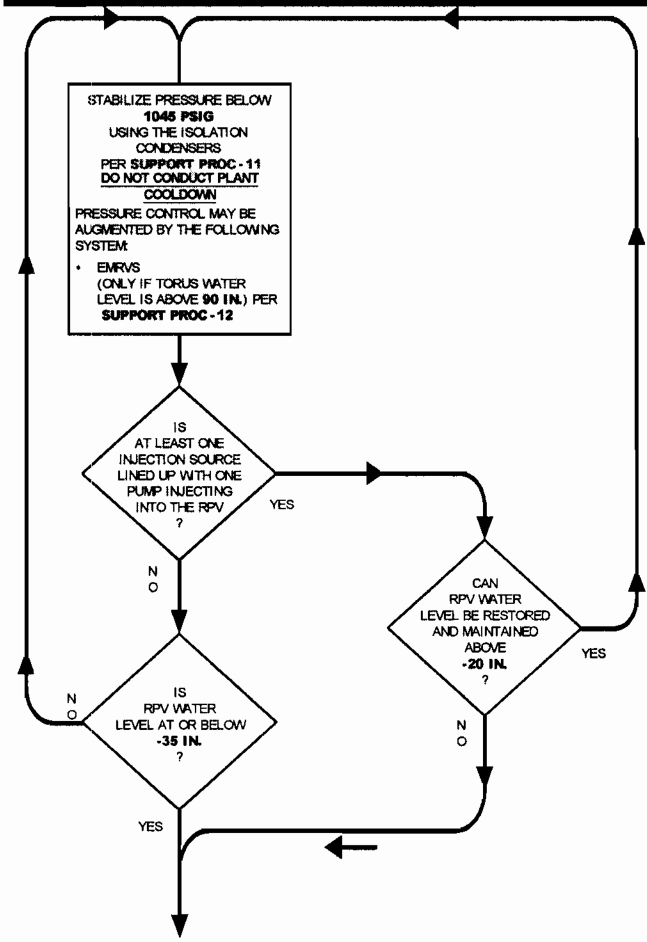
EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	1	Group	1
General References	**EOP Users Guide**				
Explanation	The Steam Cooling EOP has been entered. Core cooling is maintained from the steam passing the uncovered portions of the fuel by one of two mechanisms: injection into the RPV is available or injection is not available. If injection is available, as long as RPV water level is ≥ -20", then cladding temperature will remain ≤ 1500 °F. If no injection is available, as long as RPV water level is ≥ -35", then cladding temperature will remain ≤ 1800 °F. With no RPV injection, an RPV water level of -33" ensures clad temperature ≤ 1800 °F. Answer C is correct. Answers A & B are also above -35", but the temperature limit is incorrect. Answers A & B are incorrect. Answer D is less than -35" and is incorrect.				
References to be provided during exam:	**None**				
Learning Objective	2621.845.0.0055 LO 3004				

Question Source (New, Modified, Bank)			Modified	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:B**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Bases or purpose			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

STEAM COOLING



DISCUSSION

If no injection source can be reestablished, steam cooling is continued and RPV water level is allowed to decrease though boil off until it drops to -35 inches, the Minimum Zero Injection RPV Water Level (MZIWL). During this period the fuel temperature in the uncovered portion of the core increases, and heat is transferred from the fuel rods to the steam. The MZIWL is defined to be the lowest RPV water level at which the covered portion of the Reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1800°F. This water level/clad temperature relationship is applicable only if there is no injection (zero injection) into the RPV. Any injection would cause sub-cooling at the core inlet, reducing steam production.

If an injection source can be reestablished, steam cooling is continued and RPV water level is allowed to decrease through boil off until it drops to -20 inches, the Minimum Steam Cooling RPV Water Level (MSCWL). The MSCWL is defined to be the lowest level at which the covered portion of the core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Because the amount of sub-cooling cannot be determined, any injection is assumed to affect the amount of steam being generated in the covered portion of the core. This means that the Minimum Zero Injection RPV Water Level discussed above cannot be

46 **ID: 09-1 NRO46** **Points: 1.00**

The plant was at rated power when a manual scram was inserted due to elevated Drywell pressure. The Operator reports the following:

- Not all control rods inserted
- Reactor power indicates 55%
- MWe indicates 350

The RPV Control - With ATWS EOP directs confirming recirculation flow at minimum. IAW the EOP Users Guide, why is recirculation flow reduced instead of tripping of the Recirculation Pumps under these conditions?

A. This prevents an RPV water level shrink which could close the MSIVs.

B. This prevents an RPV water level swell which could trip the main turbine.

C. This prevents an RPV water level swell which could flood the main steam lines.

D. The Unit RO can perform this action while simultaneously inserting control rods.

Answer: B

Answer Explanation:

QID: 09-1 NRO46		
Question # / Answer	**46**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

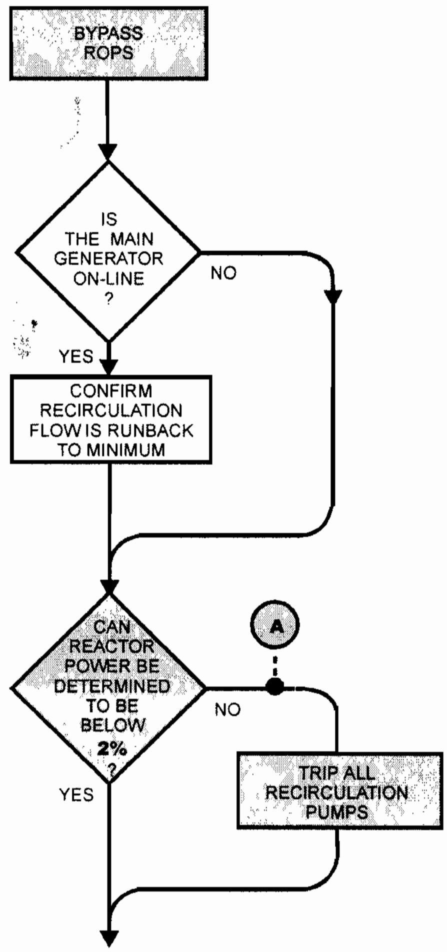
295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown **EK3.01 - Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Recirculation pump trip/runback**	**4.1**	**4.2**

Level	**RO**	**Tier**	**1**	**Group**	**1**

General References	**EOP Users Guide**		
Explanation	IAW the reference, the action is performed to prevent an RPV water level swell which could trip the main turbine. It is desirable to keep the turbine on-line under ATWS conditions for energy removal from the reactor because reactor power may exceed the capacity of the turbine bypass valves. All other answers are plausible but are incorrect IAW the reference. An RPV water level shrink to lo-lo would close the MSIVs, but Answer A is incorrect. Flooding the main steam lines is an undesirable condition as it may result in water hammer and steam line damage, but answer A is incorrect. While inserting control rods, the URO can reach the recirculation flow controller, but would have to stop inserting control rods to trip the recirculation Pumps. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.845.0.0053 LO 3055A		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:B**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Bases or purpose			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

POWER CONTROL



DISCUSSION

This question is asked to determine whether or not a runback of the Recirculation pumps will be required. Under ATWS conditions the Reactor Recirculation pumps are tripped to reduce Reactor power. If the Main Turbine is on-line, the Recirculation pump speeds are reduced prior to tripping them to prevent a large RPV level swell, or moisture separator drain tank level increase which could trip the Main Turbine.

It is desirable to keep the Main Turbine on-line under ATWS conditions for energy removal from the Reactor because Reactor power may exceed the capacity of the Bypass Valves. Under those conditions, a trip of the Main Turbine will result in a Reactor pressure increase and power spike. EMRV operation will occur as will the start of Primary Containment heat up. The Primary Containment is not designed for ATWS conditions.

If the answer to this question is YES (turbine still on-line), the operator will be directed to reduce Recirculation pump speed. The intention of these steps is to run back Recirculation pump speed to its minimum possible value as quickly as possible without causing a turbine trip on high level. The minimum flow for power operations is not applicable for this procedure.

If the answer to the question is NO (turbine is tripped or the MSIVs are closed), the operator will bypass the step directing Recirculation pump runback.

47 **ID: 09-1 NRO47** **Points: 1.00**

The plant was shutdown with fuel shuffling in progress on the refuel floor.

The fuel hoist was loaded with a new fuel bundle in the spent fuel pool and was to be inserted into the second core quadrant. The following annunciator then alarmed while the bridge was maneuvering in the Spent Fuel Pool:

- ROD DRIFT

Which of the following states the response of the refuel bridge under the conditions provided and the basis for the refuel bridge response?

	Refuel Bridge Response	Basis
A.	The refuel bridge will be prevented from being moved anywhere over the core	Prevents a large reactivity addition into the core
B.	The refuel bridge will experiences a bridge fault and **all** refuel bridge motion will be halted	Prevents a large reactivity addition into the core
C.	The refuel bridge will be prevented from being moved anywhere over the core	Prevent damage to the fuel bundle and the drifting control rod if both were inserted at the same time in the same fuel cell
D.	The refuel bridge will experiences a bridge fault and **all** refuel bridge motion will be halted	Prevent damage to the fuel bundle and the drifting control rod if both were inserted at the same time in the same fuel cell

Answer: A

Answer Explanation:

QID: 09-1 NRO47

Question # / Answer	47	Developer/Date: NTP 12/22/09

Knowledge and Ability Reference Information						
K&A					Importance Rating RO	Importance Rating SRO
295023 Refueling Acc Cooling Mode AK3.02 - Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS : Interlocks associated with fuel handling equipment					3.4	3.8
Level	RO	Tier	1	Group	1	

General References	UFSAR Table 7.7-1	UFSAR 9.1.4.3	
Explanation	The question describes a fuel-loaded refuel platform, currently in the fuel pool, to be moved to the second core quadrant, while a Rod Drift alarm annunciates. The annunciator shows that a control rod is no longer at position 00. When the bridge is moved toward the core area, the refuel interlock will auto stop the bridge before the core area is reached. This is to prevent a large reactivity addition to the core (largest reactivity addition would be in the cell with the control rod is drifting). Answer A is correct. As stated, bridge motion towards the core is halted, but other bridge motions are not impacted and there is no bridge fault. Answer B is incorrect. It is possible that inserting a fuel bundle into the core location with the drifting control rod, were it selected and being manually driven in by the operator (which is the expected control room action), that damage to the bundle or control rod blade could occur. None the less, it is not the basis for the refuel interlock. Answers C & D are incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.812.0.0003 LO 2391		

Question Source (New, Modified, Bank)	New

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Cognitive Level	Memory or Fundamental Knowledge	X 1: B	Comprehension or Analysis	
	NUREG 1021 Appendix B: Basis or purpose			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

9.1.4.3 Safety Evaluation

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur. A Senior Reactor Operator must supervise all refueling operations.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the REFUEL position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core with fuel on the hoist control rod motion is blocked by the interlocks. With the mode switch in the REFUEL position only one control rod can be withdrawn.

The rod withdrawal interlock may be bypassed in order to allow multiple control rod removal for repair, modifications, or core unloading. The requirements for simultaneous removal of more than one control rod are more stringent than the requirements for removal of a single control rod since, in the latter case, Technical Specification requirements assure that the core will remain subcritical.

The refueling interlocks may be inoperable provided that all 137 control rods are verified to be fully inserted and control rod withdrawal has been disabled prior to commencing or recommencing fuel handling operations with the head off the reactor vessel. This will ensure that all control rods remain fully inserted during fuel handling operations with the head off the reactor vessel. Therefore, Technical Specification requirement are met and the core will remain subcritical during fuel handling operations.

It is not the intent of the alternative option, found in the above paragraph, to eliminate the first performance of the refueling interlock Technical Specification Surveillance prior to in-vessel fuel movement. It is expected that the refueling interlocks would be operable during fuel moves except for equipment failures or during maintenance that would otherwise result in false indications of rod withdrawal during which all rods will be verified as fully inserted and rod withdrawal prevented.

Fuel handling is normally conducted with the fuel grapple hoist. The lowest possible load on this hoist when the interlock is required consists of the weight of the fuel grapple, bottom mast section and the fuel assembly. This total is approximately 680 lbs in the extended position. The load trip settings on the auxiliary hoist motors are adequate to trip the overload interlocks on the motors, if an attempt is made to handle a fuel bundle during refueling.

TABLE 7.7-1
(Sheet 1 of 4)

CONTROL ROD BLOCK INTERLOCKS

Display	Trip Device(s)	Interlock Description
SCRAM DUMP VOLUME	CRD hydraulic system level switch RD86 & RD90	Blocks rod withdrawal if water level in scram discharge volumes exceeds 30 inches during normal reactor operation; ensures adequate discharge volume in event of reactor scram.
ACCUMULATOR LEVEL/PRESS	Hydraulic control unit pressure switches 305-130, level detectors 305-129; control system relays 4K5-	Blocks rod withdrawal if: a. low gas pressure exists in any two scram accumulators, b. water exists on the gas side of any two scram accumulators series and 4K7; or c. one of either condition exits simultaneously on any two accumulators.
REFUEL INTERLOCK	Refueling platform position switches K2, K3, K5; control system relay 4K17	Blocks rod withdrawal in all modes except RUN when refueling platform hoist or fuel grapple is fuel-loaded and over the reactor, or if the service platform hoist is fuel-loaded.

48 **ID: 09-1 NRO48** **Points: 1.00**

The plant was at rated power with each indicated recirculation loop flow matched at 300 x 100 GPM, when the following annunciator for Recirculation Pump E alarmed:

- DRV MOT BRKR TRIP E

30 seconds later when the plant was stable, the Operator performed the IMMEDIATE OPERATOR ACTIONs IAW ABN-2, Recirculation System Failures.

Which of the following states the response of indicated RECIRC PUMP E PUMP FLOW during the following times:

1: from receipt of the annunciator until the plant stabilized 30 seconds later
2: as a result of the Operator actions

	1	2
A.	Drop to 0 GPM, then rise	Remain the same
B.	Drop to about 10 x 100 GPM	Lower to 0 GPM
C.	Drop to 0 GPM	Remain at 0 GPM
D.	Drop to 0 GPM, then rise	Lower to 0 GPM

Answer: D

Answer Explanation:

QID: 09-1 NRO48		
Question # / Answer	**48**	**Developer/Date: NTP 12/22/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

295001 Partial or Complete Loss of Forced Core Flow Circulation AA1.01 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Recirculation system					**3.5**	**3.6**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**ABN-2**		**301.2**			
Explanation	The plant was at rated power when indications show that recirculation pump E tripped. For the first 30 seconds and when the plant becomes stable, recirculation flow through the loop will lower to 0 indicated until flow through the loop goes in reverse due to the driving head of the other operating pumps. The possibility of reversed flow is noted several times in the normal procedure for the recirculation system. Both forward flow and reverse flow through a loop are treated the same and indicate identically. Thus as the tripped pump loses speed, loop flow will lower to 0. Then as reverse flow through the tripped loop is established, the tripped loop flow will rise. ABN-2 requires that the tripped pump discharge valve be closed. As the valve goes closed, the reverse flow will lower until flow has been stopped when the valve is closed. Therefore, loop E recirculation flow will lower initially, then rise somewhat, then lower as the operator action is performed to close the discharge valve. Answer D is correct. All other answers are plausible but incorrect.					
References to be provided during exam:	**None**					
Learning Objective	2621.828.0.0040 LO 209					

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

3.0 OPERATOR ACTIONS

3.1 **PERFORM** the appropriate section of this procedure as follows:

- Recirculation Pump Trip – **PERFORM** section 4.0
- Recirculation Pump Speed Controller Malfunction – **PERFORM** section 5.0
- Potential 1st Stage Seal Failure – **PERFORM** section 6.0
- Potential 2nd Stage Seal Failure – **PERFORM** section 7.0
- Total Seal Failure – **PERFORM** section 8.0

4.0 RECIRCULATION PUMP TRIP

4.1 IMMEDIATE OPERATOR ACTIONS

4.1.1 IF < 3 recirculation pumps are running,

OR

Multiple recirculation pump trips have occurred,

THEN **PERFORM** the following:

1. **SCRAM** the reactor in accordance with ABN-1, Reactor Scram. [ ]
2. **CONFIRM** operating recirculation pump speed at 20 to 30 Hz. [ ]

4.1.2 IF Any Recirculation pumps are operating,

THEN **PERFORM** the following:

1. **CONFIRM** open the DISCH BYPASS valve for the tripped pump(s).
   - V-37-11 (A Pump) [ ]
   - V-37-22 (B Pump) [ ]
   - V-37-33 (C Pump) [ ]
   - V-37-44 (D Pump) [ ]
   - V-37-55 (E Pump) [ ]

2.

> NOTE
>
> Discharge Valves can take up to two minutes to close.

CLOSE the DISCHARGE valve for the tripped pump(s).

- V-37-10 (A Pump) [ ]
- V-37-21 (B Pump) [ ]
- V-37-32 (C Pump) [ ]
- V-37-43 (D Pump) [ ]
- V-37-54 (E Pump) [ ]

4.2 SUBSEQUENT OPERATOR ACTIONS

4.2.1 IF any operating Recirculation Pump MG-Set is in Local-Manual control,

THEN **CONTROL** recirculation pump speed using Procedure 301.2, Reactor Recirculation System. [ ]

4.2.2 IF Any Recirculation pumps are operating,

AND the DISCHARGE valve **cannot** be closed for the tripped Pump,

THEN **PERFORM** the following:

1. **CLOSE** the pump SUCTION valve for the tripped Pump:
   - V-37-9 (A Pump) [ ]
   - V-37-20 (B Pump) [ ]

5.2.5.2 With reactor recirc loop water temperatures below 200°F,

- maximum pump speed will be less than 36.5 Hz
- when starting or stopping a recirc pump, maximum in-service pump speed is limited to 30 Hz.

5.2.5.3 With reactor recirc loop water temperature 200°F or greater, recirc pump speed limitations vary with reactor recirc loop water temperature in accordance with Attachment 301.2-7.

- Operation above 46 Hz is limited until reactor recirc loop water temperature is the normal operating range (515-545°F) because the pump motors are sized for pumping hot water.

5.2.6 When placing a recirculation loop into service, the pump speed must be regulated in order to prevent reverse flow and possible stalling of the pump when other pumps are in service.

5.2.7 A recirculation pump shall **not** be operated without closed cooling water, except as permitted by this procedure, in order to prevent damage to the pump seals.

5.2.8 Total core flow (Recirculation, Shutdown Cooling, Cleanup, and Control Rod Drive) is limited to 17,360 gpm, unless all SRM's, IRM's and LPRM's in the reactor vessel are surrounded on all four sides by fuel assemblies or blade guides. This is to prevent damage to the nuclear instrumentation due to flow induced vibration. This precaution does **not** apply if the Standby Liquid Control System has been initiated for reactivity control. (Ref GE SIL 406)

5.2.9 Changing the temperature of a MG set exciter winding causes an erroneous MG set speed signal change which results in a Rx recirc. flow change. Minimize the magnitude and rate of temperature changes in the MG set room. Monitor Rx recirc. flow during evolutions which change the MG set room temperature.

5.2.10 Minimize the time that reactor recirculation pumps are operated with the discharge valve closed. Vibration levels significantly increase during this mode and any extended operation should be avoided.

49 **ID: 09-1 NRO49** **Points: 1.00**

The plant was at rated power with known nuclear fuel leakers and an elevated reactor coolant activity level.

A single unisolable steam/water leak occurred which resulted in **rising** indications on the TB RAGEMS, **but** the Stack RAGEMS remained **steady**.

Which of the following states the location of the leak and the type of offsite release?

	Leak Location	Release Type
A.	Condenser Bay	Ground level
B.	Reheater Protection Area	Elevated
C.	Steam Jet Air Ejector Room	Elevated
D.	Feed Pump and Condensate Pump Room	Ground level

Answer: D

Answer Explanation:

QID: 09-1 NRO49		
Question # / Answer	**49**	**Developer/Date: NTP 12/22/09**

Knowledge and Ability Reference Information					
K&A				Importance Rating RO	Importance Rating SRO
295038 High Off-site Release Rate EA1.01 - Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Stack-gas monitoring				3.9	4.2
Level	RO	Tier	1	Group	1

General References	BR 2009 sh.1, 2 BR 2011	GU 3E-661-21-1001, sh. 1	EOP Users Guide
Explanation	The plant is at power with rising trend on the TB RAGEMS but not on the stack RAGEMS. Of the locations listed, the feed/condensate pump room atmosphere discharges to the atmosphere and the reheater protection area discharge to the TB stack (and not the main stack) and both are monitored by TB RAGEMS. IAW the EOP Users Guide, a radiological release from the TB Stack (or other release point in the TB) is considered a ground level release. Answer D is correct and answer B is incorrect. The other locations listed are in the TB but discharge to the main plant stack, which is monitored by (Main) Stack RAGEMS: Condenser bay goes to the main stack (answer A is incorrect); SJAE room goes to main stack (answer C is incorrect).		
References to be provided during exam:	None		
Learning Objective	2621.828.0054 LO 288-10437		

Question Source (New, Modified, Bank)				Bank
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize a relationship			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

RADIOACTIVITY RELEASE CONTROL

ENTRY CONDITIONS

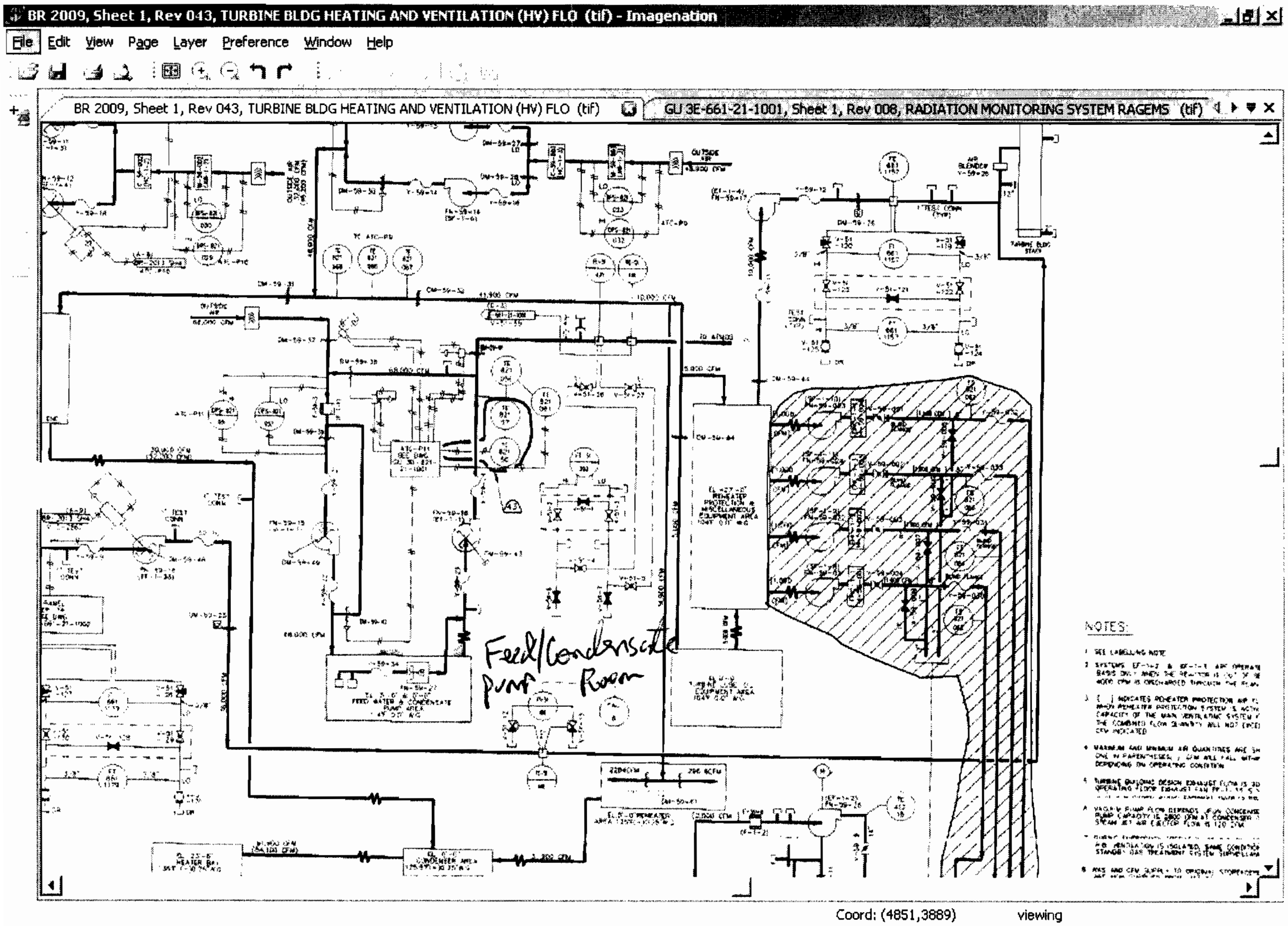
PARAMETER	CONDITION
OFF-SITE RADIOACTIVITY RELEASE RATE	ABOVE THAT WHICH REQUIRES AN **ALERT**
ISOLATION CONDENSER	TUBE LEAK

IF	THEN
THE RELEASE IS FROM THE TURBINE BUILDING	OPERATE AVAILABLE TURBINE BUILDING VENTILATION PER **SUPPORT PROC - 51**
ANY PRIMARY SYSTEM IS DISCHARGING OUTSIDE PRIMARY AND SECONDARY CONTAINMENT	ISOLATE THE AFFECTED PRIMARY SYSTEMS EXCEPT SYSTEMS REQUIRED BY ANY EOP

DISCUSSION

Continued personnel access to the Turbine Building may be essential for responding to emergencies, or transients which may degrade into emergencies. Since the Turbine Building is not an airtight structure, any radioactivity released inside the Turbine Building would not only limit personnel access but could eventually lead to an unmonitored ground level release.

This Conditional Statement directs the operator to maintain the Turbine Building Ventilation System in service to preserve Turbine Building accessibility, and ensure that any radioactivity is discharged through a monitored release point, either the Main Stack for an elevated release, or via the Turbine Building Stack, which is considered a ground level release. When required, Support Procedure - 51 provides the necessary directions for restarting the Turbine Building Ventilation System.


BR 2009, Sheet 1, Rev 043, TURBINE BLDG HEATING AND VENTILATION (HV) FLO (tif) - Imagenation
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BR 2009, Sheet 1, Rev 043, TURBINE BLDG HEATING AND VENTILATION (HV) FLO (tif)
GU 3E-661-21-1001, Sheet 1, Rev 008, RADIATION MONITORING SYSTEM RAGEMS (tif)
Feed/Condensate Pump Room
NOTES:
Coord: (4851,3889)
viewing

50 **ID: 09-1 NRO50** **Points: 1.00**

The plant was at rated power when the Shift Manager declared that a Control Room Evacuation was required and ABN-30, Control Room Evacuation, was entered.

The Operator verified a successful reactor scram and turbine trip but off-site power failed to energize its busses. The Operator also observed the following prior to leaving the Control Room:

- EDG 1 UNIT START **and** UNIT IDLING lights are de-energized
- EDG 2 UNIT START **and** UNIT IDLING lights are de-energized

IAW 346, Operation of the Remote and Local Shutdown Panels, which of the following is required to supply power to the Station?

A. Start EDG1 at the RSP.

B. Start EDG 2 at LSP-DG2.

C. Close the S1A Startup Breaker at the RSP.

D. Close the SBO Breaker at the SBO Panel.

Answer: B

Answer Explanation:

QID: 09-1 NRO50		
Question # / Answer	50	Developer/Date: NTP 12/23/09

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
295016 Control Room Abandonment AA1.04 - Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : A.C. electrical distribution				3.1	3.2
Level	RO	Tier	1	Group	1
General References	346	ABN-30			

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant was at rated power when a control room evacuation was required. The operator verified a successful scram and turbine trip. The stem also provided that off-site power was lost and the EDG indications provided show that neither EDG started and loaded. IAW 346, if EDG2 did not start, then manually start the EDG at the LSP-DG2. Answer B is correct. The other answers are plausible but incorrect. EDG1 cannot be started from the Remote Shutdown Panel. Answer A is incorrect. Startup breaker S1A cannot be operated from the Remote Shutdown Panel. Answer C is incorrect. Closing the SBO breaker could align the plant with the combustion turbine, it is not directed in 346.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0064 LO 308-10446

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:S**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Structures and locations			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

5.0 OPERATION OF LOCAL SHUTDOWN PANEL LSP-DG2
(located DG2 vault on north wall)

5.1 Prerequisites

5.1.1

> NOTE
>
> The controls and indications located in the Control Room **do not** need to be aligned for LSP-DG2 operation.

EDG-2 is lined up for operation IAW Procedure 341, Standby Diesel Generator Operation, Section 3.0. [ ]

5.2 Precautions and Limitations

5.2.1 EDG-2 shall **not** be operated in parallel with offsite power during fire conditions. If offsite power is available and the "D" 4160V bus can be energized from offsite power, then **do not** start EDG-2.

5.2.2 The emergency start sequence nullifies all engine or engine temperature fault automatic shutdowns, except the engine overspeed trip.

5.2.3 The Precautions and Limitations in Section 4.0 apply.

5.3 Transferring Control to LSP- DG2

5.3.1 **ESTABLISH** communication between LSP-DG2 and the Control Room or the RSP. [ ]

5.3.2 **CONFIRM** Tie Breaker ED Open and racked out. [ ]

5.3.3 **CONFIRM** Emergency Bus Breaker 1D Open. [ ]

5.3.4

CAUTION

Steps 5.3.4, 5.3.5, 5.3.6, and 5.3.7 must be performed in sequence to protect circuit integrity.

PLACE EDG 2 Alternate Mode Selection Switch on LSP-DG2 to "DEADLINE" [ ]

5.3.5 **CONFIRM** NORMAL-ALTERNATE switch #1 on LSP-DG2 is in ALTERNATE. [ ]

5.3.6 **PLACE** NORMAL-ALTERNATE switch #2 on LSP-DG2 to ALTERNATE [ ]

5.3.7 **PLACE** NORMAL-ALTERNATE switch #3 on LSP-DG2 to ALTERNATE [ ]

5.3.8 IF EDG is **not** running,

THEN **PLACE** the DG2 ALTERNATE EMERGENCY START switch on LSP-DG2 momentarily to START. (EDG-2 will deadline start and pickup load in 15 seconds) [ ]

5.3.9

NOTE

Monitoring of EDG-2 can only be performed for a few minutes if LSP-1B3 and/or LSP-1B32 have to be also initiated.

MONITOR operation of EDG-2 locally (IAW Procedure 341) [ ]

5.3.10 **PROCEED** to the next Local Shutdown Panel if necessary. [ ]

51 **ID: 09-1 NRO51** **Points: 1.00**

The plant was at rated power.

A leak in the service air system downstream of V-6S-2, Service Air Isolation valve, caused a drop in service air pressure and the Operator reported the following observations:

- The backup air compressors are running
- INSTR AIR SUPPLY PRESS was rising

When INSTR AIR SUPPLY PRESS reached 100 psig, the service air leak worsened. It resulted in a **lowering** of service air pressure at the rate of 2 psig/minute.

IAW ABN-35, Loss of Instrument Air, which of the following is correct under the conditions provided?

A. **No** manual scram will be required due to lowering air pressure.

B. A manual scram will be required in 19 minutes from lowering air pressure.

C. A manual scram will be required in 23 minutes from lowering air pressure.

D. A manual scram will be required in 27 minutes from lowering air pressure.

Answer: A

Answer Explanation:

QID: 09-1 NRO51		
Question # / Answer	**51**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

295019 Partial or Total Loss of Inst. Air AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Instrument air system pressure						**3.5**	**3.6**
Level	**RO**		**Tier**	**1**	**Group**	**1**	
General References	**ABN-35**		**RAP-M2b**		**BR 2013, sh. 1**		
Explanation	The plant is at power when an event caused the standby air compressors to start on low air pressure. After the compressors start, air pressure begins to recover. At 100 psig, the malfunction worsens and begins to lower air pressure at a rate of 2 psig/minute. When air pressure lowers to 75 psig, the service air isolation valve, V-6s-2, will automatically close. This will isolate air going to service air from the air compressors and instrument air. Once isolated, the running air compressors will restore instrument air pressure to normal and no scram will be required due to lowering air pressure. Answer A is correct. IAW ABN-35, when instrument air pressure drops to 55 psig, then a manual scram is required. All other answers are plausible if the candidates forget about the service isolation valve automatic action and the setpoint on which to perform the manual scram.						
References to be provided during exam:		**None**					
Learning Objective	2624.828.0.0043 LO 279-10445						

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

3.0 IMMEDIATE OPERATOR ACTIONS

3.1 IF Instrument Air Supply pressure, PT-3 (7F) drops to 55 psig,

OR

2 or more control rods begin to drift into the core,

THEN **SCRAM** the Reactor IAW ABN-1, Reactor Scram. [ ]

3.2 IF Instrument Air is lost due to a fire,

THEN **PERFORM** actions stipulated in ABN-29 and associated Fire Support Procedure (FSP) first and then as time permits, perform actions in this procedure. [ ]

4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 IF RCVR 2/INSTR AIR PRESS LO alarm (M-3-b) actuates

OR

Instrument Air Supply pressure, PT-3 (7F) drops to 95 psig,

THEN **DISPATCH** an operator to **CHECK** the following:

- Air Dryer malfunctions [ ]
- Valve lineup errors [ ]
- Air leaks in the system [ ]
- Malfunction of the loading switch on the operating air compressor [ ]

4.2 **ALERT** Station personnel by making the following announcement on the Plant PA system:

"Attention all personnel, attention all personnel, anyone presently utilizing the Service Air system shall secure all work at this time. I repeat, anyone presently utilizing the Service Air system shall secure all work at this time." [ ]

- V-6-193, Prefilter Inlet Isolation Valve [ ]
- V-6-195, Prefilter Inlet Isolation Valve [ ]

c. **OPEN** V-6-206, Air Drying Towers Bypass Valve. [ ]

d. **CLOSE** V-6-205, Common Inlet to Air Drying Towers. [ ]

e. **OPEN** V-6-242, Post Filter Bypass Valve. [ ]

f. **CLOSE** the following valves:

- V-6-243, Post Filter Isolation Valve [ ]
- V-6-245, Post Filter Isolation Valve [ ]

3. **VERIFY** all available air compressors are operating normally. [ ]

4.8 IF Instrument Air Supply pressure, PT-3 (7F), drops to 75 psig,

THEN **PERFORM** the following:

- **CONFIRM** Service Air Valve V-6S-2 has isolated [ ]

  AND

  Service Air Valve V-6S-2 is **not** bypassed. [ ]

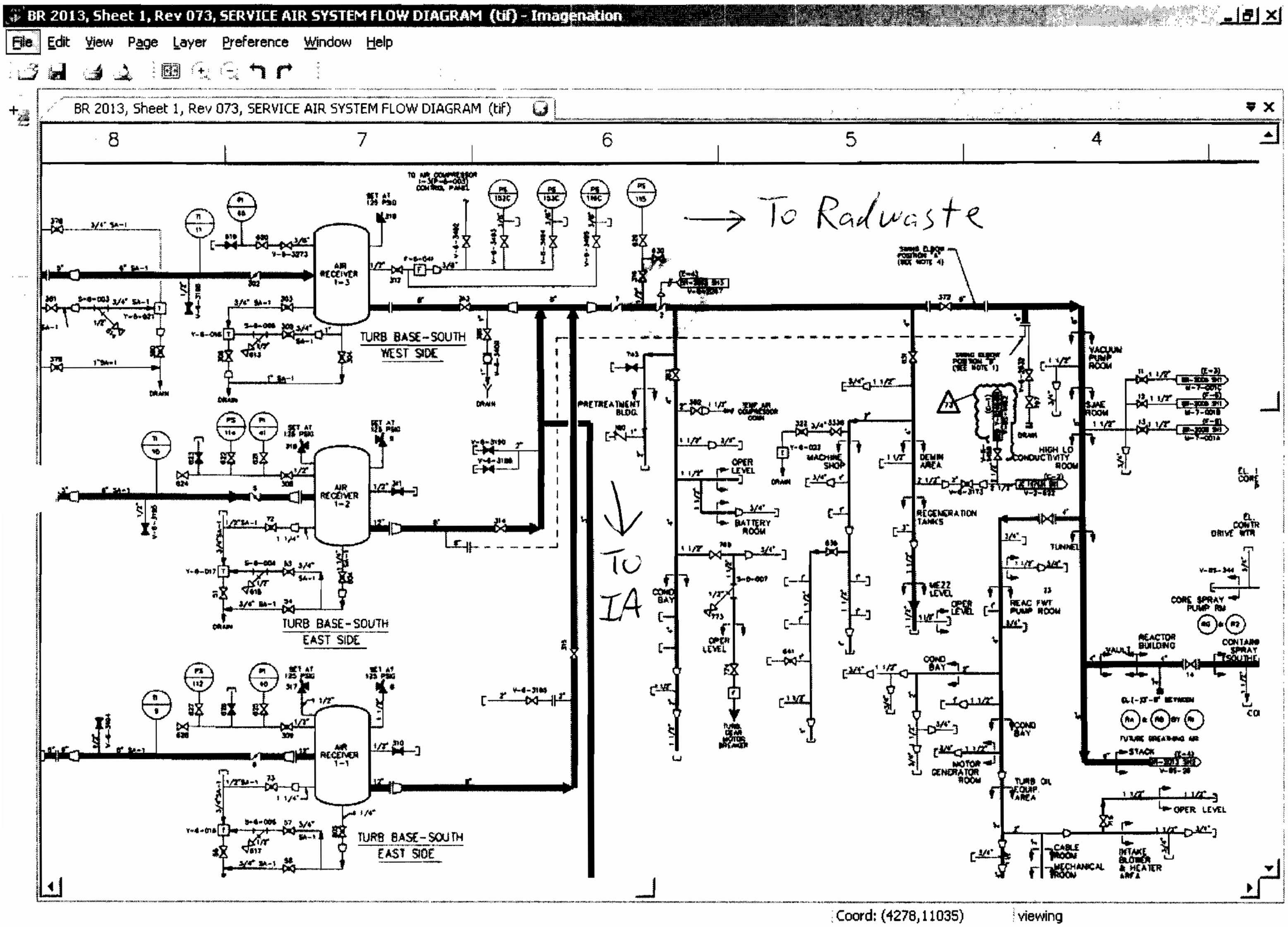
- **VERIFY** the following alarms are received:

  a. SVC AIR DISCH VLV CLOSED (M-2-b) [ ]

  b. CONTROL AIR PRESS (H-1-a) [ ]

4.9 **EXECUTE** the operator actions listed in Attachment ABN-35-1, Major Systems Affected by Loss of Instrument Air. [ ]

4.10 **REFER** to Attachment ABN-35-2, Other Plant Systems Affected. [ ]


BR 2013, Sheet 1, Rev 073, SERVICE AIR SYSTEM FLOW DIAGRAM (tif) - Imagenation
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BR 2013, Sheet 1, Rev 073, SERVICE AIR SYSTEM FLOW DIAGRAM (tif)
8
7
6
5
4
To Radwaste
To IA
AIR RECEIVER 1-3
AIR RECEIVER 1-2
AIR RECEIVER 1-1
TURB BASE-SOUTH WEST SIDE
TURB BASE-SOUTH EAST SIDE
TURB BASE-SOUTH EAST SIDE
DRAIN
PRETREATMENT BLDG.
MACHINE SHOP
OPER LEVEL
BATTERY ROOM
COND BAY
DEMIN AREA
REGENERATION TANKS
MEZZ LEVEL
VACUUM PUMP ROOM
SJAE ROOM
HIGH LO CONDUCTIVITY ROOM
TUNNEL
REAC FWT PUMP ROOM
MOTOR GENERATOR ROOM
TURB OIL EQUIP AREA
CABLE ROOM
MECHANICAL ROOM
INTAKE BLOWER & HEATER AREA
REACTOR BUILDING
VAULT
STACK
CORE SPRAY PUMP RM
CONTAINS SPRAY SOUTHE
Coord: (4278,11035)
viewing

52 **ID: 09-1 NRO52** **Points: 1.00**

The plant is starting up following an outage. Present plant conditions include the following:

- Reactor power is about 4%
- The turbine is being warmed through Stop Valve #2 internal bypass
- Feedwater Pump A is in service on the LFRV A

Two minutes later, RPV water level rose to 183".

Which of the following states the plant impact from this level excursion?

A. The main turbine **only** trips.

B. Feedwater Pump A **only** trips.

C. The turbine trips and the reactor scrams **only**.

D. The turbine trips, the reactor scrams, and Feedwater Pump A trips.

Answer: A

Answer Explanation:

QID: 09-1 NRO52		
Question # / Answer	**52**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
295006 Scram AA2.03 - Ability to determine and/or interpret the following as they apply to SCRAM : Reactor water level						4.0	4.2
Level	RO	Tier	1	Group	1		
General References	ABN-10 ABN-1		RAP-H7d RAP-H5d		201 317		

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant is starting up with power at about 4%, and the main turbine is being heated. In order to preheat the main turbine through stop valve #2, the turbine (trip) must be reset. Once reset, the turbine will trip from a turbine trip signal. As RPV water rises to 175”, a turbine trip signal is generated. As water level continues to rise to 181”, Reactor Overfill Protection System (ROPS) can be activated, unless bypassed. When activated, all operating feedwater pumps trip. ROPS is automatically bypassed when total feedwater flow is < 2.23×10^6 lb/hr. IAW procedures 201 and 317, feedwater flow should be transferred to the main flow regulation valves (MFRV) at about 0.6×10^6 lb/hr. Therefore, with only 1 feedwater pump operating at < 0.6×10^6 lb/hr, then ROPS is automatically bypassed on low feedwater flow. As the ROPS high RPV water level setpoint is reached, the operating feedwater pump will not trip since ROPS is bypassed. Answer A is correct. Answer B is incorrect since the feedwater pump does not trip. As discussed, the turbine will trip but the reactor will not scram on the high water level nor will it trip from the turbine trip (bypassed at <40%). Answer C & D are incorrect. The reactor will scram on RPV low water level of 138" There is no direct scram from RPV high water level.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0051 LO 249-10445		

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading REACTOR PRESS	H-5-f

RX LVL HI I AND	

CONFIRMATORY ACTIONS:

- ❑ **CHECK** for Reactor Vessel high water level. (Panel 5F/6F) [ ]

AUTOMATIC ACTIONS:

Turbine trip, if coincident with Channel II trip.

MANUAL CORRECTIVE ACTIONS:

- ❑ **CHECK** feedwater controls for proper operation. [ ]
- ❑ **REFER** to ABN-17, Feedwater System Abnormal Conditions. [ ]
- ❑ **REFER** to ABN-59, RPV Level Instrument Failures [ ]
- ❑ IF turbine trips,

  THEN **PERFORM** the following:

  - ❑ IF Reactor power was >30% (580 MWt),

    THEN **REFER** to ABN-1, Reactor Scram. [ ]
  - ❑ **PERFORM** followup actions IAW ABN-10, Turbine Generator Trip. [ ]

Group Heading REACTOR PRESS	H-5-f

TT RX LVL HI I AND	

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Reactor water level greater than 175 inches TAF.	175 inches TAF	Panel 18R Module RE05BY6 or RE05/19AY6 Via Relay PNL-624-5FCR2
		Reference Drawings: GU 3E-611-18-024 BR 3022, Sh. 2 GU 3E-611-17-010

Subject NSSS Alarm Response Procedures	Procedure No. RAP-H5f	Page 2 of 2	H-5-f
	Revision No: 1		

Group Heading DW PRESS	H-7-d

ROPS BYPASSED	

CONFIRMATORY ACTIONS:

- ❑ **CHECK** total feed flow indications. (Recorder ID-75; PCS point HB-FWFLN) [ ]
- ❑ **CHECK** ROPS Bypass switch position. (PNL-629-4FCS11) [ ]

AUTOMATIC ACTIONS:

NONE

MANUAL CORRECTIVE ACTIONS:

❑ IF ROPS manual bypass is **not** required,

THEN **CONFIRM** the ROPS Manual Bypass Switch is in NORMAL position. (4F) [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Total Feedwater Flow Low	Auto Bypass: $\leq 2.23 \times 10^6$ lb/hr	PNL-629-14XRCR3
ROPS manual bypass switch 4F in BYPASS position	Reset: $\geq 2.4 \times 10^6$ lb/hr	PNL-629-4FCS11
		Reference Drawings: GU 3D-629-17-002 GU 3E-611-17-010

ATTACHMENT ABN-1-3

REACTOR SCRAM TRIPS

NOTE

All of the following trips, except "Manual Scram" and "Recirc Flow Monitoring Inop", operate with one-out-of-two twice logic.

Trip	Setpoint	Bypasses
APRM Hi-Hi or Inop	Flow dependent	Joystick (single Channel)
IRM Hi-Hi or Inop	118 on 125 scale 37.8 on 40 scale	Mode switch in RUN (if APRM >2 percent)
Recirc Flow Monitoring Inop	Loss of Flow Signal	None
High RPV Pressure	1045 psig	None
Low RPV Water Level (Analog Trip System)	139.5 in.	None
High Drywell Pressure	3.0 psig	None
MSIV Closure	10 percent closure	<600 psig, mode switch in STARTUP or REFUEL; or mode switch in SHUTDOWN any pressure
Turbine Trip	10 percent closure of stop valves	<40 percent turbine power
Load Reject	Acceleration relay	<40 percent turbine power
Main Condenser Low Vacuum	22" Hg. vacuum	<600 psig if mode switch in STARTUP or REFUEL; or mode switch in SHUTDOWN any pressure
Scram Discharge Volume High level	26 gal.	Keylock switch if mode switch in shutdown or refuel
Mode Switch in SHUTDOWN		Automatic after 20 sec.
Manual Scram		None

53 **ID: 09-1 NRO53** **Points: 1.00**

The plant was at rated power. A timeline of events is shown below (hh:mm):

0800 Breaker 1A **inadvertently** opened
0815 Annunciator MN BRKR 1C 86 LKOUT TRIP alarmed
0830 The reactor scrammed on high Drywell pressure

Which of the following is correct for the given conditions?

A. Feedwater Pump A is **not** available for RPV injection.

B. Core Spray loops A **and** B are available for RPV injection.

C. **Both** Service Water Pumps are available to provide cooling water.

D. Containment Spray Pumps 51C **and** 51D are available for Drywell Sprays.

Answer: D

Answer Explanation:

QID: 09-1 NRO53		
Question # / Answer	**53**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295003 Partial or Complete Loss of AC AA2.04 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : System lineups					**3.5**	**3.7**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**RAP-S1e**		**BR 3002, sh. 2**		**RAP-T2a**	

Explanation	The plant was at rated power when breaker 1A inadvertently opened. When this breaker opens, Startup Breaker S1A automatically closes and all busses remain energized. At 0815, a lockout on Bus 1C causes Bus 1C occurs and Bus 1C will de-energize and the downstream busses (USS 1A1, USS 1A2, and USS 1A3) will also de-energize. Because there is a fault on the 1C bus, EDG1 will not fast start and load onto the bus, but remains off. When the plant scrams, Bus 1B power is automatically transferred to Startup breaker S1B. So over the events, Bus 1C is de-energized (and the downstream busses). Containment Spray Pumps are available (powered from USS 1B2) and the associated ESW Pumps are also available (powered from Bus 1D). Answer D is correct. As stated, when breaker 1A opens, the S1A breaker closes and Bus 1A remains energized (with no impact from the scram). Thus Feedwater Pump A (powered from Bus 1A) is available. Answer A is incorrect. Core Spray Main Pump A (powered from Bus 1C) and Booster Pump A (powered from USS 1A2) are not available. The Core Spray B loop does have power. Answer B is incorrect. Service Water Pump 1-1 (powered from USS 1A3) is not available, whereas Service Water Pump 1-2 (powered from USS 1B3) is available, but not both. Answer C is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0009 LO 226-10453	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describing or recognizing relationships			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading 4160V STATION POWER BUS 1A	S-1-e

MN BRKR 1A TRIP	

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** trip of 4160V Breaker 1A. [ ]
  - ❑ **VERIFY** closure of 4160V Startup Breaker S1A. [ ]

AUTOMATIC ACTIONS:

Closes 4160V Startup Breaker S1A.

MANUAL CORRECTIVE ACTIONS:

❑ IF transfer to Startup Breaker is not successful or if Startup Transformer SA is not available,

THEN **VERIFY** fast start of DG 1 to assume load on the 1C Bus. [ ]

❑ **REFER** to Procedure 337, 4160 Volt Electrical System. [ ]

❑ **CHECK** for any relay targets at breaker or in control room. [ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 2)

MN BRKR 1C 86 LKOUT TRIP

CONFIRMATORY ACTIONS:

NOTE

Lockout of Bus 1C will prevent the fast start of Emergency Diesel Generator #1 and diesel generator breaker closure on faulted Bus 1C.

- ❑ **CHECK** 1C Bus voltage and current. (8F/9F) [ ]
- ❑ **VERIFY** trip of 4160V Breaker 1C. [ ]
- ❑ **VERIFY** trip of 4160V Bus Tie Breaker EC (if closed). [ ]

AUTOMATIC ACTIONS:

Trip of 4160 V Breaker 1C **and** trip of 4160 V Bus Tie Breaker EC.

As result of the trip of these Breakers the following will occur:

Trip of:

- Emergency Service Water Pumps A and B
- Core Spray Pumps A and D (if running)
- 480 V Substation loads fed by Bus 1C
- 4160 V Bus Tie Breaker EC if Closed.

