Draft Submittal

VOGTLE MAY 2005 EXAM
50-424, 425/2005-301

MAY 17 - 25, 2005
MAY 27, 2005 (WRITTEN)

1. Senior Reactor Operator Written Exam

VOGTLE

INITIAL LICENSE EXAM

2005-301

SENIOR REACTOR OPERATOR QUESTIONS

Exam Date: May 2005

Draft

Facility: Vogtle Date of Exam: May 2005

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 & Abnormal Plant Evolutions	1	3	1	5				3	3			3	18	3	3	6
	2	2	4	1		N/A		0	0	N/A		2	9	1	3	4
	Tier Totals	5	5	6				3	3			5	27	4	6	10
2. Plant Systems	1	4	3	2	3	2	1	3	3	2	4	1	28	2	3	5
	2	1	1	1	1	0	2	1	1	1	0	1	10	2	1	3
	Tier Totals	5	4	3	4	2	3	4	4	3	4	2	38	4	4	8
3. Generic Knowledge and Abilities Categories					1		2		3		4			1 2 3 4		
					3		3		2		2		10	1 2 1 3		7

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; operability-important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.

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

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for the RO and SRO-only portions, respectively, unless the entire exam will be administered only to SRO applicants, in which case the SRO ratings should be used throughout.

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.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance rating (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. 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, importance ratings, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

ES-401 PWR Examination Outline Form ES-401-2
Vogtle 2005-301 Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO-R / **SRO-S**)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000007 (BW/E02&E10; CE/E02) Reactor Trip - Stabilization - Recovery / 1							Not selected.		
000008 Pressurizer Vapor Space Accident / 3			R		S		R 008AK3.05 Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: ECCS termination or throttling criteria. **S 008AA2.22 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Consequences of loss of pressure in RCS; methods for evaluating pressure loss.**	4.0 **4.2**	1 **1**
000009 Small Break LOCA / 3				R			R 009EA1.15 Ability to operate and monitor the following as they apply to a small break LOCA: PORV and PORV Block Valve.	3.9	1
000011 Large Break LOCA / 3					R S		R 011EA2.07 Ability to determine or interpret the following as they apply to a Large Break LOCA: That equipment necessary for functioning of critical pump water seals is operable. **S 011EA2.08 Ability to determine or interpret the following as they apply to a Large Break LOCA: Conditions necessary for recovery when accident reaches stable phase.**	3.2 **3.9**	1 **1**
000015/17 RCP Malfunctions / 4						R	R 015/017G2.4.10 Knowledge of annunciator response procedures.	3.0	1
000022 Loss of Rx Coolant Makeup / 2						S	**S 022G2.1.12 Ability to apply technical specifications for a system.**	**4.0**	**1**
000025 Loss of RHR System / 4	R						R 025AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation.	3.9	1
000026 Loss of Component Cooling Water / 8			R			S	R 026AK3.02 Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS. **S 026G2.4.48 Ability to interpret control room indications to verify the status and operation of the system, and understand how operator actions and directives affect plant and system conditions.**	3.6 **3.8**	1 **1**
000027 Pressurizer Pressure Control System Malfunction / 3			R				R 027AK3.03 Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Actions contained in EOP for PZR PCS malfunction.	3.7	1
000029 ATWS / 1					S		**S 029EA2.05 Ability to determine or interpret the following as they apply to an ATWS: System component valve position indications.**	**3.4**	**1**
000038 Steam Gen. Tube Rupture / 3					R	S	R 038EA2.17 Ability to determine or interpret the following as they apply to a SGTR: RCP restart criteria. **S 038G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.**	3.8 **3.6**	1 **1**
000040 (BW/E05; CE/E05; W/E12) Steam Line Rupture - Excessive Heat Transfer / 4						R	R W/E12 G2.1.8 Uncontrolled Depressurization of all Steam Generators - Ability to coordinate personnel activities outside the control room.	3.8	1
000054 (CE/E06) Loss of Main Feedwater / 4			R				R 054AK3.02 Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Matching of feedwater and steam flows.	3.4	1
000055 Station Blackout / 6	R						R 055EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural Circulation Cooling.	4.1	1
000056 Loss of Off-site Power / 6			R				R 056AK3.02 Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Actions contained in EOP for loss of offsite power.	4.4	1

	K1	K2	K3	A1	A2	G			
000057 Loss of Vital AC Inst. Bus / 6				R			R 057AA1.04 Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus: RWST and VCT valves.	3.5	1
000058 Loss of DC Power / 6					R		R 058AA2.02 Ability to determine and interpret the following as they apply to Loss of DC Power: 125 V dc bus voltage, low/critical low, alarm.	3.3	1
000062 Loss of Nuclear Svc Water / 4						R	R 062G2.1.32 Ability to explain and apply all system limits and precautions.	3.4	1
000065 Loss of Instrument Air / 8				R			R 065AA1.03 Ability to operate and/or monitor the following as they apply to Loss of Instrument Air: Restoration of systems served by instrument air when pressure is regained.	2.9	1
W/E04 LOCA Outside Containment / 3	R						R W/E04EK1.1 Knowledge of the operational implications of the following concepts as they apply to the (LOCA Outside Containment): Components, capacity, and function of emergency systems.	3.5	1
W/E11 Loss of Emergency Coolant Recirc. / 4		R					R W/E11EK2.2 Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	3.9	1
BW/E04; W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4							Not selected.		
	K 1	K 2	K 3	A 1	A 2	G			
K/A Category Totals:	3 0	1 0	5 0	3 0	3 3	3 3	Group Point Total:		18 / 6