BR 3002, Sheet 2, Rev 011, 480V SYSTEM ONE LINE DIAGRAM 460V UNIT SUBSTA (tif) - Imagenation

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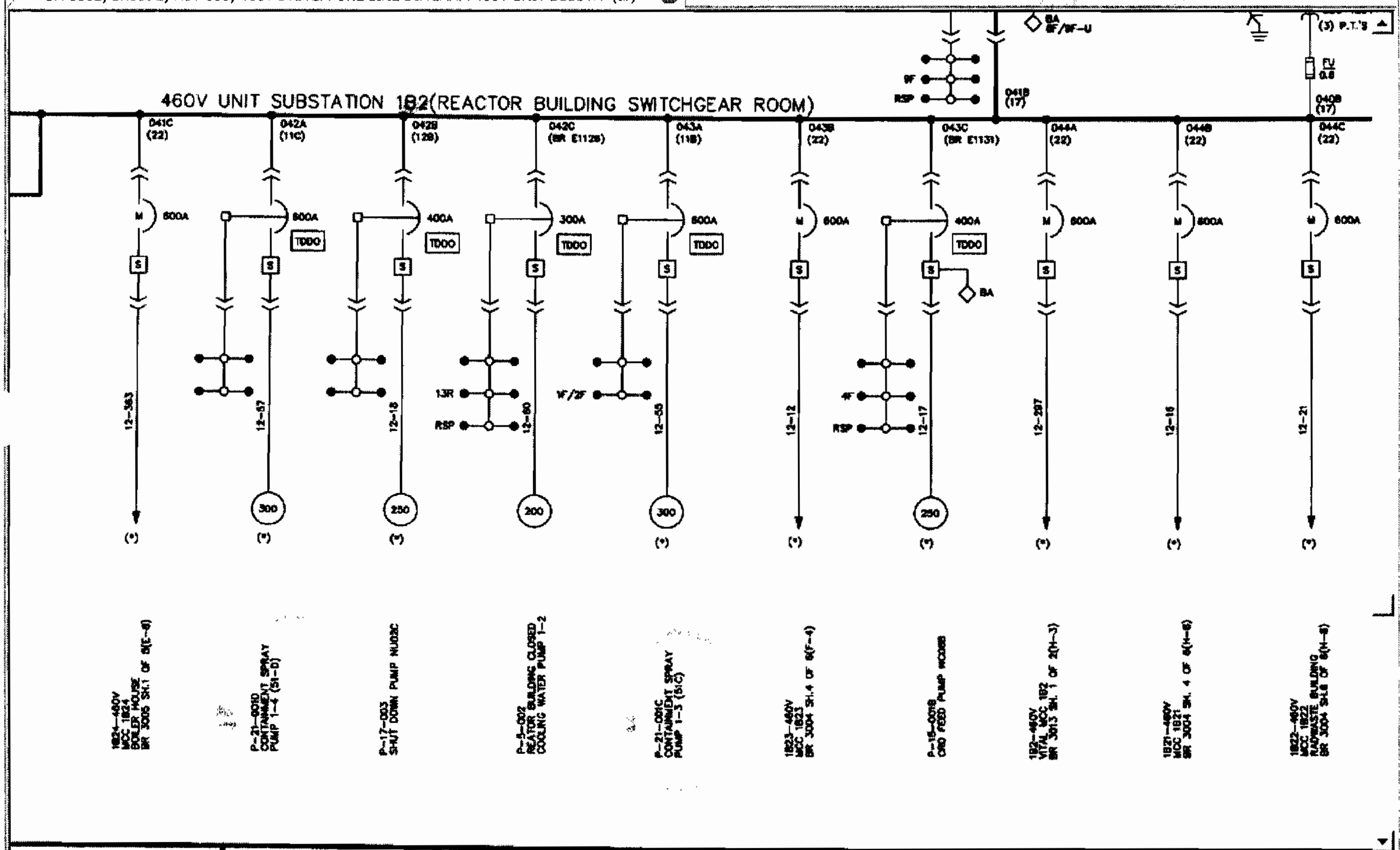
BR 3002, Sheet 2, Rev 011, 480V SYSTEM ONE LINE DIAGRAM 460V UNIT SUBSTA (tif)



Coord: (9749,10099) viewing

54 **ID: 09-1 NRO54** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions include the following:

- An ATWS is in progress
- RPV pressure is 1000 psig and stable
- EMRV NR108A is stuck full open
- 3 Turbine Bypass Valves indicate full open
- APRM indication has been lost
- The Isolation Condensers are in Standby
- Torus water temperature is 100 °F and is rising at 2 °F/minute
- Torus Cooling is inoperable

IAW the EOP Users Guide, which of the following states the **maximum** time until SLC injection is required (Assume that reactor power remains unchanged over the times listed)?

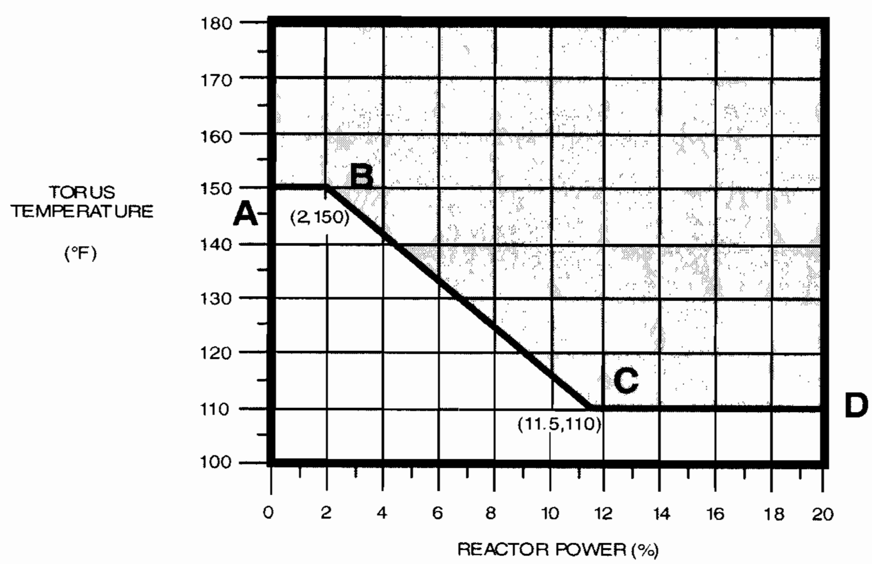


FIG. L BIIT
BORON INJECTION
INITIATION TEMPERATURE

A. 5 minutes

B. 6 minutes

C. 7 minutes

D. 8 minutes

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer: A

Answer Explanation:

QID: 09-1 NRO54		
Question # / Answer	**54**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295026 Suppression Pool High Water Temp 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**	**4.2**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**EOP Users Guide**	**UFSAR Table 5.1-1**		**UFSAR 7.7.1.5**		
Explanation	The plant was at power when an event occurred: an ATWS with a rising torus water temperature. APRM indication has been lost. But power can be estimated enough to determine when SLC injection is required. The capacity of the turbine bypass valves is 40%, or 5% per bypass valve. With 3 turbine bypass valves open, and RPV pressure stable at 1000 psig, then reactor power is about 15%. But a full open EMRV at 1250 psig allows over 600,000 lb/hr, and a reduced flow rate at 1000 psig. Therefore, the total steam flow would equate to > 15% reactor power. SLC injection is required when the Torus temperature cannot be maintained below the BIIT temperature for the given reactor power. From the BIIT Curve provided, at any power > 12%, the Curve will be met when torus water temperature reaches 110 °F. Thus, after 5 minutes, torus temperature will be 110 °F at a power > 12% and the BIIT Curve will be met, requiring SLC injection. Answer A is correct. The other answers are plausible if reactor power is estimated incorrectly or graph misread.					
References to be provided during exam:	**None**					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Learning Objective	2621.845.0.0053 LO 3055A

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**x 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using references			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

POWER CONTROL (BORON)

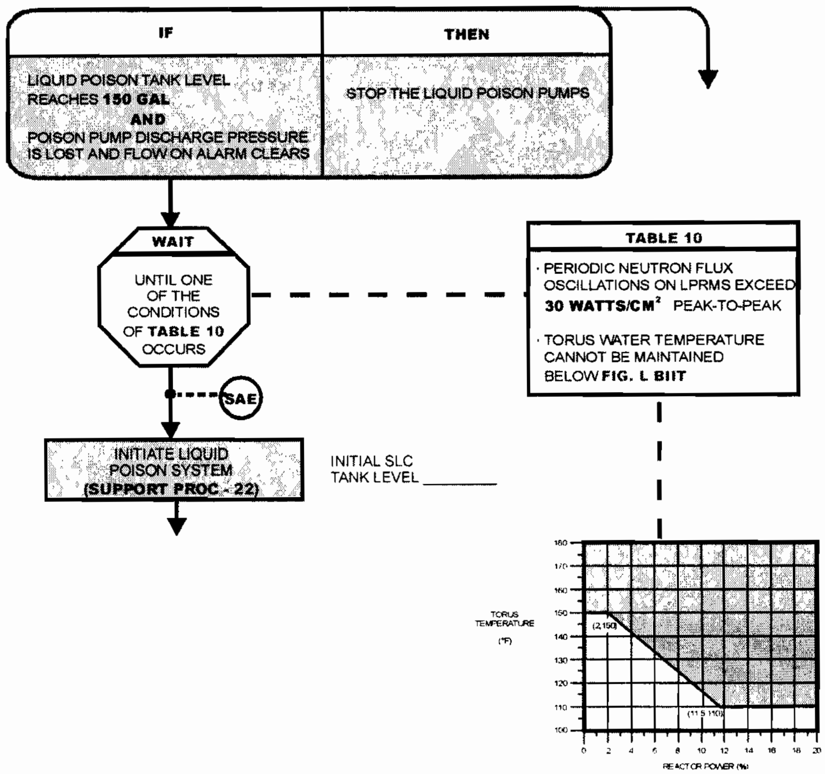


FIG. L BIIT
BORON INJECTION INITIATION TEMPERATURE

DISCUSSION

As long as the Main Turbine remains on-line or Reactor power remains within the heat removal capability of the Bypass Valves and/or Isolation Condensers during higher power ATWS conditions, the Primary Containment is not immediately threatened. If the Main Condenser is not available and heat removal is beyond the capability of the ICs, the primary method of energy removal from the Reactor is via EMRVs discharging to the Torus. This will result in a rapid heat up of the Torus (approximately 2°F per min. for each EMRV open), and thus a threat to Primary Containment integrity. The challenge to Primary Containment thus becomes one of the limiting factors that define the requirement for boron injection.

Heat up of the Torus causes Plant parameters to approach the Heat Capacity Temperature Limit (HCTL). If Torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit, an emergency depressurization of the RPV will be required. To avoid depressurizing the RPV with the Reactor at power, it is desirable to shut down the Reactor prior to reaching the Heat Capacity Temperature Limit, thus minimizing the quantity of heat rejected to the Torus. The Boron Injection Initiation Temperature is defined to achieve this when practical.

TABLE 5.1-1
(Sheet 3 of 3)

REACTOR COOLANT SYSTEM DESIGN DATA

Electromatic Relief Valves	
Number	5 (three on one steam header, two on the other)
Capacity	**602,900 lb/hr each at 1250 psig**
Pressure setting	See Technical Specifications
Design code	ASA B31.1 (original valves) ASME Sect. III (replacements)
Safety Valves	
Number	9 (5 on one steam header, 4 on the other)
Capacity	634,000 lb/hr each
Pressure setting	See Technical Specifications
Design code	ASME B&PV, Code, Section I ASA B31.1 (original valves) ASME Sect. III (replacements)
Feedwater Piping	
Design code	ASME Section I (up to first isolation valve) ASA B31.1 (balance)
Isolation Condensers	
Number	2
Design Capacity per isolation condenser (3 percent of 1930 MWt)205 x 10^6Btu/hr at 1000 Psig and 546°F	
Number of isolation	Two (2) normally open valves in inlet line (one ac operated, one dc operated)
Number of isolation	One (1) normally open valves in outlet line (ac operated) One (1) normally closed (dc operated)
Design codes shell	
Shell	ASME B&PV Code Section VIII
Tubes	ASME B&PV Code Section III C1.A
Design pressures	
Shell	15 psig internal 1 psig external, 300°F
Tube	1250 psig, saturated

7.7.1.5 Turbine Generator Controls

The Turbine Generator is provided with a complete control system for startup, shutdown, and changes in load. This system is discussed in Section 10.2. This section discusses only the Turbine-Generator interaction with the Reactor Protection System.

If the water level in the reactor vessel exceeds the height of the top of the steam separators, excessive moisture carryover occurs, resulting in turbine blade erosion. Level sensors RE 05/19A, RE05B (RPS Channel 1) RE05A and RE05/19B (RPS Channel 2) trip the turbine on high level to protect against blade damage. Note that these are safety related switches which also provide the reactor low water level trip.

If the turbine trips while operating at power levels above approximately 200 MW the reactor will be tripped.

Load rejection within the Turbine Bypass System (Section 10.4) capacity will cause the control valves to close and the bypass valves to open and dump steam to condenser. The design mismatch of five percent rated flow under these conditions should not be enough to cause a high flux scram. Load rejections beyond the bypass system capacity will cause a high flux scram, but the bypass system will normally limit the pressure rise to keep the safety valves from opening.

Three separate turbine trip sensors, listed in Table 7.7-2, anticipate the reactor power increase and start rod motion (scram) before the pressure excursion begins in order to minimize the flux peak. Although the resultant increase in reactor power would result in a trip from the pressure increase directly, the anticipatory trip feature lessons the reactor power and pressure excursion. The anticipatory trips are bypassed when the Reactor Mode Switch is not in RUN and the reactor pressure is less than 600 psig, to permit reactor startup. The Turbine Trip and Generator Trip anticipate the need for a reactor trip if a turbine trip or generator trip occurs over 40% **reactor thermal power.** Below this power level, a scram is not required since the bypass system is capable of passing this flow rate.

7.7.1.6 Reactor Overfill Protection System (ROPS)

If the water level in the reactor vessel exceeds the height of the main steam lines, a potential for Main Steam Line Break (MSLB) exists. The Reactor Overfill Protection System (ROPS) is designed to minimize the potential for such conditions. Existing level sensors (RE05A, RE05/19A, RE05B, RE05/19B) used in Reactor Protection System (RPS) are utilized in ROPS to trip all three (3) feedwater pumps on reactor high level provided the total feedwater flow is not low and the "normal/bypass" switch located in control room panel 4F is not in bypass position.

55 **ID: 09-1 NRO55** **Points: 1.00**

Consider the events listed below with the reactor at power. The reactor **remained** at power for each event.

Which of the following events represents a violation of Tech Specs?

A. RPV water level of 139" at 25% reactor power.

B. Drywell pressure of 3.2 psig at 15% reactor power.

C. Turbine Stop Valve closure at 45% reactor power.

D. RPV pressure of 1054 psig at 55% reactor power.

Answer: C

Answer Explanation:

QID: 09-1 NRO55		
Question # / Answer	**55**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
295005 Main Turbine Generator Trip 2.2.42 - Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.						**3.9**	**4.6**
Level	**RO**	**Tier**	**1**	**Group**	**1**		
General References	**TS Table 3.1.1**	**TS 2.3**					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	A turbine stop valve closure (which occurs during a turbine trip) scram is not required below 40% rated thermal power. Thus, if the reactor does not scram at 45% power, this would result in a TS violation and LCO entry. Answer C is correct. TS requires a scram when RPV water level is > 11'5" (137"). With RPV water level at 139" while at power, no Tech Spec has been violated. Answer A is incorrect. TS requires a scram from Drywell pressure of ≤ 3.5 psig. Thus, no scram with Drywell pressure of 3.2 psig does not violate TS. Answer B is incorrect. TS requires a scram at ≤ 1060 psig. With RPV pressure at 1054 psig while at power, no Tech Spec has been violated. Answer D is incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.850.0.0090 LO 1658

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:D**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Definitions			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

TABLE 3.1.1 - PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Minimum Number of OPERABLE or OPERATING [tripped] Trip Systems	Minimum Number of Instrument Channels Per OPERABLE Trip System	Action Required*
8. Average Power Range Monitor (APRM)	**		X(c,s)	X(c)	X(c)	2	3(nn)	
9. Intermediate Range Monitor (IRM)	**		X(d)	X(d)		2	3(nn)	
10. Main Steamline Isolation Valve Closure	**		X(b,s)	X(b)	X	2	4(nn)	
11. Turbine Trip Scram	**				X(j)	2	4(nn)	
12. Generator Load Rejection Scram	**				X(j)	2	2(nn)	
13. APRM Downscale/IRM Upscale	**				X(c)	2	3(nn)	
B. Reactor Isolation								
1. Low-Low Reactor Water Level	**	X	X	X	X	2	2(oo)	Close Main Steam Isolation Valves and Closed Isolation Condenser Vent Valve or PLACE IN COLD SHUTDOWN
2. High Flow in Main Steamline A	≤120% rated	X(s)	X(s)	X	X	2	2(oo)	

Individual electromatic relief valve control switches shall not be placed in the "Off" position for more than 8 hours (total time for all control switches) in any 30-day period and only one relief valve control switch may be placed in the "Off" position at a time.

i. With two core spray systems OPERABLE:

1. A maximum of two core spray booster pump differential pressure (d/p) switches may be inoperable provided that the switches are in opposing ADS trip system [i.e., only: either RV-40 A&D or RV-40 B&C]. Place the relay contacts associated with the inoperable d/p switch(es) in the de-energized position, within 24 hours. Restore the inoperable d/p switch(es) within 8 days, or declare ADS inoperable and take the action required by Specification 3.4.B.3;

or,

2. If two inoperable d/p switches are in the same ADS trip system [i.e., RV-40 A&B or RV-40 C&D], place the relay contacts associated with the inoperable d/p switch(es) in the de-energized position, within 24 hours. Restore the inoperable d/p switches within 4 days, or declare ADS inoperable and take the action required by Specification 3.4.B.3.

With only one core spray system OPERABLE:

If one or more d/p switches become inoperable in the OPERABLE core spray system, declare ADS inoperable and take the action required by Specification 3.4.B.3.

j. Not required below 40% of rated reactor THERMAL POWER.

k. All four (4) drywell pressure instrument channels may be made inoperable during the integrated primary containment leakage rate test (See Specification 4.5), provided that the plant is in the COLD SHUTDOWN condition and that no work is performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the TOP OF THE ACTIVE FUEL.

l. Bypass in IRM Ranges 8, 9, and 10.

m. There is one time delay relay associated with each of two pumps.

n. One time delay relay per pump must be OPERABLE.

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

56 **ID: 09-1 NRO56** **Points: 1.00**

The plant was at rated power when the Control Room was notified that Drywell pressure switches PS RV46A and PS RV46B, which inputs into the starting circuit for the Core Spray System, have failed in its current state such that they will not detect a high Drywell pressure condition.

Which of the following states the ability of the Core Spray System to function during a high Drywell pressure condition?

A. Core Spray Pumps A **and** B will auto start as designed, with no manual Operator actions required.

B. Core Spray Pump A will **not** auto start, but **may** be manually started. Core Spray Pump B will auto start as designed.

C. Core Spray Pump A will **not** start and **cannot** be manually started. Core Spray Pumps B **and** C auto start as designed.

D. **Neither** Core Spray Pump A **or** B will auto start, but can be manually started. All other Core Spray components operate as designed.

Answer: A

Answer Explanation:

QID: 09-1 NRO56		
Question # / Answer	**56**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295024 High Drywell Pressure 2.2.37 - Ability to determine operability and/or availability of safety related equipment.					**3.6**	**4.6**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**NU 5060E6003, sh. 1-4**	**2621.828.0.0010**		**RAP-C2f**		

Explanation	With no failures, a single high Drywell pressure signal will start the Core Spray System normally. This includes the Core Spray System A and B. There are 4 Drywell high pressure switches. If any two fails, there are still 2 others to start the Core Spray System in its normal start mode. Answer A is correct. Two instrument failures in RPS could render that RPS channel inoperable, but the Core Spray start logic is inter-mixed among systems. The other answers are plausible but incorrect since no manual operator actions are required.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0010 LO 209-10439	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

DW PRESS
HI-HI
RV 46 C/D

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** high drywell pressure. (Panel 1F/2F and 12XR) [ ]
- ❑ **VERIFY** start of core spray pumps and diesel generators. [ ]

AUTOMATIC ACTIONS:

Starts core spray pumps and diesel generators.

MANUAL CORRECTIVE ACTIONS:

- ❑ **ENTER** EMG-3200.01A, RPV Control - No ATWS [ ]

  OR

  - ❑ EMG-3200.01B, RPV Control with ATWS [ ]

  AND

  - ❑ EMG-3200.02 Primary Containment Control. [ ]

NOTE

This alarm indicates that a parameter has exceeded or has the potential to exceed an Emergency Action Level (EAL).

  - ❑ **REFER** to Procedure EP-AA-1010, Exelon Nuclear Radiological Emergency Plan Annex for Oyster Creek Station to determine EAL classification. [ ]

DW PRESS HI-HI RV 46 C/D

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Drywell pressure greater than 2.9 psig	2.9 psig	16K 115B (PS-RV46C) OR 16K115D (PS-RV46D)
		Reference Drawings: NU5060E6003 Sht. 2 & 4 GU 3E-611-17-005 Sh. 1

d. If valve is shut, it automatically opens and is interlocked open if initiation signal is received and reactor pressure < 285 psig (310 psig).

e. The discharge valve can be opened using the control switch if both Parallel valves in that system are closed.

7. Parallel Isolation Valves (V-20-15, 40, 21 & 41)

   a. Operated by individual control switches (close/normal/open).

      1) Close position shuts valves if initiation signal not present.

      2) Open position opens valves if reactor pressure < 285 psig (310 psig) or discharge valve is shut.

   b. Valves are interlocked open when an initiation signal is received and reactor pressure is < 285 psig (310 psig).

8. Testable Check Valves (V-20-151, 152, 153 & 154)

   a. System 1 Control Switch (Panel 1F/2F)

      1) Open A/Normal/Open C

   b. System 2 Control Switch (Panel 1F/2F)

      1) Open B/Normal/Open D

   c. Valve actuator not connected to disc.

      1) Combination of disc size, disc weight and actuator forces assure disc can be shut on reverse flow.

9. Core Spray Control Logic

   a. Channels A and B powered from 125 VDC Panel D.

   b. Channels C and D powered from 125 VDC Panel F.

   c. Located in cabinets ER18A & B in 480V Switchgear Room.

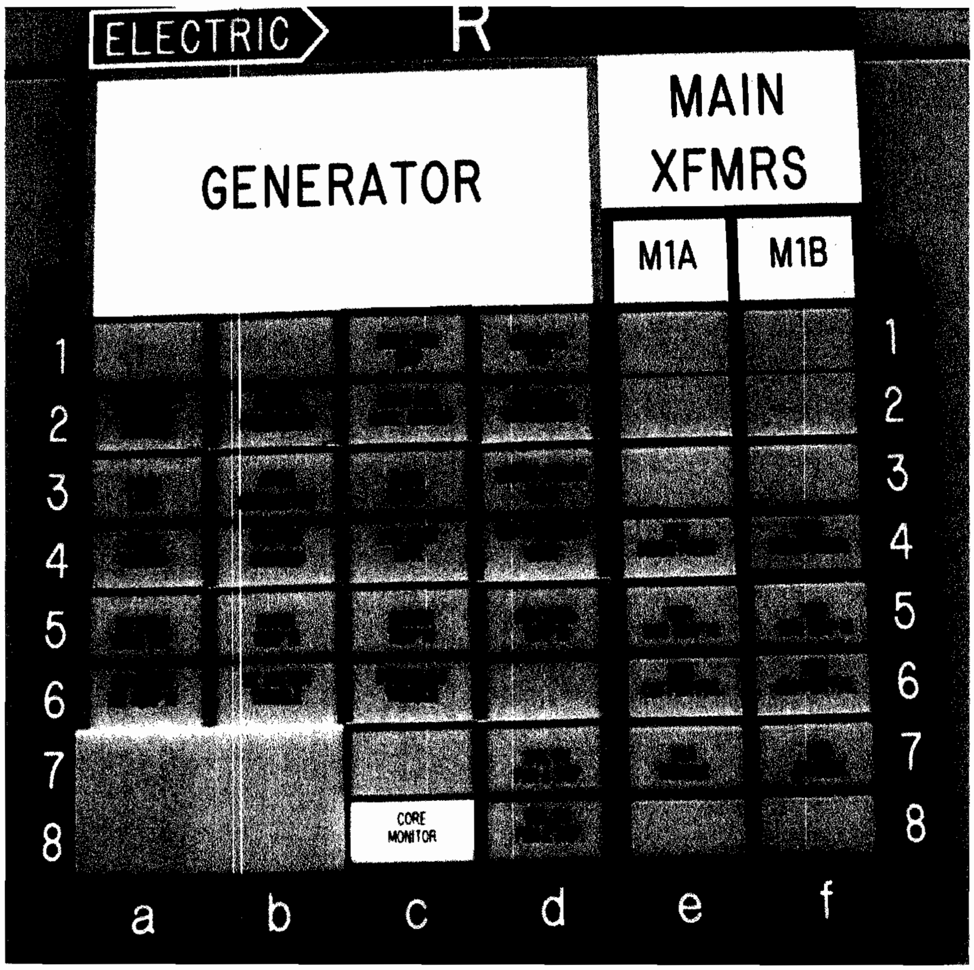
   d. Actuation of <u>any</u> channel starts both core spray systems and idle starts each EDG after a 10 sec. time delay in anticipation of a LOOP (Core Spray Control logic sensors are divisionally cross-linked: See CS logic mimic on 1F/2F).

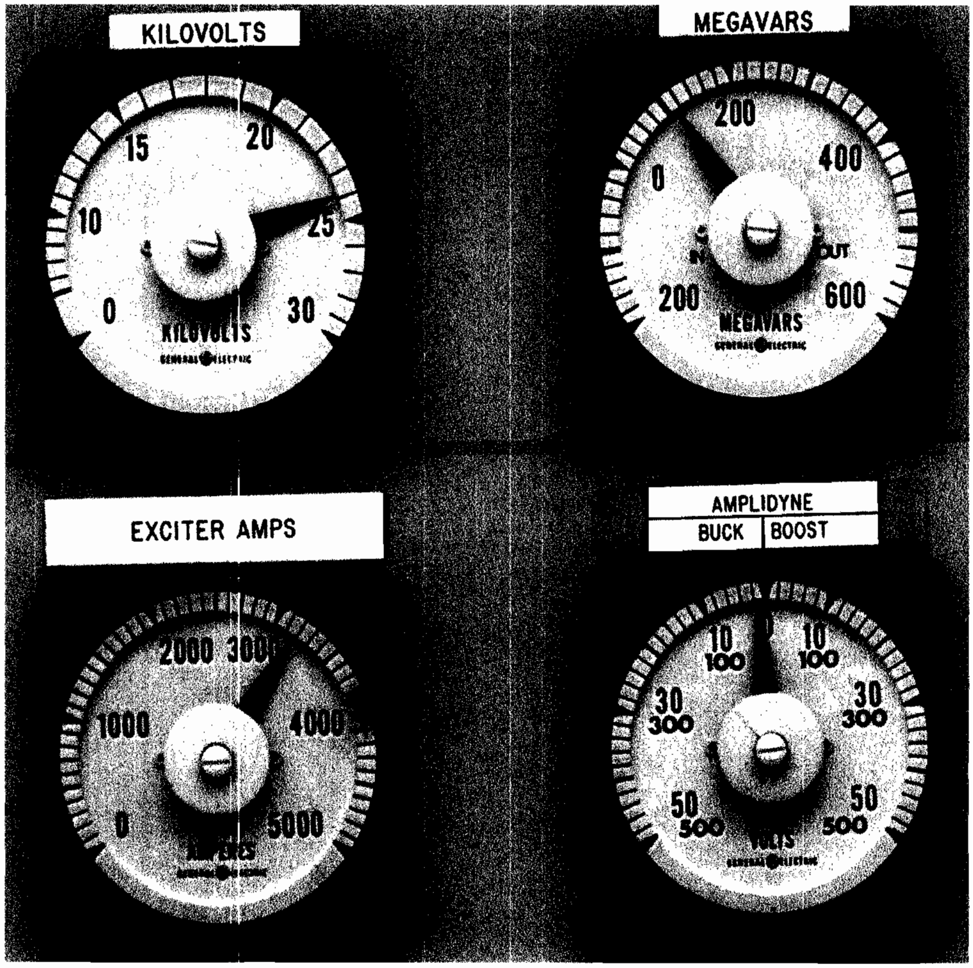
57 **ID: 09-1 NRO57** **Points: 1.00**

The plant is at rated power, and the electrical plant is in a normal lineup.

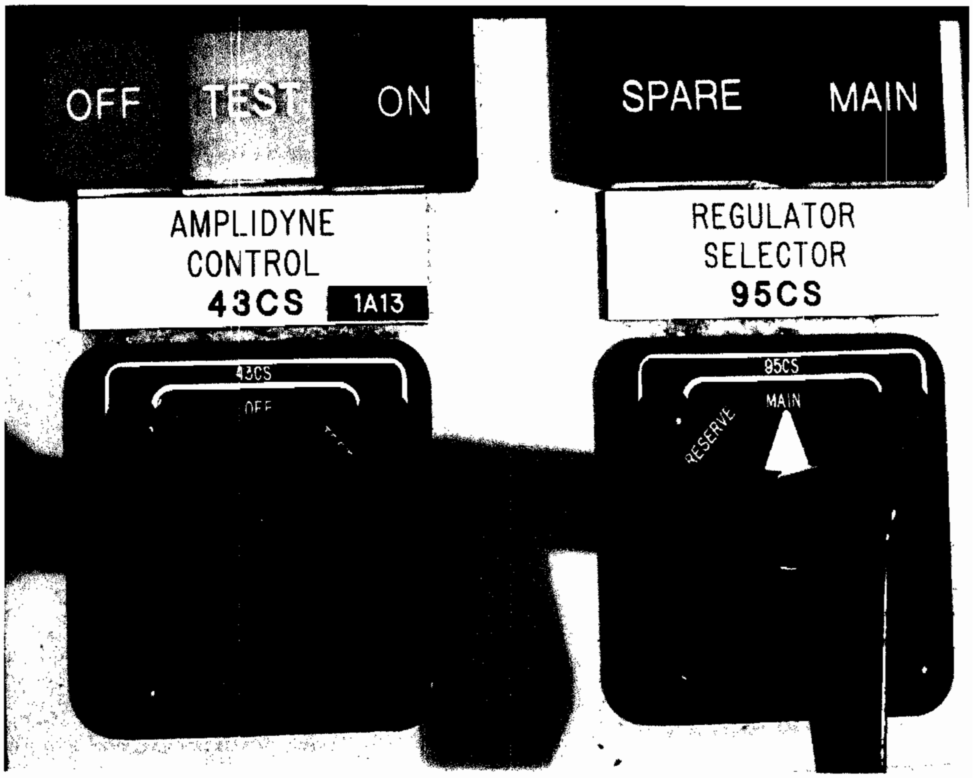
Which of the following indications or alarms, by themselves, would require entry into and performance of ABN-12, Generator Excitation Equipment Malfunction?

A.



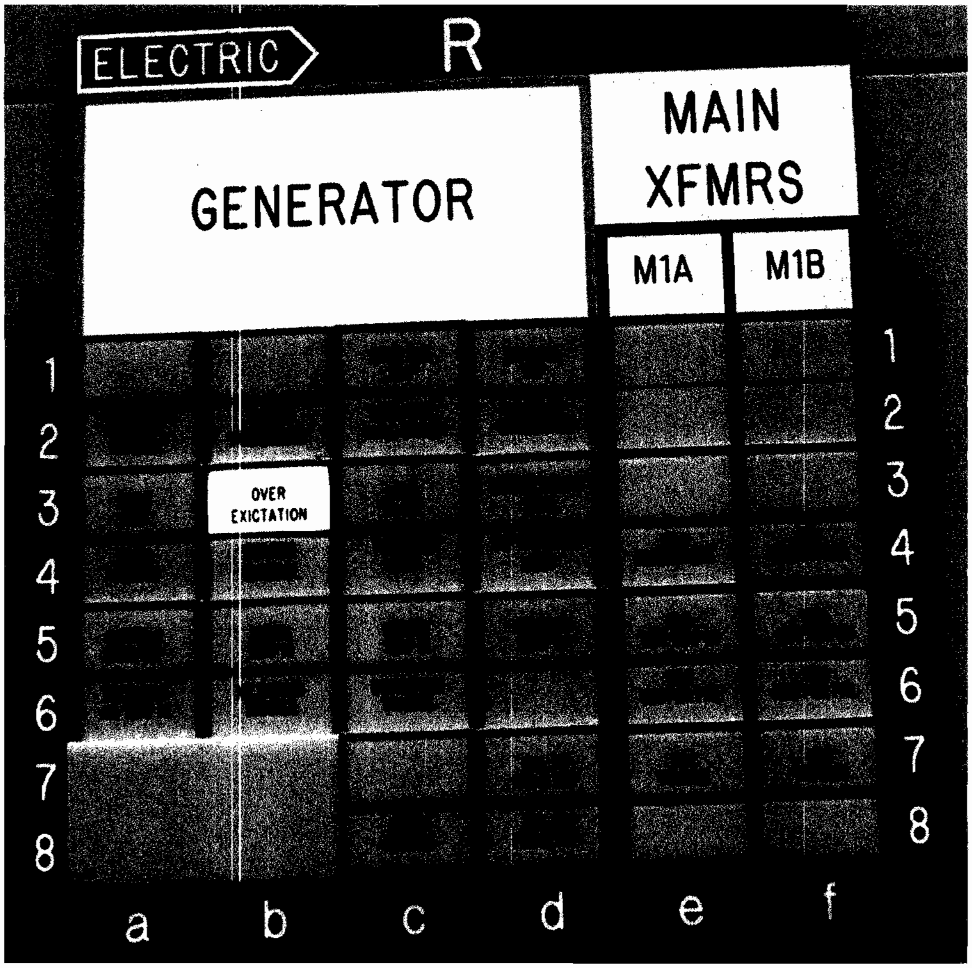
B. 
KILOVOLTS
15
20
10
25
0
30
KILOVOLTS
MEGAVARS
200
0
400
OUT
200
600
MEGAVARS
EXCITER AMPS
2000
3000
1000
4000
0
5000
AMPLIDYNE
BUCK
BOOST
10
100
10
100
30
300
30
300
50
500
50
500

C.



OFF
TEST
ON
SPARE
MAIN
AMPLIDYNE
CONTROL
43CS
1A13
REGULATOR
SELECTOR
95CS
43CS
95CS
RESERVE
MAIN

D.



Answer: C

Answer Explanation:

QID: 09-1 NRO57		
Question # / Answer	**57**	**Developer/Date: NTP 12/26/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

700000 Generator Voltage and Electric Grid Disturbances **2.4.4 - Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.**						**4.5**	**4.7**
Level	**RO**		**Tier**	**1**	**Group**	**1**	
General References	**ABN-12**		**336.1**			**RAP-R3b**	
Explanation	ABN-12 applies to the following events: 1) trip of the generator voltage regulator (amplidyne); 2) erratic operation of generator voltage regulation equipment (amplidyne); 3) Loss of 125 VDC control power to excitation switchgear. Answer C shows the amplidyne control switch in the position for automatic control but the red light is off and the green light is on which indicates the amplidyne has tripped. Answer C is correct. Answer A shows that the core monitor is in alarm and can be an indication of over-heating or insulation breakdown in the generator. This can be caused by load irregularities related to the amplidyne. ABN-12 does list this annunciator as an indication once the ABN is entered. But by itself, it is not an entry. Answer A is incorrect. Answer B shows normal full load indications for volts and vars. If the amplidyne were faulty, it could result in generator voltage out of normal limits (23.3 - 24.7 KV). Answer B is incorrect. Answer D could be a result of a faulty amplidyne, and it is noted in ABN012 as an indication. But by itself, it is not an entry into ABN-12. Answer D is incorrect.						
References to be provided during exam:		**None**					
Learning Objective	2621.828.0.0025 LO 248-10445						

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X** **3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Exelon Nuclear	OYSTER CREEK GENERATING STATION PROCEDURE	Number **ABN-12**
Title **GENERATOR EXCITATION EQUIPMENT MALFUNCTION**		Revision No. 2

1.0 APPLICABILITY

This procedure is applicable to the following events: Section

- Trip of Generator Voltage Regulator (Amplidyne) 4.1
- Erratic operation of Generator voltage regulation equipment (Amplidyne) 4.2
- Loss of 125 VDC control power to excitation switchgear. 4.3

2.0 INDICATIONS

2.1 Annunciators

Engraving	Location	Setpoint
MN XCITER AIR TEMP HI	R-6-a	60 °C
VARS HI/LO	R-3-c	Variable
ROTOR TEMP HI	R-5-b	Various
CORE MONITOR	R-8-c	50% ion current
POT XFMR LOST	R-4-c	N/A
FIELD GROUND	R-4-a	Variable
OVER EXCITATION	R-3-b	Variable

2.2 Plant Parameters

Parameter	Location	Change
Main Generator KILOVOLTS	8F/9F	Various
Main Generator MEGAVARS	8F/9F	Various
EXCITER AMPS	8F/9F	Various

58 **ID: 09-1 NRO58** **Points: 1.00**

The plant was at rated power, when a large LOCA inside the Primary Containment occurred.

IAW the EOP Users Guide, elevated temperatures in the Drywell can lead to elevated temperatures in the RPV water level instruments' ________(1)_______ leg which may result in a false indicated RPV water level. This false indicated RPV water level could ________(2)_______ .

	(1)	(2)
A.	reference	result in Core Spray initiation
B.	variable	require manual tripping of CRD Pumps
C.	reference	result in ROPS initiation
D.	variable	result in Isolation Condenser initiation

Answer: C

Answer Explanation:

QID: 09-1 NRO58		
Question # / Answer	**58**	**Developer/Date: NTP 12/26/09**

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
295028 High Drywell Temperature EK2.02 - Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Components internal to the drywell				3.2	3.3
Level	RO	Tier	1	Group	1

General References	**EOP Users Guide**	**RAP-H5d**	
Explanation	The plant was at rated power when a large LOCA occurred inside the Drywell. Elevated Drywell temperatures can lead to elevated temperatures in the RPV water level instrument reference legs and a higher than normal indicated water level due to the change in water density. ROPS initiation occurs on an RPV high water level signal (181"). Answer C is correct. Since the variable leg is incorrect, then answers B & D are incorrect. Also, ABN-1 requires tripping CRD Pumps if RPV water level cannot be maintained below 170". Answer A has the correct level leg but Core Spray starts on an RPV low water level signal, not high. Answer A is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2624.845.0.0052 LO 3053		

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:RI**
	NUREG 1021 Appendix B: Recognize interaction between systems, including consequences and implications			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

DRYWELL TEMPERATURE CONTROL OVERVIEW

CONTROL DW TEMP BELOW 150°F USING AVAILABLE DW COOLING

↓

OPERATE ALL DW COOLERS WHEN DW TEMP CAN'T BE MAINTAINED BELOW 150° F

↓

ENTER RPV CONTROL AND INITIATE DRYWELL SPRAYS BEFORE 281° F

↓

DEPRESSURIZE WHEN DW TEMP CAN'T BE RESTORED AND MAINTAINED < 281° F

DISCUSSION

A general overview of the major steps in the Drywell Temperature Control leg is illustrated above.

Some events which may affect Drywell temperatures are loss of Drywell cooling, EMRV operation, and steam leaks into the Drywell.

Adverse effects of high Drywell temperature may include but are not limited to:

- Increasing containment pressure with increasing Drywell temperature (ideal gas law PV=mRT.)
- Potential failure of safety-related equipment, non-safety-related equipment, and containment structural components if temperatures exceed their qualification or design temperatures.
- High Drywell temperatures may affect RPV water level instrumentation. Elevated temperatures at the instrument reference legs will cause the instrument to read higher than normal due to a change in density. Instrument run temperatures in excess of saturation conditions for a given RPV pressure will make the instrument inoperable due to possible boiling in the instrument run.

Group Heading DW PRESS	H - 5 - d

ROPS ACTUATE A AND	

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Rx Water Level rising.	181" TAF	PNL-629-14XRCR1
		Reference Drawings: GU 3D-629-17-002 GU 3E-611-17-010

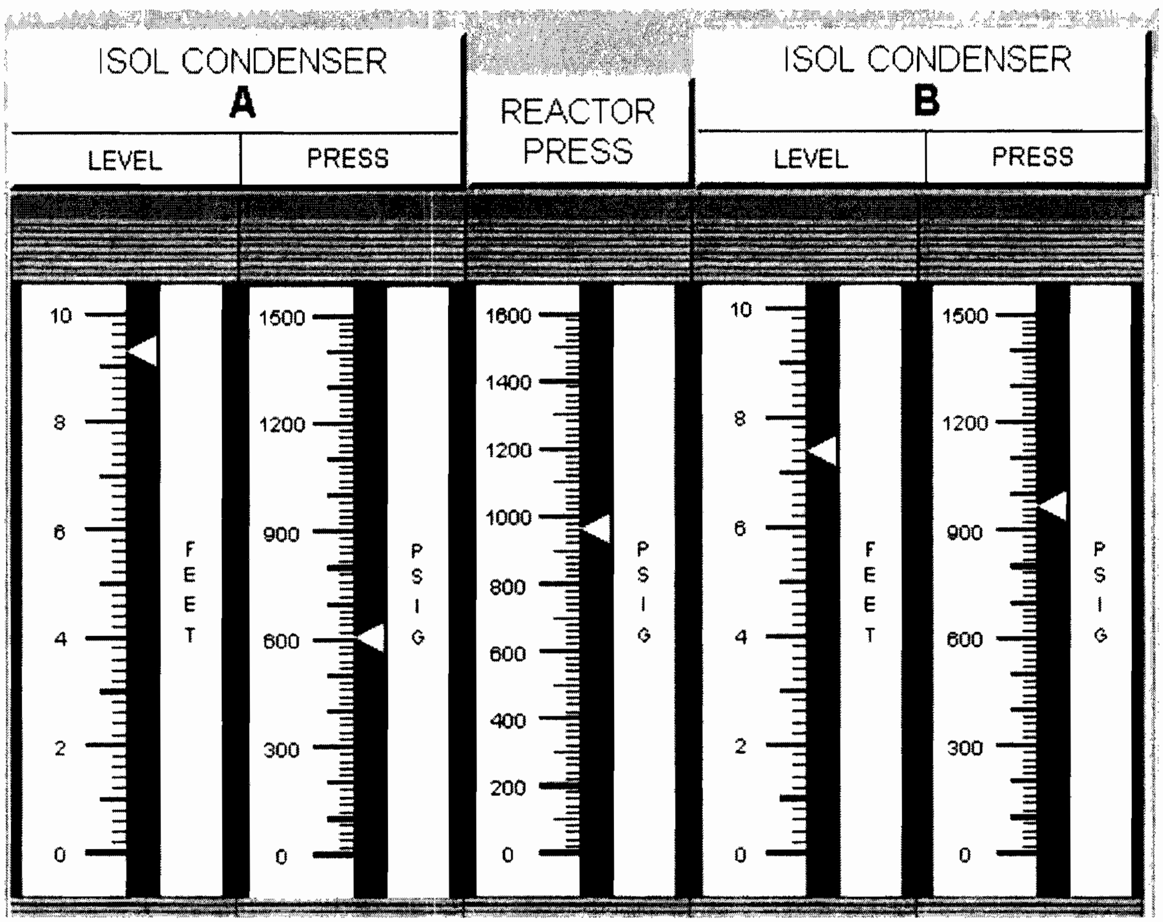
Subject	Procedure No.		
NSSS Alarm Response Procedures	RAP-H5d	Page 2 of 2	H - 5 - d
	Revision No: 0		

59 **ID: 09-1 NRO59** **Points: 1.00**

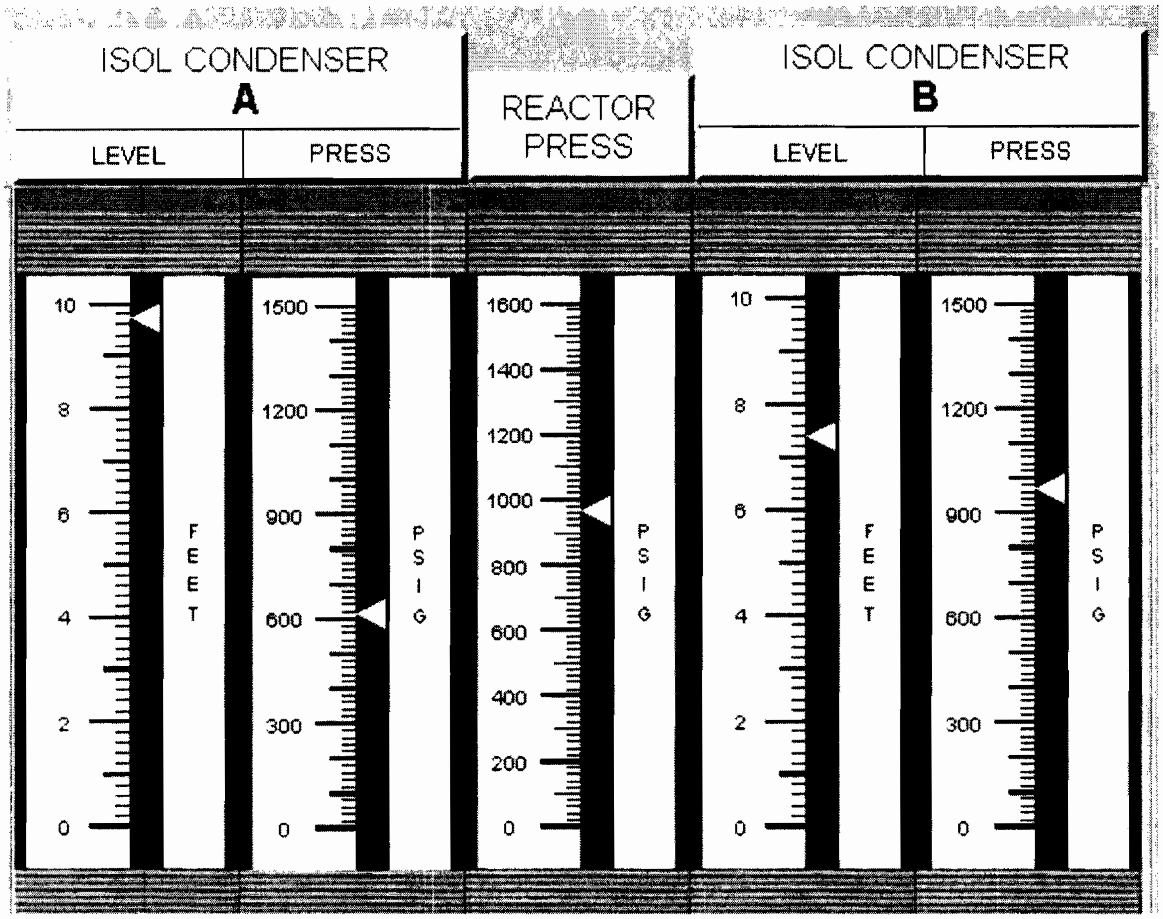
The plant was at rated power with elevated offgas radiation readings, when an event occurred.

The Operator reported the following alarms and indications:

- Annunciator AREA MON HI has alarmed and the ISOLATION COND AREA ARM indicated 80 mr/hr and steady
- Annunciator COND A FLOW HI POSSIBLE RUPTURE has alarmed
- See the indications below



Five minutes later, the Operator observes the following indications:



Which of the following is correct?

A. A radiological release **was** in progress and Isolation Condenser A **had** successfully isolated.

B. A radiological release **is** in progress and Isolation Condenser A **has** failed to automatically isolate and should be manually isolated.

C. An Isolation Condenser A steam leak into the reactor Building occurred and Isolation Condenser A **had** successfully isolated.

D. An Isolation Condenser A steam leak into the Reactor Building occurred and Isolation Condenser A **has** failed to automatically isolate.

Answer: B

Answer Explanation:

QID: 09-1 NRO59		
Question # / Answer	59	Developer/Date: NTP 12/26/09

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
295017 High Off-site Release Rate AK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE : Protection of the general public				3.8	4.3
Level	RO	Tier	1	Group	2
General References	RAP-C3a				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant was at power when indications are provided which show the following: Isolation Condenser A shell water level is high and pressure is low. The Isolation Condenser area radiation monitor is alarming, and the Possible Rupture alarm is in alarm. Two minutes later, indications are shown again. This indication shows that Isolation Condenser A shell water level is even higher. When the Possible Rupture comes in, after a 27 second time delay, the Isolation Condenser A isolation valves will go closed. Thus, about 1.5 minutes after this alarm, the valves must still be open since the shell water level is still rising. Therefore, there is a tube leak on Isolation Condenser A resulting in an offsite release (the shell directly vents outside of the reactor Building), the isolation has failed, and the Operator should manually perform the actions to isolate the Isolation Condenser. Answer B is correct. If the candidate does not realize that shell water level is still rising, then it will appear that the condenser automatically isolated. Answer A is incorrect. The possible Rupture annunciator could also be indicative of a steam line break in the Reactor Building, especially with the Isolation Condenser ARM in alarm. But since a steam line break would not impact shell water level as in the question, answers C & D are incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0023 LO 2338	

Question Source (New, Modified, Bank)			**New**		
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**		**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning				
10CRF55 Content	**55.41**	**10**	**55.43**		
	(SRO Only)				
Time to Complete: 1-2 minutes					

COND A FLOW HI POSSIBLE RUPTURE

CONFIRMATORY ACTIONS:

- **VERIFY** Closed System A Isolation Valves. [ ]

Check for indication of pipe break:

- Annunciator C-8-b, COND AREA TEMP HI alarmed. [ ]
- Rise in area temperatures. (Panel 10R) [ ]
- **CHECK** level changes. (Panel 2F) [ ]
- **CHECK** shell temperature rise on TR IG02. (Panel 2F) [ ]

Check for indication of tube leak:

- **CHECK** level changes. (Panel 2F) [ ]
- **CHECK** shell temperature rise on TR IG02. (Panel 2F) [ ]

COND A FLOW HI POSSIBLE RUPTURE

AUTOMATIC ACTIONS:

Closes Isolation Condenser System A Valves.

- V-14-30, Steam Inlet Valve to ‘A” Emergency Condenser
- V-14-31, Steam Inlet Valve to ‘A’ Emergency Condenser
- V-14-34, Emergency Condenser NE01A Condensate Return Valve
- V-14-36, Isolation Valve Emergency Condenser NE01A

MANUAL CORRECTIVE ACTIONS:

NOTE

This alarm indicates that a parameter has exceeded or has the potential to exceed an Emergency Action Level (EAL).

❑ **REFER** to Procedure EP-AA-1010, Exelon Nuclear Radiological Emergency Plan Annex to determine EAL classification. [ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 3 of 4)

COND A
FLOW HI
POSSIBLE
RUPTURE

MANUAL CORRECTIVE ACTIONS: **(continued from Page 2 of 4)**

❑ IF pipe break tube leak is verified,

THEN **PERFORM** the following:

- **EVACUATE** Reactor Building. [ ]
- **PLACE** V-14-30, V-14-31, V-14-34, V-14-36 A Iso Cond Isolation Valves Control Switches to CLOSE. [ ]
- **PLACE** V-14-5 and V-14-20, Emergency Condenser NE01A High Point Vent Valves Control Switches to CLOSE. [ ]

❑ IF no pipe break tube leak is indicated,

THEN **RETURN** A ISO COND to Service as follows:

- **OPEN** V-14-30 and V-14-31. (Panel 1F/2F) [ ]
- **POSITION** V-14-34 as directed by the Unit Supervisor. (Panel 1F/2F) [ ]
- **OPEN** V-14-36. (Panel 1F/2F [ ]
- **RESET** the Isolation Condenser isolation signal using the Isolation Condenser Reset pushbutton. (Panel 4F) [ ]
- **PLACE** all Isolation Condenser A valve control switches to AUTO. [ ]

COND A FLOW HI POSSIBLE RUPTURE

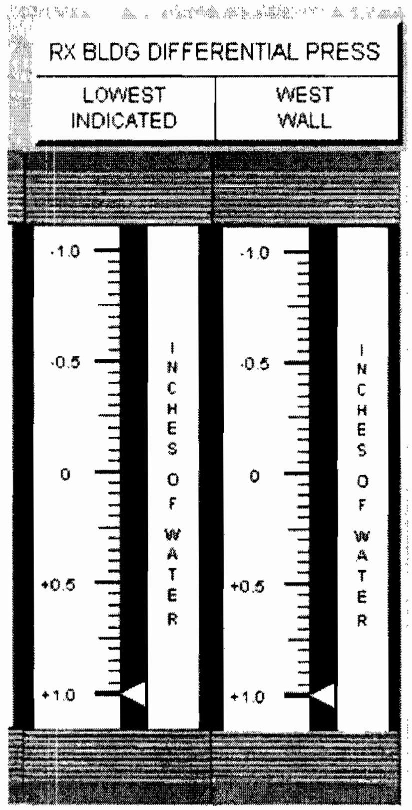
CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Sustained (27 seconds) High Steam Flow	15 psig	6K5A via 6K7B from: IB05A2 or IB11A2
OR		OR
Sustained (27 seconds) High Condensate Flow	24" H_2O DP	6K3A via 6K7B from: IB05A1 or IB11A1

Reference Drawings:

BR 3029 Sh.2
GE 148F262
GU 3E-611-17-005 Sh. 1

60 **ID: 09-1 NRO60** **Points: 1.00**

The plant was at rated power when an unisolable steam leak began in the Reactor Building. The Operator reports the following observations (see below):



Which of the following states the status of Reactor Building HVAC and the Standby Gas Treatment System (SGTS)? (Assume **no** Operator actions)

	RB HVAC	SGTS
A.	Tripped	In Standby
B.	Tripped	Running
C.	Running	In Standby
D.	Running	Running

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer: A

Answer Explanation:

QID: 09-1 NRO60		
Question # / Answer	**60**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating RO**	**Importance Rating SRO**
295035 Secondary Containment High Differential Pressure EK2.01 - Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following: Secondary containment ventilation						**3.6**	**3.6**
Level	**RO**	**Tier**	**1**	**Group**	**2**		
General References	**GE 157B6350 sh. 72A**	**BR 3017**	**2621.828.0.0042**				
Explanation	The plant is at power when an unisolable leak starts in the RB. The Operator reports that RB Dp is +1.0 inches/water. At this level, the normal RB HVAC trips to prevent over-pressurizing the RB. The same signal has no input into the auto start of SGTS and it remains in standby. Answer A is correct. All other answers are plausible but incorrect. Note that RB HVAC and SGTS can run simultaneously.						
References to be provided during exam:	**None**						
Learning Objective	2621.828.0.0042 LO 261-10445						

Question Source (New, Modified, Bank)				**Modified**
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			

Time to Complete: 1-2 minutes

1) Functions as air locks, are interlocked to prevent opening both doors together which would open Secondary Containment.

2) An interlock override push button located inside the air lock.

   a) Allows opening both doors simultaneously.

   b) Used only when authorized by the SM.

2. RBVAC

   a. RB Supply Fans SF-1-12, 13, 14.

      1) Controlled by individual ON, NEUTRAL, OFF switches on Panel 11R.

      2) Starting any supply fan interlocks open all ten pairs of RB supply header valves and the individual fan automatic inlet damper.

      3) Fans trip and the supply header valves and fan discharge dampers close on any of the following signals:

         a) Rx Op. Floor high radiation - 50 mR/hr w/2 min. TD.

         b) Fuel pool area high radiation - 50 mR/hr w/2 min. TD.

         c) RB ventilation exhaust high radiation - 9 mR/hr w/no TD. Either 1 or 2 sensors.

         d) Lo-Lo wtr. lvl. - 90" TAF

         e) High DW press. - 2.9#

         f) RB high pressure - 1" H_2O

         g) Either V-28-21 or V-28-22 Not full open.

         h) High Temp in Ventilation Duct (1 of 4 at 300°F.)

   b. RB Exhaust Fans EF-1-5 or 6

      1) Controlled by individual PULL-TO-LOCK, STOP, START switches on Panel 11R.

b. Individual train outlet temperatures are indicated on Panel 11R in the Control Room.

c. Temperature elements and controllers are provided for operational control of the electric heating coils.

2. Flow

a. Flow in the common suction piping is indicated on the local Train A panel only (ATC-P-15). Caution should be taken when running Train B to take data from both local panels.

b. Individual train flow rates are indicated at the associated local panels and provide Control Room alarms.

3. Differential pressure

a. Each of the HEPA and charcoal filters has both local and remote (at the local panels) indications for differential pressure.

b. A train differential sensed across the entire filter train provides indication at the local panels and an alarm in the Control Room.

F. Controls & Interlocks

1. SGTS

a. Lead Train Select Switch

1) One SGTS train is normally selected as the lead train with a SYSTEM 1\SYSTEM 2 Control sw. on 11R.

2) When an initiation signal is received, both trains start and run until the lead train attains normal flow.

3) When the lead train reaches rated flow, the backup train stops and the inlet and outlet valves shut after a 148 second time delay.

4) Initiating signals:

a) Lo-Lo Lvl - 90" TAF

b) Hi DW Press – 2.9#

c) Hi RB vent Rad - 9 mR/hr (No TD)

d) Hi Fuel pool area rad. - 50 mR/hr w/2 min. TD

e) Hi Refuel Floor Rad - 50 mR/hr w/2 min. TD

b. Exhaust Fans EF-1-8, 9

1) Controlled by HAND/OFF/AUTO switches on 11R.

2) In HAND the fan starts and the train valves automatically line up for operation.

a) Low Flow alarm is inoperative while in HAND.

3) Returning the switch to OFF stops the fans and realigns the valves for standby service.

4) In AUTO, SGTS starts and the valves align for operation on any of the trip signals.

a) In addition to the lead train the standby fan starts, and all valves except the cross-tie shift to a normal operating position.

b) When normal flow is sensed, the standby train will shutdown.

c) If the lead, train fan trips, the standby train will take over and the cross-tie will remain open to cool the shutdown train through the orifice purge inlet.

c. Cross-tie V-28-48

1) Controlled by a CLOSE/AUTO switch on 11R.

2) In AUTO the cross tie remains open when SGTS initiates and closes when normal flow is sensed in the lead train.

3) In CLOSE the valve shuts.

4) If the lead train trips after an auto initiation the cross-tie opens to provide cooling for the tripped train.

d. EHC-1-5, 6 and Strip Heaters

1) EHC-1-5, 6 are automatically energized when normal flow is sensed in the lead train.

2) Cycle automatically to maintain optimum running train temperatures for the removal of radioactive materials from the process flow.

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

61 **ID: 09-1 NRO61** **Points: 1.00**

The plant was at rated power when an unisolable steam leak began in the Reactor Building, and the Secondary Containment Control EOP was entered.

Conditions worsened and the SRO has directed Emergency Depressurization due to the radiation levels in the Reactor Building.

IAW the EOP Users Guide, which of the following states the bases for directing an Emergency Depressurization?

1. It places the RPV in the lowest energy state
2. It reduces the driving head on the leak
3. It allows low pressure systems to inject into the RPV
4. It reduces the amount of energy available to be deposited inside the Primary Containment

A. 1 **only**

B. 1 and 2

C. 2 and 4

D. 1, 2, and 4

Answer: B

Answer Explanation:

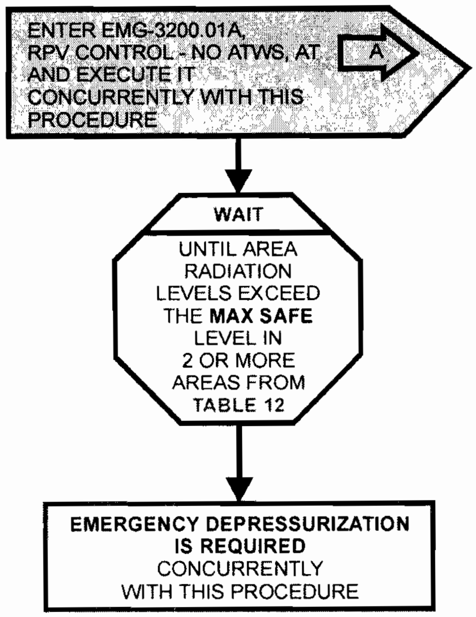
QID: 09-1 NRO61		
Question # / Answer	**61**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
295033 High Secondary Containment Area Radiation Levels EK3.01 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Emergency depressurization	**3.3**	**3.5**

Level	RO	Tier	1	Group	2
General References	EOP Users Guide				
Explanation	IAW the EOP Users Guide, the radiation increases is so wide spread that is poses a direct threat to secondary containment integrity, equipment located in the secondary containment or continued safe operation. ED will place the plant in its lowest energy state and will reduce the driving head and flow from primary systems that are discharging into the secondary containment. Answer B (1 and 2) is correct. Answer A is incorrect since it does not state both reasons for ED. ED is performed by opening the EMRVs which releases the energy from the RPV into the Torus. ED does not reduce the amount of energy to be released to the primary containment but into the Secondary Containment. Thus selection 4 is incorrect and answer D is incorrect. It is true that as RPV pressure lowers from the ED, more injection systems become available to inject into the RPV to compensate for the leak. But, this is not a reason for the ED in the question. Selection 3 is incorrect and answer C is incorrect. **Note**: A question on the most previous ILT NRC exam asked why ED is required due to secondary containment temperatures (and this question refers to radiation levels, and the question is basically the same question)				
References to be provided during exam:	None				
Learning Objective	2621.845.0.0057 LO 3082				

Question Source (New, Modified, Bank)			Bank	
Cognitive Level	Memory or Fundamental Knowledge	X 1:B	Comprehension or Analysis	
	NUREG 1021 Appendix B: Bases or purpose			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

SECONDARY CONTAINMENT RADIATION CONTROL



DISCUSSION

Should Secondary Containment radiation levels continue to increase and exceed their Maximum Safe Operating values in more than one area, an emergency depressurization is directed to limit further increases in area radiation levels.

Emergency RPV depressurization rapidly places the RPV and its attached primary systems in the lowest possible energy state and reduces the driving head and flow from primary systems that are unisolated and discharging into Secondary Containment.

The "2 or more areas" criterion ensures that the rise in Secondary Containment radiation level is a wide-spread problem posing a direct and immediate threat to personnel both onsite and offsite.

One parameter (e.g., radiation) above its Maximum Safe Operating value in one area and a different parameter (e.g., temperature or water level) above its Maximum Safe Operating values in the same or another area is not a condition requiring emergency depressurization. A combination of parameters exceeding Maximum Safe Operating values in one area does not necessarily indicate that control of a given parameter cannot be maintained or that previous actions have not been effective in confining the trouble to one area. Expanding the application of "more than one area" to encompass multiple parameters unnecessarily complicates the procedure and might lead to depressurization of the RPV when such action is not, in fact, appropriate or needed.

62 **ID: 09-1 NRO62** **Points: 1.00**

Refueling was in progress when an accident occurred on the refuel floor. The Standby Gas Treatment System initiated **immediately** when sensed radiation levels went above the setpoint.

Which of the following states the logic necessary for the SGTS automatic start described above?

A. Radiation levels ≥ 9 mr/hr as sensed by **both** RB Vent Radiation Monitors.

B. Radiation levels ≥ 9 mr/hr as sensed by **either** RB Vent Radiation Monitor.

C. Radiation levels ≥ 50 mr/hr as sensed by **both** Refuel Floor Radiation Monitors.

D. Radiation levels ≥ 50 mr/hr as sensed by **either** Refuel Floor Radiation Monitors.

Answer: B

Answer Explanation:

QID: 09-1 NRO62		
Question # / Answer	**62**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295034 Secondary Containment Ventilation High Radiation **EA1.04 - Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : SBGT/FRVS**					**4.1**	**4.2**
Level	**RO**	**Tier**	**1**	**Group**	**2**	
General References	**651.4.001**					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	SGT will auto start from $\geq$9 Mr/hr on EITHER RB vent radiation monitor (with no time delay), OR high radiation (50 Mr/hr) on the refuel floor rad monitors (with a 2-minute time delay). Since there was an immediate SGTS start, the RB vent rad monitors must have sensed rad levels above their setpoint. Therefore, only one RB vent radiation monitor had to reach its setpoint of 9 mr/hr to automatically auto start SGTS. Answer B is correct. The other answers are plausible if the SGTS start logic is not understood.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0042 LO 261-10450		

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints, or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

6.4 Reactor Building Ventilation Exhaust Monitor No. 1 Test (RN04-A1)

6.4.1

NOTE

Test SGTS I with monitor No. 1 and SGTS II with monitor No. 2. If one SGTS is **not** operable, test the available system with both monitors.

NOTE

Strip heaters need **not** be reset until completion of all testing.

CAUTION

When normal Reactor Building ventilation system is **not** operating and the water evaporator is running, there is a potential for radioactive contamination due to water condensing out in the ductwork and dripping through the duct joints located in the CRD Rebuild Room.

USE Data Display Mode on recorder R006B (panel 10F) to monitor RB Vent Manifold No. 1 - Chan. No. 7 [ ]

6.4.2 **RECORD** the following on Attachment 651.4.001 –1:

6.4.2.1 Trip and Indicator Unit Meter Indication Monitor No. 1 (RN04-A1) [ ]

6.4.2.2 Recorder Indication to the nearest 0.1 mr/hr. [ ]

6.4.3 **CONFIRM** Standby Gas Select Switch position as follows:

SYS 1 [ ]

OR

SYS 2 (in case system 1 is **not** available for testing) [ ]

6.4.4 **PERFORM** Upscale Trip of RN04-A1 as follows:

6.4.4.1 **DEPRESS** and **HOLD** the "Trip Check" button on the Trip and Indicator Unit RN04-A1. [ ]

6.4.4.2 **ROTATE** the "Trip Check Adjust" potentiometer on RN0037 clockwise to the upscale trip point (9 ±1 mr/hr). [ ]

6.4.4.3 **RECORD** the "As Found" meter trip point on Attachment 651.4.001 –1. [ ]

6.4.4.4 **VERIFY** "VENT HI" (10F-1-f) alarm is received on Attachment 651.4.001 –1. [ ]

6.4.4.5 **ROTATE** the "Trip Check Adjust" potentiometer on RN0037 counterclockwise to below the trip point. [ ]

6.4.4.6 **PERFORM** the following simultaneously:

- **RELEASE** the "Trip Check" button [ ]
- **START** the Stopwatch [ ]

6.4.4.7 WHEN SGTS initiates,

THEN **STOP** the Stopwatch. [ ]

6.4.5 **RECORD** the time delay prior to initiation of the SGTS (initiation should be instantaneous) on Attachment 651.4.001 –1. [ ]

6.4.6 **VERIFY** the following occurs as described in Attachment 651.4.001-2, Normal Operation of SGTS System 1.

- Reactor Building ventilation system isolation [ ]
- SGTS initiation [ ]

6.4.6.1 **RECORD** on Attachment 651.4.001 –1. [ ]

6.4.7

> NOTE
>
> EF1-8 will continue to run after a low flow signal with its associated inlet and outlet valves shut. Manual shutdown of the fan is required.

RESET the Radiation Monitor by depressing the "TRIP RESET" button on the Trip and Indicator Unit RN04-A1. [ ]

6.4.7.1 **VERIFY** that the SGTS has shutdown on Attachment 651.4.001 –1. [ ]

63 **ID: 09-1 NRO63** **Points: 1.00**

The plant was starting up after an outage. The following conditions currently exist:

- All IRMs are mid-range on Range 8 and **stable**
- Turbine warming is in-progress

The Operator reports the following:

- Annunciator COND VAC LO 25 INCHES has alarmed
- CONDENSER VACUUM 1A, 1B and 1C indicated 24" HG and are **degrading** at a rate of ½ "HG/minute
- RPV pressure is 580 psig and **stable**

Which of the following is correct given the above conditions (assume **no** operator actions)?

A. The reactor will **not** scram on low condenser vacuum.

B. The turbine trip will generate a reactor scram in 4 minutes.

C. A low condenser vacuum will generate a reactor scram in 4 minutes.

D. An RPV high pressure will generate a reactor scram in 8 minutes when the Turbine Bypass Valves close.

Answer: A

Answer Explanation:

QID: 09-1 NRO63		
Question # / Answer	**63**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
295002 Loss of Main Condenser Vac AA2.02 - Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM : Reactor power	**3.2**	**3.3**

Level	RO	Tier	1	Group	2
General References	RAP-G4c	RAP-J1b			
Explanation	The question stem shows that the reactor is starting up (with the mode switch in STARTUP) with RPV pressure at 580 psig and steady. At this low pressure, the main condenser low vacuum scram signal **and** the turbine stop valve closure scram signal are bypassed. The low vacuum scram setpoint is 22" hg, and the turbine bypass valves close at 20" hg. In 4 minutes, condenser vacuum drops to 22" hg, which is the scram and turbine trip setpoint (currently bypassed). At this low RPV pressure, the low vacuum signal is bypassed and the reactor will not scram from a low vacuum scram signal or from the turbine trip, and power remains at its current power level. Answer A is correct. The turbine will receive a trip signal in 4 minutes (when condenser vacuum lowers to 22") but the scram is bypassed since RPV pressure is < 600 psig. Answer B is incorrect. Condenser vacuum will lower to 22" in 4 minutes, but this scram signal is also bypassed. Answer c is incorrect. When condenser vacuum reaches 10" (or in 28 minutes later), the turbine bypass valves will close which could result in a rising RPV pressure to the scram setpoint. But, in 8 minutes only, the low condenser vacuum trip #2 to close the turbine bypass valves has not yet been reached. Answer D is incorrect.				
References to be provided during exam:	None				
Learning Objective	2621.828.0.0037 LO 212-10445				

Question Source (New, Modified, Bank)				Modified
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading REACTOR/RPS	G-4-c

RPS 600#/ SD BYPASS	

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** reactor pressure is less than 600 psig. (Panel 5F/6F) [ ]

  OR

  MODE switch is in SHUTDOWN. (Panel 4F) [ ]

- ❑ **VERIFY** reactor mode switch is **not** in RUN position. [ ]

AUTOMATIC ACTIONS:

Condenser low vacuum and main steam isolation valve closure scram bypassed.

MANUAL CORRECTIVE ACTIONS:

> NOTE
>
> This alarm is normal for REFUEL and STARTUP modes of operation with reactor pressure less than 600 psig **or** whenever mode switch is in SHUTDOWN mode after a 20 second time delay.

[redacted]

CONFIRMATORY ACTIONS:

For half scram condition

- ❑ **CHECK** Offgas Flow Recorder indicator at Panel 10F. [ ]

AUTOMATIC ACTIONS:

NOTE

Scram functions for Low Vacuum, Stop Valve closure, and Low Turbine Control Oil Pressure are bypassed:

- With mode switch in STARTUP or REFUEL, with reactor pressure less than 600 psig;
- With mode switch in SHUT DOWN at any reactor pressure after 20 second time delay.

Scram functions for Stop Valve closure and Low Turbine Control Oil Pressure are bypassed whenever steam flow to the Turbine is less than 30%.

Reactor scram with coincident Reactor Protection System Channel II trip.

[redacted]

MANUAL CORRECTIVE ACTIONS

For descended vacuum

- ❑ **REFER** to ABN-14, Loss of Condenser Vacuum. [ ]

For full scram condition

- ❑ **REFER** to ABN-1, Reactor Scram. [ ]

For Turbine Trip

- ❑ **REFER** to ABN-10, Turbine Trip. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Less than 22.0" Hg vacuum in the condenser (RSCS-11, -21) or turbine trip as indicated by closure or the main stop valves or low turbine control oil pressure. These trips are input to Reactor Protection System Channel I.	22.0" Hg vacuum or turbine tripped	Relays 1K11 and 1K12 Reference Drawings: GE 237E566, Sht. 1 GE 233R309, Sht. 2 GU 3E-611-17-011

64 **ID: 09-1 NRO64** **Points: 1.00**

The plant was at 60% power during power ascension following a 30-day refuel outage. The A Recirculation Pump was out of service and was in an idle configuration.

An event then occurred which resulted in a rising RPV pressure. The Operator reported the following:

- RPV pressure resulted in several EMRVs opening for 5 seconds. All AUTO DEPRESS VALVE indications currently indicate **green** light on
- The reactor scrammed and **all** IMMEDIATE OPERATOR ACTIONs of ABN-1, Reactor Scram, were completed

With **no** further Operator action, which of the following is correct?

A. RPV water level is controlling at 142" with the MFRV in AUTO.

B. RPV pressure will **lower** due to the initiation of the Isolation Condensers.

C. RPV pressure will be maintained by the Electronic Pressure Regulator (EPR).

D. **All** reactor Recirculation Pumps tripped **immediately** upon reaching the high pressure setpoint.

Answer: B

Answer Explanation:

QID: 09-1 NRO64		
Question # / Answer	**64**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
295007 High Reactor Pressure **2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.**	**3.94.4**	**4.7**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	1	Group	2

General References	**ABN-1**	**RAP-C1a** **RAP-E1a**	**RAP-B4g**
Explanation	The plant was at 60% power following a refuel outage when an event occurred which resulted in several EMRVs opening for 5 seconds, then closing. The lowest pressure setpoint to open EMRVs is 1065 psig. The isolation condensers auto initiate at an RPV pressure of 1051 psig. Thus the Isolation Condensers have initiated. with the isolation condensers in service and a small decay heat load, the RPV will depressurize. Answer B is correct. IAW ABN-1, when RPV water level begins to rise, it directs selecting 1 Feedwater Pump and tripping the others. It then directs placing all MFRVs in manual and taking them to close (all LFRVs would be closed at this power level). This will terminate Feedwater injection into the RPV. Thus, RPV water level will not be controlled with the MFRV in AUTO. Thus answer A is incorrect. Sine the reactor was starting up after a refuel outage, there is little decay heat to maintain RPV pressure following the scram and RPV pressure is lowering uncontrollably due to the Isolation Condensers, RPV pressure is not being maintained by the EPR. Answer C is incorrect. The reactor recirculation pumps trip on RPV high pressure of 1051 psig. At this pressure, pumps A, B & E trip immediately, and pumps C & D trip if the high pressure is sustained for 10.5 seconds. Therefore, all pumps may have tripped but not all immediately. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0023 LO 2338		

Question Source (New, Modified, Bank)				**New**
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X** **2:DR**
	NUREG 1021 Appendix B: Describe or recognize relationships			

10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

3.0 IMMEDIATE OPERATOR ACTIONS

3.1

> NOTE
>
> When a manual scram is to be performed with time permitting, Attachment ABN-1-4 should be performed.

IF an Automatic Scram occurs or is imminent,

OR

a Manual Scram is required,

THEN **PERFORM** the following:

1. **DEPRESS** both Manual Scram Pushbuttons. [ ]
2. **PLACE** the Reactor Mode Selector switch in SHUTDOWN. [ ]
3. > NOTE
   >
   > Steps 3 and 4 may be performed in any order due to operator availability and assigned duties.

   WHEN RPV level begins to rise following a scram,

   THEN **PERFORM** the following:

   a. **SELECT** one of the Feed Pumps, preferably 'A' or 'C', to be the operating pump. [ ]

   b. **TRIP** the other Feed Pumps. [ ]

   c. **PLACE** all MFRVs in MANUAL. [ ]

   d. **CLOSE** all MFRVs. [ ]
4. **VERIFY** the reactor is shutdown by performing the following:
   - All rods are fully inserted less than or equal to position 04. [ ]
   - Reactor power is lowering. [ ]

Group Heading ISOL COND	C-1-a

LOGIC TRAIN I ACTUATED	

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** high Rx pressure. (Panel 5F/6F) [ ]

  OR

- ❑ **VERIFY** Lo-Lo Rx water level (Panel 5F/6F). [ ]

AUTOMATIC ACTIONS:

❑ IF Logic Train I **and** Logic Train II actuate on Low Low Reactor water level,

THEN V-14-34, Emergency Condenser Ne01a Condensate Return Valve and V-14-35, Iso Cond "B" Condensate Return Valve Open, and all five Recirc Pumps Trip.

❑ IF Logic Train I **and** Logic Train II actuate on High Reactor Pressure,

THEN:

- V-14-34 and V-14-35 Open.
- Recirc Pumps A, B, E immediately trip.
- Recirc Pumps C and D Trip if High Reactor Pressure is sustained for greater than 10.5 seconds.

LOGIC TRAIN I ACTUATED

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Sustained high Rx pressure	1051 psig	6K9 From: RE15A or 16K110A or 6K57
OR		OR
Lo-Lo Rx water level	90" above TAF	6K10 From: RE15C or 16K110C or 6K57
OR		
V-14-34 manually opened	6K57 Energized	

Reference Drawings:

JC 19529 Sh. 1
BR 3029 Sh. 2
GE 157B6397 Sh. 15
GU 3E-611-17-005 Sh. 1

Group Heading	
ADS SV/EMRV	**B-4-g**

SV/EMRV
NOT CLOSED

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
NOTE Relief valves automatically open on high reactor pressure (1065 or 1085 psig) or auto-depressurization signal; safety valves begin opening at 1212 psig. One or more safety or relief valves not closed.	Valve not closed	Valve monitoring system master alarm Units #1 and #2 Panel 15R Reference Drawings: GU 3E-611-17-004 Sh. 1 BR 2002 BX 1106078

65 **ID: 09-1 NRO65** **Points: 1.00**

The plant was at 775 psig during a startup. Power ascension was halted to determine the cause of the over-current trip of CRD Pump B.

15 minutes later, the following annunciators then alarmed:

- 1A2 MN BRKR TRIP
- 1A2 MN BRKR OL TRIP

Which of the following actions is required?

A. Place RPS MG Set 1 on Transformer PS-1.

B. Secure Reactor Building normal HVAC and start Standby Gas Treatment System I.

C. Depress both Manual Scram pushbuttons and place the Reactor Mode Selector switch in SHUTDOWN immediately.

D. Depress both Manual Scram pushbuttons and place the Reactor Mode Selector switch in SHUTDOWN when 2 HCU accumulator alarms are received.

Answer: C

Answer Explanation:

QID: 09-1 NRO65		
Question # / Answer	65	Developer/Date: NTP 12/28/09

<table>
<tr><th colspan="6">Knowledge and Ability Reference Information</th></tr>
<tr><th colspan="4" rowspan="2">K&A</th><th colspan="2">Importance Rating</th></tr>
<tr><th>RO</th><th>SRO</th></tr>
<tr><td colspan="4">295022 Loss of CRD Pumps
2.2.2 - Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.</td><td>4.6</td><td>4.1</td></tr>
<tr><td>Level</td><td>RO</td><td>Tier</td><td>1</td><td>Group</td><td>2</td></tr>
<tr><td>General References</td><td colspan="2">RAP-H1c</td><td colspan="2">ABN-45</td><td></td></tr>
</table>

Explanation	The plant is at 775 psig during a startup, which has been suspended to diagnose the trip of CRD Pump B. With this pump OOS, CRD Pump A will be started. Then, a lockout occurs on USS 1A2, which powers CRD Pump A. Thus, both CRD Pumps are lost and will not be immediately restored. IAW RAP-H1c, if RPV pressure is <850 psig, and charging pressure cannot be immediately re-established, then scram IAW ABMN-1, reactor Scram. Answer C is correct. Placing RPS MG Set 1 loads on Transformer PS-1 is required IAW ABN-45, but the MG itself is not re-started on the transformer. Answer A is incorrect. ABN-45 directs shutdown on RB normal ventilation and manual start of SGTS II - not System I. Answer B is incorrect. IAW RAP-H1c, with RPV pressure >850 psig and no CRD Pumps available, if charging pressure cannot be immediately re-established, and 2 or more accumulator trouble alarms are received, then a manual scram IAW ABN-1 is required. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0011 LO 10465		

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	**55.41**	**6**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

PUMP A TRIP

MANUAL CORRECTIVE ACTIONS: (continued from Page 1 of 3)

❑ IF Reactor pressure is less than 850 psig	
AND charging water pressure **cannot** be immediately re-established,	
THEN **SCRAM** the Reactor in accordance with ABN-1, Reactor Scram.	[ ]
❑ IF Reactor pressure is greater than 850 psig	
AND charging water pressure **cannot** be immediately re-established,	
AND two or more accumulator trouble alarms are received (accumulator Level/press rod block light is lit),	
THEN **SCRAM** the reactor in accordance with ABN-1, Reactor Scram.	[ ]
❑ **CHECK** CRD system flow.	[ ]
❑ **CHECK** position of CRD pump minimum flow valve.	[ ]
❑ **CHECK** motor and pump bearings for loss of lubrication.	[ ]
❑ **CHECK** CRD motor and pump bearing for excessive temperatures.	[ ]
❑ **DISPATCH** an operator to check for system rupture or leaks.	[ ]
❑ **CHECK** CRD pump suction valve position.	[ ]
❑ **REFER** to the following Procedure:	
• 235, Determination and Correction of Control Rod Drive System Problems.	[ ]
❑ **CONFIRM** compliance with Tech Spec 3.4.D.	[ ]

66 **ID: 09-1 NRO66** **Points: 1.00**

The plant is at rated power.

The URO, BOP, US, SM and an inactive licensed RO from the tagging desk were in the Control Room. The URO will be leaving the Control Room to eat lunch in the cafeteria.

IAW OP-OC-101-111-1001, Strategies for Successful Transient Mitigation, which of the following is correct?

A. A turnover from the URO to the BOP is **not** required.

B. The URO shall turnover to the BOP, **but not** to the tagging RO.

C. The URO **cannot** leave the Control Room since minimum shift staffing will not be met.

D. The URO turnover shall be documented in the Control Room Log.

Answer: B

Answer Explanation:

QID: 09-1 NRO66		
Question # / Answer	66	Developer/Date: NTP 12/28/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Conduct of Operations 2.1.18 Ability to make accurate, clear and concise logs, records, status boards, and reports.					3.6	3.8
Level	RO	Tier	3	Group		
General References	OP-OC-101-111-1001	OP-OC-100-1001				

Explanation	IAW the reference, the URO shall turnover his responsibilities to the BOP if the URO will leave the 'at the controls area' for an extended time. Although another licensed RO is in the control room at the time, this RO is not an active license holder and cannot stand watch alone. Therefore, B is correct and A is incorrect. IAW OP-OC-100-1001, only 1 RO is required for minimum shift staffing. Answer C is incorrect. There is no requirement to log the URO to BOP turnover IAW OP-OC-101-111-1001. Logging shift turnover at the start of each shift is required. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

2. Unit Supervisor (US) – will maintain a command and control role of all activities in the control room. He should position himself to gain the best view of ongoing activities and to ensure clear communications. During periods when non-EOP actions are being taken the US should direct activities from the operators desk. When events occur or conditions are present that warrant ABN procedure entry the US will either direct actions from the procedures or assign the execution of the procedure to an operator. But in all cases the US is responsible for execution of the ABN's and is required to follow along in the procedure checking off steps when performed.

3. Shift Technical Assessor / Independent Assessor (STA / IA) – provides support to the SM and US by maintaining a big picture focus. The STA / IA should report to the control room when off normal conditions occur. In all cases they will report to the control room when an ABN is entered and announce (update) that they are "stationed as the STA or IA". The SM will assess the conditions and decide if the STA / IA is required to remain in the control room or may be released to perform field activities. The STA will also relieve the SM of the responsibility of the plant oversight function when the SM's attention needs to be focused on EP activities. The STA / IA shall not be assigned any critical parameters or specific duties that would distract them from maintaining a big picture overview.

   Once the STA/IA function is no longer required, the STA/IA shall have responsibility to initiate the Post Transient Review per OP-AA-108-114, if not already delegated to another individual.

4. Unit Reactor Operator (URO) – is responsible for reactivity manipulations and monitoring of primary critical parameters (Power, Pressure, and Level). The URO will remain in the "At the Controls Area (ACA)" for the majority of their time on shift, leaving the area for short periods to check procedures, AOG system, Rad monitors, Containment Isolation indications, or peer-checking is acceptable. If the URO is going to leave the ACA for extended periods of time they shall turnover the URO position to the BPO. The turnover will include a status of critical parameters and any activities that are taking place. At the end of the turnover the relieving operator will provide an update stating, "I have assumed the URO position". The URO will report Critical Parameters to the US at least every two hours or as directed by the US. At a minimum the report will include the primary critical parameters. Based on evolutions or active ACMPs additional critical parameters may be assigned. These parameters will be discussed and assigned at the pre-shift briefing.

5. Balance of Plant Operator (BPO) – maintains and controls the balance of plant systems, coordinates activities, and updates the control room log. The BOP will provide relief for the URO. During periods when the BOP leaves the front panel area they will announce such to the control room staff and turn over any critical parameter monitoring or control to the URO and provide an

67 **ID: 09-1 NRO67** **Points: 1.00**

The plant was at rated power when an event occurred which required the Operator to perform a rapid power reduction to 80% power. This action stabilized the plant.

IAW 202.1, Power Operation, which of the following shall be notified of the power reduction, in order to perform a Tech Spec required action?

A. Licensing

B. Chemistry

C. Rad Protection

D. Reactor Engineering

Answer: B

Answer Explanation:

QID: 09-1 NRO67		
Question # / Answer	**67**	**Developer/Date: NTP 12/29/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
Conduct of Operations **2.1.38 Knowledge of the station's requirements for verbal communications when implementing procedures.**					**3.7**	**3.8**
Level	**RO**	**Tier**	**3**	**Group**		
General References	**202.1**		**TS 3.6.A.4**			
Explanation	The plant was at rated power when a 20% power reduction was performed by a rapid power reduction. IAW 201 and TS, if reactor power changes by $\geq$ 15% (289.5 MWth) in an hour, then Chemistry must be notified in order to sample and analyze reactor coolant. Answer B is correct. The other answers are plausible but incorrect.					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

References to be provided during exam:	**None**	
Learning Objective		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

5.3.7 **NOTE**

This procedure section is written assuming an initial power of approximately 25%. If reactor power is above that, go to the next applicable procedural step in the power ascension.

MONITOR the following parameters during the power change to verify expected responses: [ ]

- LPRM and/or APRM levels
- Reactor pressure
- Steam line flow
- Turbine generator output
- Turbine control valve and/or bypass valve position
- Feedwater flow
- Core thermal power
- FLLLP

5.3.8 IF reactor power changes by 289.5 MWth or more in one hour,

THEN **NOTIFY** Chemistry to initiate reactor coolant sampling in accordance with Technical Specification 3.6.A.4. [ ]

5.3.9 WHEN feedwater flow is exceeds 2.0×10^6 1bm/hr,

THEN **PERFORM** the following:

5.3.9.1 **PLACE** two (2) additional condensate demineralizers in service in accordance with Procedure 319, Condensate Demineralizer Resin Regeneration & Transfer System. [ ]

5.3.9.2 **PLACE** a second Condensate Pump in service in accordance with Procedure 316, Condensate System [ ]

4. With the reactor mode switch in Run or Startup position, with:

   1. Thermal power changed by more than 15% of rated thermal power in one hour*, or

   2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

   3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

   take sample and analyze at least one sample, between 2 and 6 hours following the change in thermal power or off-gas level and at least once per four hours thereafter, until the specific activity of the primary coolant is restored to within limits.

3.6.B Liquid Radwaste Treatment - RELOCATED TO THE ODCM

3.6.C Radioactive Liquid Storage

Applicability: Applies at all times to specified outdoor tanks used to store radioactive liquids.

1. The quantity of radioactive material, excluding tritium, noble gases, and radionuclides having half-lives shorter than three days, contained in any of the following outdoor tanks shall not exceed 10.0 curies:

   a. Waste Surge Tank, HP-T-3

   b. Condensate Storage Tank

2. In the event the quantity of radioactive material in any of the tanks named exceeds 10.0 curies, begin treatment as soon as reasonably achievable, continue it until the total quantity of radioactive material in the tank is 10 curies or less, and describe the reason for exceeding the limit in the next Annual Effluent Release Report.

3. Specification 3.0.A and 3.0.B do not apply.

3.6.D Condenser Offgas Treatment - RELOCATED TO THE ODCM

3.6.E Main Condenser Offgas Radioactivity

1. The gross radioactivity in noble gases discharged from the main condenser air ejector shall not exceed 0.21/E Ci/sec after the holdup line where E is the average gamma energy (Mev per atomic transformation).

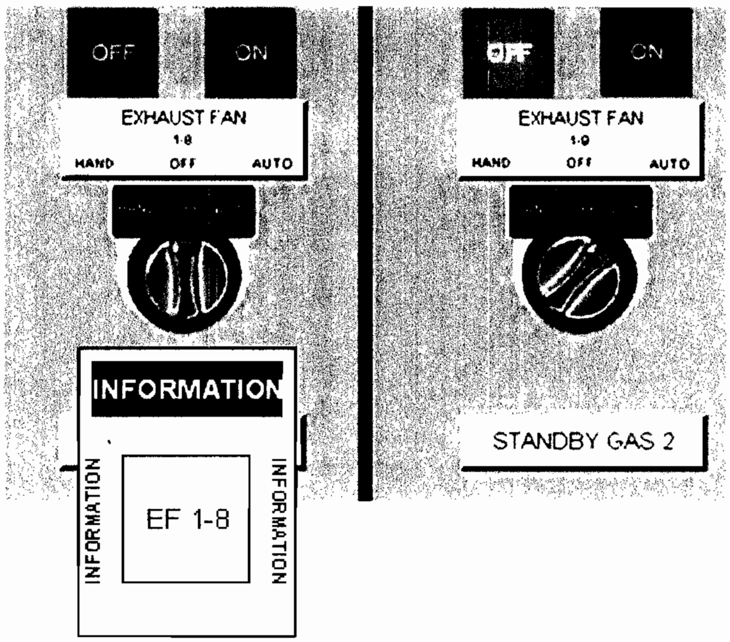
2. In the event Specification 3.6.E.1 is exceeded, reduce the discharge rate below the limit within 72 hours or be in at least SHUTDOWN CONDITION within the following 12 hours.

---

* If there are consecutive thermal power changes by more than 15% per hour, take sample and analyze at least one sample between 2 and 6 hours following the change and at least once per four hours thereafter, until the specific activity of the primary coolant is restored to within limits.

68 **ID: 09-1 NRO68** **Points: 1.00**

The plant was at rated power. You have just come in for your second day shift. On your Control Room tour during turnover, you note the change in status in the Standby Gas Treatment System (SGTS) as shown below:



Which of the following maintenance activities, if allowed, has the potential to impact the LCO for the SGT System?

Troubleshooting on the feeder breaker to ...

A. USS 1A2

B. USS 1B2

C. USS 1B3

D. VMCC 1B2

Answer: B

Answer Explanation:

QID: 09-1 NRO68

Question # / Answer	68	Developer/Date: NTP 12/29/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Equipment Control **G2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.**					3.1	4.2
Level	RO	Tier	3	Group		

General References	TS 3.5.B.7	330	BR 3002 sh. 2
Explanation	The plant is at rated power with indications showing that SGTS Fan 1 is inoperable with its breaker open and tagged out of service. TS 3.5.B.6 allows a 7-day LCO with one SGTS train inoperable. If both trains were inoperable, then TS 3.5.B.7 will require a 24-hour shutdown. SGTS fan B (EF 1-9) is powered from MCC 1B24, which is fed from USS 1B2. Thus, troubleshooting on the feeder breaker to USS 1B2 has the potential to de-energize USS 1B2, which would inop the only remaining SGTS fan, and TS 3.5.B.7 would apply for SGTS. Answer B is correct. All other answers are plausible but incorrect.		
References to be provided during exam:	None		
Learning Objective	2624.828.0.0042 LO 261-10445		

Question Source (New, Modified, Bank)				New
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

ATTACHMENT 330-2

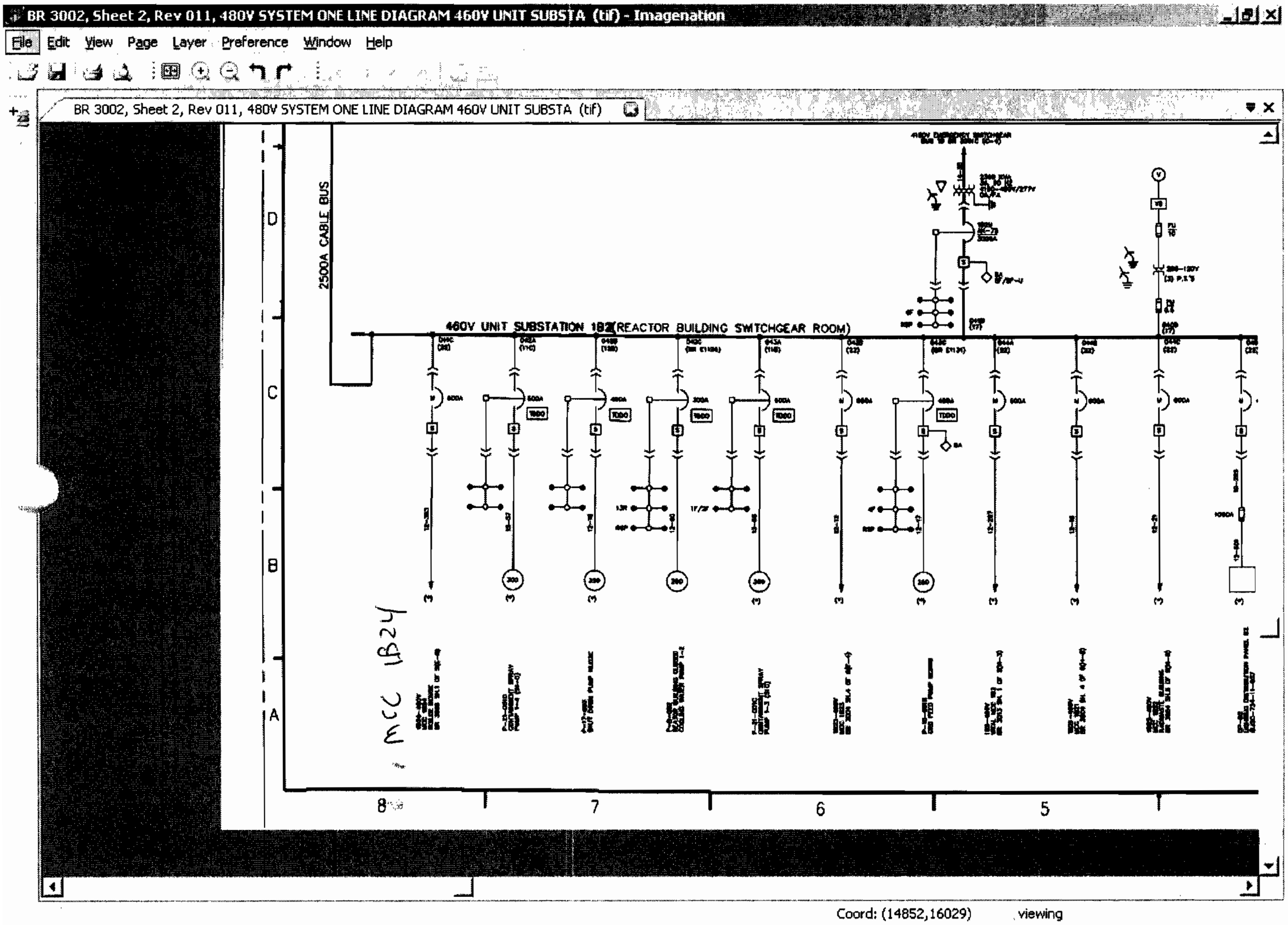
ELECTRICAL CHECKOFF LIST FOR SBGT SYSTEM

Power Supply	Item	Location	Bkr. Pos.	Perform/Verify
VAC P-1* Bkr 20	Solenoid Valve for V-28-19, V-28-21,V-28-22, V-28-48	460 Swgr Room	ON	____/____
CIP-3* Bkr 13	Solenoid Valve for V-28-17 V-28-18,V-28-47,V-27-1,V-27-2	460 Swgr Room	ON	____/____
PAIPP-1 Bkr 6	Position Indication for V-28-18 and V-27-2	Lower CSR	ON	____/____

NOTE: Only position indication for valves applicable to this procedure are listed. Other equipment is powered from this breaker.

Power Supply	Item	Location	Bkr. Pos.	Perform/Verify
460V MCC 1A24	Motor for EF-1-8	Boiler House	ON	____/____
EF-1-8 Control Power Trans	V-28-23, 24 and 26	Boiler House	ON	____/____
460V MCC 1B24	Motor for EF-1-9	Boiler House	ON	____/____
EF-1-9 Control Power Trans	V-28-27, 28 and 30	Boiler House	ON	____/____
460V MCC 1A24	Contacts for electric heating coils EHC-1-5	Boiler House	ON	____/____
460V MCC 1B24	Contacts for electric heating coils EHC-1-6	Boiler House	ON	____/____
Instr. Pnl 4C Bkr.2	Solenoids for V-23-21, V-23-22	460 Swgr Room	ON	____/____
Dist Panel P3-3	Feed to ATC-P16 (Logic Control)	Boiler House	ON	____/____

*NOTE: Only those solenoid valves applicable to this procedure are listed for the indicated power supplies. Additional solenoid valves/equipment may also be powered from the indicated power supply.


BR 3002, Sheet 2, Rev 011, 480V SYSTEM ONE LINE DIAGRAM 460V UNIT SUBSTA (tif) - Imagenation
File Edit View Page Layer Preference Window Help
BR 3002, Sheet 2, Rev 011, 480V SYSTEM ONE LINE DIAGRAM 460V UNIT SUBSTA (tif)
2500A CABLE BUS
460V UNIT SUBSTATION 1B2 (REACTOR BUILDING SWITCHGEAR ROOM)
D
C
B
A
MCC 1B24
8
7
6
5
Coord: (14852,16029)
viewing

2. Upon the accidental loss of SECONDARY CONTAINMENT INTEGRITY, restore, SECONDARY CONTAINMENT INTEGRITY within 4 hours, except as provided in specification 3.5.B.3.

3. With one or more of the automatic secondary containment isolation valves inoperable:

   a. Maintain at least one automatic secondary containment isolation valve in each affected penetration OPERABLE.

   b. Within 8 hours restore the inoperable automatic secondary containment isolation valve(s) to OPERABLE status or isolate each affected penetration with at least one valve secured in the closed position.

4. If Specifications 3.5.B.2 or 3.5.B.3 cannot be met:

   a. During Power Operation:

      (1) Have the reactor mode switch in the shutdown mode position within the following 24 hours.

      (2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

      (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

   b. During refueling:

      (1) Cease fuel handling operations or activities which could reduce the shutdown margin (excluding reactor coolant temperature changes).

      (2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

      (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

5. Two separate and independent standby gas treatment system circuits shall be operable when secondary containment is required except as specified by Specification 3.5.B.6.

6. With one standby gas treatment system circuit inoperable:

   a. During Power Operation:

      (1) Verify the operability of the other standby gas treatment system circuit within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and

      (2) Continue to verify the operability of the standby gas treatment system circuit once per 24 hours until the inoperable standby gas treatment circuit is returned to operable status.