ES-401 PWR Examination Outline Form ES-401-2
Vogtle 2005-301 Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO-R / **SRO-S**)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
000001 Continuous Rod Withdrawal / 1							Not selected.		
000003 Dropped Control Rod / 1							Not selected.		
000005 Inoperable/Stuck Control Rod / 1					S		**S 005AA2.03 Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable.**	**4.4**	**1**
000024 Emergency Boration / 1		R					R 024AK2.01 Knowledge of the interrelations between the Emergency Boration and the following: Valves.	2.7	1
000028 Pressurizer Level Malfunction / 2							Not selected.		
000032 Loss of Source Range NI / 7		R					R 032AK2.01 Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: Power supplies, including power switch positions.	2.7	1
000033 Loss of Intermediate Range NI / 7							Not selected.		
000036 (BW/A08) Fuel Handling Accident / 8	R						R 036AK1.02 Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents: SDM.	3.4	1
000037 Steam Generator Tube Leak / 3	R						R 037AK1.02 Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Leak rate vs. pressure drop	3.5	1
000051 Loss of Condenser Vacuum / 4							Not selected.		
000059 Accidental Liquid RadWaste Rel. / 9							Not selected.		
000060 Accidental Gaseous Radwaste Rel. / 9							Not selected.		
000061 ARM System Alarms / 7							Not selected.		
000067 Plant Fire On-site / 8						S	**S 067G2.4.28 Knowledge of procedures relating to emergency response to sabotage.**	**3.3**	**1**
000068 (BW/A06) Control Room Evac. / 8							Not selected.		
000069 (W/E14) Loss of CTMT Integrity / 5						S	**S 069G2.1.33 High Containment Pressure - Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.**	**4.0**	**1**
000074 (W/E06&E07) Inad. Core Cooling / 4						S	**S 074G2.4.6 Knowledge of symptom based EOP mitigation strategies.**	**4.0**	**1**
000076 High Reactor Coolant Activity / 9							Not selected.		
W/EO1 & E02 Rediagnosis & SI Termination / 3						R	R W/E01G2.1.9: Ability to direct personnel activities inside the control room.	2.5	1
W/E13 Steam Generator Over-pressure / 4							Not selected.		
W/E15 Containment Flooding / 5							Not selected.		
W/E16 High Containment Radiation / 9		R					R W/E16EK2.1 Knowledge of the interrelations between the (High Containment Radiation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	3.0	1
BW/A01 Plant Runback / 1							Not selected.		
BW/A02&A03 Loss of NNI-X/Y / 7							Not selected.		
BW/A04 Turbine Trip / 4							Not selected.		
BW/A05 Emergency Diesel Actuation / 6							Not selected.		
BW/A07 Flooding / 8							Not selected.		
BW/E03 Inadequate Subcooling Margin / 4							Not selected.		

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO-R / **SRO-S**)

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
BW/E08; W/E03 LOCA Cooldown - Depress. / 4			R				R W/E03EK3.3 Knowledge of the reasons for the following responses as they apply to the (LOCA Cooldown and Depressurization): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	3.9	1
BW/E09; CE/A13; W/E09&E10 Natural Circ. / 4		R					R W/E09EK2.2 Knowledge of the interrelations between the (Natural Circulation Operations) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, and decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	3.6	1
BW/E13&E14 EOP Rules and Enclosures							Not selected.		
CE/A11; W/E08 RCS Overcooling - PTS / 4						R	R W/E08G2.4.20 Knowledge of the operational implications of EOP cautions, warnings, and notes.	3.3	1
CE/A16 Excess RCS Leakage / 2							Not selected.		
CE/E09 Functional Recovery							Not selected.		
	K 1	K 2	K 3	A 1	A 2	G			
K/A Category Point Totals:	2 0	4 0	1 0	0 0	0 1	2 3	Group Point Total:		9/4

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
003 Reactor Coolant Pump					R							R 003K5.03 Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on T-ave., including the reason for the unreliability of T-ave. In the shutdown loop.	3.1	1
004 Chemical and Volume Control						R		R				R 004K6.13 Knowledge of the effect of a loss or malfunction on the following CVCS components: Purpose and function of the boration / dilution batch controller. R 004A2.07 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Isolation of letdown / makeup.	3.1 3.4	1 1
005 Residual Heat Removal							R	S				R 005A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Heatup/cooldown rates. **S 005A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction.**	3.5 **2.9**	1 **1**
006 Emergency Core Cooling								R				R 006A2.11 Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture of ECCS header.	4.0	1
007 Pressurizer Relief/Quench Tank									R			R 007A3.01 Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT.	2.7	1
008 Component Cooling Water										R		R 008A4.01 Ability to manually operate and / or monitor in the control room: CCW indications and controls.	3.3	1
010 Pressurizer Pressure Control	R										S	R 010K1.06 Knowledge of the physical connections and / or cause-effect relationships between the PZR PCS and the following systems: CVCS. **S 010G2.2.22 Knowledge of limiting conditions for operations and safety limits.**	2.9 **4.1**	1 **1**
012 Reactor Protection	R	R										R 012K1.07 Knowledge of the physical connections and / or cause effect relationships between the RPS and the following systems: SDS. R 012K2.01 Knowledge of bus power supplies to the following: RPS channels, components, and interconnections.	3.2 3.3	1 1
013 Engineered Safety Features Actuation			R				R					R 013K3.03 Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Containment. R 013A1.04 Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ESFAS controls including: S/G level.	4.3 3.4	1 1
022 Containment Cooling		R		R								R 022K2.01 Knowledge of power supplies to the following: Containment cooling fans. R 022K4.01 Knowledge of CCS design feature(s) and / or interlock(s) which provide for the following: Cooling of containment penetrations.	3.0 2.5	1 1
025 Ice Condenser												Not applicable.		
026 Containment Spray				R								R 026K4.09 Knowledge of CSS design feature(s) and / or interlock(s) which provide for the following: Prevention of path for escape of radioactivity from containment to the outside (interlock on RWST isolation after swapover).	3.7	1