      (3) Restore the inoperable standby gas treatment circuit to operable status within 7 days.

   b. During Refueling:

      (1) Verify the operability of the other standby gas treatment system within 2 hours. If testing is required to demonstrate operability and significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours, then demonstration by testing shall take place within 1 hour of the expiration of the 12 hour period, and

      (2) Continue to verify the operability of the redundant standby gas treatment system once per 7 days until the inoperable system is returned to operable status.

      (3) Restore the inoperable standby gas treatment system to operable status within 30 days or cease all spent fuel handling, core alterations or operation that could reduce the shutdown margin (excluding reactor coolant temperature changes.

7. If Specifications 3.5.B.5 and 3.5.B.6 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the condition of Specification 3.5.B.1 shall be met.

69 **ID: 09-1 NRO69** **Points: 1.00**

The plant was at rated power when a control power fuse blew inside a Control Room Panel. The Shift Manager has declared that the event did **not** create an emergency situation.

The original fuse was labeled as being manufactured by the Littelfuse Corporation. A similar fuse, labeled as being manufactured by the Bussmann Corporation, has been obtained from a controlled fuse location.

PIMS shows that the new fuse is a like-for-like replacement for the blown fuse.

IAW CC-AA-206, Fuse Control, which of the following is correct?

A. **Only** the Craft may install the fuse with **no** further engineering evaluation.

B. The Operator may install the fuse with **no** further engineering evaluation.

C. The Operator may **not** install the fuse since it is **not** an emergency situation.

D. The Operator may install the fuse **only after** the fuse is evaluated by the Fuse Engineer.

Answer: B

Answer Explanation:

QID: 09-1 NRO69		
Question # / Answer	69	Developer/Date: NTP 12/29/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Equipment Control 2.2.14 - Knowledge of the process for controlling equipment configuration or status.					3.9	4.3
Level	RO	Tier	3	Group		
General References	CC-AA-206					

Explanation	The plant was at rated power when a control panel fuse blew. It has been determined that the new fuse is like-for-like with the old fuse. IAW the reference, the Operator may install the fuse and no further engineering evaluation is required. Answer B is correct. Since the Operator is able to install the fuse, answer A is incorrect. IAW the reference, Operators may install fuses in both emergency and non-emergency situations. Answer C is incorrect. Since the fuse has been determined to like-for-like replacement, no further fuse evaluation is required. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective			

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps or cautions			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

2.9. **Non like-for-like fuse:** A fuse that has at least one characteristic different from the fuse being replaced. These characteristics include manufacturer, type, amp rating, voltage rating (AC or DC), physical size, or safety classification.

2.10. **Safety related fuse:** Fuses are classified safety related if they protect or isolate circuits, components, or equipment that are safety related or serve a safety function. Fuses are also classified safety related if they provide an isolation function between a safety related and non-safety related circuit.

3. RESPONSIBILITIES

3.1. Fuse Engineer

3.1.1. Maintains ownership of Fuse Control.

3.1.2. Resolve Fuse Discrepancies initiated with the PassPort/PIMS Request process.

3.1.3. Update PassPort/PIMS for Fuse Replacement using the design change process.

3.1.4. Prepare a Request using PassPort/PIMS to initiate the replacement of fuses where required.

3.1.5. Corporate Engineering performs a periodic review of corrective action databases to identify any fleet wide trends related to fuses or fuses blocks as required per MA-AA-716-210.

3.2. Maintenance Department

3.2.1. Replace fuses as designated in Work Request/Work Order and as required by this procedure.

3.2.2. Review appropriate vendor documentation when verifying as-built conditions of "Black Box" fuses.

3.2.3. Review PassPort/PIMS to verify the acceptability of as found fuses.

3.2.4. Initiate Issue Reports (for fuses that would not perform their design function) and PassPort/PIMS Requests to identify non-like-for-like fuse replacements requiring evaluation.

3.2.5. Prepare a Request using PassPort/PIMS to initiate the replacement of fuses where required.

3.2.6. Shall provide sufficient information in a PASSPORT/PIMS Request for engineering to resolve non-like-for-like fuse issues.

3.3. Operations

3.3.1. May replace fuses that are blown or in a degraded condition in any equipment.

4.2. Like-for-Like Fuse Replacements

NOTES: (1) A like-for-like fuse is identical in manufacturer, type, amp rating, voltage rating (AC or DC), physical size, and safety class to the fuse that is being replaced.

(2) Bussman rejection fuse types FRN-R, KTK-R, KTN-R, KWN-R and FRS-R may be used as substitutes for fuse types FRN, KTK, KTN, KWN and FRS respectively. Based on this information, the above listed fuses shall be considered Like-For-Like replacements where previously evaluated (Refer to MWROG Chron #179149 dated January 21, 1991 and MWROG Chron #182536 dated March 16, 1992).

(3) It is recommended and considered a good practice to replace each fuse in a circuit that contains multiple fuses (e.g. positive and negative leg of a DC circuit).

(4) If a fuse is dropped at any time, the installer should consider replacing the dropped fuse.

4.2.1. **OBTAIN** a like-for-like fuse from a controlled storage location. **(CM-1)**

4.2.2. **INITIATE an update of the** PassPort/PIMS database when required with the installed fuse Mfr/Model. This step is for like-for-like fuses that are not currently included in PASSPORT/PIMS.

4.2.3. **Prior to installing the fuse DO the following:**

- Perform a fuse continuity test
- Verify end caps are not loose
- Visually ensure cracks are not present on end caps
- Verify fuse block clip is not corroded
- Verify fuse block clip is not showing signs of heat deformation
- Verify fuse block is not missing parts (i.e. screws, nuts, retainers, etc.)
- Verify fuse block has no physical deformations (chips or cracks)
- Verify that the fuse label is legible

4.2.4. **INSTALL** fuse **and ORIENT** to position shown in Attachment 1, when practical. This allows for verifying as-built conditions without having to remove or rotate the fuse while in service.

4.2.5. **VERIFY** the fuse clips are tight and make firm contact with the fuse end caps. The fuse should not rotate once installed. If the fuse is able to rotate, then remove the fuse and reinspect the clips to determine if they need to be tightened or replaced.

4.2.6. **VERIFY** power is re-supplied to the circuit or continuity exists. This may be accomplished by observing indicating lights, hearing relay pick-up, or by using volt-ohm meter to check circuit energization/continuity.

4.2.7. **ENSURE** that there are no abnormalities. **If** an abnormality such as relay chatter is observed, **then REMOVE** fuse from the circuit **and CONTACT** the NSO or Shift Manager.

4.3. Notes: (1) A non like-for-like fuse has at least one characteristic different from the fuse being replaced. These characteristics include manufacturer, type, amp rating, voltage rating (AC or DC), physical size, or safety classification.

(2) It is recommended and considered a good practice to replace each fuse in a circuit that contains multiple fuses (e.g. positive and negative leg of a DC circuit).

(3) If a fuse is dropped at any time, the installer should consider replacing the dropped fuse.

4.3.1. **REVIEW** PassPort/PIMS approved Mfr/Model data for fuse to determine if different fuse has been evaluated and approved.

1. **If** alternate fuse replacement is identified, **then** fuse Mfr/Model may be installed as Like-for-Like fuse. **GO TO** 4.2, **otherwise CONTINUE**.

4.3.2. **INITIATE** a PassPort/PIMS Request for engineering evaluation. **(CM-1)**

Note: Operations may perform fuse replacements in emergency situations prior to having a PASSPORT/PIMS Request initiated.

4.3.3. The PassPort/PIMS Request shall include sufficient information for engineering to perform the evaluation including where applicable, EPN, system designation, drawing numbers, safety classification, description of discrepant condition, and as found fuse manufacturer, type and ampere rating. When applicable, document the replacement fuse in the Request.

4.3.4. **PERFORM** the following tasks [Fuse Engineer]:

1. **DETERMINE** a technically acceptable replacement fuse.

   A. The engineering determination of replacement fuses shall include as a minimum, where applicable, fuse amp rating, coordination, voltage rating, short circuit rating, temperature rating/characteristic, reliable circuit operation, and physical dimensions.

2. If existing design documentation does not provide acceptable solution, **then** **IDENTIFY** in PassPort/PIMS Request that a DCR is required (when applicable).

3. **APPROVE** PassPort/PIMS Request.

70 **ID: 09-1 NRO70** **Points: 1.00**

The plant was at rated power when an ATWS occurred.

IAW SP-21, Alternate Insertion of Control Rods, which of the following alternate control rod insertion methods has the potential to **raise** the airborne contamination levels in the Reactor Building?

A. Venting the Scram Air header.

B. Opening the Individual Scram Test Switches.

C. Placing the 100 amp Main RPS Breakers in OFF.

D. Placing the RPS Subchannel Test Keylock switches in TEST.

Answer: B

Answer Explanation:

QID: 09-1 NRO70		
Question # / Answer	70	Developer/Date: NTP 12/30/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Radiation Control 2.3.14 - Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.					3.4	3.8
Level	RO	Tier	3	Group		
General References	EMG-SP21					

Explanation	The plant was at power when an ATWS occurred. All answers are methods to insert control rods IAW SP-21. When a scram test switch is placed in the scram position, this de-energizes the scram solenoids for the selected control rod. This will allow reactor coolant to travel to the scram discharge volume, which is not isolated, and onto the reactor Building Equipment Drain Tank. On a normal scram, the SDV is isolated from the RBEDT. SP-21 provides a caution while using the scram test panel. Answer B is correct. All answers listed are alternate methods to insert control rods during an ATWS, but none will raise RB contamination levels.
References to be provided during exam:	**None**
Learning Objective	2621.845.0.0053 LO 3056A

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**12**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

4.5 Open Individual Scram Test Switches

4.5.1 **CONFIRM** all available CRD pumps are running. (Panel 4F) [ ]

4.5.2 **CONFIRM** open SDV vent and drain valves. (Panel 4F) [ ]

4.5.3 **OBTAIN** Key for 6XR Rod Scram Test Panel [ ]

4.5.4 **OPEN** Rod Scram Test Panel [ ]

4.5.5

> CAUTION
>
> While performing this procedure, potentially radioactive steam may be released and Reactor Building airborne contamination levels may increase.

Individually **OPEN** the scram test toggle switch for a control rod **not** inserted as follows. (Panel 6XR) [ ]

1. Attempt to **INSERT** Cram Array Control Rods first. [ ]

2. IF Cram Array Control Rods are all inserted

   OR

   cannot be moved with this method,

   THEN **INSERT** any other Control Rod as directed by the US. [ ]

4.5.6 **MONITOR** Reactor Building airborne radiation levels [ ]

4.5.7 WHEN the control rod stops moving,

THEN **CLOSE** the scram test toggle switch. [ ]

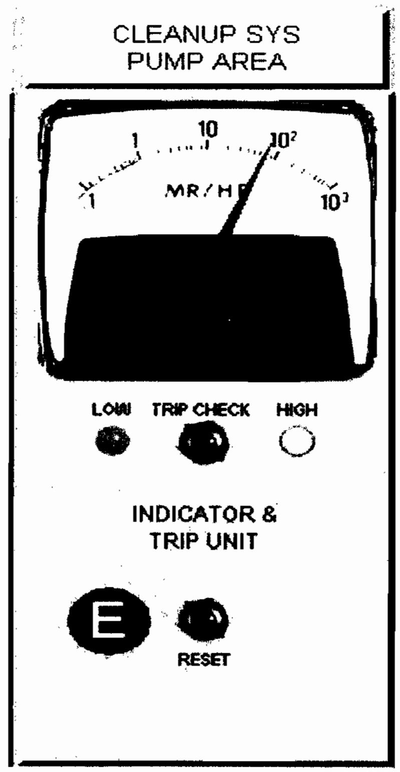
4.5.8 **REPEAT** steps 4.5.5 through 4.5.7 as required in order to insert control rods. [ ]

OVER

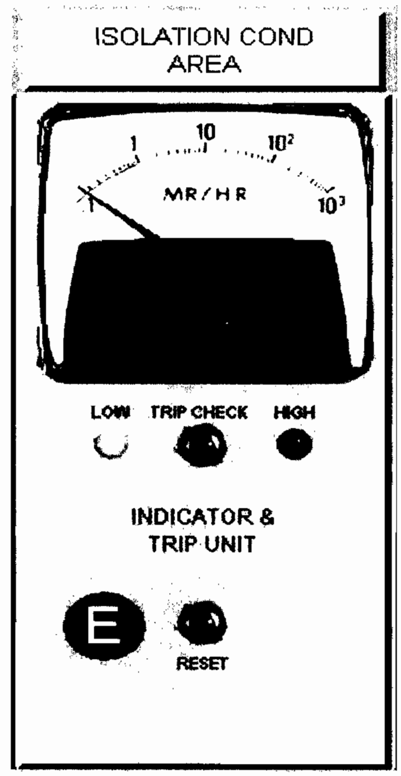
71 **ID: 09-1 NRO71** **Points: 1.00**

Which of the following indications below require entry into an EOP?

A.

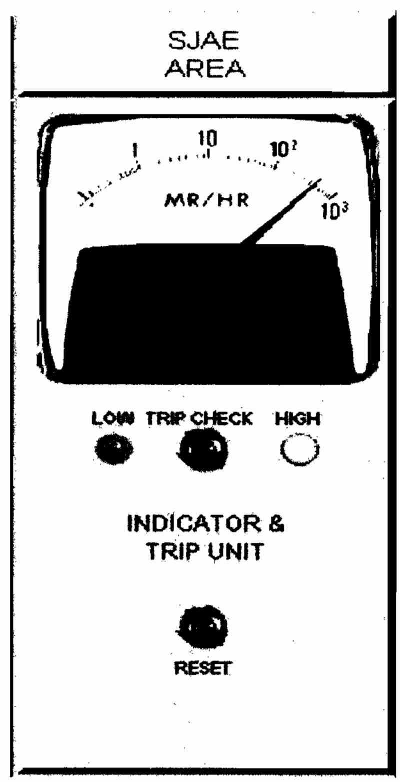


B.



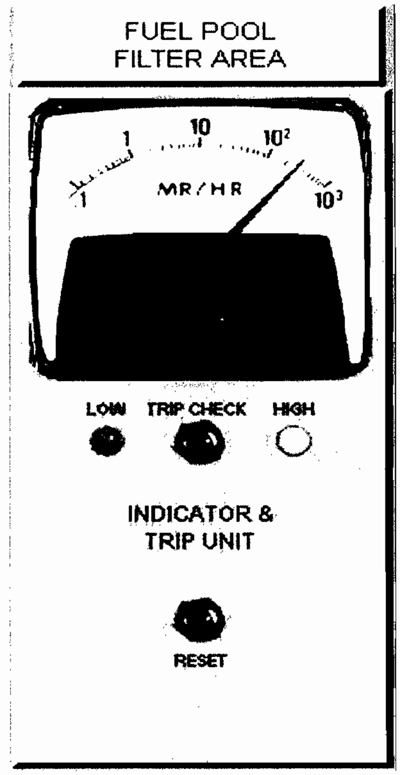
ISOLATION COND
AREA
1
10
10^2
10^3
MR/HR
LOW
TRIP CHECK
HIGH
INDICATOR &
TRIP UNIT
E
RESET

C.



SJAE
AREA
1
10
10^2
10^3
MR/HR
LOW
TRIP CHECK
HIGH
INDICATOR &
TRIP UNIT
RESET

D.



Answer: A

Answer Explanation:

QID: 09-1 NRO71		
Question # / Answer	**71**	**Developer/Date: NTP 12/30/09**

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Radiation Control 2.3.5 - Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.					2.9	2.9
Level	RO	Tier	3	Group		

General References	**Secondary Containment Control EOP**		

Explanation	IW the reference, the Cleanup System Pumps Area ARM (ARM C-1), when above the high alarm setpoint, requires entry into the Secondary Containment Control EOP. Answer A is correct. If the Isolation Condenser area ARM were indicating high, then it would require entry into the Secondary Containment Control EOP. But the ARM is indicating low - not high, then entry into the EOP is not required. Answer B is incorrect. The SJAE ARM in answer C is high, but the ARM is not located in the Secondary Containment and is thus not an EOP entry. Answer C is incorrect. The Fuel Pool Filter Area ARM sounds like it is located inside the Secondary Containment since the Fuel Pool is located there, but the filter is located outside of the Secondary Containment. It also shows that this ARM is high. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.845.0.0057 LO 1667	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**11**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

ENTRY CONDITIONS

ENTRY CONDITIONS	
PARAMETER	**CONDITION**
Rx BUILDING DIFFERENTIAL PRESSURE	AT OR ABOVE **0 IN.** OF WATER
AN AREA TEMPERATURE LISTED IN **TABLE 11**	ABOVE THE HIGH TEMPERATURE ALARM SETPOINT
A REACTOR BUILDING VENTILATION EXHAUST RADIATION LEVEL	ABOVE **9 MR/HR**
AN AREA RADIATION LEVEL LISTED IN **TABLE 12**	ABOVE THE HIGH RADIATION ALARM SETPOINT
FLOOR DRAIN SUMP 1-7 WATER LEVEL	AT OR ABOVE THE HIGH LEVEL ALARM SETPOINT
AN AREA WATER LEVEL LISTED IN **TABLE 13**	ABOVE THE **MAX NORMAL** WATER LEVEL

DISCUSSION

The six conditions that require entry into the SECONDARY CONTAINMENT CONTROL procedure are symptomatic of an abnormal condition or conditions which, if not corrected, could degrade into an emergency. Adverse effects on the operability of equipment located in the Secondary Containment and conditions directly challenging Secondary Containment integrity were specifically considered in the selection of the entry conditions.

Reactor Building Differential Pressure at or above 0 in. of water indicates loss of Secondary Containment integrity due to overpressurization and may result in an unmonitored, uncontrolled release of radioactivity to the environment.

An Area Temperature above the high temperature alarm setpoint (Maximum Normal Operating value) could be an indication of a fire in the Secondary Containment, or of steam from a primary system discharging into Secondary Containment. No matter what the source is, high area temperatures are an equipment operability concern. As temperatures continue to increase, the continued operability of equipment needed to carry out EOP actions may be compromised. High area temperatures also present a danger to personnel, a consideration of significance since access to Secondary Containment may be required by actions specified in the EOPs.

Reactor Building Ventilation Exhaust Radiation Level above 9 mr/hr is an indication that radioactivity is being released to the Secondary Containment and the environment. The Reactor Building Ventilation system should automatically isolate at 9 mr/hr.

An Area Radiation Level above the high radiation alarm setpoint (Maximum Normal Operating value) is an indication that water or steam from a primary system (or from a primary to secondary system leak) may be discharging into Secondary Containment.

Floor Drain Sump 1-7 water Level at or above high level alarm setpoint is indication that steam or water may be discharging into the Secondary Containment. Floor Drain Sump 1-7 is specifically referenced because it is the only floor drain sump in the Reactor Building which is instrumented with an alarm. Floor Drain Sump 1-6 drains into Sump 1-7.

High Area Water Level in the Reactor Building corner rooms is a concern whether the source is a primary system or not, since submergence of equipment important to safety (Core Spray, Containment Spray, etc.) may make them inoperable.

DISCUSSION (CONTINUED)

TABLE 12 SECONDARY CONTAINMENT RADIATION SETPOINTS				
AREA	INST	LOCATION	MAX NORMAL (HIGH RAD ALARM)	MAX SAFE
1	B-9 C-5 C-9 C-10	119' Rx OPER. FLOOR AREA 119' SPENT FUEL POOL 119' FUEL POOL 119' FUEL POOL	50 MR/HR 5 MR/HR 50 MR/HR 50 MR/HR	1000 MR/HR 1000 MR/HR 1000 MR/HR 1000 MR/HR
2	C-3 C-6	95' ISO COND. AREA 95' LIQ. POISON	5 MR/HR 5 MR/HR	1000 MR/HR 1000 MR/HR
3	C-1	51' CLEANUP PUMPS	15 MR/HR	1000 MR/HR
4	C-4	51' S/D HX AREA	50 MR/HR	1000 MR/HR
5	B-7 C-7	35' TIP EQUIP AREA 23' CRD MODULES AREA	10 MR/HR 10 MR/HR	1000 MR/HR 1000 MR/HR

Table 12: SECONDARY CONTAINMENT RADIATION SETPOINTS

TABLE 13 SECONDARY CONTAINMENT WATER LEVEL SETPOINTS			
WATER LEVELS		MAX NORMAL	MAX SAFE
AREA	FLOOR DRAIN SUMP		
	SUMP 1-7	HIGH LEVEL ALARM	N/A
AREA	AREA WATER LEVELS		
1	NE CORNER ROOM CONTAINMENT SPRAY PUMPS 1-1, 1-2	0 IN.	16 IN.
2	SE CORNER ROOM CONTAINMENT SPRAY PUMPS 1-3, 1-4	0 IN.	16 IN.
3	SW CORNER ROOM CORE SPRAY PUMPS B & D	0 IN.	16 IN.
4	NW CORNER ROOM CORE SPRAY PUMPS A & C	0 IN.	16 IN.

Table 13: SECONDARY CONTAINMENT WATER LEVEL SETPOINTS

72 **ID: 09-1 NRO72** **Points: 1.00**

The plant was shutdown for an outage when the fire detection in the EDG 1 enclosure caused a single fire alarm on the Main Fire Alarm Panel A. **No** other annunciators alarmed.

IAW ABN-29, Plant Fires, which of the following is the **first** Control Room response to the event?

A. Sound the Station Fire alarm for 10 seconds.

B. Evacuate personnel from the area of the alarm.

C. Dispatch an Operator to the area of the alarm.

D. Dispatch the Fire Brigade to the area of the alarm.

Answer: C

Answer Explanation:

QID: 09-1 NRO72		
Question # / Answer	72	Developer/Date: NTP 12/60/09

Knowledge and Ability Reference Information						
K&A					Importance Rating RO	Importance Rating SRO
Emergency Procedures/Plans G2.4.25 - Knowledge of fire protection procedures.					3.3	3.7
Level	RO	Tier	3	Group		
General References	ABN-29					

Explanation	The plant was shutdown when the control room received a fire alarm on the main fire alarm panel. No other alarms came in, which could indicate fire suppression system activation. Because of this, the fire is not confirmed, and the correct first action for a non-confirmed fire is to dispatch an operator to the area. Answer C is correct. The first action for a confirmed fire is to sound the fire alarm for 10 seconds, then announce the location of the fire and request the Fire Brigade to respond over the PA system. All other answers are incorrect but plausible.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0019 LO 286-10445	

Question Source (New, Modified, Bank)			**Bank**		
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**		
	NUREG 1021 Appendix B: Procedure steps and cautions				
10CRF55 Content	**55.41**	**10**	**55.43**		
	(SRO Only)				
Time to Complete: 1-2 minutes					

3.0 IMMEDIATE OPERATOR ACTIONS

None

4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 Actions for a fire alarm, but a fire is **not** confirmed

IF a fire alarm is received in the Control Room on the MFAP

OR

on any local fire alarm panel,

AND

the alarm has **not** been confirmed,

THEN **PERFORM** the following:

4.1.1 **DISPATCH** a radio-equipped operator to the alarming location to investigate. [ ]

4.1.2 **CONFIRM** the exact location and extent of the fire. [ ]

4.1.3 IF a fire has been confirmed locally,

THEN **CONTINUE** in this procedure at Section 4.2. [ ]

4.1.4 IF investigation reveals **no** fire exists,

THEN **REQUEST** fire detection circuit troubleshooting/repairs and exit this procedure. [ ]

73 **ID: 09-1 NRO73** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions include the following:

- RPV water level indicates 160" and steady
- RPV pressure indicates 1000 psig and steady
- Generator electric indicates 125 MWe and steady
- Control rod position indication has been lost
- Annunciators DW PRESS HI-HI I **and** DW PRESS HI-HI II have been acknowledged
- Core Spray indicates running
- Annunciator RB ΔP LO is alarming **and** RX BLDG DIFFERENTIAL PRESS indicates 0 inches/water

Which of the following states **all** Emergency Operating Procedures that must be entered and executed under the conditions above?

A. RPV Control - No ATWS **and** Primary Containment Control

B. RPV Control - With ATWS **and** Primary Containment Control

C. RPV Control - No ATWS, Primary Containment Control, **and** Secondary Containment Control

D. RPV Control - With ATWS, Primary Containment Control **and** Secondary Containment Control

Answer: D

Answer Explanation:

QID: 09-1 NRO73		
Question # / Answer	**73**	**Developer/Date: NTP 12/30/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
Emergency Procedures/Plan G2.4.46 - Ability to verify that the alarms are consistent with the plant conditions.					**4.2**	**4.2**
Level	**RO**	**Tier**	**3**	**Group**		

General References	EOP Users Guide	RAP-H1d	RAP-H2d
Explanation	The plant was at power when an event occurred. Indications (annunciators) show that Drywell pressure is above the scram setpoint, but the reactor is still at power as evidenced by a + MWe. Thus, the ATWS EOP is entered. With a high Drywell condition (annunciators) and core spray running, the Primary Containment Control EOP is entered. With RB Dp alarm in and Dp indicators 0", entry into the Secondary Containment Control EOP is required. Thus, the RPV Control - with ATWS EOP, Primary Containment Control EOP and Secondary Containment Control EOP will be entered. Answer D is correct. All other answers are plausible but incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.845.0.0053 LO 3052A		

Question Source (New, Modified, Bank)				Modified
Cognitive Level	Memory or Fundamental Knowledge	X 1:P	Comprehension or Analysis	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

ENTRY CONDITIONS

ENTRY CONDITIONS

PARAMETER	CONDITION
RPV WATER LEVEL	BELOW **138 IN.**
RPV PRESSURE	ABOVE **1045 PSIG**
DRYWELL PRESSURE	ABOVE **3.0 PSIG**
REACTOR POWER	ANY CONDITION WHICH REQUIRES A REACTOR SCRAM AND ANY CONTROL ROD IS NOT INSERTED TO OR BEYOND POSITION 04
ANY CONDITION WHICH IN THE JUDGEMENT OF THE OPERATOR REQUIRES ENTRY INTO THIS PROCEDURE TO EITHER CONSERVE RPV INVENTORY OR REDUCE THE RELEASE OF RADIOACTIVITY TO THE ENVIRONMENT	

A

CONFIRM THE FOLLOWING:
- Rx SCRAM INITIATED
- MODE SWITCH IN SHUTDOWN

DISCUSSION

The entry conditions to RPV CONTROL - WITH ATWS and RPV CONTROL - NO ATWS are identical. The difference between the procedures is that RPV CONTROL - WITH ATWS procedure provides RPV control instructions when it cannot be determined that the Reactor will remain shutdown under all conditions on control rod insertion alone.

When an entry condition occurs and it is questionable as to whether the Reactor will remain shutdown, the operator may choose to enter the RPV CONTROL - WITH ATWS procedure directly. However, entry to the RPV CONTROL - NO ATWS procedure under the same circumstances is also acceptable, since the initial actions are identical and the RPV CONTROL - NO ATWS procedure will direct the operator to the appropriate procedural instructions. Likewise, if the operator entered the RPV CONTROL - WITH ATWS procedure when control rod insertion was sufficient to maintain the Reactor shutdown under all conditions, this procedure will direct the operator back to the RPV CONTROL - NO ATWS procedure.

Repeating the entry conditions in the RPV CONTROL - WITH ATWS procedure is also necessary because, like all the other EOPs, reoccurrence of an entry condition requires reentry to the procedure.

Like the RPV CONTROL - NO ATWS procedure, the conditions that require entry to the RPV CONTROL - WITH ATWS procedure are symptomatic of a condition or conditions which, if not corrected, could degrade into an emergency. During failure-to-scram conditions, the entry conditions are related to the parameters controlled by the procedure. Refer to the "ENTRY CONDITIONS" section of RPV CONTROL - NO ATWS for additional discussion regarding these entry conditions.

ENTRY CONDITIONS

ENTRY CONDITIONS

PARAMETER	CONDITION
Rx BUILDING DIFFERENTIAL PRESSURE	AT OR ABOVE **0 IN.** OF WATER
AN AREA TEMPERATURE LISTED IN **TABLE 11**	ABOVE THE HIGH TEMPERATURE ALARM SETPOINT
A REACTOR BUILDING VENTILATION EXHAUST RADIATION LEVEL	ABOVE **9 MR/HR**
AN AREA RADIATION LEVEL LISTED IN **TABLE 12**	ABOVE THE HIGH RADIATION ALARM SETPOINT
FLOOR DRAIN SUMP 1-7 WATER LEVEL	AT OR ABOVE THE HIGH LEVEL ALARM SETPOINT
AN AREA WATER LEVEL LISTED IN **TABLE 13**	ABOVE THE **MAX NORMAL** WATER LEVEL

DISCUSSION

The six conditions that require entry into the SECONDARY CONTAINMENT CONTROL procedure are symptomatic of an abnormal condition or conditions which, if not corrected, could degrade into an emergency. Adverse effects on the operability of equipment located in the Secondary Containment and conditions directly challenging Secondary Containment integrity were specifically considered in the selection of the entry conditions.

Reactor Building Differential Pressure at or above 0 in. of water indicates loss of Secondary Containment integrity due to overpressurization and may result in an unmonitored, uncontrolled release of radioactivity to the environment.

An Area Temperature above the high temperature alarm setpoint (Maximum Normal Operating value) could be an indication of a fire in the Secondary Containment, or of steam from a primary system discharging into Secondary Containment. No matter what the source is, high area temperatures are an equipment operability concern. As temperatures continue to increase, the continued operability of equipment needed to carry out EOP actions may be compromised. High area temperatures also present a danger to personnel, a consideration of significance since access to Secondary Containment may be required by actions specified in the EOPs.

Reactor Building Ventilation Exhaust Radiation Level above 9 mr/hr is an indication that radioactivity is being released to the Secondary Containment and the environment. The Reactor Building Ventilation system should automatically isolate at 9 mr/hr.

An Area Radiation Level above the high radiation alarm setpoint (Maximum Normal Operating value) is an indication that water or steam from a primary system (or from a primary to secondary system leak) may be discharging into Secondary Containment.

Floor Drain Sump 1-7 water Level at or above high level alarm setpoint is indication that steam or water may be discharging into the Secondary Containment. Floor Drain Sump 1-7 is specifically referenced because it is the only floor drain sump in the Reactor Building which is instrumented with an alarm. Floor Drain Sump 1-6 drains into Sump 1-7.

High Area Water Level in the Reactor Building corner rooms is a concern whether the source is a primary system or not, since submergence of equipment important to safety (Core Spray, Containment Spray, etc.) may make them inoperable.

ENTRY CONDITIONS

ENTRY CONDITIONS

PARAMETER	CONDITION
TORUS WATER TEMP	ABOVE **95°F**
BULK DRYWELL TEMP	ABOVE **150°F**
DRYWELL PRESSURE	ABOVE **3.0 PSIG**
TORUS WATER LEVEL	BELOW **143 IN.**
TORUS WATER LEVEL	ABOVE **154 IN.**
PRIMARY CONTAINMENT HYDROGEN CONCENTRATION	ABOVE **1.5%**

DISCUSSION

The six conditions requiring entry into the Primary Containment Control procedure are symptomatic of an emergency condition or conditions which, if not corrected, could degrade into an emergency. The bases for the entry condition values are as follows:

- TORUS WATER TEMP ABOVE 95°F The high Torus temperature entry condition is based on the Technical Specification limit for Torus temperature.

- BULK DRYWELL TEMP ABOVE 150°F The entry condition setpoint is the maximum temperature expected to exist in the Drywell during normal operating conditions and is consistent with applicable system operating procedure requirements for controlling Drywell temperature.

- DRYWELL PRESSURE ABOVE 3.0 PSIG The entry condition is based on the high Drywell pressure scram setpoint (annunciators H-1-d/e, DW PRESS HI-HI I/II.) Note -occurrence of this entry condition also requires entry to RPV CONTROL - NO ATWS and RPV CONTROL - WITH ATWS.

- TORUS WATER LEVEL BELOW 143 IN. The entry condition setpoint is consistent with Technical Specification LCO 3.5.A.1.b which limits the volume of water in the Torus to a minimum of 82,000 cubic feet. This setpoint is also consistent with applicable operating procedure requirements for controlling Torus water level.

- Torus WATER LEVEL ABOVE 154 IN. The entry condition setpoint is consistent with Technical Specification LCO 3.5.A.1a, which limits the volume of water in the Torus to a maximum of 92,000 cubic feet. This setpoint is also consistent with applicable system operating procedure requirements for controlling Torus water level.

- PRIMARY CONTAINMENT HYDROGEN CONCENTRATION ABOVE 1.5%. The entry condition is based on the minimum detectable hydrogen concentration of the Drywell H_2O_2 monitors (panel 16R), which is also the alarm setpoint for high hydrogen. Detection of concentrations above this level indicates that hydrogen is building up in the containment and should be controlled. Since the post-accident hydrogen/oxygen monitoring system is not normally operating, it must be placed in service when Plant conditions approach those capable of producing hydrogen to permit the operator to monitor hydrogen buildup.

Group Heading

DW PRESS | **H - 1 - d**

[redacted]

CONFIRMATORY ACTIONS:

- ❑ IF half scram signal is present,

  THEN **CHECK** drywell pressure. (Panels 1F/2F and 12XR) [ ]

- ❑ **CHECK** drywell temperatures and sumps for an indication of leakage. [ ]

AUTOMATIC ACTIONS:

Reactor scram and primary and secondary containment isolation coincident with Channel II trip.

MANUAL CORRECTIVE ACTIONS:

- ❑ **ENTER** Procedures EMG-3200.01A, RPV Control - No ATWS, **and** EMG-3200.02, Primary Containment Control, and execute them concurrently. [ ]

- ❑ IF trip is erroneous or spurious,

  THEN **RESET** tripped channel. [ ]

MANUAL CORRECTIVE ACTIONS: (continued on Page 2 of 2)

74 **ID: 09-1 NRO74** **Points: 1.00**

The plant was at rated power when an event occurred. A summary of the event is provided below:

- Total seal leak of Recirculation Pump C
- Recirculation Pump C was shutdown
- Recirculation Pump C PUMP SUCTION Valve V-37-31 switch was placed in CLOSE but **both** the red and green lights remained energized
- The SRO directed venting the Torus through Reactor Building normal ventilation IAW SP-31, Venting the Primary Containment to Maintain Pressure Below 3.0 psig

Five minutes after the venting process began, the following annunciator alarmed:

- STACK EFFLUENT HI

Chemistry has confirmed the elevated radiation readings are from the Primary Containment.

IAW the above RAP, which of the following is required to continue venting?

A. Vent the Primary Containment through the Standby Gas Treatment System.

B. Primary Containment venting may continue in the current lineup since it's directed by the EOPs.

C. Place the keylock DRYWELL VENT-PURGE INTERLOCK BYPASS switch in BYPASS.

D. Place the keylock CNTMT VENT AND PURGE ISOLATION BYPASS switch in BYPASS.

Answer: A

Answer Explanation:

QID: 09-1 NRO74		
Question # / Answer	**74**	**Developer/Date: NTP 12/30/09**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Radiation Control 2.3.11 - Ability to control radiation releases					3.8	4.3
Level	RO	Tier	3	Group		
General References	EMP-SP31		RAP-10F2d			
Explanation	The plant was at rated power when a recirculation pump seal failed (resulting in elevated Drywell pressures and temperatures) and the pump isolation was not complete. The primary Containment Control EOP was entered and Primary Containment venting through the RB normal HVAC system was initiated. The Stack Hi alarm came in and was confirmed to be from the Primary Containment. IAW RAP-19F2d, vent the primary Containment through the Standby Gas treatment System, which is performed in SP-31. Venting through STGS will filter the effluent prior to discharge to the stack. Answer A is correct. Even though the venting is directed IAW the EOPs, SP-31 does also provide venting through the SGTS, which would not be in conflict with the RAP. Answer B is incorrect. The switches in answer C & D sound very close. If the switch in answer D is placed in bypass, this would override the Drywell high pressure isolation of Drywell valves. This would only be done if venting would prevent Primary Containment failure. The switch in answer C is used when inerting and the reactor mode switch is to be placed in RUN. Answers C & D are plausible but incorrect.					
References to be provided during exam:	EMG-SP31					
Learning Objective	2621.845.0.0056 LO 200-10450					

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge	X 1:P	Comprehension or Analysis	
	NUREG 1021 Appendix B: Procedure steps and cautions			

10CRF55 Content	55.41	12	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

Group Heading RADIATION MONITORS PROCESS STACK EFFLUENT	10F - 2 - d

STACK EFFLUENT HI	REFLASH

CONFIRMATORY ACTIONS:

- ❑ **VERIFY** the high radiation level at the Stack RAGEMS noble gas effluent monitors on Panel 1R or Stack RAGEMS effluent recorders on Panel 10F. [ ]

- ❑ IF the alarm is from a high concentration of noble gas in main stack effluent as verified from the Panel 10F Recorders,

  THEN **FOLLOW** the manual corrective actions. [ ]

- ❑ **CONTACT** Chemistry to obtain a stack effluent noble gas sample. [ ]

> NOTE
>
> Primary Containment was being vented, the source of the high stack activity may be from the Primary Containment.

- ❑ IF source of the activity is confirmed to be from Primary Containment,

  THEN **ENSURE** Primary Containment is vented through Standby Gas Treatment System. [ ]

AUTOMATIC ACTIONS:

NONE

75 **ID: 09-1 NRO75** **Points: 1.00**

The plant was at rated power when a small steam leak resulted in a reactor scram due a rising Drywell pressure.

Following the completion of the ABN-1, Reactor Scram, IMMEDIATE OPERATOR ACTIONs, the Operator reports the following indications:

- Drywell pressure had risen to 4 psig and is steady
- Isolation Condensers remain in Standby
- RPV pressure is 800 psig and steady

Which of the following states the correct indications from this event?

A. MAIN STEAM NS-03A, NS-04A, NS-03B and NS-04B indicate green lights energized.

B. EDG 1 UNIT START and UNIT IDLING lights energized and EDG 1 BREAKER red light energized.

C. Annunciators CORE SPRAY - SYSTEM 1 AUTOSTART and SYSTEM 2 AUTOSTART are in alarm.

D. Annunciators CORE SPRAY - SYSTEM 1 FLOW PERMISSIVE and SYSTEM 2 FLOW PERMISSIVE are in alarm.

Answer: C

Answer Explanation:

QID: 09-1 NRO75		
Question # / Answer	75	Developer/Date: NTP 12/31/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Emergency Procedures/Plans 2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.					4.5	4.6
Level	RO	Tier	3	Group		

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	**EMG-SP1**	**RAP-B1e** **RAP-B1f**	

Explanation	The plant was at rated power when a small steam leak occurred resulting in a high Drywell pressure scram. Indications after the ABN-1 immediate operator actions are complete show that RPV pressure is 800 psig, and RPV water level did not lower to the Lo-Lo setpoint (Isolation Condensers auto initiate on RPV water level Lo-Lo of 86"). Core Spray will auto start on a high Drywell pressure or RPV Lo-Lo water level signal, and the Core Spray Autostart annunciators will be in alarm. The auto start of Core Spray corresponds to an entry condition in Primary Containment Control EOP. Answer C is correct. The MSIVs close on an RPV lo-lo water level signal and RPV low pressure 850 psig with the Reactor Mode switch in RUN, but since the Lo-Lo water level was not reached, MSIV closure did not occur from RPV water level. Also, since the scram actions have been performed, which includes placing the Mode switch in SHUTDOWN, the RPV low pressure MSIV closure is bypassed. Thus all MSIVs are still open. Answer A is incorrect. A Drywell high pressure signal will idle start both EDGs (idling light on) but the output breakers will be open. Answer B is incorrect. The annunciators in answer D will not be in alarm since they show that Core Spray injection setpoints have been reached, which includes the RPV pressure permissive of 305 psig, and RPV pressure is currently 800 psig. Answer D is incorrect.

References to be provided during exam:	**None**	

Learning Objective	2621.828.0.0010 LO 209-10445

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X** **3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			

10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

SYSTEM 1
AUTOSTART

MANUAL CORRECTIVE ACTIONS: (continued from Page 1 of 2)

NOTE

This alarm indicates that a parameter has exceeded or has the potential to exceed an Emergency Action Level (EAL).

- ❑ **REFER** to Procedure EP-AA-1010, Exelon Nuclear Radiological Emergency Plan Annex for Oyster Creek Station to determine EAL classification. [ ]

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
Low low reactor water level	90" above TAF	RE02AY5 RE02BY5 RE02CY5 RE02DY5
OR		
High drywell pressure	Drywell press. 2.9 psig	(Panel 18R & 19R Relay Modules) P.S. RV46 A, B, C, D

Reference Drawings:

NU 5060E6003
GU 3E-611-17-004 Sh. 1

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

1 | **ID: 09-1 NSRO1** | **Points: 1.00**

The plant was at rated power preparing for a shutdown due to an overload trip of FEEDER BREAKER 1A2P.

An event then occurred. Present plant conditions include the following:

- Most control rods indicate full-in with several control rods at various positions between 24 and 44
- Reactor power is slowly oscillating between 3% and 5% on APRMs, with LPRMs oscillating 5 watts/cm^2 peak-to-peak
- RPV water level indicates 80" and lowering slowly
- Drywell pressure indicates 19 psig and rising slowly
- Drywell temperature indicates 279 °F and rising slowly
- Containment Spray System 2 DRYWELL SPRAY DISCHARGE V-21-5 is stuck closed
- Drywell OXYGEN CONCENTRATION indicates 3.4%
- Torus water temperature indicates 92 °F and rising very slowly

Which of the following states the **next** SRO direction?

A. Initiate the Standby Liquid Control System IAW SP-22, Initiating the Liquid Poison System.

B. Emergency Depressurize the RPV IAW the Emergency Depressurization - With ATWS EOP.

C. Initiate Torus Cooling IAW SP-25, Initiation of the Containment Spray System in the Torus Cooling Mode.

D. Lineup and initiate Drywell Sprays IAW SP-29, Initiation of the Containment Spray System for Drywell Sprays.

Answer: B

Answer Explanation:

QID: 09-1 NSRO1		
Question # / Answer	**1**	**Developer/Date: NTP 12/31/09**

Knowledge and Ability Reference Information	
K&A	**Importance Rating**

	RO	SRO
295028 High Drywell Temperature EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Reactor pressure		3.9

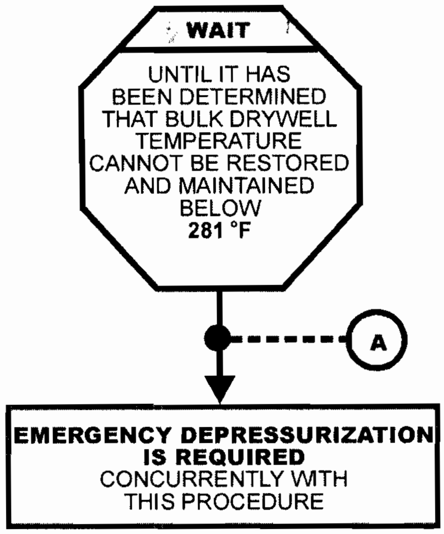
Level	SRO	Tier	1	Group	1

General References	Primary Containment Control EOP	RPV Control - With ATWS EOP	EOP Users Guide

Explanation	The plant was at rated power with USS Bus 1A2 lost. Drywell temperature is 279 F and rising. The Primary Containment Control EOP states that when it has been determined that bulk Drywell temperature cannot be maintained below 281 F, then ED is required. The question stem states that power is 3-5% on APRMs and several rods are not fully inserted. Drywell sprays are designed to reduce Drywell temperature and pressure in cases of elevated Drywell temperatures/pressures such as this. But, with USS 1A2 de-energized (which normally powers Containment Spray Pumps A & B), and the stuck valve V-21-5 (which prevents Drywell Sprays in Containment Spray System 2), there are no containment sprays available while Drywell temperature is rising. Thus, ED - With ATWS is required. Answer B is correct. Starting Standby Liquid Control is not a correct action in this ATWS condition. The RPV Control - With ATWS directs starting SLC in 2 cases: 1)periodic oscillations on LPRMs exceed 30 watts/cm^2 peak-to-peak; 2) When Torus water temperature cannot be maintained less that the BIIT Curve. The most restrictive Torus water temperature is 110 F (at $\geq$ 10% power). At the current reactor power level, the BIIT limit is > 130 F. Thus, under the given conditions, initiating SLC is not appropriate. Answer A is incorrect. The Primary Containment Control EOP says to maintain Torus water temperature below 95 F, and to place a loop of Torus cooling in service. Under the given conditions, Torus water temperature is being maintained with no further actions. And, comparing the actions to ED or Torus cooling, then Torus cooling is of lower importance and thus not the first action. Answer C is incorrect. As stated previously, it would be beneficial to be spraying the Drywell to lower temperatures and Pressures. If a Drywell Spray system were available, then initiating it to lower Drywell parameters to prevent an ED would be a viable option. But since no Drywell Sprays are available, then Answer D is incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.845.0.0053 LO 3055A	

<table>
<tr><td colspan="3">Question Source (New, Modified, Bank)</td><td colspan="2">New</td></tr>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X
3:SPK</td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td></td><td>55.43</td><td>5</td></tr>
<tr><td colspan="4">(SRO Only) Assessment of conditions and selection of appropriate procedure</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
</table>

DRYWELL TEMPERATURE CONTROL



DISCUSSION

If Drywell bulk temperature cannot be restored and maintained below 281°F, continued Drywell heatup is reduced by rapid depressurization of the RPV. This action transfers energy from the RPV to the water in the Torus and reduces the steam flow to the Drywell from any existing break in the RPV, thereby, terminating or minimizing and Drywell temperature increase. Further, RPV depressurization cools the RPV, which reduces the differential temperature between the RPV and Drywell atmosphere, reducing the heat transfer rate from the RPV to the Drywell.

Emergency RPV Depressurization is required when Drywell bulk temperature cannot be restored and maintained below the Drywell design temperature of 281°F. Otherwise, Primary Containment integrity and operability of equipment located in the Drywell required for safe shut down of the Plant can no longer be assured.

Consistent with the definition of "restore," Emergency Depressurization is not required until it has been determined that Drywell sprays initiated in the previous step are ineffective is reducing drywell temperature. It is not expected that containment integrity will be immediately challenged when the temperature limit of 281°F is reached.

If Drywell temperature is already above 281°F when the previous step to initiate Drywell sprays is reached, Drywell sprays may still be used, if available, in preference to Emergency Depressurization. If sprays are effective in reducing Drywell temperature, the depressurization need not be performed. However, extended operation above 281°F is <u>not</u> permitted.

The wording of this step allows the flexibility to attempt reducing Drywell temperature with Drywell sprays. At the same time, there is no requirement to wait until 281°F is exceeded before performing the Emergency Depressurization. While it is prudent to perform the Emergency Depressurization only when absolutely necessary, Plant conditions may warrant an earlier decision. This would be based upon LOS determination of system availability and the likelihood of success in initiation Drywell sprays.

"EMERGENCY DEPRESSURIZATION IS REQUIRED" is printed in bold, uppercase letters enclosed in a red box to emphasize the need to override RPV pressure control actions carried out concurrently in the RPV CONTROL procedure. Conditional Statements will direct depressurization according to the applicable EMERGENCY DEPRESSURIZATION procedure. The operator remains in the Primary Containment Control procedure and performs it concurrently with the Emergency Depressurization procedure.

ENTRY CONDITIONS

ENTRY CONDITIONS

PARAMETER	CONDITION
RPV WATER LEVEL	BELOW **138 IN.**
RPV PRESSURE	ABOVE **1045 PSIG**
DRYWELL PRESSURE	ABOVE **3.0 PSIG**
REACTOR POWER	ANY CONDITION WHICH REQUIRES A REACTOR SCRAM AND ANY CONTROL ROD IS NOT INSERTED TO OR BEYOND POSITION 04
ANY CONDITION WHICH IN THE JUDGEMENT OF THE OPERATOR REQUIRES ENTRY INTO THIS PROCEDURE TO EITHER CONSERVE RPV INVENTORY OR REDUCE THE RELEASE OF RADIOACTIVITY TO THE ENVIRONMENT	

A

CONFIRM THE FOLLOWING:

- Rx SCRAM INITIATED
- MODE SWITCH IN SHUTDOWN

DISCUSSION

The entry conditions to RPV CONTROL - WITH ATWS and RPV CONTROL - NO ATWS are identical. The difference between the procedures is that RPV CONTROL - WITH ATWS procedure provides RPV control instructions when it cannot be determined that the Reactor will remain shutdown under all conditions on control rod insertion alone.

When an entry condition occurs and it is questionable as to whether the Reactor will remain shutdown, the operator may choose to enter the RPV CONTROL - WITH ATWS procedure directly. However, entry to the RPV CONTROL - NO ATWS procedure under the same circumstances is also acceptable, since the initial actions are identical and the RPV CONTROL - NO ATWS procedure will direct the operator to the appropriate procedural instructions. Likewise, if the operator entered the RPV CONTROL - WITH ATWS procedure when control rod insertion was sufficient to maintain the Reactor shutdown under all conditions, this procedure will direct the operator back to the RPV CONTROL - NO ATWS procedure.

Repeating the entry conditions in the RPV CONTROL - WITH ATWS procedure is also necessary because, like all the other EOPs, reoccurrence of an entry condition requires reentry to the procedure.

Like the RPV CONTROL - NO ATWS procedure, the conditions that require entry to the RPV CONTROL - WITH ATWS procedure are symptomatic of a condition or conditions which, if not corrected, could degrade into an emergency. During failure-to-scram conditions, the entry conditions are related to the parameters controlled by the procedure. Refer to the "ENTRY CONDITIONS" section of RPV CONTROL - NO ATWS for additional discussion regarding these entry conditions.

RPV PRESSURE CONTROL

PRESSURE

IF	THEN
EMERGENCY DEPRESSURIZATION IS OR HAS BEEN REQUIRED	EXIT RPV PRESSURE CONTROL AND ENTER EMG-3200.04B, EMERGENCY RPV DEPRESSURIZATION-WITH ATWS
CORE SPRAY PUMPS ARE RUNNING	**BEFORE** RPV PRESSURE REACHES **310 PSIG,** STOP INJECTION INTO THE RPV FROM CORE SPRAY PUMPS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING PER **SUPPORT PROC - 10**

CAUTION

YARWAY LEVEL INSTRUMENTS SHALL NOT BE USED TO DETERMINE RPV WATER LEVEL DURING RAPID DEPRESSURIZATION BELOW **500 PSIG**

DISCUSSION

The Pressure Control section of this procedure provides the operator with direction for controlling RPV pressure. Whenever Plant conditions degrade to the point where an emergency depressurization is or has been required, the operator must exit the RPV Pressure Control leg of the RPV CONTROL - WITH ATWS procedure and enter the EMERGENCY DEPRESSURIZATION - WITH ATWS procedure. This is done since actions to rapidly depressurize the RPV with EMRVs take precedence over normal actions for controlling RPV pressure in the RPV Pressure Control leg.

2 **ID: 09-1 NSRO2** **Points: 1.00**

The plant was preparing to initiate a shutdown from 83% power after being on-line for 297 days, when a catastrophic loss of instrument air occurred. Present plant conditions include the following:

- INST AIR SUPPLY PRESS indicates 0 psig
- **All** control rods indicate fully inserted **except** 18-15 and 18-19 which indicate position 48, with their individual red scram lights de-energized
- **All** LPRM amber lights on Panel 4F are energized
- RPV water level is 128" and rising
- RPV pressure is 1018 psig and rising

Which of the following shall the SRO direct?

A. Using RMCS, manually insert control rods 18-15 and 18-19 IAW SP-21, Alternate insertion of Control Rods.

B. Control RPV pressure using the Isolation Condensers IAW SP-11, Alternate Pressure Control Systems - Isolation Condensers.

C. Maintain RPV water level 138" - 175" controlling with the Feedwater MFRVs IAW SP-19, Feedwater/Condensate and CRD System Operation.

D. Augment RPV pressure control by operating RWCU in the Letdown Mode IAW SP-14, Alternate Pressure Control Systems, Clean-up in Letdown Mode.

Answer: B

Answer Explanation:

QID: 09-1 NSRO2		
Question # / Answer	**2**	**Developer/Date: NTP 1/2/10**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

295019 Partial or Total Loss of Inst. Air AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure							**3.6**
Level	**SRO**	**Tier**	**1**	**Group**	**1**		
General References	**RPV Control - w/ATWS EOP**	**EMG-SP21 EMG-SP11**		**EMG-SP19 EMG-SP14**			
Explanation	The plant was at power when catastrophic loss of instrument air occurred. From this, outboard MSIVs will close and control rods will scram from MSIV position. 2 control rods failed to scram and are at position 48 and the ATWS EOP is entered. Since the MSIVs are closed, the Isolation Condensers are available for use to control RPV pressure IAW SP-11. Answer B is correct. SP-21 does allow manual control rod insertion during an electric ATWS, but with instrument air gone, both CRD FCVs auto close. With these closed, there is no driving water flow to manually insert the control rods. Answer A is incorrect. The ATWS EOP does direct an RPV water level of 138"-175" using SP-19, but with a loss of instrument air, the MFRVs lockup (but may drift open/closed) and thus the valves cannot be controlled from the control room. Answer C is incorrect. The RPV Control - w/ATWS EOP allows use of the RWCU in the letdown mode, but with a loss of instrument air, RWCU has isolated and the letdown FCV ND22 has failed closed. Answer D is incorrect.						
References to be provided during exam:	**None**						
Learning Objective	2624.845.0.0053LO 3055A						

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of appropriate procedures			

Time to Complete: 1-2 minutes

ATTACHMENT ABN-35-1

Major Systems Affected by Loss of Instrument Air

SYSTEM	EFFECT	OPERATOR ACTION	OPERATOR ACTION BEFORE AIR IS RESTORED	
Isolation Condensers	IC Vent Valves V-14-1, -5, -19 and –20 close. IC Makeup Valves V-11-34 and –36 close. Condensate Transfer AOV Valve for ICMU, V-11-257 fails open after accumulator air is depleted. Makeup capability from DWST is available. Makeup capability from Condensate Transfer System is lost if the CST is drained.	1. V-11-34 and V-11-36 each have an accumulator sized for 6 strokes of its respective valve. 2. After depleting the accumulators, **OPERATE** V-11-34 and V-11-36 manually at RB 95' el., OR **RECHARGE** the accumulators using Procedure 307. 3. V-11-257 has an accumulator sized for 5 strokes.	None	[ ] [ ] [ ]
Main Steam	MSIVs NS04A and NS04B close. NS03A, NS03B, and V-6-395 will close **if** being supplied by instrument air.	**CONTROL** RPV pressure using Isolation Condensers and/or EMRVs in accordance with ABN-1 and/or the EOPs.	**PLACE** NS03A, NS03B, NS04A, and NS04B control switches to CLOSE.	[ ] [ ] [ ] [ ]
Main Turbine	Backup Turning Gear **cannot** be remotely engaged.	Engage the Turning Gear manually.	None	[ ]
Off-Gas	SJAE Air Inlet Valves V-7-17 through –28 open. Offgas Outlet Valves V-7-1 through –6 close. Main Condenser vacuum will degrade.	Commence Load Reduction to maintain vacuum.	None	[ ]

ATTACHMENT ABN-35-1

Major Systems Affected by Loss of Instrument Air

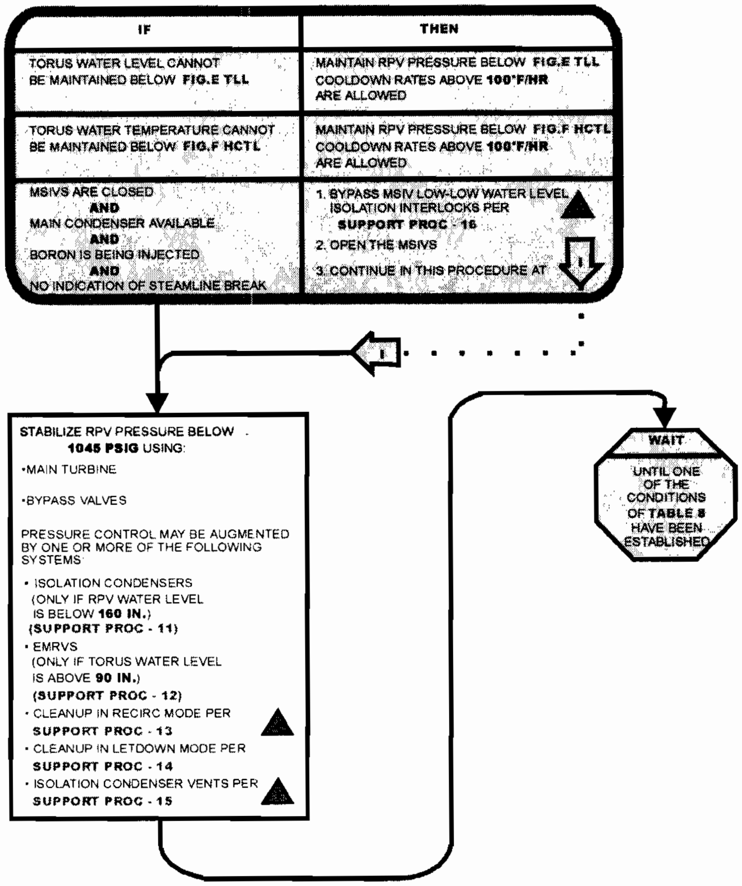
SYSTEM	EFFECT	OPERATOR ACTION	OPERATOR ACTION BEFORE AIR IS RESTORED	
Drywell and Suppression	Reactor Building-to-Torus Vacuum Breakers (V-26-16 and –18) open.	None	IF all alarms are cleared, THEN **PLACE** control switch for V-26-16 and V-26-18 to the desired position (open or closed.)	[ ] [ ]
Feedwater	Feedwater Control Valves lock up (but may slowly drift open or closed) Minimum Flow Valves Fail Open.	1. Place feedwater control valves in local-manual control in accordance with Procedure 317. 2. **PERFORM** the following actions as needed to control RPV water level: • **THROTTLE** Heater Bank Outlet Valve(s) V-2-10, -11, -12 • **TRIP** feedwater and condensate pumps as necessary	None	[ ] [ ] [ ]
Fuel Pool Cooling	Filter isolates Pumps trip	None	None	[ ]
#2 Heating Boiler	Feed reg valve closes Boiler trips	None	None	[ ]

ATTACHMENT ABN-35-1

Major Systems Affected by Loss of Instrument Air

SYSTEM	EFFECT	OPERATOR ACTION	OPERATOR ACTION BEFORE AIR IS RESTORED	
Circulating Water	Loss of traveling screen dP control	Screens should operate on timed cycle. **CYCLE** manually as required.	None	[ ]
Condensate	Spill valves V-2-15, -17, and makeup valves V-2-16 and V-2-235 open. V-2-90, CST Isolation Valve fails closed on Loss of Air.	None required. V-2-90 closes on Loss of air or Loss of power.		
Condensate Prefilter	Vessel inlet and outlet isolation valves V-425-101, 102, 201 and 202 lockup in current position. Bypass valve V-425-301 fails open	No immediate operator action required. Valve controllers fail to manual on loss of air. They require reset upon restoration of air and remain at last position following reset.	None	[ ]
CRDH	Loss of air to scram inlet and scram outlet valves. SDV isolates. Flow Control Valves NC30A and NC30B (V-15-128 and –129) close.	RPV inventory shall be carefully **MONITORED** due to failure of RWCU valves (lose ability to let down from the reactor.)	None	[ ]
Drywell purge and inerting	Supply and exhaust dampers close	None. V-27-1, 2, 3 and 4 accumulators are sized for one stroke to close and remain closed. Hardened Vent Valves V-23-13, 14, 15 and 16 accumulators are sized for 6 cycles.	None	[ ]

RPV PRESSURE CONTROL



DISCUSSION

This step directs the operator to control RPV pressure below the high pressure scram setpoint. Controlling pressure below the scram setpoint allows the scram logic to be reset (provided no other scram signals exist) and avoids EMRV actuation since the lowest EMRV lifting pressure is above the 1045 psig scram setpoint. The Turbine Bypass Valves are the preferred choice for controlling RPV pressure because heat is passed outside of Primary Containment and the Turbine Control System gives good control of Reactor pressure.

A low end pressure is purposely not listed because, depending on the transient, RPV pressure may start out much lower than the 1045 psig specified maximum. The intention of the step is to gain control of pressure as soon as possible. A stabilized, relatively constant, RPV pressure will make control of RPV water level and power less difficult. Note that even if the operator stabilizes the RPV at high pressures, this procedure and other EOP procedures will direct control of RPV pressure to the appropriate band for available water injection systems to assure adequate core cooling.

If the Main Turbine or Bypass Valves are unavailable, or are insufficient for controlling RPV pressure as desired, one or more of the alternate RPV pressure control systems may be implemented. Since symptom-oriented procedures must accommodate a full spectrum of Plant conditions and events, no prioritization regarding use of the alternate RPV pressure control systems is specified by this step. The LOS should choose the pressure control system(s) based on system capacity, degree of pressure control, heat sink, availability, and potential for release of radioactivity to the environment. A brief summary of the more significant advantages, limitations, and other pertinent factors associated with each system is given on the following pages.

3 **ID: 09-1 NSRO3** **Points: 1.00**

The plant was at rated power when the following annunciator alarmed:

- RECIRC PUMP A – CCW FLOW LO A (E7d)

The Operator reports the following indications:

- Recirculation Pump A SEAL CAVITIES PRESSURE NO. 1 indicates 1020 psig and steady
- Recirculation Pump A SEAL CAVITIES PRESSURE NO. 2 indicates 1020 psig and steady

Two minutes later, the Operator reports the following:

- Recirculation Pump A Cavity Temperatures indicate a rising trend

Which of the following actions shall the SRO direct?

A. Trip and isolate Recirculation Pump A due to the failure of the NO. 2 seal IAW ABN-2, Recirculation System Failures.

B. Immediately trip and close the discharge valve for Recirculation Pump A due to the loss of cooling IAW RAP-E7d, CCW FLOW LO A.

C. Due to the loss of cooling, trip Recirculation Pump A when cavity temperatures rise to the values specified IAW ABN-19, RBCCW Failure Response.

D. In order to minimize the seal cavity heatup, place Recirculation Pump A in MAN and reduce Recirculation Pump A speed IAW 302.1, Reactor Recirculation System.

Answer: C

Answer Explanation:

QID: 09-1 NSRO3		
Question # / Answer	**3**	**Developer/Date: NTP 1/4/10**

Knowledge and Ability Reference Information

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

K&A				Importance Rating	
				RO	SRO
295018 Partial or Total Loss of CCW AA2.03 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Cause for partial or complete loss					**3.5**
Level	**SRO**	**Tier**	**1**	**Group**	**1**
General References	**ABN-19**	**RAP-E7d**	**ABN-2**		
Explanation	The plant was at rated power when a low cooling water flow/loss of cooling water flow event to Recirculation Pump A is annunciated. When this applies a single recirculation pump, the associated RAP directs that the actions in ABN-19, RBCCW Failure Response, apply. When either seal cavity temperatures, motor bearing temperatures or motor winding temperature limits are reached, then ABN-19 requires tripping the pump and referring to ABN-2, Recirculation System Failures. Answer C is correct. The question stem also shows a problem with the recirculation pump seals. The indications show a failure of the No. 1 seal, and actions IAW ABN-2 apply. These actions include removing the pump from service and evaluating isolating the recirculation loop. Answer A provides these actions, but due to a failure of the No. 2 seal - not the No. 1 seal. Thus, answer A is incorrect. Up until recently, a single loss of CCW alarm for a recirculation pump required tripping the recirculation pump after 1 minute IAW the procedure regardless of pump temperatures. But now the temperature limits dictate when to trip the pump. Answer B is incorrect. Generally, when cooling to a component is reduced, good engineering practice is to reduce load on the component to reduce the heatup of the component. But, ABN-2, says in a note that recirculation pump speed changes should be minimized as much as possible, and does not provide any direction to reduce pump speed. Answer D is incorrect.				
References to be provided during exam:	**None**				
Learning Objective	2621.828.0.0038 LO 202-10445				

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of appropriate procedures			
Time to Complete: 1-2 minutes				

4.6 **NOTE**

The indicated actions for the following systems, upon reaching their limits, may be performed in any order or concurrently.

• Recirculation Pumps	Step 1
• Cleanup System	Step 2
• Drywell Coolers	Step 3
• Fuel Pool Cooing	Step 4
• Shutdown Cooling	Step 5

IF the temperature/conditions in any of the following systems/components reaches the specified limits below,

THEN **PERFORM** the indicated actions:

- Recirculation Pumps

1. IF any of the following temperature limits are exceeded:

   - RCP motor bearing (TR-IA55, Panel 8R) 185°F [ ]
   - RCP motor winding (TR-IA70, Panel 8R) 230°F [ ]
   - RCP Upper Seal (#2) (TR-IA71, Panel 3F) 160°F [ ]
   - RCP Lower Seal (#1) (TR-IA71, Panel 3F) 180°F [ ]

   THEN **PERFORM** the following actions:

   1. **TRIP** the affected pump(s) [ ]
   2. **EXECUTE** ABN-2, Recirculation System Failures. [ ]

4 **ID: 09-1 NSRO4** **Points: 1.00**

The plant was at 425 psig during a startup when the following annunciator alarmed:

- TORUS LEVEL HI/LO

5 minutes later, the following reports were made:

- A large Torus leak has developed inside the Torus Room
- Both TORUS LEVEL WIDE RANGE indicators show 120" and lowering
- The STA reports that Torus water level will be < 90"" by the time that Torus makeup is injecting IAW SP-37, Makeup to the Torus via Core Spray System

IAW the EOP Users Guide, which of the following states the **next** required action and the bases for the action?

	Required Action	Basis
A.	Initiate a reactor shutdown as directed in the TORUS LEVEL HI/LO alarm response	This allows a normal RPV depressurization before the pressure suppression function of the Primary Containment is lost
B.	Emergency Depressurize the RPV as directed in the Primary Containment Control EOP	This allows a rapid RPV depressurization before the pressure relief function of the EMRVs is lost
C.	Scram as directed in the Primary Containment Control EOP	This allows a rapid RPV depressurization with Turbine Bypass Valves which reduces the burden on the Torus suppression capability
D.	Scram as directed in the Primary Containment Control EOP	This allows a rapid RPV depressurization with Turbine Bypass Valves before losing NPSH to the Core Spray Pumps

Answer: C

Answer Explanation:

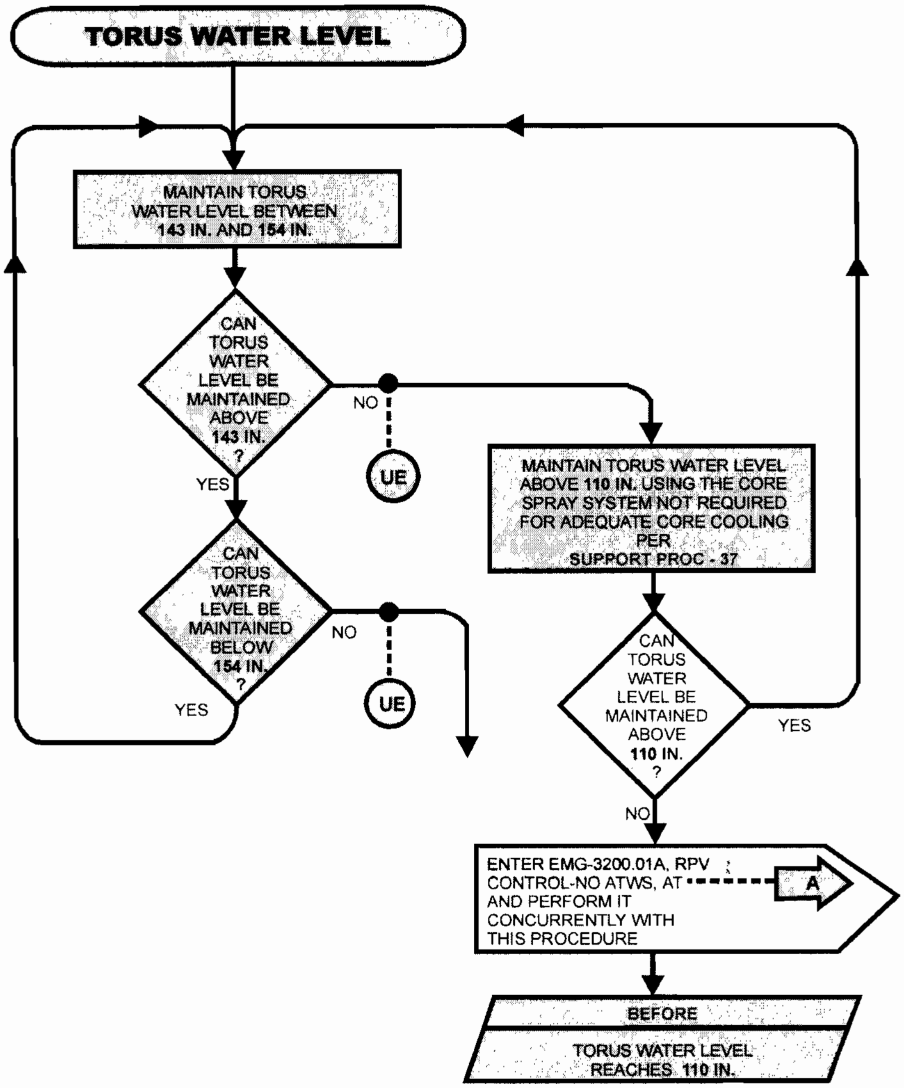
QID: 09-1 NSRO4		
Question # / Answer	4	Developer/Date: NTP 1/4/10

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295006 SCRAM 2.4.18 - Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.						4.0
Level	SRO	Tier	1	Group	1	
General References	EOP Users Guide					
Explanation	The plant was at 425 psig when a Torus water level hi/lo alarm came in. 5 minutes later, Torus water level is 120" and lowering, and the STA has determined that Torus water level will be below 90" before Torus makeup injection will start. Therefore, it can be determined that Torus water level will go below 110", and the Primary Containment Control EOP requires a manual scram. At <110', the downcomers become uncovered and steam from any LOCA will directly pressurize the Torus air space instead of being forced and quenched in the Torus water. IAW the reference, reducing RPV pressure through the Turbine Bypass valves reduces the burden on the Torus syppression capability, which is already diminished due to the low water level. Answer C is correct. The basis for answers A & B are logical, as both of these can occur if Torus water level continues to lower. The very next step in the Primary Containment Control EOP, after scram of the reactor, is to Emergency Depressurize. But, a manual scram is to take place first. Answers A & B are incorrect. Answer D also has a logical basis, but it is not the basis for the step to scram. Answer D is incorrect.					
References to be provided during exam:	None					

Learning Objective	2621.845.0.0056 LO 200-10445

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:B**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Basis or purpose			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

TORUS WATER LEVEL CONTROL



DISCUSSION

This question is asked to determine if the operator can be successful in maintaining Torus level above the bottom of the downcomers (110 in..) If the operator can be successful, the procedure will attempt to restore Torus level in the normal band. If it appears the operator will not be successful in maintaining Torus water level above 110 in., a Reactor scram is initiated via entry to RPV CONTROL - NO ATWS. Execution of RPV CONTROL - NO ATWS allows the operator to initiate a rapid RPV depressurization with the Turbine Bypass Valves and Isolation Condensers before the requirement to Emergency Depressurize with EMRVs.

Use of Turbine Bypass Valves and Isolation condensers removes energy from the primary system, which would be transferred to the Torus if an emergency depressurization is subsequently required. This reduces the burden on the Torus suppression capability, which is already currently diminished because of the decreasing Torus water inventory.

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

5 **ID: 09-1 NSRO5** **Points: 1.00**

The plant was at rated power when a series of events occurred. The time line of the events is provided below:

0800 The Control Room receives notification of a fire in the Reactor Building
0801 ABN-29, Plant Fires, is entered
0805 Annunciator ROD DRIFT alarms
0805 The Operator reports control rod 26-27 is drifting outward
0806 ABN-6, Control Rod Malfunctions, is entered
0814 The Fire Brigade Leader reports the fire has been extinguished
0815 Annunciator OFFGAS HI alarms
0816 ABN-26, High Main Steam/Off-Gas/Stack Effluent Activity, is entered
0830 Annunciator OFFGAS HI-HI alarms
0831 The Operator reports that STACK EFFLUENT High Range Monitor indicates 2 μCi/cc and rising slowly
0845 The Operator reports that Offgas has successfully isolated
0846 The Operator reports that **no** ARMs in the Turbine Building or Reactor Building have reached the MAX SAFE value
0847 The Operator reports that STACK EFFLUENT High Range Monitor indicates 3 μCi/cc and lowering slowly

Which of the following states the correct emergency plan declaration?

A. Alert (Radiological Effluent)

B. Alert (Abnormal Rad Levels)

C. Unusual Event (Fire/Explosion)

D. Unusual Event (Abnormal Rad Levels)

Answer: A

Answer Explanation:

QID: 09-1 NSRO5		
Question # / Answer	5	**Developer/Date: NTP 1/4/10**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

295038 High Offsite Release Rate **2.4.11 - Emergency Procedures / Plan: Knowledge of abnormal condition procedures.**							**4.2**
Level	**SRO**	**Tier**	**1**	**Group**	**1**		
General References	**ABN-29** **EP-AA-1010**		**ABN-2**		**ABN-26**		
Explanation	The plant is at rated power when several events occur: a fire, and a control rod drift which results in fuel failures, and a rise on offgas activity and stack activity. It is shown that the Stack activity exceeds the Alert level at time 0831. At 16 minutes later, it shows that the Stack activity, although lowering, is still above the alert level. The radiation level must be above the Alert Table for $\geq$ 15 minutes. Emergency classification for the alert (RA1, Radiological Effluent) is correct. Answer A is correct. Radiation readings > 2000 mr/hr in areas of the Turbine or Reactor Building would require an Alert emergency classification (RA3). The question states that no RB or TB ARM has reached the Max Safe value, which is 1000 mr/hr. Therefore, answer B is incorrect. A UE for a fire requires that the fire in the reactor Building not extinguished within 15 minutes. The question shows that the reactor Building fire is extinguished in 14 minutes. Answer C is incorrect. Unusual Event RU3 classification is required if offgas isolation occurs or should occur from a valid offgas radiation monitor signal, which has occurred. As a minimum, this UE would be declared if it were the only classification. Answer D is incorrect.						
References to be provided during exam:		**Hot Matrix from EP-AA-1010**					
Learning Objective	G-101 DBIG LO G-101 DBIG-01						

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X** **3:SPR**
	NUREG 1021 Appendix B: Solve a problem with a reference			

10CRF55 Content	55.41		55.43	1
	(SRO Only) Assessment of facility conditions and selection of appropriate procedures			
Time to Complete: 1-2 minutes				

Table OCGS 3-1: Emergency Action Level (EAL) Matrix **Modes:** **1 -** Power Ops **2 -** Hot Shutdown (≥ 212 °F) **3 -** Cold Shutdown (< 212 °F) **4 -** Refuel **D -** Defuel

GENERAL EMERGENCY | **SITE AREA EMERGENCY** | **ALERT** | **UNUSUAL EVENT**

Abnormal Rad Levels / Radiological Effluent

Radiological Effluent

GENERAL EMERGENCY

RG1 Offsite Dose Resulting from An Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. 1 2 3 4 D

__EAL Threshold Values:__

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do __not__ delay declaration awaiting dose assessment results.

1. VALID reading on one or more of the Table R1 radiation monitors that exceeds or is expected to exceed the reading shown (Table R1) for **≥ 15 minutes**.

OR

2. Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**:
   a. **> 1000 mRem TEDE**

   OR

   b. **> 5000 mRem CDE Thyroid**

OR

3. Field survey results at or beyond Site Boundary indicate **EITHER**:
   a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour.

   OR

   b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.

SITE AREA EMERGENCY

RS1 Offsite Dose Resulting from An Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. 1 2 3 4 D

__EAL Threshold Values:__

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do __not__ delay declaration awaiting dose assessment results.

1. VALID reading on one or more of the Table R1 radiation monitors that exceeds or is expected to exceed the reading shown (Table R1) for **≥ 15 minutes**.

OR

2. Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**:
   a. **> 100 mRem TEDE**

   OR

   b. **> 500 mRem CDE Thyroid**

OR

3 Field survey results at or beyond Site Boundary indicate **EITHER**:
   a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour.

   OR

   b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.

ALERT

RA1 Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer. 1 2 3 4 D

__EAL Threshold Values:__

1. VALID reading on any of the following effluent monitors **> 200 times alarm setpoint** established by a current radioactivity discharge permit for **≥ 15 minutes**:
   - Radwaste Overboard Discharge effluent monitor
   - Discharge Permit specified monitor

OR

2. VALID reading on one or more of the Table R1 radiation monitors that exceeds the Table R1 values for **≥ 15 minutes**.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.

UNUSUAL EVENT

RU1 Any UNPLANNED Release Of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. 1 2 3 4 D

__EAL Threshold Values:__

1. VALID reading on any of the following effluent monitors **> 2 times alarm setpoint** established by a current radioactivity discharge permit for **≥ 60 minutes**:
   - Radwaste Overboard Discharge effluent monitor
   - Discharge Permit specified monitor

OR

2. VALID reading on one or more of the Table R1 radiation monitors that exceeds the Table R1 values **≥ 60 minutes**.

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Table R1 – Effluent Monitor Thresholds

	General Emergency	Site Area Emergency	Alert	Unusual Event
Main Stack RAGEMS	4.0 E+01 μCi/cc HRM **OR** 1.6 E-08 amps HRM	4.0 E+00 μCi/cc HRM **OR** 1.6 E-09 amps HRM	1.93 E+00 μCi/cc HRM **OR** 7.8 E-10 amps HRM	7.92 E+03 cps LRM
Turbine Bldg RAGEMS	5.0 E-01 μCi/cc HRM **OR** 2.0 E-10 amps HRM	2.51 E+05 cpm LRM	8.11 E+04 cpm LRM	8.11 E+02 cpm LRM

6 **ID: 09-1 NSRO6** **Points: 1.00**

The plant was at rated power when an automatic scram setpoint was exceeded. Present plant conditions include the following:

- Generator load indicates 135 MWe
- The REACTOR MODE SELECTOR switch is in SHUTDOWN
- The MASTER RECIRCULATION SPEED CONTROLLER indicates 35 hertz
- ROPS is in BYPASS
- ALT ROD INJECTION SYS has been initiated
- RPV water level indicates 127" and rising
- SP-1, Confirmation of Automatic Initiation and Isolations, is being performed

Which of the following states the **next** SRO EOP direction and basis for this direction IAW the EOP Users Guide?

	EOP Direction	EOP Basis
A.	Bypass the RPV Lo-Lo water level MSIV closure	Prevents loss of the primary heat sink and the potential impact on the Primary Containment
B.	Bypass the RPV low pressure MSIV closure	Prevents loss of the primary heat sink and the potential impact on the Primary Containment
C.	Trip all Recirculation Pumps	Reduces reactor power and minimizes the potential for power oscillations
D.	Inhibit ADS by placing all EMRV keylock switches to DISABLE	Prevents an unnecessary RPV depressurization and injection of cold unborated water into the RPV

Answer: A

Answer Explanation:

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

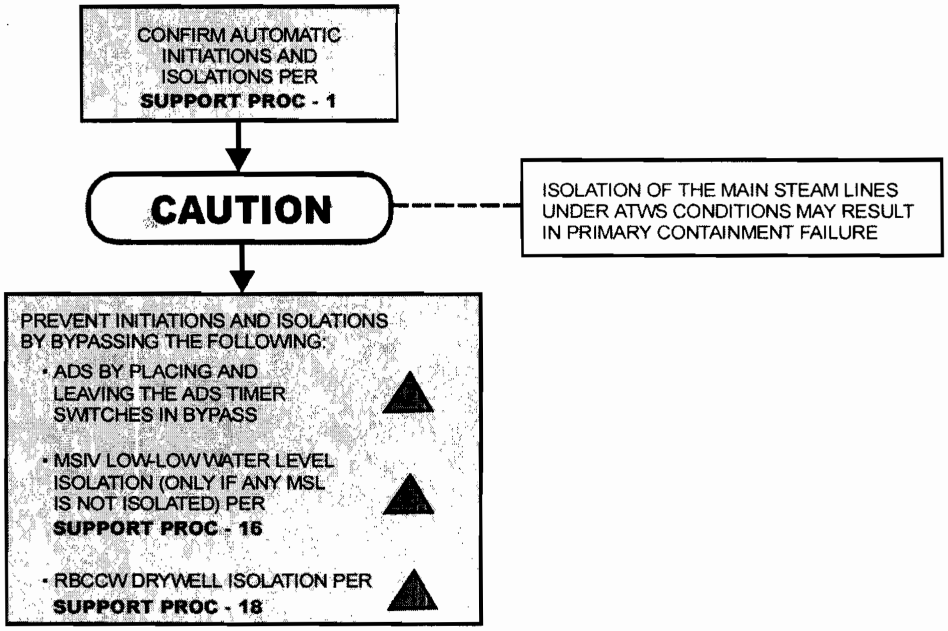
QID: 09-1 NSRO6		
Question # / Answer	6	Developer/Date: NTP 1/5/10

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown 2.4.20 - Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes.						4.3
Level	SRO	Tier	1	Group	1	
General References	EIOP Users Guide					

Explanation	The plant was at power when an auto scram setpoint occurred and a high power ATWS (>2% power) occurred. The next action in the ATWS EOP level/power leg is to bypass the RPV lo-lo water level MSIV closure. The caution associated with this step says that isolation of the MSIVs under ATWS conditions may result in Primary Containment failure. In the following step, because power is >2%, is to lower RPV water level which could result in closure of the MSIVs and will require the use of the EMRVs which discharges into the Primary Containment. Answer A is correct. Answer B is incorrect since the RPV low pressure MSIV closure is already bypassed when the Reactor Mode Switch is taken to shutdown. Keeping the MSIVs open does ensure the primary heat sink (condenser) stays available. Answer B is incorrect. In the power leg, it asks if the generator is still on-line. If yes, the action is to runback flow to minimum, and then trip the recirculation pumps if power is > 2%. This will reduce power and prevent a turbine trip during the RPV water level excursion. It can be seen that at 35 Hz, recirculation flow is not yet at minimum (about 11 Hz). Thus, running recirculation flow to minimum is performed prior to tripping the recirculation pumps. Answer C is incorrect. Defeating ADS is also a correct next action, but the method of performing this is incorrect in answer D. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.845.0.0053 LO 200-10445A		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only)			
Time to Complete: 1-2 minutes				

LEVEL/POWER CONTROL

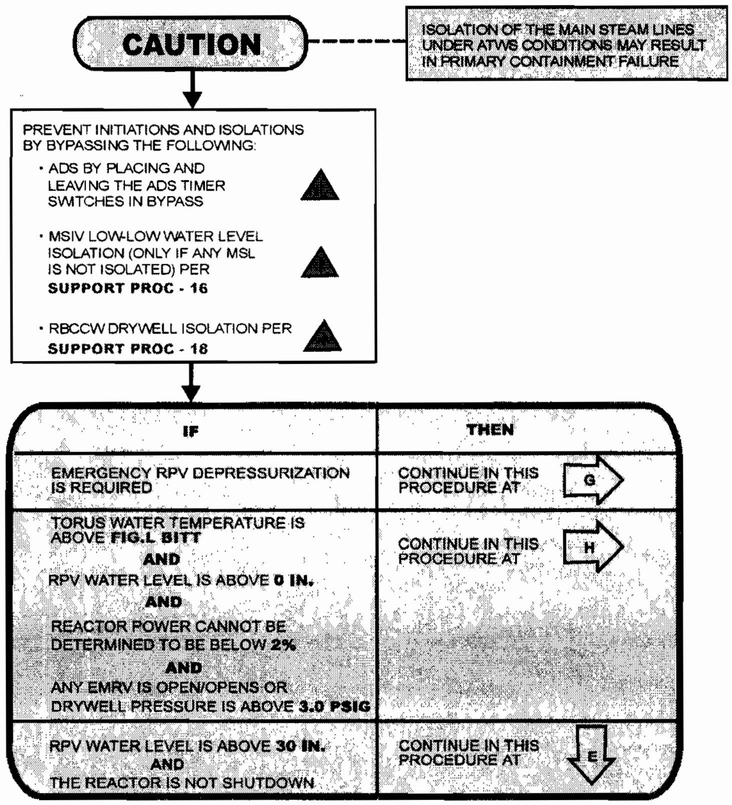


DISCUSSION

This caution addresses the concern for Primary Containment integrity during ATWS conditions. Under 100% ATWS conditions, the initial actions to trip the Reactor Recirculation pumps will result in a power reduction to approximately 45 to 55%. The relief capability of the EMRVs is approximately 45%, after which the Safety Valves will lift unless the ability to use the Main Condenser is maintained.

Primary Containment heat up is significant even when only the EMRVs are being used to provide heat removal from the RPV, therefore it is important to maintain the Main Condenser as a heat sink as long as possible during an ATWS.

LEVEL/POWER CONTROL



DISCUSSION

In order to reduce Reactor power or mitigate the effects of power oscillations, the Level/Power Control leg will require RPV water level to be deliberately lowered. The point to which level is lowered is dependent on the reason for lowering level. If the concern is power oscillations, then level will be lowered to approximately 2 feet below the Feedwater spargers. If level is being lowered to reduce overall power for containment protection, then level could be anywhere between 2 feet below the Feedwater spargers and TAF. The only situation where RPV water level will not be deliberately lowered is an ATWS condition where Reactor power can be determined to be less than 2%. Under this condition, water level will be maintained in the normal water level control band.

When the RPV water level must be deliberately lowered, it will always be lowered to at least 24 in. below the Feedwater spargers. Since the Feedwater spargers are located at 68 in. TAF, water level will always be lowered to below the Lo-Lo and Lo-Lo-Lo setpoints, initiating the automatic actions associated with those levels. Several of these automatic actions are undesirable and even detrimental to overall control of the Plant during an ATWS. Included in this group are: automatic initiation of ADS; automatic isolation of the MSIVs; and automatic isolation of RBCCW to the Drywell.

If ADS should actuate during this action, in addition to the severe thermal transient on the RPV, level control would be greatly complicated. Depressurization may also cause uncontrolled injection of cold, unborated water via low pressure systems not injecting previously because RPV pressure was greater than their discharge pressure. The result would be a dilution of in-core boron and reduction of coolant temperature. Both of these effects could give enough positive reactivity to cause a power excursion sufficient to severely damage the core. Additionally, power instabilities are more likely at lower pressures together with the chance of having large, core damaging power oscillations. Because of these concerns, ADS is prevented by bypassing the ADS timers.

DISCUSSION (CONTINUED)

During an ATWS it is most desirable to maintain the use of the Main Condenser as a heat sink, thereby reducing the amount of energy that must be absorbed by the Primary Containment. The Lo-Lo water level isolation is bypassed to preclude MSIV closure and subsequent loss of this heat sink. It should be noted that these steps do not provide authorization to defeat other MSIV isolation interlocks. Automatic isolation logic for protection against main steam line breaks remains operable.

Support Procedure - 16 provides the necessary instructions for bypassing the MSIV Lo-Lo RPV water level isolation. Additionally, Support Procedure - 16 bypasses the isolations to the Drywell pneumatic supply, since lowering RPV water level may also cause Instrument Air/Nitrogen to the Drywell to isolate. Maintaining a Drywell pneumatic supply is important for maintaining the inboard MSIVs in the open position.

Explicit instructions are contained in Support Procedure – 1 and ABN-27 where Main Steam line radiation levels are rising due to potential fuel damage. The operator is directed to close the MSIVs when MSL radiation levels reach 800 units, but <u>only if</u> no ATWS condition exists. If an ATWS is in progress, the MSIVs will remain open, preserving the Main Condenser as the heat sink. The Plant Off-Gas system will be relied upon in this case to minimize the radioactivity release.

If the MSIVs are closed when the operator reaches these steps, directions to reopen are not given by this leg. Instructions for reestablishing the Main Condenser as a heat sink are included in the RPV Pressure Control section of this procedure, which is being executed concurrently.

Together with the instructions to lower RPV level and bypass ADS are the instructions to prevent the isolation of RBCCW to the Drywell. The RBCCW System supports operation of the Drywell coolers that aids in Primary Containment heat removal. Therefore, bypassing these isolation signals to the RBCCW Drywell isolation valves is appropriate.

Support Procedure - 18 directs the following to support continued Drywell cooler operation:

1. Defeats all isolation signals to the RBCCW isolation valves.

2. Confirms open the RBCCW isolation valves.

3. Starts all available Drywell cooler fans.

4. Bypasses the "Instrument Air Isolation Valve, V-6-395," isolation signal.

All three of these bypass actions are contained in the same flowchart step. Based on the availability of personnel and Plant conditions, the LOS may direct one or more operators to perform these steps. Subsequent steps will delay lowering level until the MSIV Lo-Lo water level isolation is bypassed. This is not expected to be a significant wait since these interlocks are easily bypassed using the EOP Jumper Panels in Panels 6R and 7R. It is not intended that there be a significant delay in lowering water level due to the potential for core damage due to power oscillations.

7 **ID: 09-1 NSRO7** **Points: 1.00**

The plant was at rated power with the following abnormal lineup:

- Core Spray Main Pump NZ01D was tagged out of service as of 0800 this morning

A grid disturbance occurred which resulted in the following annunciator at 1100:

- LKOUT RELAY 86/S1B TRIP

IAW Tech Specs and procedure OP-OC-108-104-1001, Guidance for Limiting and Administrative Conditions for Operations, which of the following states the required action?

A. The reactor shall be placed in cold shutdown within the next 30 hours.

B. The reactor shall be placed in cold shutdown within 7-days from the pump inoperability.

C. The reactor shall be placed in cold shutdown within 15-days from the pump inoperability.

D. The reactor shall be placed in cold shutdown within 7 days from receipt of the above annunciator.

Answer: A

Answer Explanation:

QID: 09-1 NSRO7		
Question # / Answer	**7**	**Developer/Date: NTP 1/5/10**

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
700000 Generator Voltage and Electric Grid Disturbances **2.2.40 - Equipment Control: Ability to apply technical specifications for a system.**							**4.7**
Level	**SRO**	**Tier**	**1**	**Group**	**1**		

General References	TS 3.7	OP-OC-108-104-1001	341
Explanation	The plant was at rated power with Core Spray Main Pump NZ01D out of service (powered from 4160 Bus 1C, which can be supplied from Startup transformer S1A). A grid disturbance then results in the loss of Startup Transformer S1B. Procedure 341 states that Core Spray Pumps are engineered safeguard loads. TS 3.7.B provides the following: The reactor shall be PLACED IN the COLD SHUTDOWN CONDITION if the availability of power falls below that required by Specification A above, except that 1. The reactor may remain in operation for a period not to exceed 7 days if a startup transformer is out of service. None of the engineered safety feature equipment fed by the remaining transformer may be out of service. If just the Startup Transformer were lost, a 7-day LCO would be correct. But since the Core Spray Pump NZ01D is powered from the remaining transformer, the plant shall be placed in cold shutdown. IAW OP-OC-108-104-1001, it states that the requirement to be in cold shutdown (with no time provided) shall be within 30 hours. Answer A is correct. The TS for Core Spray does allow a 7-day and a 15-day LCO. But since the times listed are incorrect for the given conditions, then answers B & C are incorrect. Answer D is incorrect since the time is incorrect.		
References to be provided during exam:	TS 3.7 (No basis)		
Learning Objective	2621.828.0.0010 LO 209-10451		

Question Source (New, Modified, Bank)		New		
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	55.41		55.43	2
	(SRO Only) Facility operating limitations in the technical specifications and their bases			
Time to Complete: 1-2 minutes				

Exelon Nuclear	OYSTER CREEK GENERATING STATION PROCEDURE	Number **341**
Title **Emergency Diesel Generator Operation**		Revision No. 91

ATTACHMENT 341-5

EDG 1

ENGINEERED SAFEGUARD LOADS AND OTHER CRITICAL LOADS

-- INDICATES ENGINEERED SAFEGUARD LOAD
∅ INDICATES PRIORITY PUMP FOR SAFETY SYSTEM
● INDICATES ALTERNATE PUMP FOR SAFETY SYSTEM
Δ INDICATES VALVE COULD RENDER SYSTEM INOPERABLE

SYSTEM	BUS	EQUIPMENT
-- CORE SPRAY SYSTEM PUMPS	1C 1C 1A2 1A2	∅ CORE SPRAY MAIN PUMP NZ01A (493KW) SYS 1 ● CORE SPRAY MAIN PUMP NZ01D (481KW) SYS 2 ∅ CORE SPRAY BOOSTER PUMP NZ03A (247KW) SYS 1 ● CORE SPRAY BOOSTER PUMP NZ03D (255KW) SYS 2-AUTO STARTS ONLY IF BOTH NZ03A AND NZ03B **NOT** RUNNING
-- CONT. SPRAY SYSTEM PUMPS	1A2 1A2 1C 1C	NOTE: START PREVENTED FOR 200 SECONDS AFTER EDG BREAKER CONTAINMENT SPRAY PUMP 51A (254KW) SYS 1 CONTAINMENT SPRAY PUMP 51B (254KW) SYS 1 ESW PUMP 52A (328KW) SYS 1 ESW PUMP 52B (328KW) SYS 1
LIQUID POISON SYSTEM PUMPS	1A21	LIQUID POISON PUMP NPO2-A AND SQUIB VALVE NPO5-A (25KW)
-- STANDBY GAS TREAT-MENT FANS	1A24	NOTE: PRIORITY SGTS DEPENDS ON SYSTEM SELECTED ON PANEL 11R - ALL ASSOCIATED VALVES ARE AIR OPERATED. EF-1-8 (SGTS I) (9KW)
CRD SYSTEM PUMPS	1A2	CRD SYS. PUMP NC08A (212KW)
SERVICE WATER SYS. PUMP	1A3	SERVICE WATER PUMP 1-1 (187KW)
RBCCW SYS. PUMP	1A2	RBCCW PUMP 1-1 (163KW)
-- CONTROL ROOM HVAC SYSTEM FAN	1A2 (DP-A2)	SUPPLY FAN FN-826-008A (9KW)
POST ACCIDENT INSTRUMENT POWER PANEL (PAIPP)	1A2	PANEL PAIPP-1, PDP-733-057 (1.9KW)

3.7 AUXILIARY ELECTRICAL POWER

Applicability: Applies to the OPERATING status of the auxiliary electrical power supply.

Objective: To assure the OPERABILITY of the auxiliary electrical power supply.

Specification:

NOTE: LCO 3.0.C.2 is not applicable to Auxiliary Electrical Power.

A. The reactor shall not be made critical unless all of the following requirements are satisfied:

1. The following buses or panels energized.

    a. 4160 volt buses 1C and 1D in the Turbine Building Switchgear Room.

    b. 460 volt buses:

    USS 1A2, USS 1B2, MCC 1A21, MCC 1B21, Vital MCC 1A2, and Vital MCC 1B2 in the Reactor Building 480 V Switchgear Room.

    USS 1A3 and USS 1B3 in the Intake Structure.

    MCC 1A21A, MCC 1A21B, MCC 1B21A, MCC 1B21B, and Vital MCC 1AB2 on Reactor Building Elevation 23' 6".

    MCC 1A24 and 1B24 in the Boiler House.

    c. 208/120 volt panels CIP-3, IP-4, IP-4A, IP-4B, IP-4C and VACP-1 in the Reactor Building Switchgear Room.

    d. 120 volt protection panels PSP-1 and PSP-2 in the Lower Cable Spreading Room.

    e. 125 VDC Distribution Centers DC-B and DC-C.
    125 VDC Power Panels DC-D and DC-F.
    125 VDC MCCs DC-1 and DC-2

    f. 24 volt DC power panels DC-A and DC-B in the Lower Cable Spreading Room.

2. One 230 KV line (N-line or O-line) is fully operational and switch gear and both startup transformers are energized to carry power to the station 4160 volt AC buses and carry power to or away from the plant.

3. An additional source of power consisting of one of the following is in service connected to feed the appropriate plant 4160 V bus or buses:

    a. A 230 KV S-line fully operational.
    b. A 34.5 KV line fully operational.

4. Station batteries B and C and an associated battery charger are OPERABLE. Switchgear control power for 4160 volt bus 1D and 460 volt buses 1B2 and 1B3 is provided by 125 VDC Distribution Center DC-B. Switchgear control power for 4160 volt bus 1C and 460 volt buses 1A2 and 1A3 is provided by 125 VDC Distribution Center DC-C.

5. Bus tie breakers ED and EC are in the open position.

B. The reactor shall be PLACED IN the COLD SHUTDOWN CONDITION if the availability of power falls below that required by Specification A above, except that

1. The reactor may remain in operation for a period not to exceed 7 days if a startup transformer is out of service. None of the engineered safety feature equipment fed by the remaining transformer may be out of service.

2. The reactor may remain in operation for a period not to exceed 7 days if 125 VDC Motor Control Center DC-2 is out of service, provided the requirements of Specification 3.8 are met.

3. The reactor may remain in operation provided the requirements of Specification 3.7.D are met.

C. Standby Diesel Generators

1. The reactor shall not be made critical unless both diesel generators are operable and capable of feeding their designated 4160 volt buses.

2. If one diesel generator becomes inoperable during power operation, repairs shall be initiated immediately and the other diesel shall be operated at least one hour every 24 hours at greater than 80% rated load until repairs are completed. The reactor may remain in operation for a period not to exceed 7 days if a diesel generator is out of service. During the repair period none of the engineered safety features normally fed by the operational diesel generator may be out of service or the reactor shall be placed in the cold shutdown condition. If a diesel is made inoperable for biennial inspection, the testing and engineered safety feature requirements described above must be met.

3. If both diesel generators become inoperable during power operation, the reactor shall be placed in the cold shutdown condition.

4. For the diesel generators to be considered operable:

A) There shall be a minimum of 14,000 gallons of diesel fuel in the standby diesel generator fuel tank,

OR

B) To facilitate inspection, repair, or replacement of equipment which would require full or partial draining of the standby diesel generator fuel tank, the following conditions must be met:

1) There shall be a minimum of 14,000 gallons of fuel oil contained in temporary tanker trucks, connected and aligned to the diesel generator fill station.