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
039 Main and Reheat Steam					R					R		R 039K5.08 Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity. R 039A4.07 Ability to manually operate and / or monitor in the control room: Steam dump valves.	3.6 2.8	1 1
059 Main Feedwater			R							R		R 059K3.03 Knowledge of the effect that a loss or malfunction of the MFW will have on the following: S/Gs. R 059A4.12 Ability to manually operate and monitor in the control room: Initiation of automatic feedwater isolation.	3.5 3.4	1 1
061 Auxiliary/Emergency Feedwater								R			S	R 061A2.08 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Flow rates expected from various combinations of AFW pump discharge valves. **S 061G2.4.18 Knowledge of the specific bases for EOPs.**	2.7 **3.6**	1 **1**
062 AC Electrical Distribution							R		R			R 062A1.01 Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits. R 062A3.05 Ability to monitor automatic operation of the ac distribution system, including: Safety-related indicators and controls.	3.4 3.5	1 1
063 DC Electrical Distribution										R		R 063A4.01 Ability to manually operate and / or monitor in the control room: Major breakers and control power fuses.	2.8	1
064 Emergency Diesel Generator								S			R	R 064G2.2.23 Ability to track limiting conditions for operations. **S 064A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loading the ED/G.**	2.6 **3.2**	1 **1**
073 Process Radiation Monitoring	R											R 073K1.01 Knowledge of the physical connections and / or cause-effect relationships between the PRM system and the following systems: Those systems served by PRMs.	3.6	1
076 Service Water		R									S	R 076K2.08 Knowledge of bus power supplies to the following: ESF-actuated MOVs. **S 076G2.4.11 Knowledge of abnormal condition procedures.**	3.1 **3.6**	1 **1**
078 Instrument Air	R											R 078K1.01 Knowledge of the physical connections and / or cause-effect relationships between the IAS and the following systems: Sensor air.	2.8	1
103 Containment				R								R 103K4.06 Knowledge of containment system design feature(s) and / or interlock(s) which provide for the following: Containment isolation system.	3.1	1
	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G			
K/A Category Point Totals:	4 **0**	3 **0**	2 **0**	3 **0**	2 **0**	1 **0**	3 **0**	3 **2**	2 **0**	4 **0**	1 **3**	Group Point Total:		28/5

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

Vogtle 2005-301 Plant Systems - Tier 2/Group 2 (RO-R / **SRO-S**)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	K/A Topic(s)	IR	#
001 Control Rod Drive		R										R 001K2.01 Knowledge of bus power supplies to the following: One-line diagram of power supply to M/G sets.	3.5	1
002 Reactor Coolant											R	R 002G2.4.31 Knowledge of annunciators alarms, and indications, and use of the response instructions.	3.3	1
011 Pressurizer Level Control												Not selected.		
014 Rod Position Indication							R					R 014A1.03 Ability to predict and / or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: PDIL, PPDIL.	3.6	1
015 Nuclear Instrumentation												Not selected.		
016 Non-nuclear Instrumentation								S				**S 016A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Interruption of transmitted signal.**	**3.3**	**1**
017 In-core Temperature Monitor			R									R 017K3.01 Knowledge of the effect that a loss or malfunction of the ITM system will have on the following: Natural circulation indications.	3.5	1
027 Containment Iodine Removal												Not selected.		
028 Hydrogen Recombiner and Purge Control												Not selected.		
029 Containment Purge												Not selected.		
033 Spent Fuel Pool Cooling												Not selected.		
034 Fuel Handling Equipment				R								R 034K4.03 Knowledge of design feature(s) and / or interlock(s) which provide for the following: Overload protection.	2.6	1
035 Steam Generator											S	**S 035G2.1.20 Ability to execute procedure steps.**	**4.2**	**1**
041 Steam Dump/Turbine Bypass Control						R						R 041K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS.	2.7	1
045 Main Turbine Generator	R											R 045K1.18 Knowledge of the physical connections and / or cause-effect relationships between the MT/G system and the following systems: RPS.	3.6	1
055 Condenser Air Removal												Not selected.		
056 Condensate												Not selected.		
068 Liquid Radwaste						R						R 068K6.10 Knowledge of the effect of a loss or malfunction on the following will have on the Liquid Radwaste System: Radiation monitors.	2.5	1
071 Waste Gas Disposal												Not selected.		
072 Area Radiation Monitoring								S				**S 072A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.**	**2.9**	**1**
075 Circulating Water												Not selected.		
079 Station Air								R				R 079A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the SAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Cross-connection with IAS.	2.9	1
086 Fire Protection									R			R 086A3.02 Ability to monitor automatic operation of the Fire Protection System including: Actuation of the FPS.	2.9	1
K/A Category Point Totals:	1 0	1 0	1 0	1 0	0 0	2 0	1 0	1 **2**	1 0	0 0	1 **1**	Group Point Total:	10/3	

ES-401	Generic Knowledge and Abilities Outline (Tier 3)					Form ES-401-3
Facility:	Vogtie	Date of Exam : May 2005				
Category	K/A #	Topic	RO		**SRO - Only**	
			IR	#	**IR**	**#**
Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements.	3.7	1		
	2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	3.7	1		
	2.1.23	Ability to perform specific system and plant procedures during all modes of plant operation.	3.9	1		
	2.1.14	**Knowledge of system status criteria which require the notification of plant personnel.**			**3.3**	**1**
	Subtotal			3		1
Equipment Control	2.2.13	Knowledge of tagging and clearance procedures.	3.6	1		
	2.2.24	Ability to analyze the affect of maintenance activities on LCO status.	2.6	1		
	2.2.28	Knowledge of new and spent fuel movement procedures.	2.6	1		
	2.2.7	**Knowledge of the process for conducting tests or experiments not described in the safety analysis report.**			**3.2**	**1**
	2.2.21	**Knowledge of pre- and post- maintenance operability requirements.**			**3.5**	**1**
	Subtotal			3		**2**
Radiation Control	2.3.1	Knowledge of 10CFR: 20 and related facility radiation control requirements.	2.6	1		
	2.3.11	Ability to control radiation releases.	2.7	1		
	2.3.6	**Knowledge of the requirements for reviewing and approving release permits.**			**3.1**	**1**
	Subtotal			2		**1**
Emergency Procedures / Plan	2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.0	1		
	2.4.12	Knowledge of general operating crew responsibilities during emergency operations.	3.4	1		
	2.4.16	**Knowledge of EOP implementation hierarchy and coordination with other support procedures.**			**4.0**	**1**
	2.4.44	**Knowledge of emergency plan protective action recommendations.**			**4.0**	**1**
	2.4.45	**Ability to prioritize the significance of each annunciator or alarm.**			**3.6**	**1**
	Subtotal			2		**3**
Tier 3 Point Total				10		**7**