"the Reactor shall be in COLD SHUTDOWN within 24 hours" "be in the SHUTDOWN CONDITION within 24 hours"	statement completion time and subtracting the appropriate allotted plant maneuvering time. A minimum of four (4) hours (or longer, to meet environmental requirements) shall be allotted to maneuver from POWER OPERATION to SHUTDOWN CONDITION (or longer, to meet environmental requirements). A minimum of eight (8) hours shall be allotted to maneuver from the SHUTDOWN CONDITION to COLD SHUTDOWN.
TIME INTERVAL SPECIFIED	**CRITERIA**
"the Reactor shall be placed in the COLD SHUTDOWN CONDITION" (no time interval specified)	The Reactor shall be placed in a cold shutdown condition in 30 hours. Initiation of a plant shutdown does NOT have to be commenced within one hour. The minimum initiation time when a plant shutdown should be commenced is derived by using the ACTION statement completion time and subtracting the appropriate allotted plant maneuvering time. A minimum of four (4) hours (or longer, to meet environmental requirements) shall be allotted to maneuver from POWER OPERATION to SHUTDOWN CONDITION (or longer, to meet environmental requirements). A minimum of eight (8) hours shall be allotted to maneuver from the SHUTDOWN CONDITION to COLD SHUTDOWN.

4.5 The following criteria shall be used for determining when a component or system shall be considered "inoperable" or "operable" for purposes of satisfying the requirements of the Technical Specifications:

4.5.1 A component or system shall be declared inoperable when events indicate inoperability or inoperability information is received.

Example: An ESW pump fails to attain required flow at 0000 hours on X/1/XX. Later, an Engineering evaluation is completed at 0800 hours on X/1/XX. The ESW pump is considered inoperable since 0000 hours and the appropriate LCO time clock also begins at 0000 hours on X/1/XX.

4.5.2 A component or system shall be considered operable from the time that its surveillance is satisfactorily completed and not upon completion of the surveillance's reviews or closeout of the job order, provided that no discrepancies that impact operability are identified during these reviews. If any discrepancies occur that impact operability, then the inoperability period shall continue from the original failure date and time.

8 **ID: 09-1 NSRO8** **Points: 1.00**

The plant was at rated power when an automatic scram setpoint was exceeded. Current plant conditions include the following:

- All control rod position indication is lost
- All individual scram lights (Panel 4F) are energized
- RPV water level is 141" and steady
- RPV pressure is 990 psig and steady
- Torus water temperature is 94 °F and steady
- Drywell pressure is 9 psig and rising slowly
- Drywell temperature is 201 °F and rising slowly
- TOTAL STEAM FLOW is 2.01 MLB/HR

IAW the EOP Users Guide, which of the following states the strategy of **highest** priority?

A. In the Pressure leg of the Primary Containment Control EOP, initiating Drywell Sprays.

B. In the Power leg of the RPV Control - With ATWS EOP, inserting control rods by venting the scram air header.

C. In the Level/Power leg of the RPV Control - With ATWS EOP, maximize the steam flow/feed flow mismatch to lower RPV water level.

D. In the Torus Water Temperature leg of the Primary Containment Control EOP, initiating Containment Spray in the Torus Cooling Mode.

Answer: C

Answer Explanation:

QID: 09-1 NSRO8		
Question # / Answer	8	Developer/Date: NTP 1/5/10

Knowledge and Ability Reference Information		
K&A	Importance Rating	
	RO	SRO

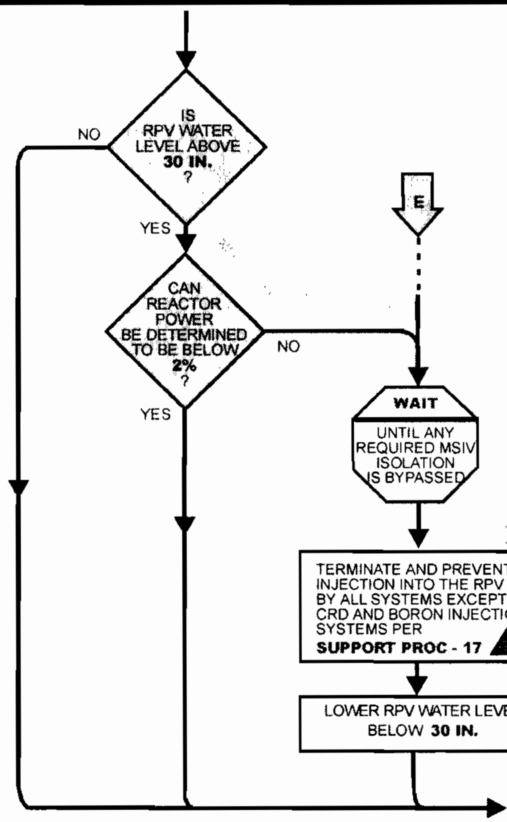
295009 Low Reactor Water Level AA2.02 - Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL : Steam flow/feed flow mismatch					3.7
Level	SRO	Tier	1	Group	2
General References	EOP Users Guide				
Explanation	The plant was at power when an ATWS occurred. Control rod position has been lost and no direct power indication is provided. But with steam flow at 2 Mlb/hr, and steady RPV pressure, power is about 28% (rated steam flow is about 7.1 Mlb/hr). Because reactor power is so high, the highest priority task is to terminate/prevent injection IAW SP-17 by stopping all Feedwater Pumps. The steam/feed flow mismatch will result in a lowering water level and the low end of the water band will be -20". Answer C is correct. Maintaining Primary Containment intact is certainly important, but Drywell Sprays cannot be initiated until Drywell/Torus pressure exceeds 12 psig (and its only 9 psig currently). Therefore, Drywell Sprays cannot be initiated from the pressure leg of the Primary Containment Control EOP. Answer A is incorrect. Inserting control rods is also of high priority, but the method to insert control rods is for an electric ATWS and indications given suggest a hydraulic ATWS. Answer B is incorrect. With Torus water temperature at 93 °F and steady, the entry condition from this parameter into Primary Containment Control EOP has not been reached. Since temperature is < 95 °F and is steady, there is no hurry in initiating Torus Cooling and is of lower priority than lowering RPV water level through termination/prevention of RPV injection. Answer D is incorrect.				
References to be provided during exam:	None				
Learning Objective	2621.845.0.0053 LO 200-10445A				

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK

	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning

10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of procedure			
Time to Complete: 1-2 minutes				

LEVEL/POWER CONTROL



DISCUSSION

Once it has been determined that injection to the RPV must be terminated to protect the core from damage due to large power oscillations, this series of steps provides guidance on how injection is to be terminated/prevented and how far RPV water level is to be lowered.

Although the Pressure Control leg of this procedure allows reopening of the MSIVs if they are closed, actions to lower RPV water level will not be performed until the actions to bypass the MSIV Lo-Lo level isolation have been completed. This ensures that closure of the MSIVs due to low RPV water level will not occur, significantly complicating an already detailed situation. There should be no significant delay time at this point because the bypassing of the MSIV Lo-Lo RPV water level isolation signal takes less than 2 minutes to perform.

Injection from boron injection systems (if boron injection is required) and CRD is not terminated because boron injection systems add negative reactivity and CRD is required to manually insert control rods.

Support Procedure – 17 directs the steps needed to terminate and prevent needed injection.

Termination and prevention of Condensate and Feedwater injection

Support Procedure - 17 directs shut down of all operating Feedwater pumps, and all but one Condensate pump. One Condensate pump is left in service to maintain cooling to the SJAE condensers to maintain vacuum. Additionally, the procedure directs the operator to close all Feed Regulating Valves, so that with subsequent restart of the system, manual operator action is required to reestablish flow.

Termination and prevention of Core Spray injection

The Core Spray System, even if not operating, will have to be terminated/prevented. The automatic start feature on this system complicates the issue of preventing flow; however, Support Procedure - 17 provides the necessary instructions for securing and preventing auto restart of the system.

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

9 **ID: 09-1 NSRO9** **Points: 1.00**

The plant was at rated power when the following annunciators alarmed:

- ROPS ACTUATE A
- RX LVL HI II

The Operator reports that indication for RE05A in Panel 19R indicates upscale at 185".

1) IAW ABN-59, RPV Level Instrument Failures, which of the following actions is required; and,
2) What action is required by Tech Specs 3.1.1?

	1) ABN-59 Action	2) TS Action
A.	Place the redundant GEMAC level instrument in control	Within 12 hours, restore the instrument or place the instrument in the Trip Condition
B.	Place RPS 1 Subchannel test Switches to TEST	Within 6 hours, restore the instrument or place the instrument in the Trip Condition
C.	Insert a manual 1/2 scram on RPS 1	Within 24 hours, restore the instrument or place the instrument in the Trip Condition
D.	Confirm all automatic actions have occurred from the failed instrument	Within 12 hours, restore the instrument or place the instrument in the Trip Condition

Answer: D

Answer Explanation:

QID: 09-1 NSRO9		
Question # / Answer	9	**Developer/Date: NTP 1/5/10**

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295008 High Reactor Water Level 2.1.20 - Conduct of Operations: Ability to interpret and execute procedure steps.						4.6
Level	SRO	Tier	1	Group	2	
General References	ABN-59		TS 3.1.1			

Explanation	The plant was at rated power when RPS RPV water level instrument RE05A failed upscale. This instrument inputs into the turbine trip circuit and Feedwater Pump trip on RPV high water level, and inputs into the reactor scram on RPV low water level. When just one level instrument fails upscale, there are no automatic actions which occur. ABN-59 says that if an instrument malfunction has occurred or is suspected (as compared to a failed indicator), then confirm all automatic actions have occurred due to the failed instrument. TS 3.3.1, note nn provides the following (from the scram on low water level portion of the Table): With one required channel inoperable in one Trip System, within 12 hours, restore the inoperable channel or place the inoperable channel and/or that Trip System in the tripped▲ condition. With two or more required channels inoperable: 1. Within one hour, verify sufficient channels remain OPERABLE or tripped▲ to maintain trip capability, and 2. Within 6 hours, place the inoperable channel(s) in one Trip System and/or that Trip System▲▲ in the tripped condition▲ , and 3. Within 12 hours, restore the inoperable channels in the other Trip System to an OPERABLE status or tripped. Otherwise, take the Action Required. Therefore, the instrument must be placed in the trip condition within 12 hours. Answer D is correct. The ABN action in answer A is correct if the failed instrument input into Feedwater Level Control, but the RE05A does not. The TS action time is correct. Answer A is incorrect. If the candidate confuses a scram setpoint with RPV high water level, then the ABN actions in answers B & C might be correct. But even though it is a scram water level instrument, there is no scram setpoint on RPV water level high. Answers B & C are incorrect.

References to be provided during exam:	**TS 3.1.1**	

Learning Objective	2621.828.0.0030 LO 1032

Question Source (New, Modified, Bank)	**New**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPR
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	55.41		55.43	2
	(SRO Only) Facility operating limitations in the technical specifications and their bases			
Time to Complete: 1-2 minutes				

3.0 IMMEDIATE OPERATOR ACTIONS

None

4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 IF an RPV level indicator exhibits erratic response, unusual inconsistency with redundant indicators, or has failed,

THEN **PERFORM** the following:

4.1.1 **DETERMINE** (by available indications) if an indicator or complete instrument failure has occurred. [ ]

4.2 IF the controlling GMAC has failed

THEN **PLACE** the other GMAC level instrument in control as follows:

4.2.1

> NOTE
> In a transient, the feed system continues to use the last Auto Signal seen until a manual adjustment is made.

PLACE the MASTER FEEDWATER LEVEL CONTROLLER on 4F in MANUAL by pressing the auto/man soft key. [ ]

4.2.2 **CONFIRM** green AUTO LED is off and red MAN LED is on [ ]

4.2.3 **CONFIRM** Feed Flow is approximately the same as Steam Flow. [ ]

4.2.4 **PLACE** the LEVEL TRANSMITTER SELECTOR on 4F in one of the following positions:

- Position A [ ]
- Position B [ ]

4.2.5 **SELECT** the S display on the MASTER FEEDWATER LEVEL CONTROLLER [ ]

4.2.6 **MATCH** the S display digital readout to the P display digital readout on the MASTER FEEDWATER LEVEL CONTROLLER [ ]

4.2.7 WHEN Deviation = 0 (S display digital readout and P display digital readout are equal, Y= 0),

THEN **PLACE** the MASTER FEEDWATER LEVEL CONTROLLER in AUTO by pressing the auto/man soft key. [ ]

4.2.8 **CONFIRM** green AUTO LED is on and red MAN LED is off [ ]

4.2.9

CAUTION
Monitor and maintain thermal power below applicable limits while restoring reactor level

ADJUST level setpoint as required to maintain reactor level between 155 and 165 inches TAF. [ ]

4.3 IF only an RPV level indicator has failed,

THEN **PERFORM** the following:

4.3.1

NOTE
ABN-59-3, RPV Level Instruments Sharing Common Reference Legs, lists available RPV level instruments and common reference leg information.

USE redundant indications for the failed indicator. [ ]

4.3.2 **INITIATE** an Issue Report (IR) for repairs. [ ]

4.4 IF an instrument failure has occurred or is suspected,

THEN **PERFORM** the following:

4.4.1 **CONFIRM** all automatic actions (half-scrams, ESF system start, etc.) associated with the failed instrument have occurred. [ ]

TABLE 3.1.1 - PROTECTIVE INSTRUMENTATION REQUIREMENTS

Sheet 1 of 13

Function	Trip Setting	Reactor Modes in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Minimum Number of OPERABLE or OPERATING [tripped] Trip Systems	Minimum Number of Instrument Channels Per OPERABLE Trip System	Action Required*
A. Scram								
1. Manual Scram		X	X	X	X	2	1	Insert control rods
2. High Reactor Pressure	**		X(s)	X(ll)	X	2	2(nn)	
3. High Drywell Pressure	≤ 3.5 psig		X(u)	X(u)	X	2	2(nn)	
4. Low Reactor Water Level	**		X	X	X	2	2(nn)	
5. a. High Water Level in Scram Discharge Volume North Side	≤ 29 gal.		X(a)	X(z)	X(z)	2	2(nn)	
b. High Water Level in Scram Discharge Volume South Side	≤ 29 gal.		X(a)	X(z)	X(z)	2	2(nn)	
6. Low Condenser Vacuum	≥ 20 in. hg.			X(b)	X	1	3(mm)(nn)	
7. DELETED								

nn. With one required channel inoperable in one Trip System, within 12 hours, restore the inoperable channel or place the inoperable channel and/or that Trip System in the tripped▲ condition.

With two or more required channels inoperable:

1. Within one hour, verify sufficient channels remain OPERABLE or tripped▲ to maintain trip capability, and

2. Within 6 hours, place the inoperable channel(s) in one Trip System and/or that Trip System▲▲ in the tripped condition▲, and

3. Within 12 hours, restore the inoperable channels in the other Trip System to an OPERABLE status or tripped▲.

Otherwise, take the Action Required.

▲ An inoperable channel or Trip System need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the Action Required shall be taken.

▲▲ This action applies to that Trip System with the most inoperable channels; if both Trip Systems have the same number of inoperable channels, the action can be applied to either Trip System.

oo. With one required channel inoperable in one Trip System, either

1. Place the inoperable channel in the tripped condition within

   a. 12 hours for parameters common to Scram Instrumentation, and
   b. 24 hours for parameters not common to Scram Instrumentation.

or

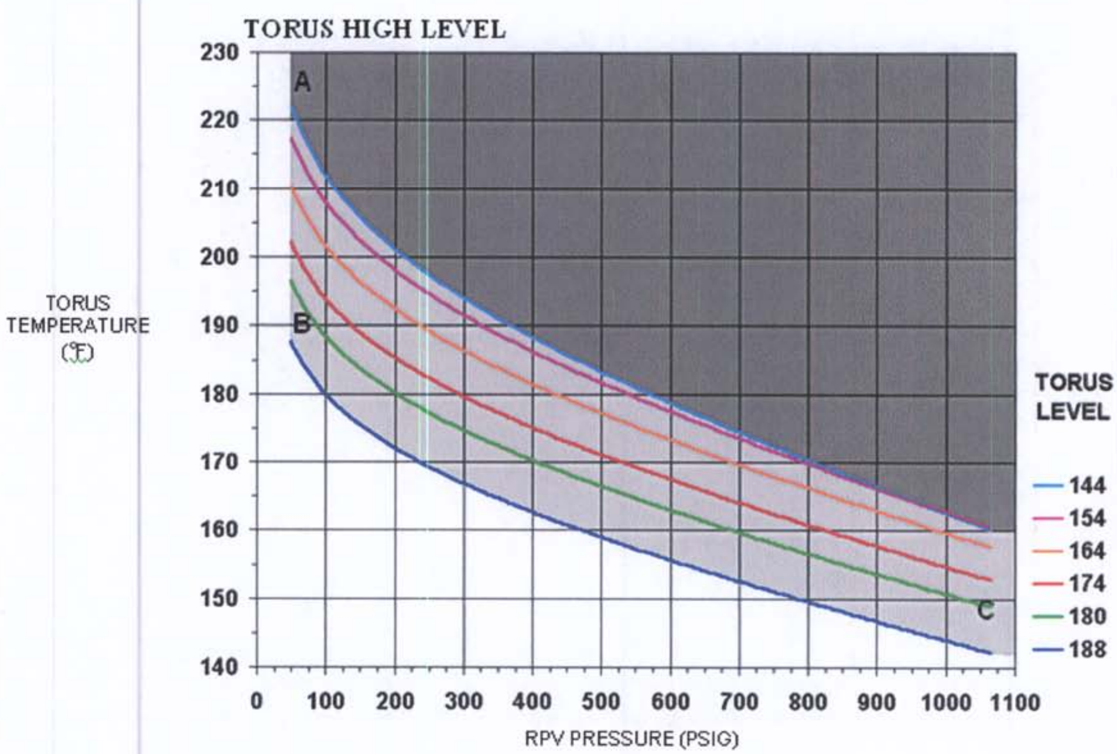
2. Take the Action Required.

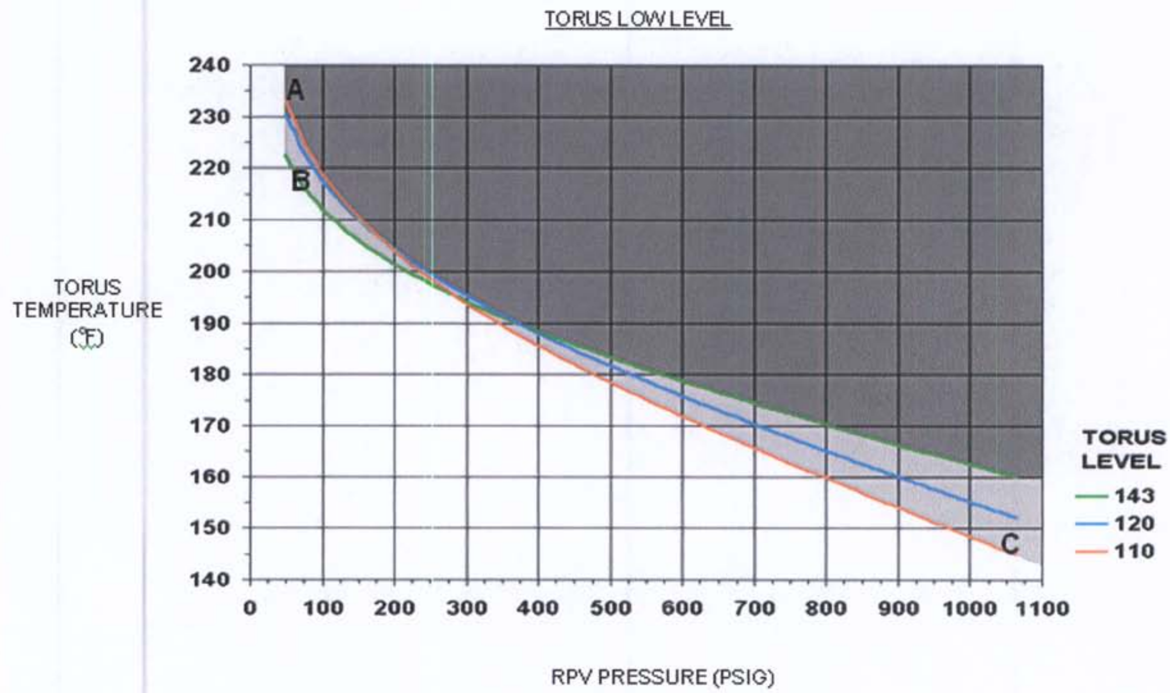
10 **ID: 09-1 NSRO10** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions include the following:

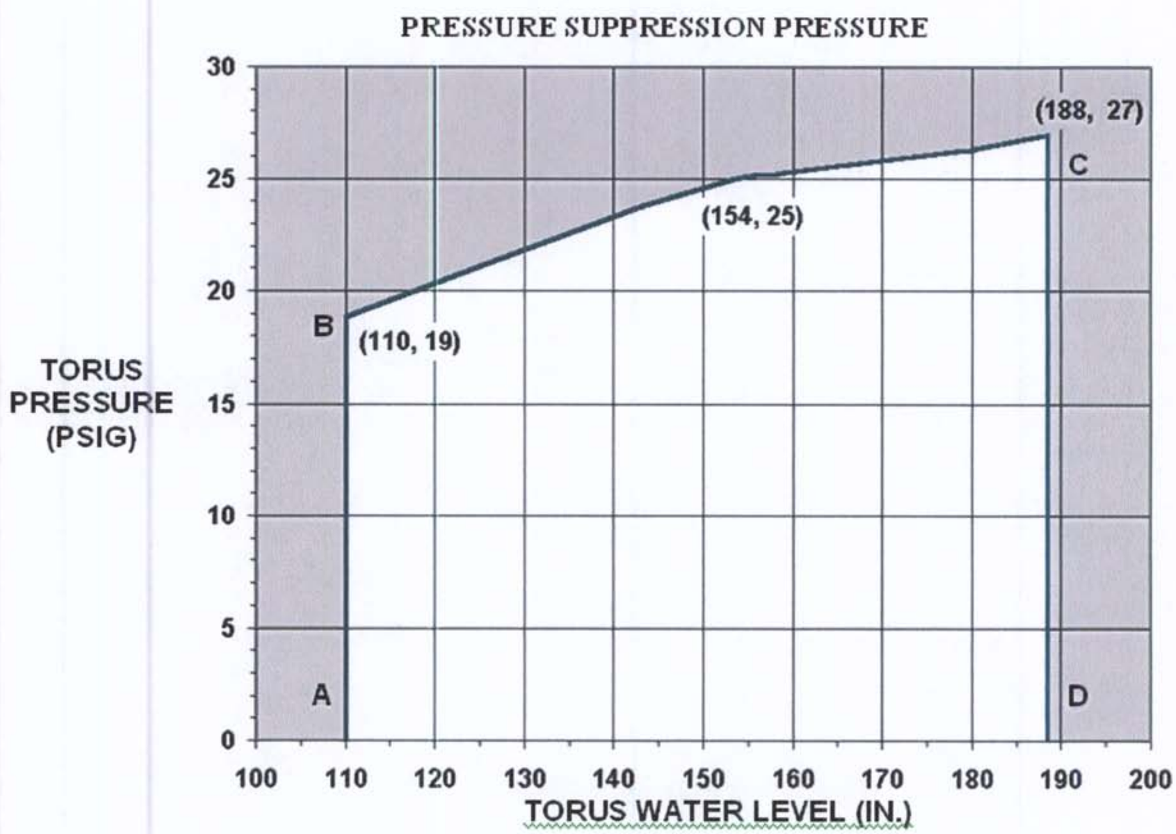
- RPV water level is 80"
- RPV pressure is 700 psig
- Drywell pressure is 12 psig
- Torus pressure is 11 psig
- Drywell temperature is 239 °F
- Torus water level is 164 "
- Torus water temperature is 155 °F

Which of the following is required given the associated parameter change? (Assume all other parameters **remain constant**)





TORUS LOW LEVEL
TORUS TEMPERATURE (°F)
240
230
220
210
200
190
180
170
160
150
140
A
B
C
0
100
200
300
400
500
600
700
800
900
1000
1100
RPV PRESSURE (PSIG)
TORUS LEVEL
143
120
110



PRESSURE SUPPRESSION PRESSURE
TORUS PRESSURE (PSIG)
30
25
20
15
10
5
0
(188, 27)
C
(154, 25)
B
(110, 19)
A
D
100
110
120
130
140
150
160
170
180
190
200
TORUS WATER LEVEL (IN.)

A. Emergency Depressurization is required if Torus water level rose and stabilized at 180" IAW the Primary Containment Control EOP.

B. Emergency Depressurization is required if Torus water temperature rose and stabilized at 175 °F IAW the Primary Containment Control EOP.

C. Rapidly depressurize the RPV with Turbine Bypass Valves in anticipation of Emergency Depressurization if Torus pressure rose and stabilized at 20 psig IAW the RPV Control - No ATWS EOP.

D. Rapidly depressurize the RPV with the Isolation Condensers in anticipation of Emergency Depressurization if Torus water level lowered and stabilized at 120" IAW the RPV Control - No ATWS EOP.

Answer: B

Answer Explanation:

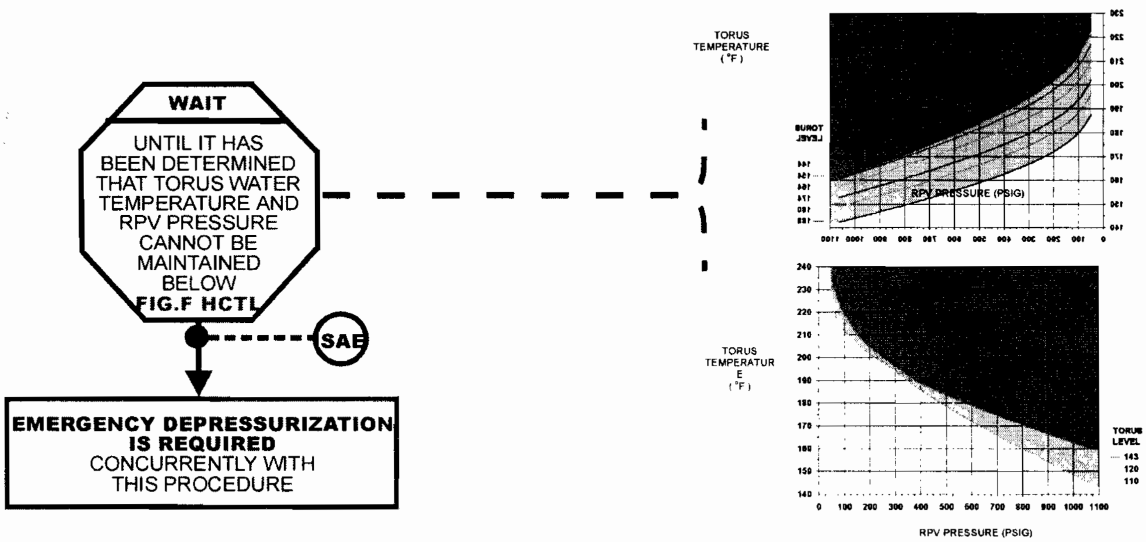
QID: 09-1 NSRO10		
Question # / Answer	**10**	**Developer/Date: NTP 1/5/10**

<table>
<tr><th colspan="6">Knowledge and Ability Reference Information</th></tr>
<tr><th colspan="4" rowspan="2">K&A</th><th colspan="2">Importance Rating</th></tr>
<tr><th>RO</th><th>SRO</th></tr>
<tr><td colspan="4">295013 High Suppression Pool Temperature 2.1.23 - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation</td><td></td><td>4.4</td></tr>
<tr><td>Level</td><td>SRO</td><td>Tier</td><td>1</td><td>Group</td><td>2</td></tr>
<tr><td>General References</td><td colspan="2">EOP Users Guide</td><td colspan="2"></td><td></td></tr>
</table>

Explanation	The plant was at power when an event occurred. Only 1 parameter is changed at a time with all others remain the same. At the RPV pressure of 700 psig, and Torus water level of 164", if Torus water temperature were to rise to 175 °F, then this would violate the Heat Capacity Limit Curve and an ED would be required. Answer B is correct. If Torus water level rose to 180", at 700 PSIG and a temperature of 155 °F, HCTL is not violated (temperature limit at 180" is 160 °F). Answer A is incorrect. At a Torus water level of 164" and Torus pressure rising to 20 psig, the Pressure Suppression Pressure is not violated and no ED would be required, thus anticipating ED is not permissible. Answer C is incorrect. At 700 psig, and a level of 120", the HCTL limit is 170 °F and the HCTL is not violated and thus ED is not required and cannot be anticipated. Also, since Torus water level is being maintained steady at > 110", ED is not required. Answer D is incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.845.0.0056 LO 200-10445

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of procedure			
Time to Complete: 1-2 minutes				

TORUS TEMPERATURE CONTROL



DISCUSSION

Earlier steps in the Torus temperature control leg prescribed actions for:

- Reducing Torus water temperature (maximizing Torus cooling)
- Eliminating unnecessary heat addition to the Torus (attempting to close any stuck open EMRVs OR closing any EMRVs not required for RPV pressure control or adequate core cooling)
- Minimizing the energy transferred from the Reactor to the Torus (Reactor scram)

Further, with entry to procedure RPV CONTROL - NO ATWS, direction may have been given to depressurize the RPV to stay below the Heat Capacity Temperature Limit (HCTL.) (Refer to Figure F of the Figures and Limits section of this document for additional details of the HCTL.)

If it becomes apparent that these efforts will fail to maintain the combination of Torus water temperature and RPV pressure below the Heat Capacity Temperature Limit, an Emergency RPV Depressurization is initiated while the Torus can still safely accommodate the blow down. If the combination of Torus water temperature and RPV pressure can be maintained below the limit, efforts to control the combination of Torus temperature and RPV pressure are continued.

"EMERGENCY DEPRESSURIZATION IS REQUIRED" is printed in bold, uppercase letters enclosed in a red box to emphasize the need to override RPV pressure control actions carried out concurrently in the RPV CONTROL procedure. Conditional Statements will direct depressurization according to the applicable EMERGENCY DEPRESSURIZATION procedure. The operator remains in the Primary Containment Control procedure and performs it concurrently with the Emergency Depressurization procedure.

11 **ID: 09-1 NSRO11** **Points: 1.00**

The plant was shutdown with fuel shuffling underway when a fire occurred in the Boiler House. The fire heavily damaged both MCCs 1A24 and 1B24 and both are de-energized.

Which of the following states the impact on the refuel floor activities?

A. Core alterations are **not** impacted by the event and may continue unrestricted.

B. Core alterations may continue for **only** the next 7 days due to the loss of SGTS Fan A.

C. Core alterations may continue **only** for the next 7 days due to the loss of SGTS Fan B.

D. Core alterations shall cease immediately due to the loss of Secondary Containment integrity.

Answer: D

Answer Explanation:

QID: 09-1 NSRO11		
Question # / Answer	11	Developer/Date: NTP 1/6/10

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
261000 SGTS A2.07 - 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: A.C. electrical failure					2.8
Level	SRO	Tier	2	Group	1
General References	205 330	TS 1.14		TS 3.5.B	

Explanation	The plant is shutdown with fuel shuffling in-progress. This activity requires Secondary Containment integrity, which includes the Standby Gas Treatment System operable. SGTS Fan 1-8 is powered from MCC 1A24 and SGTS Fan 1-9 is powered from MCC 1B24. Refuel activities could continue for a limited time if only 1 fan were operable, but it is given that both MCCs 1A24 and 1B24 are de-energized, and are thus must be declared inoperable. Thus, there is no operable SGT Fan. TS definition 1.14 for Secondary Containment integrity includes SGTS operable. Procedure 205 also requires Secondary Containment integrity. With no SGTS, there is no Secondary Containment integrity and core alterations must cease immediately. Answer D is correct. Answers B & C would be correct for the loss of a single SGTS Fan but are incorrect for the loss of both. If the candidate does not realize the relationship between the MCCs and SGTS and Secondray Containment integrity, then answer A might seem correct. But, answer A is incorrect.	
References to be provided during exam:	**TS 3.5**	
Learning Objective	2621.812.0.0003 LO 234-10451	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	**55.41**		**55.43**	**2**
	(SRO Only) Facility operating limitations in the Tech Specs and their bases			
Time to Complete: 1-2 minutes				

4.1.8

> NOTE
>
> Procedure 205.62, Refueling Bridge Check-Off is performed daily or as specified by the Shift Manager while the Refueling Platform is in operation.

Refueling Bridge Check-Off has been completed IAW Procedure 205.62. [ ]

4.1.9

> NOTE
>
> If one Drywell Radiation Monitor (criticality monitor) becomes inoperable, fuel movement may continue, provided all personnel are evacuated from the Drywell.

The following radiation monitors are operable:

- Area Radiation Monitors on the RB 119' elevation [ ]
- Refueling Platform (a minimum of one) [ ]
- Drywell Radiation Monitors [ ]

4.1.10 Source Range Monitors (SRMs) have been calibrated IAW Procedure 620.3.006. [ ]

4.1.11

> CAUTION
>
> Fuel transfers into the Fuel Pool are **not** permitted if pool temperature exceeds 115°F.

Fuel Pool Cooling is adequate to remove the decay heat load due to core offloading. [ ]

4.1.12 Secondary Containment integrity has been established IAW Procedure 312.10. [ ]

4.1.13 Shutdown Margin (SDM) requirements and SDM code bias have been determined by Reactor Engineering. [ ]

4.2 Precautions and Limitations

4.2.1 Core alterations are **not** allowed unless directly supervised by a FHD (SRO Licensed Operator) and authorized by a Fuel Move Sheet.

4.2.2 Limit core plate ΔP to 4 psid or less to prevent blade guide lift (GE SIL 406).

ATTACHMENT 330-2

ELECTRICAL CHECKOFF LIST FOR SBGT SYSTEM

Power Supply	Item	Location	Bkr. Pos.	Perform/Verify
VAC P-1* Bkr 20	Solenoid Valve for V-28-19, V-28-21,V-28-22, V-28-48	460 Swgr Room	ON	____/____
CIP-3* Bkr 13	Solenoid Valve for V-28-17 V-28-18,V-28-47,V-27-1,V-27-2	460 Swgr Room	ON	____/____
PAIPP-1 Bkr 6	Position Indication for V-28-18 and V-27-2	Lower CSR	ON	____/____

NOTE: Only position indication for valves applicable to this procedure are listed. Other equipment is powered from this breaker.

Power Supply	Item	Location	Bkr. Pos.	Perform/Verify
460V MCC 1A24	Motor for EF-1-8	Boiler House	ON	____/____
EF-1-8 Control Power Trans	V-28-23, 24 and 26	Boiler House	ON	____/____
460V MCC 1B24	Motor for EF-1-9	Boiler House	ON	____/____
EF-1-9 Control Power Trans	V-28-27, 28 and 30	Boiler House	ON	____/____
460V MCC 1A24	Contacts for electric heating coils EHC-1-5	Boiler House	ON	____/____
460V MCC 1B24	Contacts for electric heating coils EHC-1-6	Boiler House	ON	____/____
Instr. Pnl 4C Bkr.2	Solenoids for V-23-21, V-23-22	460 Swgr Room	ON	____/____
Dist Panel P3-3	Feed to ATC-P16 (Logic Control)	Boiler House	ON	____/____

*NOTE: Only those solenoid valves applicable to this procedure are listed for the indicated power supplies. Additional solenoid valves/equipment may also be powered from the indicated power supply.

1.14 SECONDARY CONTAINMENT INTEGRITY

Secondary containment integrity means that the reactor building is closed and the following conditions are met:

A. At least one door at each access opening is closed. (Note: Momentary opening and closing of the trunnion room door does not constitute a loss of secondary containment intergrity.)

B. The standby gas treatment system is operable.

C. All automatic secondary containment isolation valves are operable or are secured in the closed position.

1.15 (DELETED)

1.16 RATED FLUX

Rated flux is the neutron flux that corresponds to a steady state power level of 1930 MW(t). Use of the term 100 percent also refers to the 1930 thermal megawatt power level.

1.17 REACTOR THERMAL POWER-TO-WATER

Reactor thermal power-to-water is the sum of (1) the instantaneous integral over the entire fuel clad outer surface of the product of heat transfer area increment and position dependent heat flux and (2) the instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components except fuel rods in the cylindrical volume defined by the active core height and the inner surface of the core shroud.

1.18 PROTECTIVE INSTRUMENTATION LOGIC DEFINITIONS

A. Instrument Channel

An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

B. Trip System

A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system (e.g., initiation of a core spray loop, automatic depressurization, isolation of an isolation condenser, offgas system isolation, reactor building isolation, standby gas treatment and rod block) or the coincident tripping of two trip systems (e.g., initiation of scram, isolation condenser, reactor isolation, and primary containment isolation).

8. Shock Suppressors (Snubbers)

a. All safety related snubbers are required to be operable whenever the systems they protect are required to be operable except as noted in 3.5.A.8.b and c below.

b. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to operable status.

c. If the requirements of 3.5.A.8.a and 3.5.A.8.b cannot be met, declare the protected system inoperable and follow the appropriate action statement for that system.

d. An engineering evaluation shall be performed to determine if the components protected by the snubber(s) were adversely affected by the inoperability of the snubber prior to returning the system to operable status.

B. Secondary Containment

1. Secondary containment integrity shall be maintained at all times unless all of the following conditions are met:

a. The reactor is subcritical and Specification 3.2.A is met.

b. The reactor is in the cold shutdown condition.

c. The reactor vessel head or the drywell head are in place.

d. No work is being performed on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive material.

e. No operations are being performed in, above, or around the spent fuel storage pool that could cause release of radioactive materials.

2. Upon the accidental loss of SECONDARY CONTAINMENT INTEGRITY, restore, SECONDARY CONTAINMENT INTEGRITY within 4 hours, except as provided in specification 3.5.B.3.

3. With one or more of the automatic secondary containment isolation valves inoperable:

   a. Maintain at least one automatic secondary containment isolation valve in each affected penetration OPERABLE.

   b. Within 8 hours restore the inoperable automatic secondary containment isolation valve(s) to OPERABLE status or isolate each affected penetration with at least one valve secured in the closed position.

4. If Specifications 3.5.B.2 or 3.5.B.3 cannot be met:

   a. During Power Operation:

      (1) Have the reactor mode switch in the shutdown mode position within the following 24 hours.

      (2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

      (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

   b. During refueling:

      (1) Cease fuel handling operations or activities which could reduce the shutdown margin (excluding reactor coolant temperature changes).

      (2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

      (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

5. Two separate and independent standby gas treatment system circuits shall be operable when secondary containment is required except as specified by Specification 3.5.B.6.

12 **ID: 09-1 NSRO12** **Points: 1.00**

The plant was starting up after an outage with reactor power monitored by the SRMs and has **not** been declared critical. The URO had just taken the shift and reported the following observations:

- SRM 21 shows a slow rise with **no** control rod selected
- SRM PERIOD CHANNEL 21 shows a slow rise (positive period getting shorter)
- All other SRMs indicate constant counts
- No annunciators have alarmed

Which of the following states the cause for the observations and the SRO direction to the URO?

	Cause	SRO Direction
A.	The applied voltage to the detector is lowering and the SRM is inoperable	Cease all control rod withdrawals due to inadequate operable SRMs
B.	The SRM 21 recorder is failing and the SRM 21 recorder is inoperable	Bypass SRM 21 and continue with control rod withdrawals
C.	The applied voltage to the detector is rising and the SRM is inoperable	Bypass SRM 21 and continue with control rod withdrawals
D.	A control rod is drifting outward from the core	Enter and execute ABN-6, Control Rod Malfunctions

Answer: C

Answer Explanation:

QID: 09-1 NSRO12		
Question # / Answer	**12**	**Developer/Date: NTP 1/6/10**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
215004 Source Range Monitor **A2.01 - Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply degraded**						2.9
Level	**SRO**	**Tier**	**2**	**Group**	**1**	
General References	**TS 3.2.B.5**	**GFES RAP-G4d**		**LP 2621.828.0.0029**		
Explanation	The plant is starting up with reactor power on the SRMs when an observation is reported that SRM 21 is rising and the associated period meter is also rising. Both the SRM indication and period meter are fed from the SRM drawer. The SRMs operate in the Proportional region of the Gas-Filled Detector Characteristic Curve. As applied voltage to the detector rises, the counts also rises. Since no control rods are currently selected and being withdrawn, the SRMs will read a constant value. Since SRM 21 is displaying aberrant behavior, it shall be declared inoperable and bypassed as directed by the RAP when it alarms from high counts. IAW TS 3.2.B.5, only 3 operable SRMs are required, and thus the SRO can allow control rod withdrawals to continue. Answer C is correct. Answer A is incorrect since a lower detector voltage will result in fewer counts and there is no need to cease control rod withdrawals. Answer A is incorrect. Since the SRM drawer feeds the SRM indicator and the period meter, if just the SRM indicator was failing, there would be no change in the period meter. Answer B is incorrect. The question stem says that no annunciators have alarmed. It is true that a drifting outward control rod would give the indications observed in the question stem, the rod drift annunciator would be in alarm. Therefore, it can't be due to a drifting control rod. Answer D is incorrect.					

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

References to be provided during exam:	None	
Learning Objective	2621.812.0.0003 LO 234-10451	

Question Source (New, Modified, Bank)				New
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

SRM
HI / INOP

MANUAL CORRECTIVE ACTIONS: (continued from Page 1 of 3)

❑	IF	count rate is high,	
	THEN	SRMs may be withdrawn to maintain a count rate between 10^3 and 10^5 if at least three operable IRM channels in each Reactor Protection System indicate a decade of overlap.	[ ]

> NOTE
>
> Ensure the Technical Specification requirements of Table 3.1.1 **and** Section 3.9 (if applicable) are met prior to bypassing the SRM.

❑	IF	unit is inoperative,	
	THEN	a unit may be bypassed to allow repair.	[ ]
❑	IF	all SRM indication is lost,	
	THEN	**PERFORM** the following:	
		• **MAINTAIN** reactor coolant temperature constant.	[ ]
		• **CHECK** for a loss of 24VDC to Panel 3R and 5R.	[ ]

❑ **CHECK** for loss of SRM High Volts. [ ]

❑ **BYPASS** the affected SRM channel per Procedure 401.4, Nuclear Instrumentation-SRM Channels Bypass Operation. [ ]

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Rod Length Inserted (%)	Insertion Time (Seconds)
5	0.398
20	0.954
50	2.120
90	5.300

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.2.A are met. Time zero shall be taken as the de-energization of the pilot scram valve solenoids.

4. In service control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. Inoperable control rods shall be valved out of service, in such positions that Specification 3.2.A is met. In no case shall the number of inoperable control rods valved out of service be greater than six during the power operation. If this specification is not met, the reactor shall be placed in the shutdown condition.

5. Control Rods shall not be withdrawn for approach to criticality unless at least two source range channels have an observed count rate equal to or greater than 3 counts per second.

C. Standby Liquid Control System

1. The standby liquid control system shall be operable at all times under the following conditions:

   (a) when the reactor is not shut down by the control rods such that Specification 3.2.A is met, except as provided in Specification 3.2.C.3, and

   (b) when the reactor is >212°F, except during REACTOR VESSEL PRESSURE TESTING.

2. The standby liquid control solution shall have a Boron-10 isotopic enrichment equal to or greater than 35 atom %, be maintained within the cross-hatched volume-concentration requirement area in Figure 3.2-1 and at a temperature not less than the temperature presented in Figure 3.2-2 at all times when the standby liquid control system is required to be operable.

3. (a) If one standby liquid control system pumping circuit becomes inoperable during the RUN mode and Specification 3.2.A is met, the reactor may remain in operation for a period not to exceed 7 days, provided the pump in the other circuit is verified daily to be operable, otherwise be in the Shutdown condition within 24 hours.

13 **ID: 09-1 NSRO13** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions include the following:

- Offsite power has been lost
- EDG 1 has experienced an overspeed condition
- The RPV has been depressurized to 0 psig
- RPV water level is -10" and steady
- Drywell pressure is 21 psig
- Drywell temperature is 268 ^{o}F
- Torus water temperature is 104 ^{o}F
- Fire Water is injecting into the RPV via Core Spray System 1
- Core Spray System 2 s injecting into the RPV at 2000 GPM

Which of the following is the SRO's **next** direction?

A. Lineup CST to the Core Spray System.

B. Confirm SLC Pump B is injecting into the RPV.

C. Enter the Primary Containment Flooding Procedure.

D. Initiate Drywell Sprays with Containment Spray System A.

Answer: B

Answer Explanation:

QID: 09-1 NSRO13		
Question # / Answer	**13**	**Developer/Date: NTP 1/6/10**

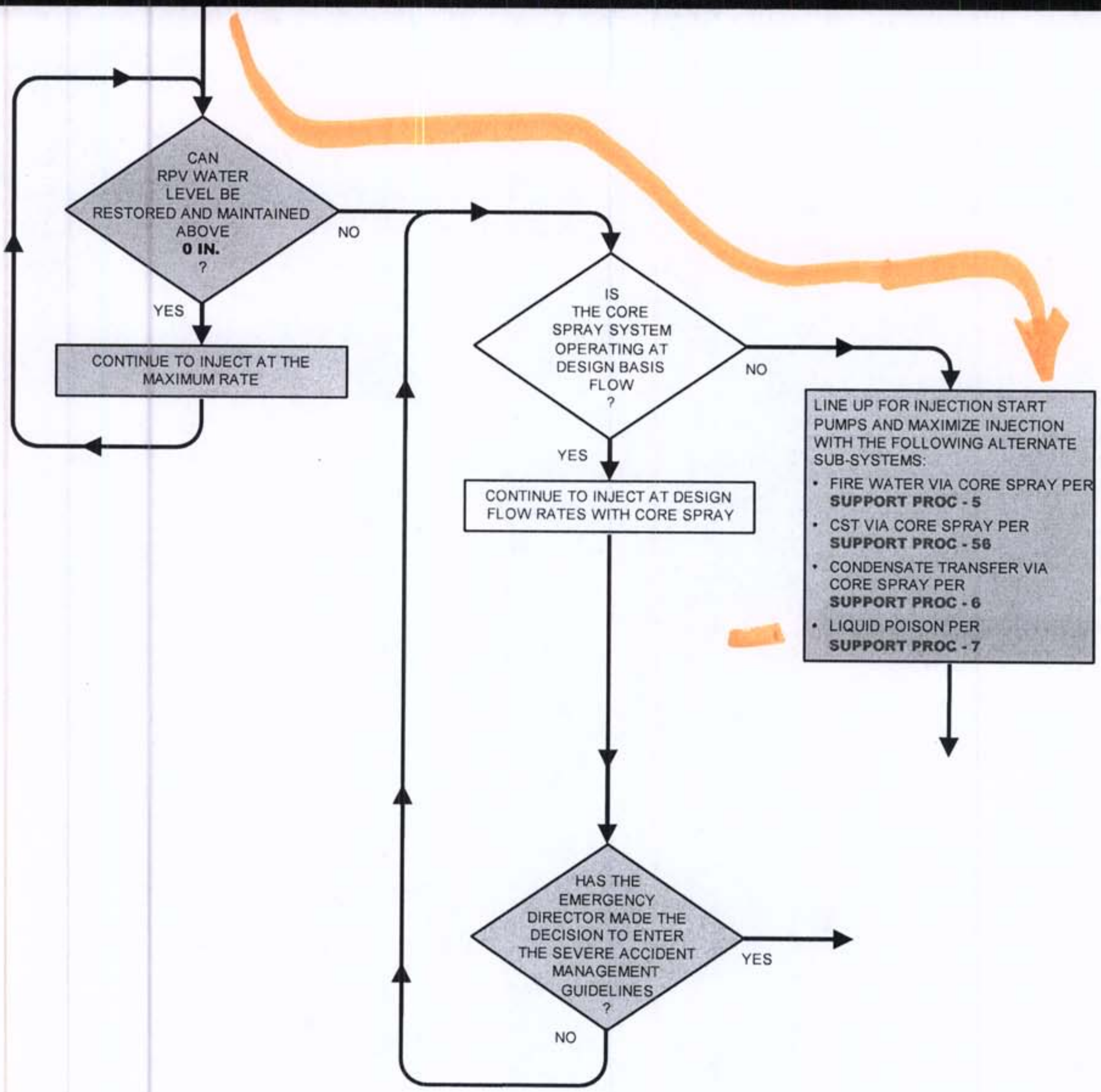
Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
211000 SLC **2.4.4 - Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.**					**4.7**
Level	**SRO**	**Tier**	**2**	**Group**	**1**

General References			
Explanation	The plant was at rated power when an event occurred, with several plant conditions given: Drywell temperature and pressure are high, Torus water temperature is high, and RPV water level is low. Offsite power is gone which removes and Feedwater/Condensate flow. EDG 1 & 2 fast started from the loss of offsite power, but EDG 1 has since tripped on overspeed. Most faults are bypassed on a EDG fast start, except overspeed. Thus Bus 1C and downstream busses are de-energized. IAW the RPV Control - No ATWS EOP, when RPV water level lowered to 0", the RPV was emergency depressurized. With RPV water level at 10", the nuclear fuel is exposed and adequate core cooling becomes the priority. Under the conditions given, with RPV water level this low, no FW/Condensate and Core Spray injecting at much less than design values, the EOP again directs SLC injection. SLC Pump 2 is powered from USS 1B2 (from EDG 2) and is available. Answer C is correct. Lining-up the CST to Core Spray is a potential path in the RPV Control - No ATWS EOP, but Core Spray B (System 2) should be and is injecting. CST could be lined up to those Core Spray Pumps not already injecting, but Fire Water is already injecting through the other Core Spray System. Fire water is a higher pressure, higher capacity that the CST flow. Thus answer A is incorrect. Under the given conditions, if RPV water level cannot be maintained > -20", then Primary Containment Flooding would be a correct answer. But with RPV water level at -10" and steady, answer C is incorrect. Spraying the Drywell is allowed under the conditions given, but with no offsite power and EDG 1 tripped, there is no power to Containment Spray A. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.845.0.0052 LO 200-10445		

Question Source (New, Modified, Bank)	**New**

<table>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X
3:SPK</td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td></td><td>55.43</td><td>5</td></tr>
<tr><td colspan="4">(SRO Only) Assessment of facility conditions and selection of appropriate procedure</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
</table>

RPV WATER LEVEL CONTROL (LEVEL RESTORATION)



DISCUSSION

The Level Restoration steps have been expanded to include the Core Spray System operating as designed as a success path for alternate level control. This change permits reliance on design basis core cooling criteria in preference to low-quality injection and Primary Containment flooding. As long as RPV water level can be restored and maintained above the Minimum Steam Cooling RPV Water Level or design basis flow from the Core Spray System can be established and maintained, the core cooling will remain within design basis and no other action is immediately required. The "design basis" core cooling criteria is derived from information contained in the Plant FSAR.