Tier / Group	Randomly Selected K/A	Reason for Rejection
2 / 1	008A4.11	Not applicable. Equipment does not exist. Replaced with 008A4.01.
2 / 1	005A1.07	Not applicable. No TS valve stroke times exist for RHR valves. Replaced with 005A1.01.
1 / 2	W/E15G2.2.25	T.S. Bases no RO knowledge level. Replaced with W/E01 G2.1.9.
1 / 1	015/017G2.2.12	Generic K/A not applicable to topic. Replaced with 2.4.10.
1 / 2	067AK3.02	Could not construct discriminating question. Replaced with 037AK1.02.
1 / 1	054AK1.02	Similar to another question already written. Replaced with 054AK3.02.
1 / 1	029EA1.12	Similar to another already written question. Replaced with 065AA1.03.
1 / 2	028AA2.07	Not conducive to SRO-only level question. Replaced with 067G2.4.28.
1 / 1	008AA2.09	Not conducive to SRO-only level question. Replaced with 008AA2.22.
1 / 1	015/017AA2.09	Not conducive to SRO-only level question. Replaced with 011EA2.08.

1. 005A2.04 001

The following Unit 1 conditions exist:

- The reactor cavity is flooded with the transfer canal open and fuel in the vessel
- The 'A' RHR pump is operating
- The 'B' RHR pump is out of service due to a breaker malfunction
- The Train 'A' Residual Heat Exchanger Outlet Valve has closed and will not re-open
- No RCS leakage is present
- 18019-C, Loss of Residual Heat Removal, has been entered.

Which ONE of the following correctly describes the actions that must be taken in accordance with 18019-C?

A.✓ Implement 91001-C, Emergency Classification and Implementing Instructions.
Shift CCP Suction to the RWST.
Place SFP Purification in service and transfer water from the SFP to the RWST.

B. Implement 91001-C, Emergency Classification and Implementing Instructions.
Ensure open RHR Train to Hot Leg Crossover Iso Valves (HV-8716A/B).
Ensure RCS temperature remains less than 200 °F.

C. Stop the 'A' RHR Pump and declare the 'A' RHR Train Inoperable.
Ensure open RHR Train to Hot Leg Crossover Iso Valves (HV-8716A/B).
Initiate actions to establish containment closure.

D. Stop the 'A' RHR Pump and declare the 'A' RHR Train Inoperable.
Start available containment cooling fans in low speed.
Ensure open HV-8804A, SIP/CCP Suction Header Discharge.

K/A
005 Residual Heat Removal
A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction.

K/A MATCH ANALYSIS
The HX outlet valve has malfunctioned and the Loss of RHR procedure needs to be used to mitigate the problem. The question is SRO level because the E-Plan must be entered.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Section C (begins on Page 31) of 18019-C provides guidance to enter the E-Plan procedure, shift CCP suctions to RWST, align SFP prurification, and xfer water from SFP to RWST.
B. Incorrect. No guidance exists for opening HV-8716A/B and they tap off downstream of the HX outlet. Plausible because these valves do cross-connect the two trains and can be operated from the CR.
C. Incorrect. See B analysis above.
D. Incorrect. No guidance exists for opening HV-8804A and it also taps off downstream of the HX . Plausible applicant may think this to be a viable flowpath to circulate water through the RHR HX.

REFERENCES
1. 18019-C, Loss of Residual Heat Removal, Rev. 24, 09/17/2003.
2. P&ID, 1X4DB122, Residual Heat Removal, Rev. 47.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D D C A C D A C D	Scramble Range: A - D
Tier:	2		Group:	1	
Key Word:	RHR VALVE MALFUNCTIO		Cog Level:	C/A 2.9	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

1. 005A2.04 001

The following Unit 1 conditions exist:

- Fuel is in the vessel.
- The reactor cavity is flooded with the transfer canal open
- The 'A' RHR pump is operating
- The 'B' RHR pump is out of service due to a breaker malfunction
- The Train 'A' Residual Heat Exchanger Outlet Valve has closed and will not re-open
- No RCS leakage is present
- 18019-C, Loss of Residual Heat Removal, has been entered.

Which ONE of the following correctly describes the actions that must be taken in accordance with 18019-C?

A. ~~Enter~~ Implement 91001-C, Emergency Classification and Implementing Instructions.
Shift CCP Suction to the RWST.
Place SFP Purification in service and transfer water from the SFP to the RWST.

B. ~~Enter~~ Implement 91001-C, Emergency Classification and Implementing Instructions.
Ensure open RHR Train to Hot Leg Crossover Iso Valves (HV-8716A/B).
Ensure RCS temperature remains less than 200 °F.

C. Stop the 'A' RHR Pump and declare the 'A' RHR Train Inoperable.
Ensure open RHR Train to Hot Leg Crossover Iso Valves (HV-8716A/B).
Initiate actions to establish containment closure.

D. Stop the 'A' RHR Pump and declare the 'A' RHR Train Inoperable.
Start available containment cooling fans in low speed.
Ensure open HV-8804A, SIP/CCP Suction Header Discharge.

K/A
005 Residual Heat Removal
A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction.

K/A MATCH ANALYSIS
The HX outlet valve has malfunctioned and the Loss of RHR procedure needs to be used to mitigate the problem. The question is SRO level because the E-Plan must be entered.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Section C (begins on Page 31) of 18019-C provides guidance to enter the E-Plan procedure, shift CCP suctions to RWST, align SFP prurification, and xfer water from SFP to RWST.
B. Incorrect. No guidance exists for opening HV-8716A/B and they tap off downstream of the HX outlet. Plausible because these valves do cross-connect the two trains and can be operated from the CR.
C. Incorrect. See B analysis above.
D. Incorrect. No guidance exists for opening HV-8804A and it also taps off downstream of the HX . Plausible applicant may think this to be a viable flowpath to circulate water through the RHR HX.

REFERENCES
1. 18019-C, Loss of Residual Heat Removal, Rev. 24, 09/17/2003.
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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A D D C A C D A C D		Scramble Range: A - D
Tier:	2		Group:		1	
Key Word:	RHR VALVE MALFUNCTIO		Cog Level:		C/A 2.9	
Source:	N		Exam:		VG05301	
Test:	S		Author/Reviewer:		MAB/RSB	

1. 005A2.04 001

The following Unit 1 conditions exist:

- The reactor cavity is flooded with the transfer canal open
- The 'A' RHR pump is operating
- The 'B' RHR pump is out of service due to a breaker malfunction
- The Train 'A' Residual Heat Exchanger Outlet Valve has closed and will not re-open
- No RCS leakage is present
- 18019-C, Loss of Residual Heat Removal, has been entered.

Which ONE of the following correctly describes the actions that must be taken in accordance with 18019-C?

A.✓ Enter 91001-C, Emergency Classification and Implementing Instructions.
Shift CCP Suction to the RWST.
Place SFP Purification in service and transfer water from the SFP to the RWST.

B. Enter 91001-C, Emergency Classification and Implementing Instructions.
Ensure open RHR Train to Hot Leg Crossover Iso Valves Open (HV-8716A/B).
Ensure RCS temperature remains less than 200 °F.

C. Stop the 'A' RHR Pump and declare the 'A' RHR Train Inoperable.
Ensure open RHR Train to Hot Leg Crossover Iso Valves Open (HV-8716A/B).
Initiate actions to establish containment closure.

D. Stop the 'A' RHR Pump and declare the 'A' RHR Train Inoperable.
Start available containment cooling fans in low speed.
Ensure open HV-8804A, SIP/CCP Suction Header Discharge.

K/A
005 Residual Heat Removal
A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction.

K/A MATCH ANALYSIS
The HX outlet valve has malfunctioned and the Loss of RHR procedure needs to be used to mitigate the problem. The question is SRO level because the E-Plan must be entered.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Section C (begins on Page 31) of 18019-C provides guidance to enter the E-Plan procedure, shift CCP suctions to RWST, align SFP prurification, and xfer water from SFP to RWST.
B. Incorrect. No guidance exists for opening HV-8716A/B and they tap off downstream of the HX outlet. Plausible because these valves do cross-connect the two trains and can be operated from the CR.
C. Incorrect. See B analysis above.
D. Incorrect. No guidance exists for opening HV-8804A and it also taps off downstream of the HX . Plausible applicant may think this to be a viable flowpath to circulate water through the RHR HX.

REFERENCES
1. 18019-C, Loss of Residual Heat Removal, Rev. 24, 09/17/2003.
2. P&ID, 1X4DB122, Residual Heat Removal, Rev. 47.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D D C A C D A C D	Scramble Range: A - D
Tier:	2		Group:	1	
Key Word:	RHR VALVE MALFUNCTIO		Cog Level:	C/A 2.9	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

C. LOSS OF RHR - Rx Cavity Flooded and Transfer Canal Open

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: 91001-C EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS should be implemented at this time.

C1. Shift charging suction to the RWST:

a. Open RWST TO CCP A&B SUCTION VALVES:

- LV-0112D
- LV-0112E

b. Shut VCT OUTLET ISOLATION valves:

- LV-0112B
- LV-0112C

c. Align RV TO RWST ISOLATION valves:

- HV-8508A CCP-A - ENABLE PTL

- HV-8508B CCP-B - ENABLE PTL

d. Shut CCP normal mini flow valves:

- HV-8110 CCP-A&B COMMON MINIFLOW

- HV-8111A CCP-A MINIFLOW

- HV-8111B CCP-B MINIFLOW

e. Verify VCT diverts to HUT on HI level (97%).

* C2. MONITOR RCS HOT LEG TEMPERATURES by trending on IPC.

C. LOSS OF RHR - Rx Cavity Flooded and Transfer Canal Open

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* C3. IDENTIFY and ISOLATE any RCS leakage.	→ Info given in stem that there was no RCS leakage present.
* C4. ADJUST charging as necessary to stabilize RCS HOT LEG TEMPERATURES and RCS level.	
C5. Place SFP Purification in service by initiating 13719, SPENT FUEL POOL COOLING AND PURIFICATION SYSTEM section 4.1.2.	
C6. TRANSFER water from the SFP to the RWST by initiating 13719, SPENT FUEL POOL COOLING AND PURIFICATION SYSTEM section 4.4.12.	
C7. ADJUST the transfer flowrate from the SFP to the RWST to 250 gpm.	
* C8. ADJUST charging to lower or maintain RCS HL temperature and to maintain Rx Cavity level constant.	* C8. MAINTAIN RCS level as necessary with any of the following: • Normal charging to intact CL (i.e., RCS loop NOT open to containment atmosphere) • Alternate charging to intact CL (i.e., RCS loop NOT open to containment atmosphere) • BIT flow (align using ATTACHMENT D) • Align SFP Purification back to the RHUT by initiating 13719, SPENT FUEL POOL COOLING AND PURIFICATION SYSTEM section 4.4.2.