14 **ID: 09-1 NSRO14** **Points: 1.00**

The plant was at rated power when an event occurred. The Operator reports the following observations:

- BUS 1B CNTRL DC LOST has alarmed
- BUS 1D CNTRL DC LOST has alarmed
- **All** Isolation Condenser A valves on Panel 1F/2F indicate green light on
- Annunciator DC-E PWR XFER has **not** alarmed
- Annunciator DC-D PWR XFER has **not** alarmed

Which of the following shall the SRO direct?

Note:

ABN-53 is DC A and Panel Failures
ABN-54 is DC B and Panel/MCC Failures
ABN-55 is DC C and Panel/MCC Failures

A. IAW ABN-54, direct an Operator to manually align DC-1 transfer switch to DC-A.

B. IAW ABN-53, direct an Operator to manually align DC-E transfer switch to DC-B.

C. IAW ABN-54, direct an Operator to manually align DC-D transfer switch to DC-A.

D. IAW ABN-55, direct an Operator to manually operate supply and load breakers at DC-2 as required.

Answer: C

Answer Explanation:

QID: 09-1 NSRO14		
Question # / Answer	**14**	**Developer/Date: NTP 1/6/10**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

263000 DC Electrical Distribution 2.1.30 - Conduct of Operations: Ability to locate and operate components, including local controls.						4.0
Level	SRO	Tier	2	Group	1	

General References	ABN-54	ABN-53	ABN-55
Explanation	The plant was at rated power when DC control power was lost to Bus 1B & 1D. This DC power comes from 125 VDC Bus DC-B. Thus, there is a loss of DC-B and ABN-54 applies. The provided information also states that DC-D & DC-E did not transfer to their alternate DC supply. Of these, only DC-D is fed from DC-B. The question stem also states that the valve positions for Isolation Condenser A indicate their normal positions. Two of the valves are powered by DC-1, which is fed by DC-B. Since the indications do show valve positions, then DC-1 has transferred to its alternate DC supply (DC-A). Thus, DC-B has been lost and DC-D did not auto transfer. IAW ABN-54, manually performing the transfer of DC-D is correct. Answer C is correct. ABN-54 does direct manually transferring the power supply for DC-1 if it didn't auto transfer. Bus as shown, it did auto transfer. Answer A is incorrect. DC-E is normally powered from DC-A and the alternate supply is DC-B. But since DC-A has not lost power, performing actions IAW ABN-53 is not appropriate. Answer B is incorrect. The action in answer D is correct IAW ABN-55, but this ABN will not be entered under the given conditions since DC-C has not lost power. Answer D is incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0012 LO 263-10445		

Question Source (New, Modified, Bank)		New		
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SKP
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			

10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

1.0 APPLICABILITY

This procedure is applicable to a loss of 125 VDC power to the B 125 VDC Distribution System and DC-D and/or MCC DC-1, which are normally powered from DC-B.

2.0 INDICATIONS

2.1 Annunciators

If power is lost on both DC-A and DC-B, the annunciators will **not** alarm because power will be lost to DC-E.

Engraving	Location
BUS 1B CNTRL DC LOST	T-5-c
BUS 1D CNTRL DC LOST	T-5-e
1B2 DC LOST	U-4-d
1B3 DC LOST	U-6-d
MCC DC-1 PWR XFER	9XF-5-e
BUS A/B UV	9XF-1-d
DC-D PWR XFER	9XF-3-e
86A/SBO DC LOST	S-2-d
BATTERY B BKR OPEN	9XF-6-e

2.2 Plant Parameters

Parameter	Location	Change
BUS B VOLTS	8F/9F	Downscale
BUS B AMPS	8F/9F	Downscale
CHARGER B AMPS	8F/9F	Downscale

2.3 Other Indications

- Possible MSIV closure and scram due to transfer of the AC and DC power supplies to the MSIV solenoid valves.
- Loss of position indication to breakers powered from 4160 VAC buses 1B and 1D and 480 VAC USS 1B2 AND 1B3.
- Isolation Condenser A isolated (only AC valves and vent valves closed).
- CRD pump B (NC08B) trips, if running.
- Reactor Recirculation Pumps B and D loss of drive motor breaker indication.
- Cleanup Pump A (ND02A) trips on loss of DC-B, if running.
- Feed Pump 1B and 1C loss of indication.
- Cleanup Isolation Valves V-16-2 and V-16-14 loss of indication.

3.0 IMMEDIATE OPERATOR ACTIONS

None

4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 **REFER** to EP-OC-1010, Radiological Emergency Plan for Oyster Creek Generating Station for EAL evaluations. [ ]

4.2 IF power panel DC-D is unavailable,

THEN **PERFORM** the following:

4.2.1 **PERFORM** the following automatic actions verifications:

- **VERIFY** 'A' Isolation Condenser isolates. [ ]
- IF running,

  THEN **VERIFY** CRD Pump B trips. [ ]
- IF Mode Switch **not** in RUN

  AND IRMs <Range 8

  THEN **VERIFY** Rod Block received. [ ]
- **VERIFY** 'B' Isolation Condenser Vent Valves close (11F)
  - V-14-5 [ ]
  - V-14-20 [ ]
- **VERIFY** Containment Spray System two valves auto reposition to Torus Cooling mode (if in Drywell Spray mode), pumps stay running
- IF in use,

  THEN **VERIFY** Breaker controls powered from Remote Shutdown Panel return to NORMAL. [ ]

4.2.2 IF Auto Transfer Switch DC-D has **not** shifted to the alternate supply,

THEN manually **ALIGN** Auto Transfer Switch to DC Distribution Center A (125 VDC A/B Battery Rm). [ ]

4.2.3 IF power **cannot** be restored to 125 VDC Power Panel DC-D,

THEN **PERFORM** the following:

4.2.3.1 **START** P-15-1A, 'A' CRD Pump (NC08A). [ ]

4.2.3.2 **INFORM** Unit Supervisor (US) of Technical Specification 3.5 due to loss of auto isolation capability for the following valves:

- V-5-147, Rx Bldg Closed Cooling Water Drywell Inlet Valve [ ]
- V-5-166, Rx Bldg Closed Cooling Water Drywell Outlet Valve. [ ]

4.2.3.3 **CONTACT** Work Week Manager to initiate Panel DC-D repairs. [ ]

4.2.3.4 WHEN power is restored to Panel DC-D,

THEN **PERFORM** the following:

1. Momentarily **PRESS** the Recirc Pump Trip/Reset Channel A RESET. [ ]
2. Momentarily **PRESS** Channel C Reset, to reset the Recirculation Pump ATWS logic. [ ]

15 **ID: 09-1 NSRO15** **Points: 1.00**

The plant was at rated power when a major unisolable TBCCW leak occurred. **All** IMMEDIATE OPERATOR ACTIONs of ABN-20, TBCCW FAILURE RESPONSE, have been performed. Present plant conditions include the following:

- RPV water level is 168" and steady
- RPV pressure is 900 psig and steady
- **All** control rods in core quadrant 1 indicate position 04 and **all** other control rods indicate full-in
- Primary Containment parameters indicate normal

The Operator reports the following observation:

- **All** operating Feedwater and Condensate Pump temperatures indicate in excess of 250 °F

Which of the following will the SRO direct **next**?

A. Swap operating Feedwater Pumps IAW procedure 317 due to high bearing temperatures.

B. Terminate and prevent Feedwater injection IAW the Support Procedure, due to the ATWS.

C. Trip **only** the operating Feedwater Pumps IAW ABN-1, Reactor Scram, due to RPV high water level.

D. Trip **both** the operating Feedwater **and** Condensate Pumps IAW the ABN, due to high bearing temperatures.

Answer: D

Answer Explanation:

QID: 09-1 NSRO15		
Question # / Answer	**15**	**Developer/Date: NTP 1/7/10**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

400000 Component Cooling Water 2.4.16 - Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.					4.4
Level	SRO	Tier	2	Group	1

General References	RAP-H8j	EOP Users Guide	ABN-20

Explanation	The plant was at rated power when a major unisolable TBCCW leak occurred. In this condition, ABN-20 requires a manual scram and trip of all recirculation pumps. Conditions show that an entire quadrant of control rods inserted only to position 04. TBCCW supplies cooling to the Feedwater and Condensate Pumps. It is expected that the temperatures of the operating Feedwater & Condensate Pumps would rise. The conditions show normal parameters following a scram for RPV water level, pressure and Primary Containment, and with RPV water level within the EOP prescribed band and no RPV leaks in the primary Containment. On a normal scram, it is expected that the RPV Control - No ATWS EOP will be entered on RPV water level. And in this case, with many control rods at position 04, the reactor can still be determined to be shutdown and the No ATWS EOP is the correct EOP. In the water level leg of the EOP, it directs controlling water level using Feedwater/Condensate, CRD, and/or Core Spray. Each of these systems is bulletized which means that no system takes precedence over any other. 1 Feedwater Pump and all 3 Condensate Pumps will be running post-scram. The ABN-20 says to secure all pumps at the given temperatures. Since there is no RPV leak, RPV water level is steady high in the band, and both CRD pumps remain available, then stopping all pumps IAW the ABN is correct. Answer D is correct. Swapping Feedwater Pumps is plausible since 2 Feedwater Pumps would already have been shutdown. But, since all running condensate pumps would also be overheated, they would require to be shutdown as well. Running a Feedwater Pump with no condensate pump is not allowed. Thus answer A is incorrect. In the ATWS EOP, it is appropriate to terminate and prevent Feedwater injection when power is >2%. Although no power level is given, the candidate may think that entry into the ATWS EOP is correct and termination of Feedwater could be correct IAW Support Procedure 17. Since the No ATWS EOP allows control rods at position 04, then the ATWS EOP is incorrect and no termination/prevention of Feedwater IAW the Support procedures applies. Answer B is incorrect. ABN-1 provides direction for RPV water level control. When water level cannot be restored and maintained

	below 170", it directs to trip all Feedwater Pumps. But with water level at 168" and steady, this is not appropriate. Answer C is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0017 LO 260-10445		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of appropriate procedures			
Time to Complete: 1-2 minutes				

4.9

NOTE

The indicated actions for the following systems, upon reaching their limits, may be performed in any order or concurrently.

- Feed and Condensate System — Step 4.9.1
- Stator Cooling Water — Step 4.9.2
- Turbine Lube Oil — Step 4.9.3
- Recirc MG Sets — Step 4.9.4

PERFORM the indicated actions for any of the following systems:

4.9.1 Feed and Condensate System

4.9.1.1 IF Condensate pump bearing temperature ≥ 185° F (J-8-f)

OR

C' Feed pump outer bearing temperature ≥ 195° F (J-8-f)

OR

Any other Feed pump bearing temperature ≥ 185° F (J-8-f),

THEN **MONITOR** bearing temperatures closely on Panel 12XR, Temperature Monitor 12XR-21. [ ]

4.9.1.2 IF all pump bearing temperatures ≥ 195° F, as indicated by Panel 12XR, Temperature Monitor 12XR-21 and therefore require all Feed and Condensate pumps to be shut down.

THEN **PERFORM** the following:

1. IF the reactor is in the STARTUP or RUN mode,

   THEN **PERFORM** the following:

   a. **SCRAM** reactor IAW ABN-1. [ ]

   b. **CONFIRM** Feed pumps shutdown. [ ]

   c. **CONFIRM** Condensate pumps shutdown. [ ]

2. IF the reactor is in the SHUTDOWN or REFUEL mode,

   THEN **PERFORM** the following:

   a. **CONFIRM** Feed pumps shutdown. [ ]

   b. **CONFIRM** Condensate pumps shutdown. [ ]

4.9.2 Stator Cooling Water

4.9.2.1 IF a turbine runback occurs,

THEN concurrently **EXECUTE** ABN-11, Loss of Generator Stator Cooling. [ ]

16 **ID: 09-1 NSRO16** **Points: 1.00**

The plant was at rated power with Offgas Radiation Monitor RN12A failed downscale due to an electrical failure.

An event then occurred that resulted in the following annunciators alarming over the next 3 minutes:

- RWCU HELB I **and** RWCU HELB II
- RADIATION MONITORS PROCESS - OFFGAS HI **and** OFFGAS HI HI
- RADIATION MONITORS AREA - AREA MON HI

The Operator reports the following observations:

- All Reactor Water Cleanup System isolation valves indicate green light on
- Offgas Radiation Monitor RN12B indicates 1100 mr/hr and **rising** slowly
- STACK EFFLUENT LRM CH#1 and CH#2 are **rising** slowly
- ARM CLEANUP SYS PUMP AREA (C-1) indicates 40 mr/hr and **lowering**

Which of the following states the impact on the Offgas System and the next SRO direction?

	Offgas Impact	Required Action
A.	Offgas will automatically isolate	Reduce reactor power IAW ABN-ABN-26, High Main Steam/Offgas/Stack Effluent Activity
B.	Offgas will automatically isolate	Scram the reactor IAW the Secondary Containment Control EOP
C.	Offgas will **not** automatically isolate	Scram the reactor IAW the Radioactivity Release Control EOP
D.	Offgas will **not** automatically isolate	Shutdown the reactor IAW the Secondary Containment Control EOP

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Answer: A

Answer Explanation:

QID: 09-1 NSRO16		
Question # / Answer	16	Developer/Date: NTP 1/7/10

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
272000 Radiation Monitoring A2.03 - Ability to 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: A.C. electrical failure					3.1
Level	SRO	Tier	2	Group	2
General References	ABN-26	EOP Users Guide	EP-AA-1010		

Explanation	The plant was at rated power with an offgas radiation monitor failed downscale due to a power failure. An event then occurs which shows a leak in the RWCU System, high radiation in the offgas system, an area radiation monitor in alarm, and increased radiation in the stack. Several procedures could be entered: the RAP for the alarms, ABN-26, Secondary Containment Control EOP, and possibly the Radioactive release Control EOP. The Operator observations show that the RWCU System isolated, as designed; 1 offgas radiation monitor is above the high setpoint, and the RWCU ARM radiation indication is lowering (confirming that the RWCU leak has been stopped). The logic for the offgas radiation monitors for isolating the offgas system is both hi-hi, or one downscale and 1 hi-hi. Thus, with the information provided, 1 radiation monitor is hi-hi and 1 is downscale. Therefore, the offgas isolation will occur (after a 15 minute timer times out). For the conditions provided, ABN-26 directs lowering reactor power in an attempt to clear the alarm to prevent the offgas system isolation. Answer A is correct. The Secondary Containment Control EOP directs a scram prior to exceeding a max safe value, provided that a primary system is discharging into the Secondary Containment. From the question, it shows that a primary system was discharging into the Secondary Containment but is no longer, and radiation levels are lowering. Thus a scram is not the appropriate action. Answer B is incorrect. Indications show that an offsite radiological release is in progress, but it does not rise to the point of the entry condition for the Radioactive Release Control EOP. If it had, then a scram would be appropriate. Also, offgas will isolate. Answer C is incorrect. A reactor shutdown is appropriate in the Secondary Containment Control EOP if a non-primary system were discharging into the Secondary Containment and radiation levels were rising. Also, offgas will isolate. Answer D is incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0004 LO 193	

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize relationship			
10CRF55 Content	55.41		55.43	5
	(SRO Only)			
Time to Complete: 1-2 minutes				

4.3.3 **MONITOR** the following:

- Off gas activity. [ ]
- Stack effluent activity. [ ]

4.3.4 **EVACUATE** the following as directed by the Unit Supervisor:

- Turbine Building [ ]
- Reactor Building [ ]

4.3.5 **REFER** to EP-OC-1010, Radiological Emergency Plan for Oyster Creek Generating Station, for EAL evaluation. [ ]

4.3.6 **NOTIFY** Reactor Engineering of Plant conditions. [ ]

4.4 Rise in Off Gas or Stack Effluent Activity

4.4.1 IF Reactor Power is greater than 40%,

AND Off gas or Stack effluent activity rises by more than 50% after factoring out any rise due to changes in thermal power,

THEN **PERFORM** the following:

1. **DIRECT** Chemistry to sample the following for activity:
   - Off gas. [ ]
   - Reactor coolant. [ ]
2. **INFORM** Unit Supervisor of Technical Specifications:
   - 3.6.E, Main Condenser Offgas Radioactivity [ ]
   - 4.6.E, Main Condenser Offgas Radioactivity (Surveillance) [ ]
3. **NOTIFY** Reactor Engineering of the rise in Off gas or Stack effluent activity. [ ]

4.4.2 IF Any of the following alarms are received:

- OFF GAS HI (10F-2-c) [ ]
- STACK EFFLUENT HI (10F-2-d) [ ]
- STACK EFFLUENT HI-HI (10F-1-d) [ ]

THEN **PERFORM** the following:

1. NOTE

   A change in any of the listed parameters may cause a fluctuation in the off gas release rate.

   REVIEW recent changes in any of the following parameters:

   - Off Gas line flow. [ ]
   - Condenser vacuum. [ ]
   - Steam seal header pressure. [ ]

2. **NOTIFY** Chemistry of any change in conditions. [ ]

3. **REDUCE** Reactor Power in accordance with Procedure 202.1, Power Operation until all three radiation alarms listed in Step 4.4.2 have cleared. [ ]

4. IF all three radiation alarms listed in Step 4.4.2 **cannot** be cleared,

   THEN **DIRECT** Chemistry to sample the following for activity:

   - Reactor coolant. [ ]
   - Off gas. [ ]

4.4.3 <u>IF</u> the OFF GAS HI-HI alarm (10F-1-c) is received,

<u>THEN</u> **PERFORM** the following:

1. **VERIFY** off gas indications on Panel 10F. [ ]

2. **REDUCE** Reactor Power in accordance with Procedure 202.1, Power Operation until the OFF GAS HI-HI alarm clears. [ ]

3. **COMMENCE** Plant shutdown in accordance with Procedure 203, Plant Shutdown. [ ]

4. <u>IF</u> the OFF GAS HI-HI alarm does <u>**not**</u> clear within 15 minutes of actuation,

   <u>THEN</u> **PERFORM** the following:

   A. **SCRAM** the Reactor IAW ABN-1, Reactor Scram. [ ]

   B. **CONFIRM** the following valves closed:

      - V-7-31, Off Gas Exhaust Isolation Valve. (Panel 10XF) [ ]

      - AOV-0001A/-0001B, AOG Inlet Valve. (Panel 10XF) [ ]

   C. **PLACE** Drain Valves V-7-29/SOV-016 control switch in the CLOSE position. (Panel 10XF) [ ]

17 **ID: 09-1 NSRO17** **Points: 1.00**

The plant was starting up after a refuel outage. Present plant conditions include the following:

- The REACTOR MODE SELECTOR switch is in STARTUP
- RPV pressure indicates 0 psig
- RECIRC PUMP SUCTION TEMPS indicates 60 °F subcooled
- The very **first** control rod in the pull sheet has been withdrawn to position 48

An event then occurred resulting in the following:

- An apparent electrical malfunction caused TIP #1 to drive to the CORE TOP and remain
- Efforts to withdraw the TIP from the Control Room were unsuccessful
- Control rod withdrawals have been halted

IAW TS 3.5, Containment, which of the following actions is correct?

A. The TIP ball valve must be closed within 4 hours or activate the shear valve.

B. The TIP ball valve must be closed within 48 hours or activate the shear valve.

C. No actions are required since Primary Containment Integrity is not currently required.

D. The TIP ball valve **or** shear valve must be closed within 48 hours or insert all control rods.

Answer: C

Answer Explanation:

QID: 09-1 NSRO17		
Question # / Answer	17	Developer/Date: NTP 1/9/10

Knowledge and Ability Reference Information		
K&A	Importance Rating	
	RO	SRO

215001 Traversing In-core Probe 2.2.22 - Equipment Control: Knowledge of limiting conditions for operations and safety limits					4.7
Level	**SRO**	**Tier**	**2**	**Group**	**2**
General References	**TS 3.5**				
Explanation	The plant is starting after a refuel outage with 1 control rod withdrawn. Shutdown margin proves that the reactor cannot go critical with a single control rod withdrawn. The stem also shows that RPV pressure is 0 psig and is 60 °F subcooled. This means that RPV coolant temperature is 212-60 = 152 °F. An event then occurs which results in TIP #1 driving in to the Core Top location, and it cannot be retracted. TS 3.5 provides the following: PRIMARY CONTAINMENTINTEGRITY shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt or during REACTOR VESSEL PRESSURE TESTING. a. With one or more of the automatic containment isolation valves inoperable: (1) Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours (48 hours for the traversing in-core probe system) either; (a) Restore the inoperable valve(s) to OPERABLE status or (b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or (c) Isolate each affected penetration by use of at least one closed manual valve or blind flange. (2) If Specification 3.5.A.3 or the provisions of Specifications 3.5.A.3.a.(1)(a), (b) or (c) can not be met, the reactor shall be PLACED IN the COLD SHUTDOWN CONDITION within 24 hours. Under the given conditions, Primary Containment Integrity is not required and thus there are no TS actions required. Answer C is correct. Had Primary Containment Integrity been required, then LCOs for an inoperable TIP would have applied. The other answers allow some time to recover primary Containment but are incorrect.				

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

References to be provided during exam:	**TS 3.5**	
Learning Objective	2624.828.0.0032 LO 422	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	**55.41**		**55.43**	**2**
	(SRO Only) Facility operating limitations in the Tech Specs and their basis			
Time to Complete: 1-2 minutes				

3. PRIMARY CONTAINMENT INTEGRITY shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 Mwt or during REACTOR VESSEL PRESSURE TESTING.

a. With one or more of the automatic containment isolation valves inoperable:

(1) Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours (48 hours for the traversing in-core probe system) either;

(a) Restore the inoperable valve(s) to OPERABLE status or

(b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or

(c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.

(2) If Specification 3.5.A.3 or the provisions of Specifications 3.5.A.3.a.(1)(a), (b) or (c) can not be met, the reactor shall be PLACED IN the COLD SHUTDOWN CONDITION within 24 hours.

(3) An inoperable containment isolation valve of the shutdown cooling system may be opened with a reactor water temperature equal to or less than 350°F in order to PLACE the reactor IN the COLD SHUTDOWN CONDITION. The inoperable valve shall be returned to the OPERABLE status prior to placing the reactor in a condition where PRIMARY CONTAINMENT INTEGRITY is required.

b. If the primary containment air lock is inoperable, per Specification 4.5.C.2, restore the inoperable air lock to OPERABLE status within the 24 hours or be in at least a SHUTDOWN CONDITION within the next 12 hours and in cold shutdown within the following 24 hours.

18 **ID: 09-1 NSRO18** **Points: 1.00**

The plant was at 15% power on a startup. Present plant conditions include the following:

- Feedwater Pump C and Condensate Pumps A & C in service
- CRD Pump NC08A is in service

An event then occurred resulting in the loss of DC Bus B.

While investigating the electrical problem, A LOCA in the Primary Containment occurred. Present plant conditions include the following:

- RPV water level is 125" and lowering slowly
- RPV pressure is 825 psig and lowering slowly
- Drywell temperature is 248 °F and rising
- Drywell pressure is 17 psig and rising slowly
- Torus water temperature is 87 °F and steady

Which of the following is the **next** SRO direction?

A. Start CRD Pump NC08B and maintain RPV water level 138" - 175" IAW the RPV Control - No ATWS EOP.

B. Start Feedwater Pump A and maintain RPV water level 138" - 175" IAW the RPV Control - No ATWS EOP.

C. Emergency Depressurize the RPV IAW the Drywell Temperature leg of the Primary Containment Control EOP.

D. Initiate Drywell Sprays using Containment Spray System 2 IAW the Drywell Pressure leg of the Primary Containment Control EOP.

Answer: B

Answer Explanation:

QID: 09-1 NSRO18		
Question # / Answer	**18**	**Developer/Date: NTP 1/8/10**

Knowledge and Ability Reference Information

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

K&A				Importance Rating RO	Importance Rating SRO
259001 Reactor Feedwater A2.08 - Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of DC electrical power					2.6
Level	SRO	Tier	2	Group	2
General References	BR 3028	RPV Control - No ATWS EOP	317		

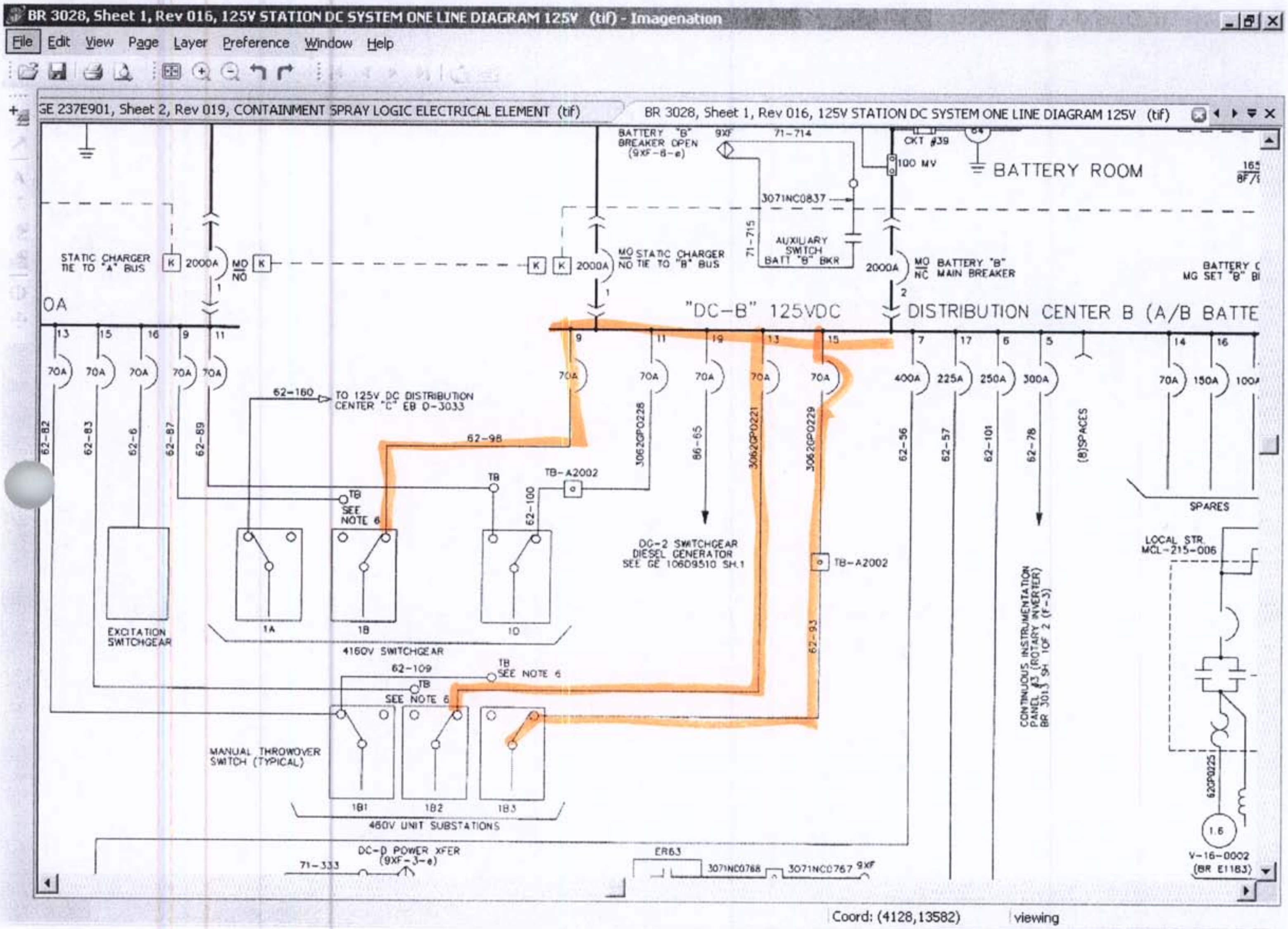
Explanation	The plant was at 15% power on a startup when DC B was lost. Then, a LOCA in the Primary Containment occurred. Indications provided show that RPV water level is low and Drywell parameters are elevated. A loss of DC B results in the loss of control power to Bus 1B, 1D, USS 1B2 and 1B3. With RPV water level low, the RPV Control EOP directs RPV water level 138-175" with Support procedure 2 (Feedwater) and SP-3 (CRD). With the loss of DC B, there is no power to close either Feedwater Pump B or C breaker from the control room. Feedwater Pump A gets DC control power from DC C, which is available. Answer B is correct. SP-3 for CRD does say to start additional CRD pumps to maintain RPV water level. But with no control power, the Panel Operator is unable to start CRD Pump NC08B (DC supply is from DC B). Answer A is incorrect. The Drywell temperature leg of the Primary Containment Control EOP directs ED if Drywell temperature cannot be maintained less than 281 °F. The DC loss prevents starting any Containment Spray System 2 pumps from the Control Room, but System 1 pumps are available and can be started from the control room. This avenue should be attempted first to see if Drywell temperature can be maintained less than the limit to prevent the need to ED as given in answer D. Answer C is incorrect. As discussed for answer C, Containment Spray System 2 pumps cannot be started from the control room to lower Drywell temperature, but System 1 pumps can be. These Containment Spray Pumps shall be attempted to lower Drywell temperature and pressure before it is determined that Drywell temperature cannot be maintained below 281 °F, and the direction to Emergency Depressurize is required. Answer D is incorrect.

References to be provided during exam:	**None**	

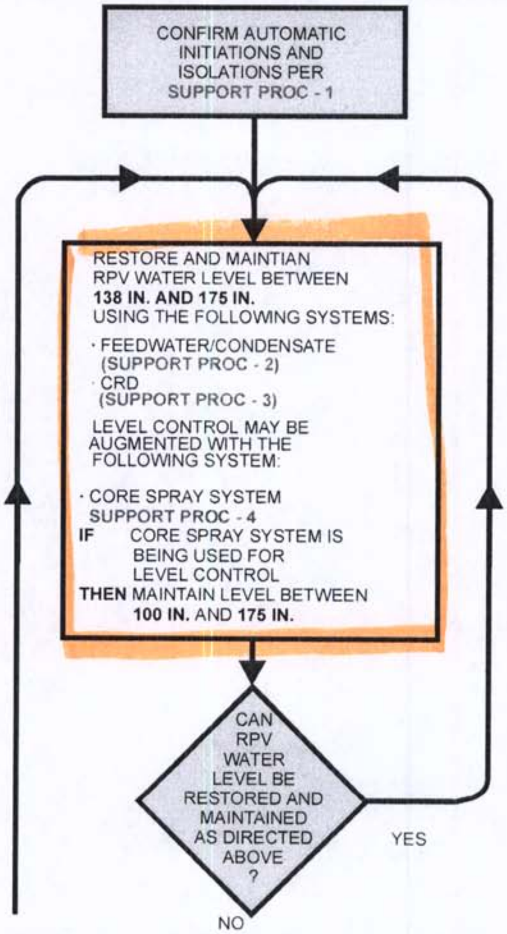
Learning Objective	2621.845.0.0052 LO 3055

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**

	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning				
	10CRF55 Content	55.41		55.43	5
		(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
	Time to Complete: 1-2 minutes				


BR 3028, Sheet 1, Rev 016, 125V STATION DC SYSTEM ONE LINE DIAGRAM 125V (tif) - Imagenation
File Edit View Page Layer Preference Window Help
GE 237E901, Sheet 2, Rev 019, CONTAINMENT SPRAY LOGIC ELECTRICAL ELEMENT (tif)
BR 3028, Sheet 1, Rev 016, 125V STATION DC SYSTEM ONE LINE DIAGRAM 125V (tif)
BATTERY "B" BREAKER OPEN (9XF-8-e)
9XF
71-714
CKT #39
100 MV
BATTERY ROOM
3071NC0837
71-715
AUXILIARY SWITCH BATT "B" BKR
STATIC CHARGER TIE TO "A" BUS
2000A
MO NO
STATIC CHARGER TIE TO "B" BUS
MO NC
BATTERY "B" MAIN BREAKER
BATTERY MG SET "B"
"DC-B" 125VDC
DISTRIBUTION CENTER B (A/B BATTE
70A
400A
225A
250A
300A
150A
62-82
62-83
62-6
62-87
62-89
62-160
TO 125V DC DISTRIBUTION CENTER "C" EB D-3033
62-98
3062GP0228
86-65
3062GP0221
3062GP0229
62-56
62-57
62-101
62-78
(8)SPACES
SPARES
TB
TB-A2002
TB SEE NOTE 6
62-100
DC-2 SWITCHGEAR DIESEL GENERATOR SEE GE 106D9510 SH.1
TB-A2002
62-93
CONTINUOUS INSTRUMENTATION PANEL #3 (ROTARY INVERTER) BR 3013 SH 1OF 2 (F-3)
LOCAL STR. MCL-215-006
EXCITATION SWITCHGEAR
1A
1B
1D
4160V SWITCHGEAR
62-109
TB SEE NOTE 6
MANUAL THROWOVER SWITCH (TYPICAL)
1B1
1B2
1B3
480V UNIT SUBSTATIONS
DC-D POWER XFER (9XF-3-e)
71-333
ER63
3071NC0768
3071NC0767
62OP0225
1.6
V-16-0002 (BR E1183)
Coord: (4128,13582)
viewing

RPV WATER LEVEL CONTROL



DISCUSSION

This step specifies the desired RPV water level range, as well as the manner in which some of the injection systems should be operated.

The upper end of the level control band is 175 in., the high level Turbine trip. Controlling RPV water level below this value avoids moisture carryover into the Main Steam Lines or IC lines. The lower end of the control band (when Core Spray is not in service) is 138 in., the low RPV water level scram setpoint. Maintaining RPV water level above this value is preferred because, barring the presence of other scram/isolation signals, doing so allows the reset of the low level scram signal.

If Core Spray is being used for RPV makeup, the lower end of the control band is expanded to 100 in. The expanded level control band reduces cycling of the Core Spray parallel isolation valves and increases the rest time of the valve operators. With extended use and without sufficient rest time, the valve operators for the parallel isolation valves may fail. With the expanded band (100 in. – 175 in.), the valve operators will be provided with the rest time necessary to insure proper long term operation.

These level control ranges are sufficient to assure adequate core cooling yet minimize unwarranted demands on an operator's attention. If unnecessarily constrained within narrower limits, an operator may be less effective in performing concurrent duties.

ATTACHMENT 317-3

FEEDWATER SYSTEM PRE-STARTUP ELECTRICAL LINEUP

POWER SUPPLY	EQUIPMENT	LOCATION	BREAKER POSITION	Perform/ IV*
4160V 1A	Feed Pump 1A	TB 4160V RM	Close	/
4160V 1B	Feed Pump 1B	TB 4160V RM	Close	/
4160V 1B	Feed Pump 1C	TB 4160V RM	Close	/
1B11A	A String Heater (V-2-10) Bank Outlet Valve	TB MEZ	Close	/
1B11A	B String Heater (V-2-11) Bank Outlet Valve	TB MEZ	Close	/
1B11A	C String Heater (V-2-12) Bank Outlet Valve	TB MEZ	Close	/
1A12A	MFRV A Block Valve V-2-740	TB MEZ	Close	/
1B12A	MFRV C Block Valve V-2-741	TB MEZ	Close	/
DC-E Bkr. 15	ROPS (Panel 14XR)	Lower Cable Spreading Rm	Close	/
IP-4B Bkr. 1	ROPS (Panel 14XR)	480V Room	Close	/

Performed By: ______________________ Date: ____________ Time: ________

Verified By: ______________________ Date: ____________ Time: ________

Approved By: ______________________ Date: ____________ Time: ________
US

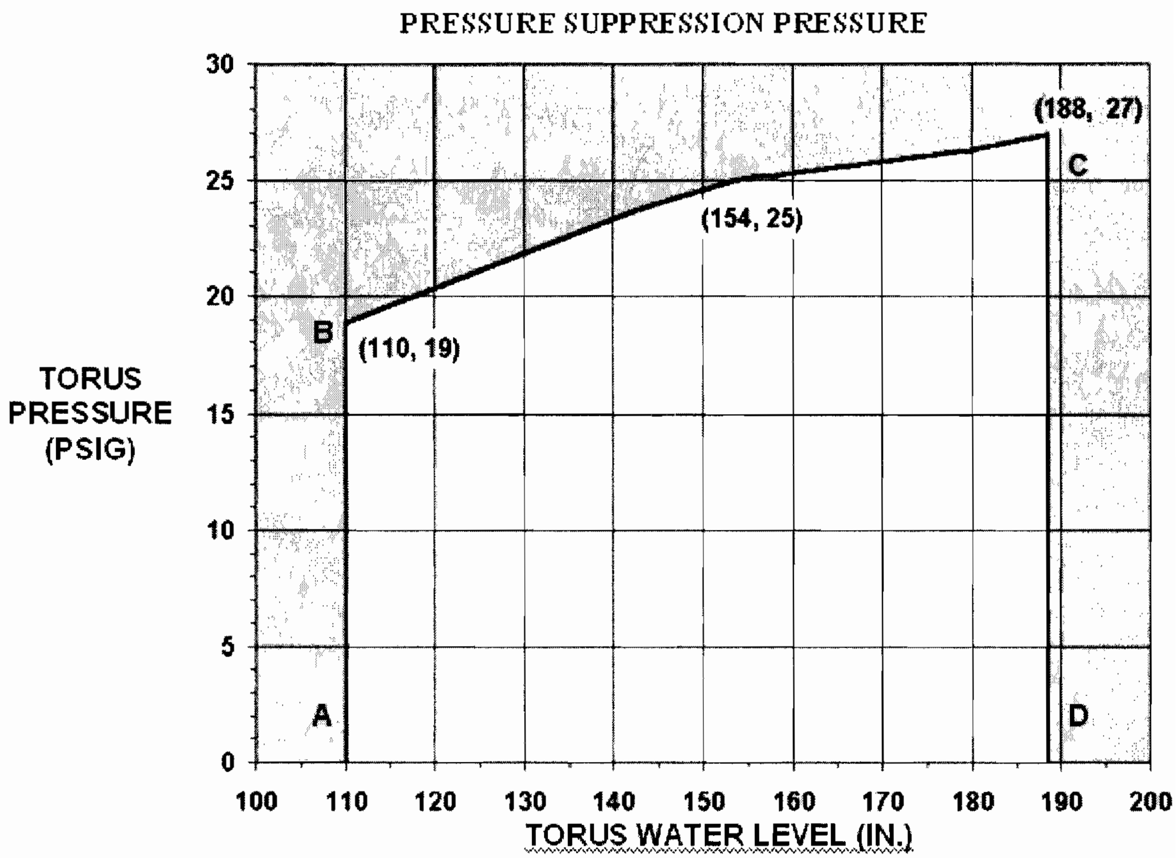
* Independent Verification (IV)

19 **ID: 09-1 NSRO19** **Points: 1.00**

The reactor was at rated power when a LOCA occurred. Present plant conditions include the following:

- All control rods inserted
- RPV pressure is 45 psig
- RPV water level is 60" and rising
- 1 Condensate Pump is injecting
- Core Spray A and B are injecting
- Torus water level is 184" and rising
- Torus pressure is 22 psig and rising slowly
- Containment Sprays are inoperable

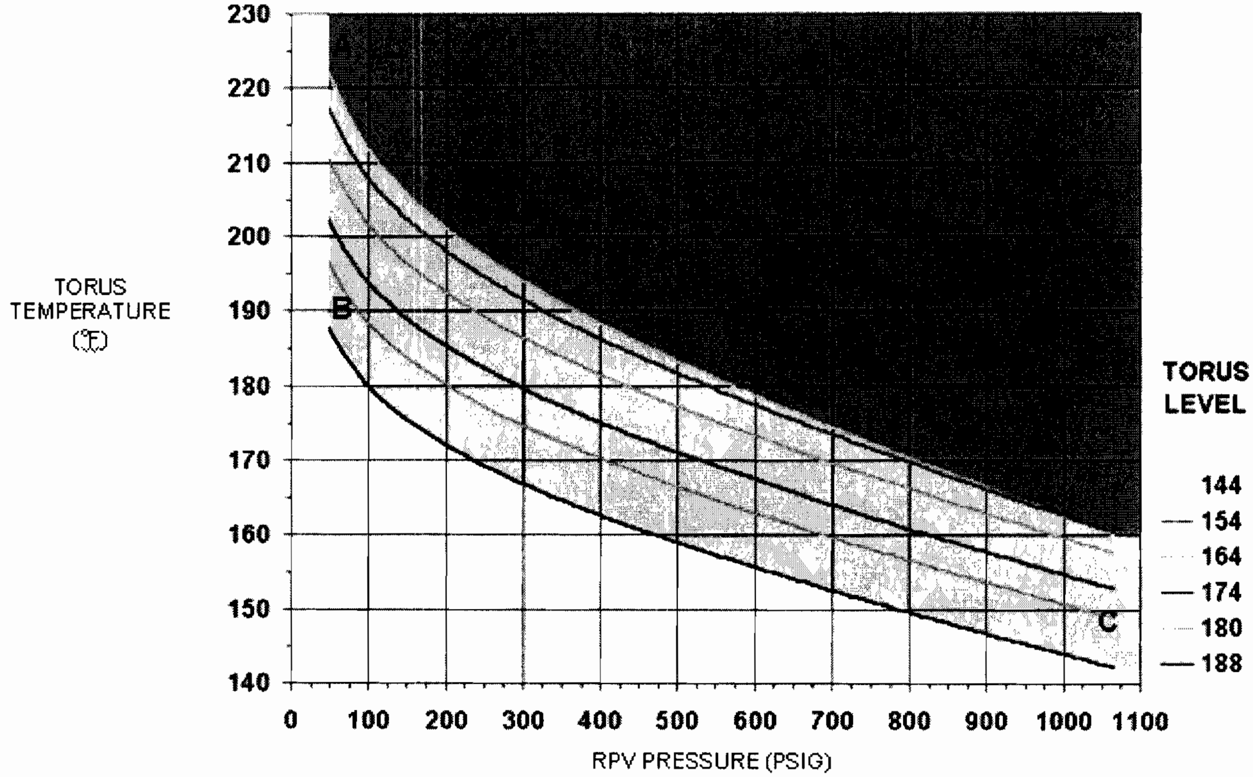
Which of the following shall the SRO direct **next**?



EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

TORUS HIGH LEVEL



TORUS LOAD LIMIT

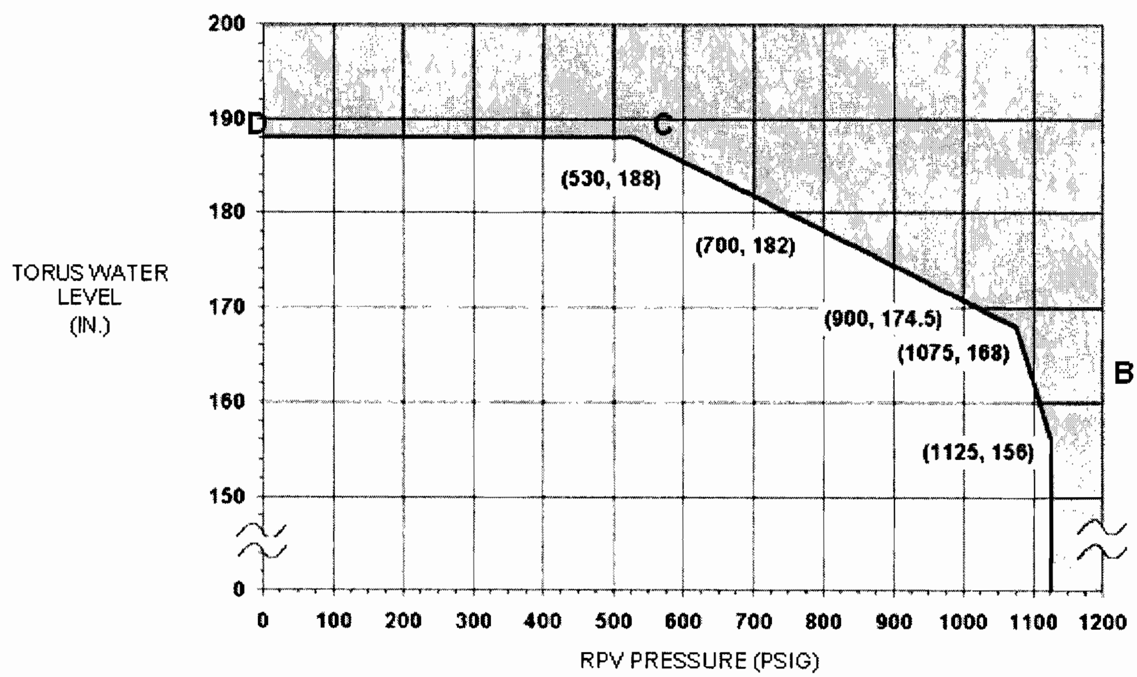
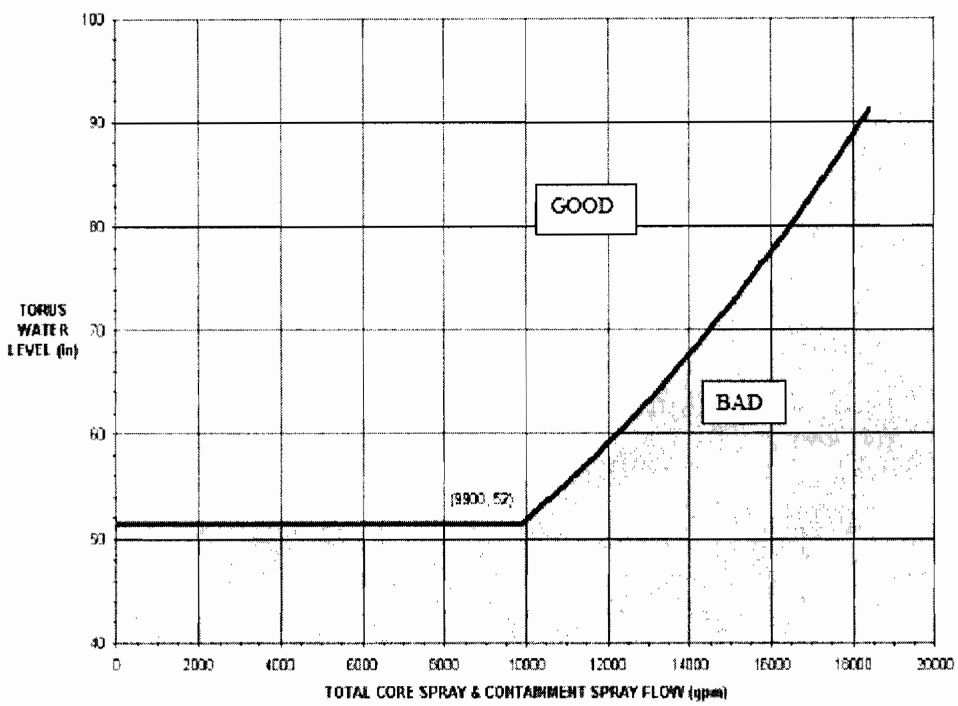


FIGURE A

CORE SPRAY VORTEX LIMIT



A. Due to Vortex limits concerns, terminate Core Spray IAW Support Procedure 4, Operation of the Core Spray System.

B. Due to Heat Capacity Temperature Limit concerns, Emergency Depressurize using the EMRVs IAW the Emergency Depressurization - No ATWS EOP.

C. Due to Torus Load Limit concerns, terminate RPV injection with Condensate and continue injection with Core Spray IAW the Primary Containment Control EOP.

D. Due to Pressure Suppression Pressure limit concerns, anticipate Emergency Depressurization and rapidly depressurize with the Turbine Bypass valves IAW the RPV Control - No ATWS EOP.