C. LOSS OF RHR - Rx Cavity Flooded and Transfer Canal Open

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C9. START available containment cooling fans in low speed.

- HS-12582A CTB CLG UNIT FAN-1 LOW SPEED
- HS-2582A CTB CLG UNIT FAN-2 LOW SPEED
- HS-12583A CTB CLG UNIT FAN-3 LOW SPEED
- HS-2583A CTB CLG UNIT FAN-4 LOW SPEED
- HS-12584A CTB CLG UNIT FAN-5 LOW SPEED
- HS-2584A CTB CLG UNIT FAN-6 LOW SPEED
- HS-12585A CTB CLG UNIT FAN-7 LOW SPEED
- HS-2585A CTB CLG UNIT FAN-8 LOW SPEED

*C10. If RCS temperature becomes unstable,
THEN initiate actions to protect personnel working in containment.

a. Evacuate non-essential personnel in containment.

b. Periodically monitor containment radiation conditions using any available RAD monitors in containment.

c. Monitor ambient containment temperature to determine if evacuation of all personnel is required.

C. LOSS OF RHR - Rx Cavity Flooded and Transfer Canal Open

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*C11. If RCS temperature becomes unstable, THEN initiate actions to establish CNMT closure by performing the following:	
• ACTUATE CIA/CVI	• IF SSPS in TEST, THEN manually shut dampers and valves.
• Verify equipment hatch shut	
• Verify personnel hatch shut	
• Review LCO Log for containment isolation	
• As time permits, perform actions required to establish Containment integrity by initiating 14210, CONTAINMENT BUILDING PENETRATION VERIFICATION - REFUELING and applicable steps of 12001-C, UNIT HEATUP TO HOT SHUTDOWN (MODE 5 TO MODE 4).	

C. LOSS OF RHR - Rx Cavity Flooded and Transfer Canal Open

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C12. Vent any RHR pump that experienced cavitation:

a. Maintain RCS level while venting RHR system.

b. Vent RHR:

1) Vent RHR pump(s) at high point vent until water is discharged:

UNIT 1

- 1-HV-10465 RHR SUCT VENT LINE TRN A (AB-B08)
- 1-HV-10466 RHR SUCT VENT LINE TRN B (FHB-B13)

UNIT 2

- 2-HV-10465 RHR SUCT VENT LINE TRN A (AB-B131)
- 2-HV-10466 RHR SUCT VENT LINE TRN B (FHB-B03)

2. 005AA2.03 001

The following Unit 1 conditions exist:

- Reactor power is 75% following a power reduction.
- During rod insertion the crew observed that three control rods failed to insert with the rest of their group and are now 14 steps further withdrawn than the group step counter demand position.
- Troubleshooting revealed the following:
  - All three control rods will not move via operation of the control rod drive system.
  - A verification has not yet been made to determine if a rod control system failure is preventing rod motion.
- The crew has verified shutdown margin to be 1.85 % delta-k / k.

Which ONE of the following correctly describes additional actions, if any, that are required by Technical Specifications and the basis for those actions?

A.✓ Reduce thermal power to place the plant in Mode 3 because the conditions are outside of the Safety Analysis assumptions.

B. Verify $F_Q(z)$ is within its steady state limit and initiate boration to restore shutdown margin within the limits of the COLR.

C. Reevaluate the Safety Analysis to confirm that the results remain valid for the duration of operation under these conditions and initiate boration to restore shutdown margin within the limits of the COLR.

D. No additional actions are required because the stated plant conditions are bounded by the Safety Analysis.

K/A
005 Inoperable/Stuck Control Rod
AA2.03 Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable.

K/A MATCH ANALYSIS
The stem describes a situation where three rods are immovable and inoperable. The question then tests the actions that are required by tech specs based on more than one rod being inoperable, therefore, the question is testing the knowledge required by the K/A. The question is SRO-only knowledge because it is asking about actions and the reasons for those actions. The question is closed reference because SROs are required to know the bases behind the Tech Specs. The question is also closed book because the completion time knowledge is not being tested.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Power must be reduced because the plant must be taken to Mode 3 based on inoperability of the control rods (LCO 3.1.4 Action A.2 and D.2).
B. Incorrect. SDM is already met, therefore boration is not required. Plausible because boration would be required if SDM were not met. Also, verification of $F_Q(z)$ is required for one misaligned rod.
C. Incorrect. SDM is already met, therefore boration is not required. Plausible because boration would be required if SDM were not met. Also plausible because reevaluation of the Safety Analysis is required for one misaligned rod.
D. Incorrect. Actions stated above are required. Plausible because SDM requirements are met.

REFERENCES
1. Technical Specification LCO 3.1.4, Rod Group Alignment.
2. Technical Specification LCO 3.1.4 Basis.
3. Technical Specification LCO 3.2.1, Heat Flux Hot Channel Factor.
4. Core Operating Limits Report, Unit 1, Cycle 12.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D B D A C B A D B	Scramble Range: A - D
Tier:	1		Group:	2	
Key Word:	STUCK CONTROL ROD		Cog Level:	C/A 4.4	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

1. 005AA2.03 001

The following Unit 1 conditions exist:

- Reactor power is 75% following a power reduction.
- During rod insertion the crew observed that three control rods failed to insert with the rest of their group and are now 14 steps further withdrawn than the group step counter demand position.
- Troubleshooting releaved the following: revealed
  - All three control rods will not move via operation of the control rod drive system.
  - There is no verification that a rod control system failure is preventing rod motion.
- The crew has verified shutdown margin to be 1.85 % delta-k / k.

Which ONE of the following correctly describes additional actions, if any, that are required by Technical Specifications and the basis for those actions?

A.✓ Reduce thermal power to place the plant in Mode 3 because the conditions are outside of the Safety Analysis assumptions.

B. Verify $F_Q(z)$ is within its steady state limit and initiate boration to restore shutdown margin within the limits of the COLR.

C. Reevaluate the Safety Analysis to confirm that the results remain valid for the duration of operation under these conditions and initiate boration to restore shutdown margin within the limits of the COLR.

D. No additional actions are required because the stated plant conditions are bounded by the Safety Analysis.

It has been verified that
There is no rod control sys failure is preventing rod motion.

K/A
005 Inoperable/Stuck Control Rod
AA2.03 Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable.

K/A MATCH ANALYSIS
The stem describes a situation where three rods are immovable and inoperable. The question then tests the actions that are required by tech specs based on more than one rod being inoperable, therefore, the question is testing the knowledge required by the K/A. The question is SRO-only knowledge because it is asking about actions and the reasons for those actions. The question is closed reference because SROs are required to know the bases behind the Tech Specs. The question is also closed book because the completion time knowledge is not being tested.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Power must be reduced because the plant must be taken to Mode 3 based on inoperability of the control rods (LCO 3.1.4 Action A.2 and D.2).
B. Incorrect. SDM is already met, therefore boration is not required. Plausible because boration would be required if SDM were not met. Also, verification of $F_Q(z)$ is required for one misaligned rod.
C. Incorrect. SDM is already met, therefore boration is not required. Plausible because boration would be required if SDM were not met. Also plausible because reevaluation of the Safety Analysis is required for one misaligned rod.
D. Incorrect. Actions stated above are required. Plausible because SDM requirements are met.

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4. Core Operating Limits Report, Unit 1, Cycle 12.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D B D A C B A D B	Scramble Range: A - D
Tier:	1		Group:	2	
Key Word:	STUCK CONTROL ROD		Cog Level:	C/A 4.4	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

2. 005AA2.03 001

The following Unit 1 conditions exist:

- Reactor power is 75% following a power reduction.
- During rod insertion the crew observed that three control rods failed to insert with the rest of their group and are now 14 steps further withdrawn than the group step counter demand position.
- Troubleshooting releaved the following:
  - All three control rods will not move via operation of the control rod drive system.
  - There is no verification that a rod control system failure is preventing rod motion.
- The crew has verified shutdown margin to be 1.85 % delta-k / k.

Which ONE of the following correctly describes additional actions, if any, that are required by Technical Specifications and the basis for those actions?

A.✓ Reduce thermal power to place the plant in Mode 3 because the conditions are outside of the Safety Analysis assumptions.

B. Verify $F_Q(z)$ is within its steady state limit and initiate boration to restore shutdown margin within the limits of the COLR.

C. Reevaluate the Safety Analysis to confirm that the results remain valid for the duration of operation under these conditions and initiate boration to restore shutdown margin within the limits of the COLR.

D. No additional actions are required because shutdown margin requirements are met.

K/A
005 Inoperable/Stuck Control Rod
AA2.03 Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable.

K/A MATCH ANALYSIS
The stem describes a situation where three rods are immovable and inoperable. The question then tests the actions that are required by tech specs based on more than one rod being inoperable, therefore, the question is testing the knowledge required by the K/A. The question is SRO-only knowledge because it is asking about actions and the reasons for those actions. The question is closed reference because SROs are required to know the bases behind the Tech Specs. The question is also closed book because the completion time knowledge is not being tested.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Power must be reduced because the plant must be taken to Mode 3 based on inoperability of the control rods (LCO 3.1.4 Action A.2 and D.2).
B. Incorrect. SDM is already met, therefore boration is not required. Plausible because boration would be required if SDM were not met. Also, verification of $F_Q(z)$ is required for one misaligned rod.
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			Answer:	A D B D A C B A D B	Scramble Range: A - D
Tier:	1		Group:	2	
Key Word:	STUCK CONTROL ROD		Cog Level:	C/A 4.4	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

1. 005AA2.03 001

The following Unit 1 conditions exist:

- Reactor power is 75% following a power reduction.
- During rod insertion the crew observed that three control rods failed to insert with the rest of their group and are now 14 steps further withdrawn than the group step counter demand position.
- Troubleshooting releaved the following:
  - All three control rods will not move via operation of the control rod drive system.
  - There is no verification that a rod control system failure is preventing rod motion.
- The crew has verified shutdown margin to be 1.85 % delta-k / k.

Which ONE of the following correctly describes additional actions, if any, that are required by Technical Specifications and the basis for those actions?

A.✓ Reduce thermal power to place the plant in Mode 3 because the conditions are outside of the Safety Analysis assumptions.

B. Verify $F_Q(z)$ is within its steady state limit and initiate boration to restore shutdown margin within the limits of the COLR.

C. Reevaluate the Safety Analysis to confirm that the results remain valid for the duration of operation under these conditions and initiate boration to restore shutdown margin within the limits of the COLR.

D. No additional actions are required because shutdown margin requirements are met.

K/A
005 Inoperable/Stuck Control Rod
AA2.03 Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Required actions if more than one rod is stuck or inoperable.

K/A MATCH ANALYSIS
The stem describes a situation where three rods are immovable and inoperable. The question then tests the actions that are required by tech specs based on more than one rod being inoperable, therefore, the question is testing the knowledge required by the K/A. The question is SRO-only knowledge because it is asking about actions and the reasons for those actions. The question is closed reference because SROs are required to know the bases behind the Tech Specs. The question is also closed book because the completion time knowledge is not being tested.