Answer: C

Answer Explanation:

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

QID: 09-1 NSRO19		
Question # / Answer	19	Developer/Date: NTP 1/9/10

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Conduct of Operations **2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.**						4.4
Level	SRO	Tier	3	Group		
General References	EOP Users Guide	LP 2621.828.0.0005				
Explanation	The plant was at rated power when a LOCA occurred. Present plant conditions include low RPV water level with 1 condensate pump and Core Spray injecting; Torus high temperature/pressure and high water level. IAW the Primary Containment Control EOP, the Torus Load Level is being approached. At the current RPV pressure, lowering RPV pressure adds no additional margin to TLL. Thus the EOP directs stopping injection into the RPV from sources external to the Primary Containment as long as adequate core cooling is maintained. Condensate is outside the Primary Containment and adequate core cooling is currently assured with water level at 60" and rising. Answer C is correct. Core Spray should be secured IAW Support Procedure 4 if vortex limits are exceeded. Even though no Core Spray flow was provided, it can be seen that the limit is not exceeded. Thus answer A is incorrect. If HCTL is exceeded, the EOP directs ED. But given an RPV pressure of only 45 psig, the EMRVs will not open. Answer B is incorrect. Because an ED limit is being approached, anticipation of ED and rapidly depressurizing the RPV is allowed in the EOPs. Bus since RPV water level is 60", and there is no allowed bypass for MSIV closure on low-low RPV water level (as in the ATWS EOP), the MSIVs have closed on low-low water level are the turbine bypass valves are not available. Answer D is incorrect.						
References to be provided during exam:	None					

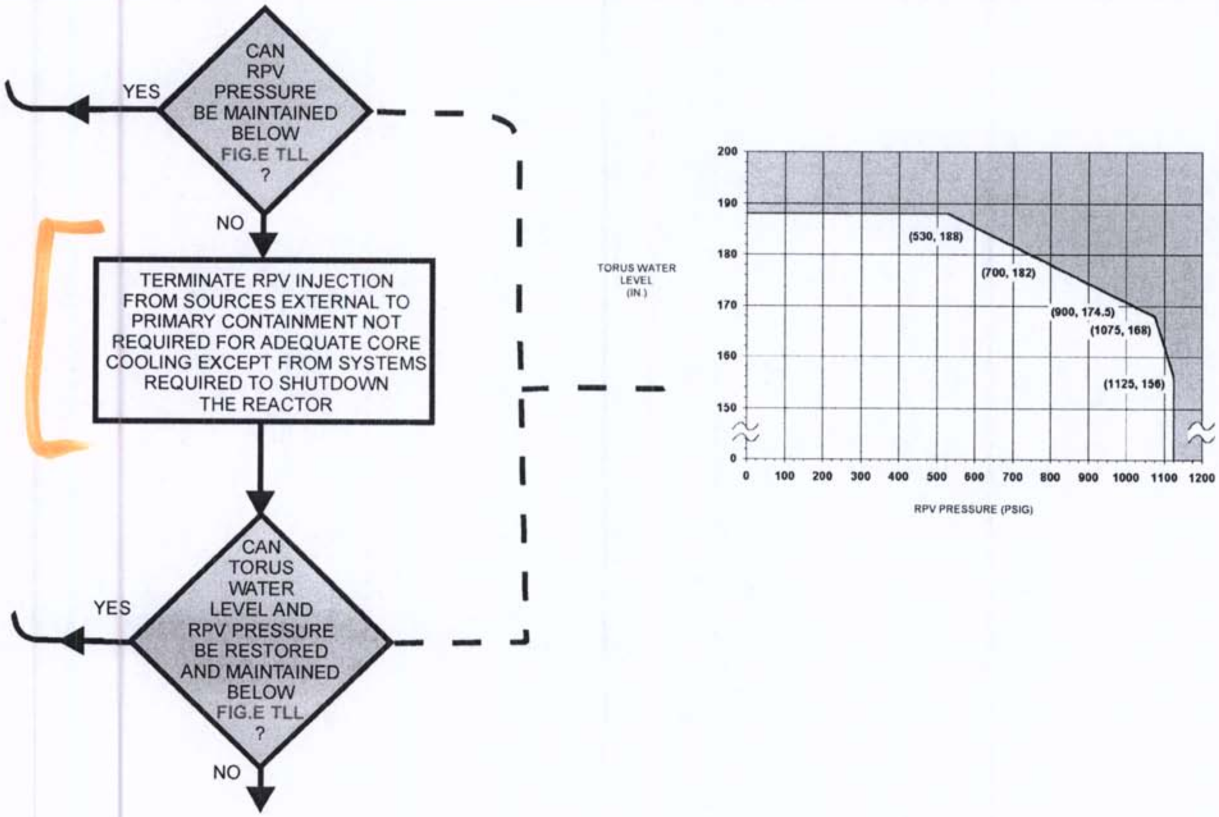
EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Learning Objective	2621.845.0.0056 LO 200-10445

Question Source (New, Modified, Bank)			Modified	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	x 3:SPK
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of procedure			
Time to Complete: 1-2 minutes				

TORUS WATER LEVEL CONTROL



DISCUSSION

A Torus high water level condition can be caused if water being injected into the vessel exits through a break and accumulates in the Torus. If injection from sources external to Primary Containment is terminated, it may prevent this from occurring. However, adequate core cooling is the priority whenever the EOPs are executed, so any actions to terminate injection from those sources being used to assure adequate core cooling is not permitted.

This step includes such actions as transferring RPV level control from the Feedwater/Condensate system to the Core Spray system. The Core Spray system will recirculate water from the Torus, through the RPV and back to the Torus via the break without causing Torus level to increase. Again, such actions are only performed if adequate core cooling can be assured.

Injection from boron injection systems and CRD is not terminated because these systems may be required to establish and maintain the Reactor in a shut down condition.

SPARGER 1 DP HI

CONFIRMATORY ACTIONS:

NOTE
Sparger Dp Alarm may be received during Plant Transients involving a Scram.

- ❑ **VERIFY** pressure differential. (Instrument rack RK04) [ ]

AUTOMATIC ACTIONS:

NONE

MANUAL CORRECTIVE ACTIONS:

- ❑ **NOTIFY** US. [ ]
- ❑ IF instrument reading is greater than or equal to 1 psid,
  THEN **CONSIDER** Core Spray System 1 inoperable. [ ]
- ❑ **VERIFY** operability of System 2. [ ]

CAUTION
Core MAPRAT must be reduced to 0.90 or below within 2 hours.

- ❑ **NOTIFY** Reactor Engineering by referencing the Core Maneuvering Daily Instructions for guidance on rod movement and power changes. [ ]

SPARGER 1 DP HI

CAUSES:	SETPOINTS:	ACTUATING DEVICES:
High-pressure differential across Core Spray System 1 sparger nozzles due to Core Spray line break in the vessel annulus.	0.3 ± 0.3 psid (Max. reset value-1.8 psid)	DPIS RV30A

Reference Drawings:

GE 148F712
GE 885D781
GE 112C2845 Sh. 3
GU 3E-611-17-004 Sh. 2

GE 885D781, ...t 1, Rev 071, CORE SPRAY SYSTEM FLOW DIAGRAM (tif) - Imagenati...

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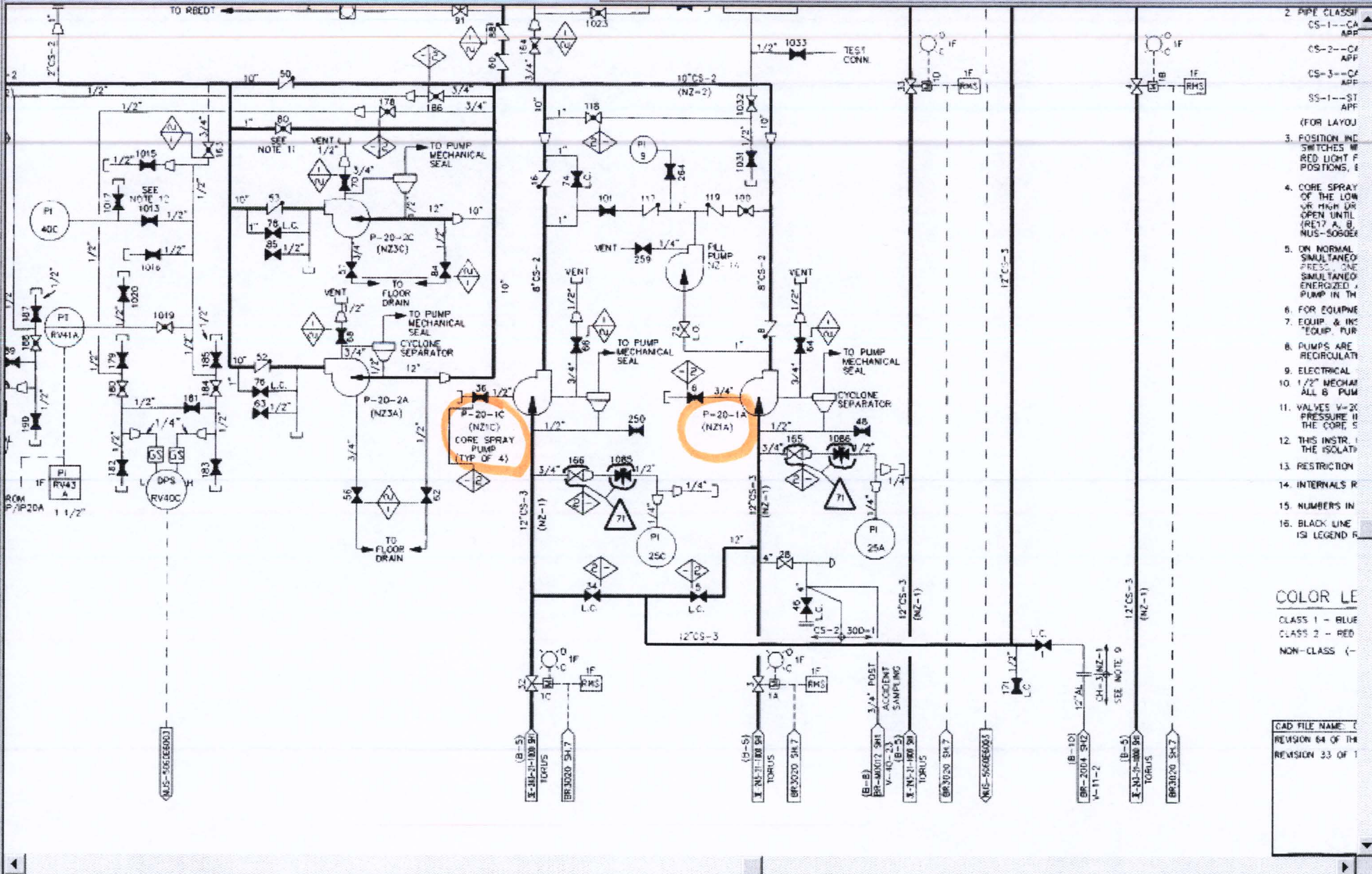
GE 885D781, Sheet 1, Rev 071, CORE SPRAY SYSTEM FLOW DIAGRAM (tif)



Coord: (5852,5528) viewing

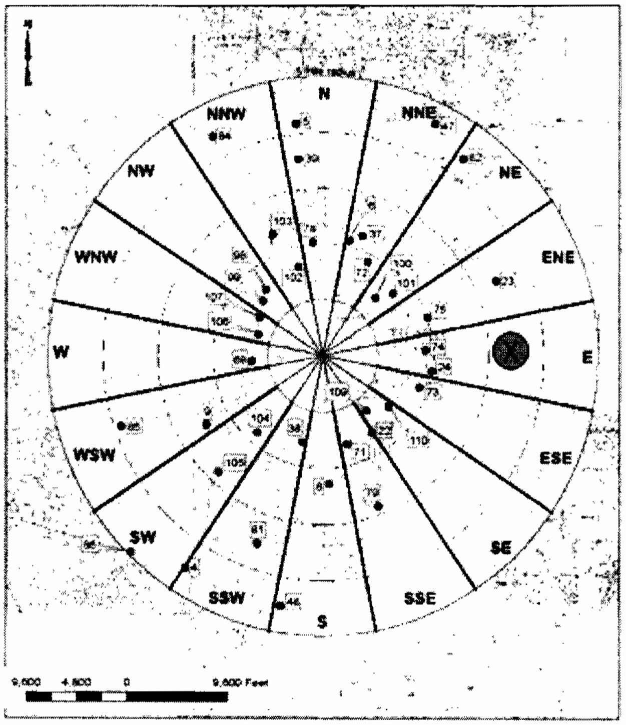
23 **ID: 09-1 NSRO23** **Points: 1.00**

The plant was at power when an event occurred resulting in an offsite radiological release.

IAW Emergency Procedures, which of the following lists the **smallest** release and the wind direction **as indicated on the PPC**, which would require the Shift Emergency Director to **recommend evacuation** of Point X on the attached map?

The EP-AA-1010, Radiological Emergency Plan for Oyster Creek Station, Radiological Effluents thresholds is shown below. (Assume there are no offsite impediments to evacuation.)

Table R1 – Effluent Monitor Thresholds				
	GE	SAE	Alert	UE
Main Stack RAGEMS	4.0 E+01 μCi/cc HRM OR 1.6 E-08 amps HRM	4.0 E+00 μCi/cc HRM OR 1.6 E-09 amps HRM	1.93 E+00 μCi/cc HRM OR 7.8 E-10 amps HRM	7.92 E+03 cps LRM
Turbine Bldg RAGEMS	5.0 E-01 μCi/cc HRM OR 2.0 E-10 amps HRM	2.51 E+05 cpm LRM	8.11 E+04 cpm LRM	8.11 E+02 cpm LRM



5-Mile Radius from Oyster Creek

Note that each radiological release listed below has been at the value indicated for > 15 minutes.

	Release	Wind Direction
A.	Main Stack RAGEMS 6.6 E+01 µCi/cc HRM	90 °
B.	Main Stack RAGEMS 4.3 E+00 µCi/cc HRM	90 °
C.	TB RAGEMS 5.9 E-01 µCi/cc HRM	270 °
D.	TB RAGEMS 2.78 E+05 cpm LRM	270 °

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Answer: C

Answer Explanation:

QID: 09-1 NSRO23		
Question # / Answer	23	Developer/Date: NTP 1/10/10

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Radiation Control 2.3.15 - Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.						3.1
Level	SRO	Tier	3	Group		
General References	EP-AA-1010		EP-AA-112-100-F-01		EP-AA-111-F-10	

Explanation	The plant was at rated power when an event occurred resulting in an offsite release. The question asks what the smallest release rate would require the Shift Emergency Director to recommend evacuation of Point X. Evacuation and sheltering are components of Protective Action recommendations (PARs). Thus, the question asks when a PAR will be recommended. PARs are recommended at the General Emergency level. Point X on the map is in the eastern section. For wind to blow in this section, it must come from a westerly direction. Indicated wind direction is "from", not "to". A wind from 270 degrees would require evacuation of sections ENE, E, & ESE. Point X is contained in the E section. Therefore, a GE must be declared and indicated wind is from the west or 270 degrees. For plant-based PARs, evacuation of a 2-mile radius and 5 miles downwind is required, and Point X is within 5 miles downwind. Answer C is lists a release rate which would require a GE declaration and indicated wind is from the west at 270 degrees. Answer C is correct. Answer A is a GE, but the wind direction is incorrect. Answer A is incorrect. Answer B is a SAE and the incorrect wind direction. Answer B is incorrect. Answer D is a SAE and the correct wind direction. Answer D is incorrect. If the candidate confuses that emergency level which requires PAR determination and the indicated wind direction, all answers are plausible.	
References to be provided during exam:	**None**	
Learning Objective	G-101 DBIG LO G-101 DBIG 01	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning.			
10CRF55 Content	**55.41**		**55.43**	**2**
	(SRO Only) Conditions and limitations in the facility license			

Time to Complete: 1-2 minutes

Exelon. Nuclear

SHIFT EMERGENCY DIRECTOR CHECKLIST

Section 1, Initial Actions
- 1.1, Unusual Event
- 1.2, Alert
- 1.3, Site Area Emergency
- 1.4, General Emergency

Section 2, Ongoing Actions with Command and Control in Control Room

Section 3, Ongoing Actions after Transfer of Command and Control

Section 4, Closeout Actions

NOTES: Steps in each section of this checklist may be performed in an order other than listed or they may be omitted if not applicable

The Shift Emergency Director may delegate plant announcements, call out of the ERO and actual communications with offsite agencies once review and approval of notification information has been made.

GE Step	SAE Step	Alert Step	UE Step	IMMEDIATE ACTIONS TABLE (Control Room in C & C)
1.4.A	1.3.A	1.2.A	1.1.A	Announce the classification
1.4.D	1.3.D	1.2.D	1.1.E	Initiate Emergency PA for classification (within 15 minutes of classification)
1.4.E	1.3.E	1.2.E	1.1.C or F **or** G	Notification or Activation of the Emergency Response Organization (ERO)
1.4.F	N/A	N/A	N/A	Determine the correct PAR per station PAR flowchart
1.4.G	1.3.F	1.2.F	1.1.H	Initiate State/Local notification (within 15 minutes of classification)
1.4.J	1.3.I	1.2.G	1.1.I	Initiate ENS notification (within 60 minutes of classification)
1.7	1.7	1.7	N/A	Activate ERDS (within 60 minutes of an Alert classification)
1.4.I	1.3.H	2.3 (Opt)	2.3 (Opt)	Initiate Personnel Accountability
2.6	2.6	2.6	2.6	Perform "Quick Assessment" (if release in progress)
1.9	1.9	1.9	1.9	Significant Events Reporting (OP-AA-106-101)
2.7	2.7	2.7	2.7	Emergency Exposure Controls (KI, exposure extensions)

(Opt) – Indicates that this action is optional at this classification level

Table OCGS 3-1: Emergency Action Level (EAL) Matrix Modes: 1 - Power Ops 2 - Hot Shutdown (≥ 212 °F) 3 - Cold Shutdown (< 212 °F) 4 - Refuel D - Defuel

Abnormal Rad Levels / Radiological Effluent

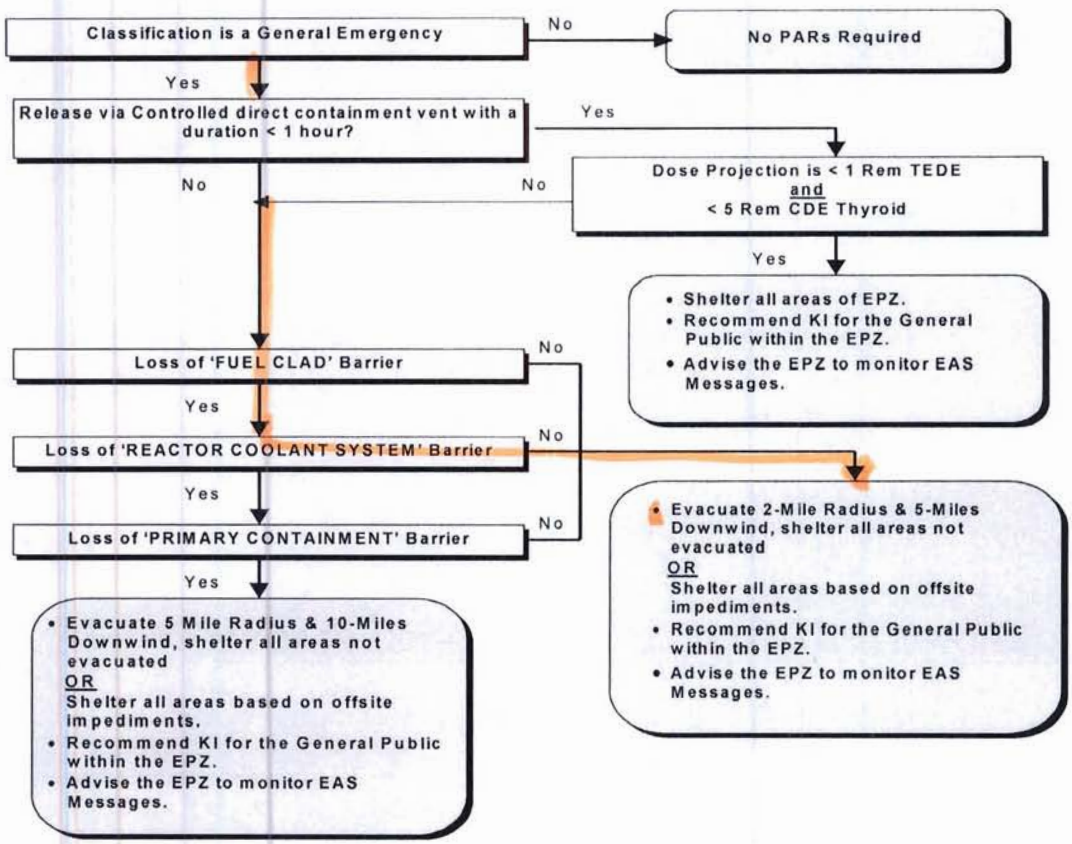
Radiological Effluent

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
RG1 Offsite Dose Resulting from An Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. 1 2 3 4 D	**RS1** Offsite Dose Resulting from An Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. 1 2 3 4 D	**RA1** Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer. 1 2 3 4 D	**RU1** Any UNPLANNED Release Of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. 1 2 3 4 D
EAL Threshold Values: NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. VALID reading on one or more of the Table R1 radiation monitors that exceeds or is expected to exceed the reading shown (Table R1) for **≥ 15 minutes.** OR 2. Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**: a. **> 1000 mRem TEDE** OR b. **> 5000 mRem CDE Thyroid** OR 3. Field survey results at or beyond Site Boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour. OR b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.	**EAL Threshold Values:** NOTE: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do not delay declaration awaiting dose assessment results. 1. VALID reading on one or more of the Table R1 radiation monitors that exceeds or is expected to exceed the reading shown (Table R1) for **≥ 15 minutes.** OR 2. Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**: a. **> 100 mRem TEDE** OR b. **> 500 mRem CDE Thyroid** OR 3 Field survey results at or beyond Site Boundary indicate **EITHER**: a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour. OR b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.	**EAL Threshold Values:** 1. VALID reading on any of the following effluent monitors **> 200 times alarm setpoint** established by a current radioactivity discharge permit for **≥ 15 minutes**: • Radwaste Overboard Discharge effluent monitor • Discharge Permit specified monitor OR 2. VALID reading on one or more of the Table R1 radiation monitors that exceeds the Table R1 values for **≥ 15 minutes.** OR 3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes.**	**EAL Threshold Values:** 1. VALID reading on any of the following effluent monitors **> 2 times alarm setpoint** established by a current radioactivity discharge permit for **≥ 60 minutes**: • Radwaste Overboard Discharge effluent monitor • Discharge Permit specified monitor OR 2. VALID reading on one or more of the Table R1 radiation monitors that exceeds the Table R1 values **≥ 60 minutes.** OR 3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes.**

Table R1 -- Effluent Monitor Thresholds

	General Emergency	Site Area Emergency	Alert	Unusual Event
Main Stack RAGEMS	4.0 E+01 µCi/cc HRM **OR** 1.6 E-08 amps HRM	4.0 E+00 µCi/cc HRM **OR** 1.6 E-09 amps HRM	1.93 E+00 µCi/cc HRM **OR** 7.8 E-10 amps HRM	7.92 E+03 cps LRM
Turbine Bldg RAGEMS	5.0 E-01 µCi/cc HRM **OR** 2.0 E-10 amps HRM	2.51 E+05 cpm LRM	8.11 E+04 cpm LRM	8.11 E+02 cpm LRM

OYSTER CREEK PLANT BASED PAR FLOWCHART



Date:___________ **Time:**___________

NOTE:	**ENSURE** dose based PARs are **EVALUATED** when a release is in progress and **EVALUATE** for a potential Sea Breeze affect.

WIND DIRECTION FROM		AFFECTED DOWNWIND SECTORS	WIND DIRECTION FROM		AFFECTED DOWNWIND SECTORS
N	350 to 11	SSW / **S** / SSE	S	170 to 191	NNW / **N** / NNE
NNE	12 to 34	S / **SSW** / SW	SSW	192 to 214	N / **NNE** / NE
NE	35 to 56	SSW / **SW** / WSW	SW	215 to 237	NNE / **NE** / ENE
ENE	57 to 79	SW / **WSW** / W	WSW	238 to 259	NE / **ENE** / E
E	80 to 101	WSW / **W** / WNW	W	260 to 281	ENE / **E** / ESE
ESE	102 to 124	W / **WNW** / NW	WNW	282 to 304	E / **ESE** / SE
SE	125 to 146	WNW / **NW** / NNW	NW	305 to 326	ESE / **SE** / SSE
SSE	147 to 169	NW / **NNW** / N	NNW	327 to 349	SE / **SSE** / S

24 **ID: 09-1 NSRO24** **Points: 1.00**

The plant was in a refuel outage. Due to the loss of SRM 24, fuel was being shuffled in core quadrants 1, 2, and 3.

While reviewing work packages for the following day, you note a maintenance activity requiring a tagout to de-energize 24 VDC Power Panel A.

If the maintenance activity were allowed to occur as scheduled, which of the following states the impact on refueling, if any?

A. There will be **no** impact on the core alterations.

B. Core alterations will be restricted to core quadrant 3 **only**.

C. **All** core alterations must cease due to the loss of the required number of operable SRMs.

D. **All** core alterations must cease due to the loss of Secondary Containment Integrity and the auto start of SGTS.

Answer: C

Answer Explanation:

QID: 09-1 NSRO24		
Question # / Answer	24	Developer/Date: NTP 1/11/10

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Equipment Control 2.2.36 - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.						4.2
Level	SRO	Tier	3	Group		
General References	TS 3.9		GE 706E812 sh. 3, 5, 6			

24 **ID: 09-1 NSRO24** **Points: 1.00**

The plant was in a refuel outage. Due to the loss of SRM 24, fuel was being shuffled in core quadrants 1, 2, and 3.

While reviewing work packages for the following day, you note a maintenance activity requiring a tagout to de-energize 24 VDC Power Panel A.

If the maintenance activity were allowed to occur as scheduled, which of the following states the impact on refueling, if any?

A. There will be **no** impact on the core alterations.

B. Core alterations will be restricted to core quadrant 3 **only**.

C. **All** core alterations must cease due to the loss of the required number of operable SRMs.

D. **All** core alterations must cease due to the loss of Secondary Containment Integrity and the auto start of SGTS.

Answer: C

Answer Explanation:

QID: 09-1 NSRO24		
Question # / Answer	**24**	**Developer/Date: NTP 1/11/10**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
Equipment Control **2.2.36 - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.**					**4.2**
Level	**SRO**	**Tier**	**3**	**Group**	
General References	**TS 3.9**	**GE 706E812 sh. 3, 5, 6**			

Explanation	The plant is in a refuel outage with SRM 24 inoperable. Core alterations are occurring in the other core quadrants with operable SRMs. If 24 VDC Power Panel A is de-energized, this will render SRMs 21 and 22 inoperable. TS 3.9.D provides the following: During CORE ALTERATIONS at least two (2) source range monitor (SRM) channels shall be OPERABLE and inserted to the normal operating level. One of the OPERABLE SRM channel detectors shall be located in the core quadrant where CORE ALTERATIONS are being performed, and another shall be located in an adjacent quadrant. TS 3.9.G provides the following: With any of the above requirements not met, cease CORE ALTERATIONS or control rod removal as appropriate, and initiate action to satisfy the above requirements. Since only 1 SRM remains operable in quadrant 3, the requirement for 2 operable SRMs will not be met and core alterations must cease. Answer C is correct. Since the refuel activities are impacted, answer A is incorrect. Since SRM 23, in core quadrant 3 is still operable, the candidate may think that fuel moves are still allowed in that single quadrant. But as shown, 2 SRMs are required. Answer B is incorrect. The loss of 24 VDC Power Panel will isolate RB normal Vent and initiate the Standby Gas treatment System (SGTS). This will not cause SGTS or Secondary Containment to be inoperable. Answer D is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0029 LO 215-10451		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**		**55.43**	**2**
	(SRO Only) Facility operating limitations in the tech Specs and the bases			
Time to Complete: 1-2 minutes				

3.9 REFUELING

Applicability: Applies to fuel handling operations during refueling.

Objective: To assure that criticality does not occur during refueling.

Specification:

A. Fuel shall not be loaded into a reactor core cell unless the control rod in that core cell is fully inserted.

B. During CORE ALTERATIONS the reactor mode switch shall be locked in the refuel position.

C. The refueling interlocks shall be OPERABLE with the fuel grapple hoist loaded switch set at ≤485 lb. during the fuel handling operations with the head off the reactor vessel. If the frame-mounted auxiliary hoist, the trolley-mounted auxiliary hoist or the service platform hoist is to be used for handling fuel with the head off the reactor vessel the load limit switch on the hoist to be used shall be set at ≤400 lb.

   Fuel Handling operations with the head off the reactor vessel can be performed with the refueling interlocks inoperable provided all the following specifications are satisfied:

   1. All control rods are verified to be fully inserted.

   2. Control rod withdrawal has been disabled.

D. During CORE ALTERATIONS at least two (2) source range monitor (SRM) channels shall be OPERABLE and inserted to the normal operating level. One of the OPERABLE SRM channel detectors shall be located in the core quadrant where CORE ALTERATIONS are being performed, and another shall be located in an adjacent quadrant.

E. Removal of one control rod or rod drive mechanism may be performed provided that all the following specifications are satisfied.

   1. The reactor mode switch is locked in the refuel position.

   2. At least two (2) sources range monitor (SRM) channels shall be OPERABLE and inserted to the normal operation level. One of the OPERABLE SRM channel detectors shall be located in the core quadrant where the control rod is being removed and one shall be located in an adjacent quadrant.

F. Removal of any number of control rods or rod drive mechanisms may be performed provided all the following specifications are satisfied:

   1. The reactor mode switch is locked in the refuel position and all refueling interlocks are OPERABLE as required in Specification 3.9.C. The refueling interlocks associated with the control rods being withdrawn may be bypassed as required after the fuel assemblies have been removed from the core cell surrounding the control rods as specified in 4, below.

   2. At least two (2) source range monitor (SRM) channels shall be OPERABLE and inserted to the normal operation level. One of the OPERABLE SRM channel detectors shall be located in the core quadrant where a control rod is

being removed and one shall be located in an adjacent quadrant.

3. All other control rods are fully inserted with the exception of one rod which may be partially withdrawn not more than two notches to perform refueling interlock surveillance.

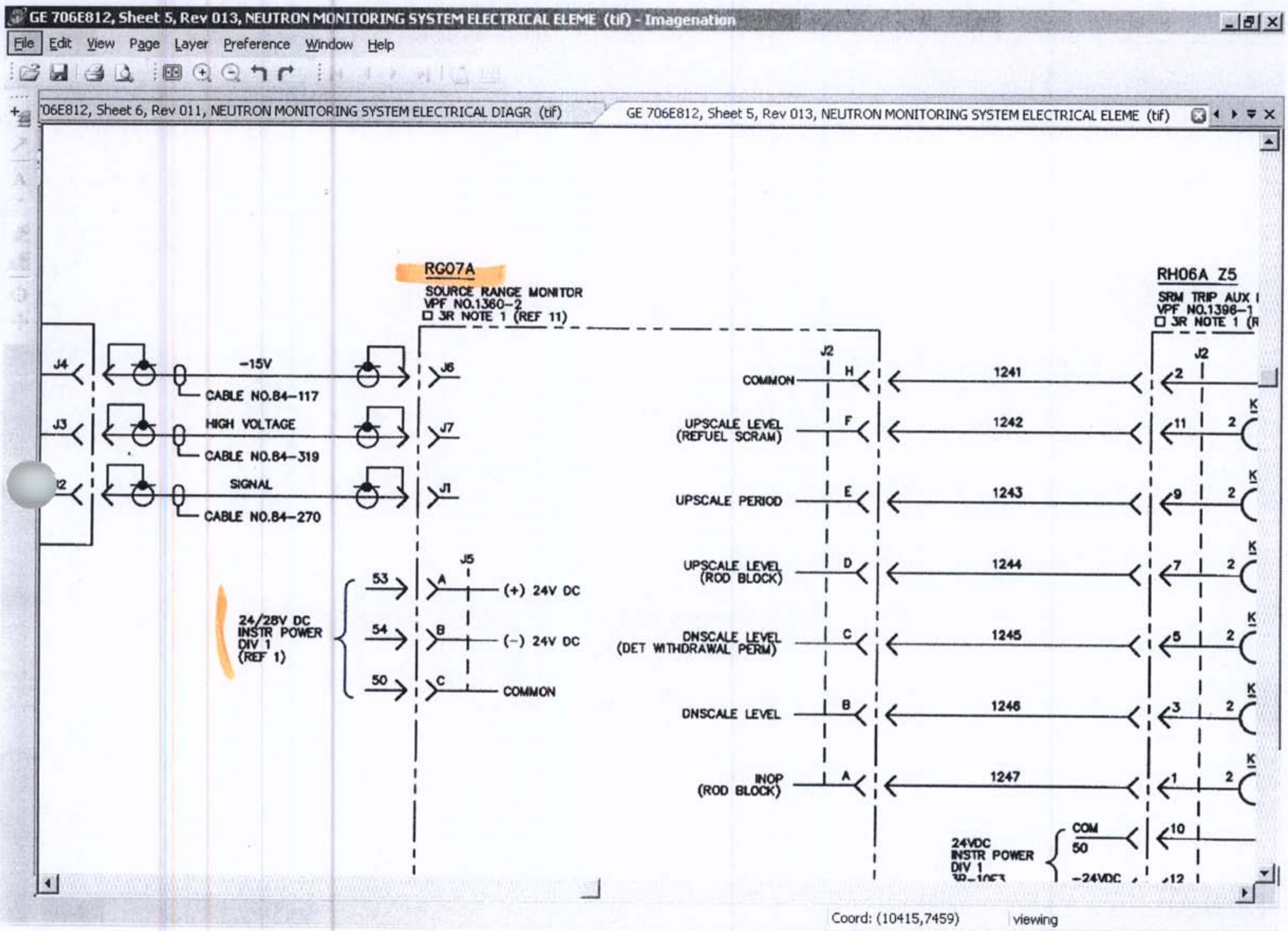
4. The four fuel assemblies are removed from the core cell surrounding each control rod or rod drive mechanism to be removed.

5. The SHUTDOWN MARGIN requirements of Specification 3.2.A are met.

6. An evaluation will be conducted for each refuel/reload to ensure that actual core criticality of the proposed order of defueling and refueling is bounded by previous analysis performed to support such defueling and refueling activities, otherwise a new analysis shall be performed.

   The new analysis must show that sufficient conservatism exists for the proposed order of defueling and refueling before such operation shall be allowed to proceed.

G. With any of the above requirements not met, cease CORE ALTERATIONS or control rod removal as appropriate, and initiate action to satisfy the above requirements.


GE 706E812, Sheet 5, Rev 013, NEUTRON MONITORING SYSTEM ELECTRICAL ELEME (tif) - Imagenation
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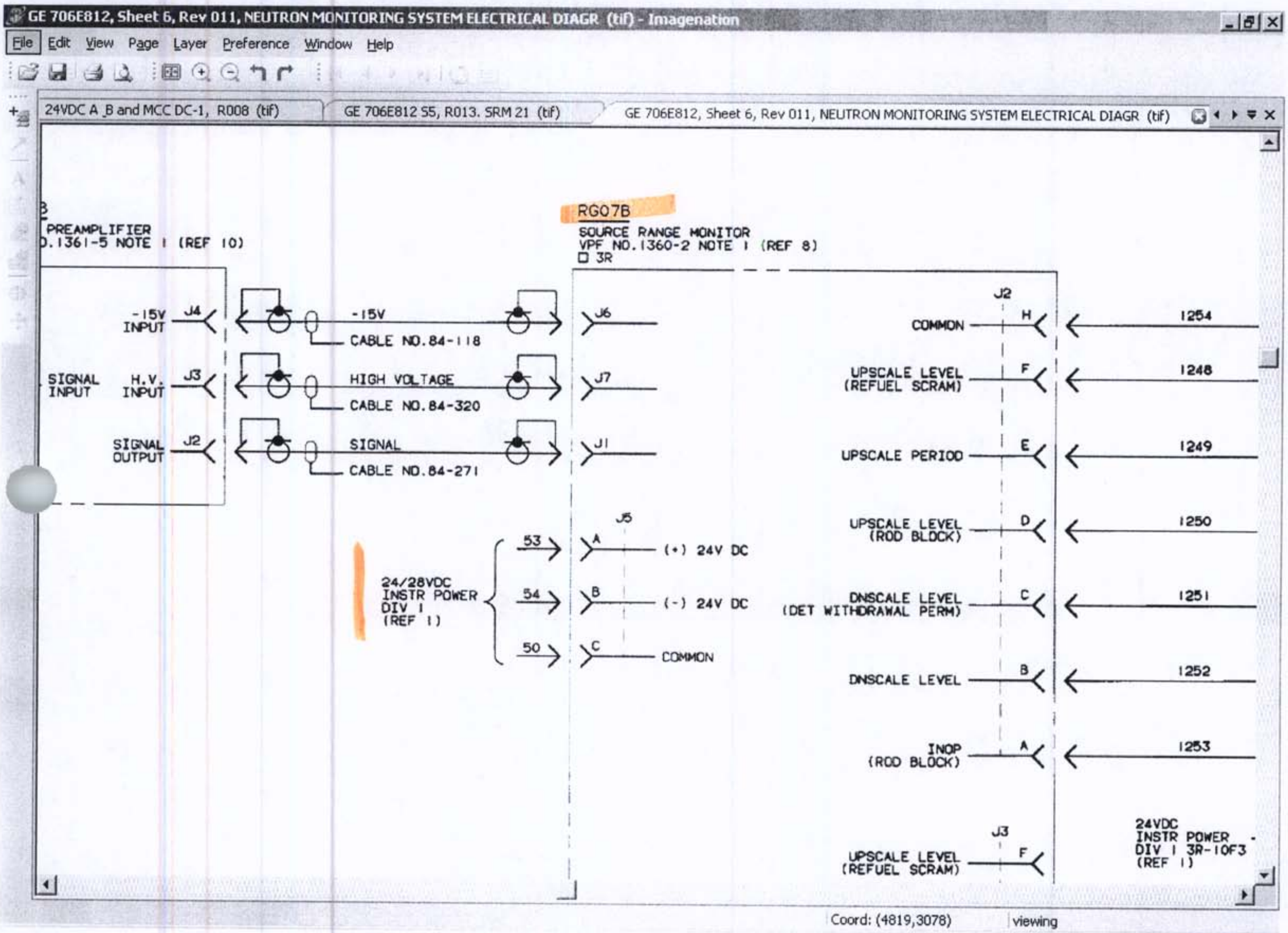
706E812, Sheet 6, Rev 011, NEUTRON MONITORING SYSTEM ELECTRICAL DIAGR (tif)
GE 706E812, Sheet 5, Rev 013, NEUTRON MONITORING SYSTEM ELECTRICAL ELEME (tif)
RG07A
SOURCE RANGE MONITOR
VPF NO.1360-2
3R NOTE 1 (REF 11)
RH06A Z5
SRM TRIP AUX
VPF NO.1396-1
3R NOTE 1
J4
J3
-15V
CABLE NO.84-117
HIGH VOLTAGE
CABLE NO.84-319
SIGNAL
CABLE NO.84-270
J6
J7
J1
J5
J2
24/28V DC
INSTR POWER
DIV 1
(REF 1)
53
54
50
(+) 24V DC
(-) 24V DC
COMMON
COMMON
UPSCALE LEVEL
(REFUEL SCRAM)
UPSCALE PERIOD
UPSCALE LEVEL
(ROD BLOCK)
DNSCALE LEVEL
(DET WITHDRAWAL PERM)
DNSCALE LEVEL
INOP
(ROD BLOCK)
1241
1242
1243
1244
1245
1246
1247
24VDC
INSTR POWER
DIV 1
COM
50
-24VDC
Coord: (10415,7459)
viewing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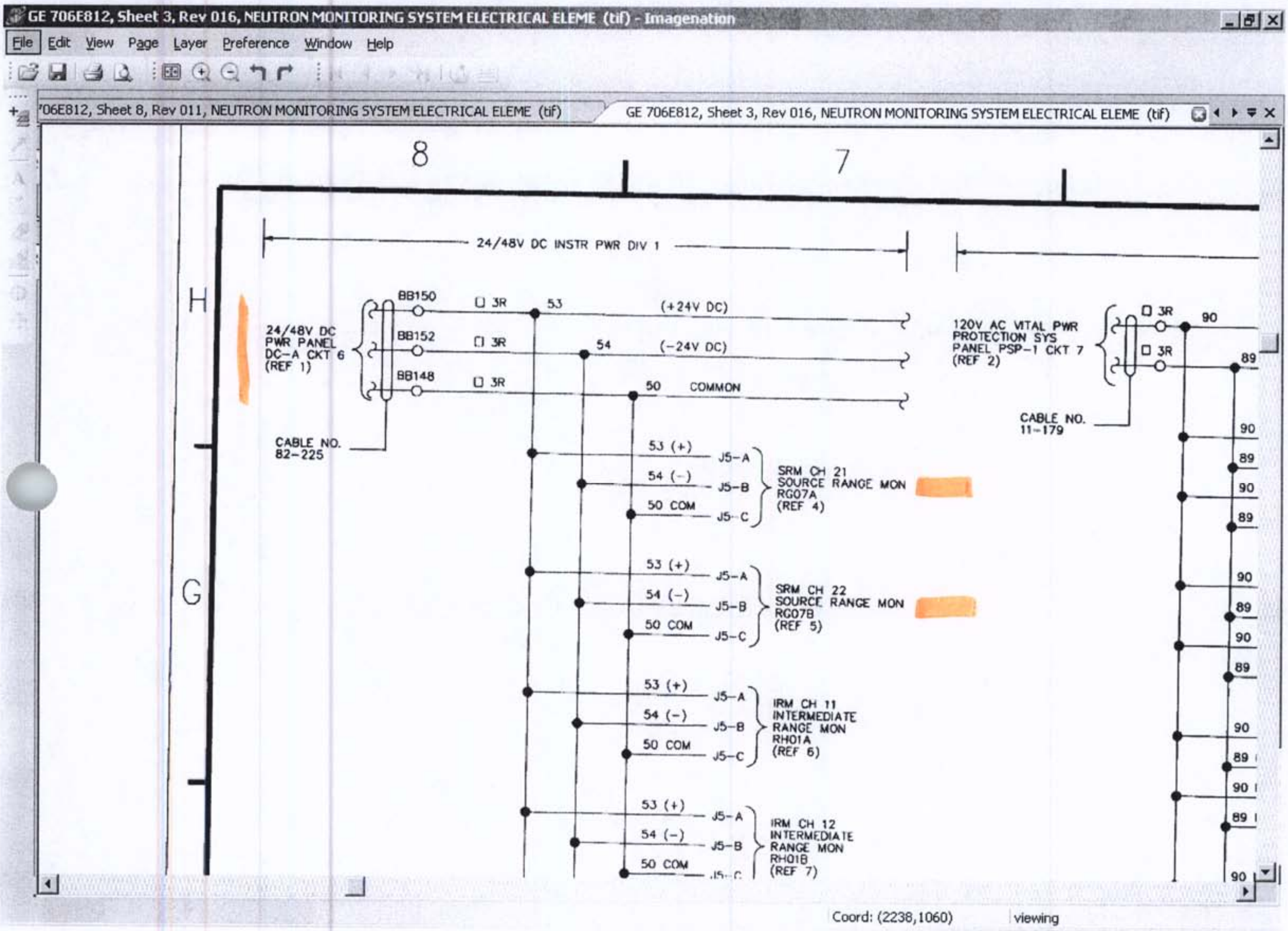
24VDC A B and MCC DC-1, R008 (tif)
GE 706E812 S5, R013. SRM 21 (tif)
GE 706E812, Sheet 6, Rev 011, NEUTRON MONITORING SYSTEM ELECTRICAL DIAGR (tif)
PREAMPLIFIER
D.1361-5 NOTE 1 (REF 10)
RG07B
SOURCE RANGE MONITOR
VPF NO.1360-2 NOTE 1 (REF 8)
3R
-15V INPUT
J4
-15V
CABLE NO.84-118
J6
SIGNAL INPUT
H.V. INPUT
J3
HIGH VOLTAGE
CABLE NO.84-320
J7
SIGNAL OUTPUT
J2
SIGNAL
CABLE NO.84-271
J1
J5
24/28VDC INSTR POWER DIV 1 (REF 1)
53
54
50
A
B
C
(+) 24V DC
(-) 24V DC
COMMON
J2
COMMON
H
1254
UPSCALE LEVEL (REFUEL SCRAM)
F
1248
UPSCALE PERIOD
E
1249
UPSCALE LEVEL (ROD BLOCK)
D
1250
DNSCALE LEVEL (DET WITHDRAWAL PERM)
C
1251
DNSCALE LEVEL
B
1252
INOP (ROD BLOCK)
A
1253
J3
UPSCALE LEVEL (REFUEL SCRAM)
F
24VDC INSTR POWER DIV 1 3R-1OF3 (REF 1)
Coord: (4819,3078)
viewing



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GE 706E812, Sheet 3, Rev 016, NEUTRON MONITORING SYSTEM ELECTRICAL ELEME (tif)
8
7
H
G
24/48V DC INSTR PWR DIV 1
24/48V DC PWR PANEL DC-A CKT 6 (REF 1)
BB150
BB152
BB148
3R
53
54
50
(+24V DC)
(-24V DC)
COMMON
CABLE NO. 82-225
120V AC VITAL PWR PROTECTION SYS PANEL PSP-1 CKT 7 (REF 2)
CABLE NO. 11-179
90
89
53 (+)
54 (-)
50 COM
J5-A
J5-B
J5-C
SRM CH 21 SOURCE RANGE MON RG07A (REF 4)
SRM CH 22 SOURCE RANGE MON RG07B (REF 5)
IRM CH 11 INTERMEDIATE RANGE MON RH01A (REF 6)
IRM CH 12 INTERMEDIATE RANGE MON RH01B (REF 7)
Coord: (2238,1060)
viewing

25 **ID: 09-1 NSRO25** **Points: 1.00**

The plant was at rated power when the Shift Manager declared an Unusual Event, EAL HU6, due to a fire.

15 minutes later, the Shift Emergency Director declared an Alert, EAL HA6, due to worsening conditions from the fire.

IAW EP-AA-112-100-F-01, Shift Emergency Director Checklist, which of the following is required at the **new** EAL which is **not** required at the original EAL?

A. Initiation of a site evacuation.

B. Notification of state/local authorities within 15 minutes.

C. Activation of the Emergency Response Organization (ERO).

D. Determination of Protective Action Recommendations (PARs).

Answer: C

Answer Explanation:

QID: 09-1 ASRO25		
Question # / Answer	**25 C**	**Developer/Date: NTP 8/27/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
2.4.29 Knowledge of the emergency plan						**4.4**
Level	**SRO**	**Tier**	**3**	**Group**		
General References	**EP-AA112-100-F-01**	**EP-AA-1010**				

Explanation	The question shows that an Unusual Event emergency was declared and then upgraded to an Alert. IAW the Shift Emergency Director Checklist, activation of the emergency response organization (ERO) is required at the alert level or higher (except a security event). Answer C is correct. Initiation of a site evacuation is performed at the Site Area Emergency or higher. Answer A is incorrect. Notification of state/local authorities is performed at all emergency levels. Answer C is incorrect. PARs are only recommended at the General Emergency level. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	G-101 DBIG LO G-101 DBIG 01	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**		**55.43**	**1**
	(SRO Only) Conditions and limitations in the facility license			
Time to Complete: 1-2 minutes				

SHIFT EMERGENCY DIRECTOR CHECKLIST

Section 1, Initial Actions
- 1.1, Unusual Event
- 1.2, Alert
- 1.3, Site Area Emergency
- 1.4, General Emergency

Section 2, Ongoing Actions with Command and Control in Control Room
Section 3, Ongoing Actions after Transfer of Command and Control
Section 4, Closeout Actions

NOTES: Steps in each section of this checklist may be performed in an order other than listed or they may be omitted if not applicable

The Shift Emergency Director may delegate plant announcements, call out of the ERO and actual communications with offsite agencies once review and approval of notification information has been made.

GE Step	SAE Step	Alert Step	UE Step	IMMEDIATE ACTIONS TABLE (Control Room in C & C)
1.4.A	1.3.A	1.2.A	1.1.A	Announce the classification
1.4.D	1.3.D	1.2.D	1.1.E	Initiate Emergency PA for classification (within 15 minutes of classification)
1.4.E	1.3.E	1.2.E	1.1.C or F **or** G	Notification or Activation of the Emergency Response Organization (ERO)
1.4.F	N/A	N/A	N/A	Determine the correct PAR per station PAR flowchart
1.4.G	1.3.F	1.2.F	1.1.H	Initiate State/Local notification (within 15 minutes of classification)
1.4.J	1.3.I	1.2.G	1.1.I	Initiate ENS notification (within 60 minutes of classification)
1.7	1.7	1.7	N/A	Activate ERDS (within 60 minutes of an Alert classification)
1.4.I	1.3.H	2.3 (Opt)	2.3 (Opt)	Initiate Personnel Accountability
2.6	2.6	2.6	2.6	Perform "Quick Assessment" (if release in progress)
1.9	1.9	1.9	1.9	Significant Events Reporting (OP-AA-106-101)
2.7	2.7	2.7	2.7	Emergency Exposure Controls (KI, exposure extensions)

(Opt) – Indicates that this action is optional at this classification level

SHIFT EMERGENCY DIRECTOR CHECKLIST

1. **INITIAL ACTIONS**

1.1. **If** the event is classified as an **UNUSUAL EVENT then**, **PERFORM** the following:	
A. **ANNOUNCE** the event classification to the Control Room staff.	❑
B. **RECORD** the EAL and declaration threshold(s) (as applicable).	EAL
	EAL Threshold(s) (as applicable)
NOTE: ERO activation is optional for non-security threat Unusual Event classifications.	
C. **IF** the event is a Security Event, Unusual Event, **THEN**, **ACTIVATE** the ERO.	❑ **Tab 2**
D. **USE** site-specific Operations/Security procedures for announcements for Security events **AND CONSIDER** limitations on personnel movement prior to sounding alarms or making announcements.	❑
E. **USE** the Emergency Public Address Announcements form to select and direct the appropriate public address announcement for an Unusual Event within 15 minutes of event classification.	❑ **Tab 1**
Time:	________
F. **IF** optional facility staffing is called for, then **PERFORM** the "ERO Response Required" steps of the ERO Notification or Augmentation form and **GO TO** 1.1.H, **OTHERWISE GO TO** next step	❑ **Tab 2**
Time:	________
G. **PERFORM** the "Management Notification Only" steps of the ERO Notification or Augmentation form.	❑ **Tab 2**
Time:	________