ANSWER / DISTRACTOR ANALYSIS
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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D B D A C B A D B	Scramble Range: A - D
Tier:	1		Group:	2	
Key Word:	STUCK CONTROL ROD		Cog Level:	C/A 4.4	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with all individual indicated rod positions within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) untrippable. Stem states Rods ARE trippable.	A.1.1 Verify SDM is ≥ the limit specified in the COLR. OR A.1.2 Initiate boration to restore SDM to within limit. AND A.2 Be in MODE 3.	1 hour 1 hour 6 hours
B. One rod not within alignment limits. K/A Requires more than one Rod.	B.1.1 Verify SDM is ≥ the limit specified in the COLR. OR	1 hour (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	<u>AND</u>		
	B.3	Verify SDM is ≥ the limit specified in the COLR.	Once per 12 hours
	<u>AND</u>		
	B.4	Perform SR 3.2.1.1.	72 hours
	<u>AND</u>		
	B.5	Perform SR 3.2.2.1.	72 hours
	<u>AND</u>		
	B.6	Reevaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is ≥ the limit specified in the COLR.	1 hour
	<u>OR</u>	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours <u>AND</u> Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.2	Verify rod freedom of movement by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the physical fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. T_{avg} ≥ 551°F; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q(Z)$ is within steady state limit.	Once after each refueling after achieving equilibrium conditions at any power level exceeding 50% RTP <u>AND</u> Once after achieving equilibrium conditions after exceeding, by $\geq$ 20% RTP, the THERMAL POWER at which $F_Q(Z)$ was last verified <u>AND</u> 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND 31 EFPD thereafter

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately ⅝ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists

(continued)

BASES

BACKGROUND (continued)

of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. There are four control banks and five shutdown banks. All control banks contain two rod groups, two shutdown banks contain two rod groups, and the remaining three shutdown banks contain one rod group.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog

(continued)

BASES

BACKGROUND (continued)

signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is six steps. However, the magnetic drive rod concentrates the magnetic lines of flux developed in the coil resulting in a change in coil output voltage when the shaft is close to it. This provides a ± 4 step accuracy with all coils operable. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy (System A failure = +10, -4 steps and System B failure = -10, +4 steps) with an effective coil spacing of 7.5 inches, which is 12 steps. The resolution of the rod position indicator channel is ± 5 percent of span (± 7.5 in. or ± 12 steps). Deviation of any RCCA from its group by 10 percent of span (15 inches or 24 steps) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span (12 steps). Therefore, since indication from one system is sufficient to maintain alignment within 24 steps, operation with one system (in the event of failure of the other) is acceptable.

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

   1. specified acceptable fuel design limits, or
   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment. With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 3).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ($F_Q(Z)$) and the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_Q(Z)$ and $F^N_{\Delta H}$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q(Z)$ and $F^N_{\Delta H}$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

(continued)

BASES (continued)

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability) are separate from the alignment requirements which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The rod OPERABILITY (i.e., trippability) requirement is satisfied provided that the rod will fully insert in the required rod drop time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod lift coil failures), but that do not impact trippability, do not result in rod inoperability. However, where rod(s) are not moving, the rod(s) must be considered untrippable unless there is verification that a rod control system failure is preventing rod motion. If the rod control system is demanding motion properly and no motion occurs, the rod is considered untrippable (i.e., inoperable).

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. When required, movable incore detectors may be used to determine rod position and verify the rod alignment requirement of this LCO is met.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which a self-sustaining chain reaction ($K_{eff} \geq 1$) occurs, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are fully inserted and the reactor is shut down, with no self-sustaining chain reaction. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

(continued)

BASES (continued)

ACTIONS

A.1.1 and A.1.2

When one or more rods are untrippable, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.

In the situation of untrippable rod(s), SDM verification must account for the absence of the negative reactivity of the untrippable rod(s), as well as a rod of maximum worth.

A.2

If the untrippable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1.1 and B.1.2

With a misaligned but trippable rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be inserted to approximately 100 to 115 steps.

Power operation may continue with one RCCA trippable but misaligned, provided that SDM is verified within 1 hour.

(continued)

BASES

ACTIONS

B.1.1 and B.1.2 (continued)

The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2, B.3, B.4, B.5, and B.6

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned shortly after the misalignment, local xenon redistribution during this short interval will not be significant, and operation in compliance with the LCO may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of Required Action B.2 gives the operator sufficient time to adjust the rod positions in an orderly manner or subsequently reduce power if the rod alignment cannot be restored to within the LCO limits shortly after the misalignment.

For continued operation with a misaligned rod, reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors (FQ(Z) and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be reevaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required.

(continued)

BASES

ACTIONS

B.2, B.3, B.4, B.5, and B.6 (continued)

A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F^N_{\Delta H}$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F^N_{\Delta H}$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

The following accident analyses require reevaluation for continued operation with a misaligned rod.

RCCA Insertion Characteristics
RCCA Misalignment
Decrease in Reactor Coolant Inventory

- Inadvertent Opening of a Pressurizer Safety or Relief Valve
- Break in Instrument Line or Other Lines From Reactor Coolant Pressure Boundary That Penetrates Containment
- Loss-of-Coolant-Accidents

Increase in Heat Removal by the Secondary System (Steam System Piping Rupture) Spectrum of RCCA Ejection Accidents.

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least

(continued)

BASES

ACTIONS

C.1 (continued)

MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned (but trippable) from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration required for potential xenon redistribution, the low probability of an accident to provide negative reactivity, as described in the Bases or LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

Supports correct answer and SRO-only closed book knowledge.

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

Exercising each individual control rod every 92 days provides confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times from the physical fully withdrawn position allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 551°F (TI-0412, TI-0422, TI-0432, TI-0442) to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, Subsection 15.4.3.

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in section 1.0 are presented in the following subsections. These limits have been developed using NRC-approved methodologies, including those specified in Technical Specification 5.6.5.

2.1 SHUTDOWN MARGIN - MODES 1 AND 2 (Technical Requirement 13.1.1)

2.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.30 percent $\Delta k/k$.

2.2 SHUTDOWN MARGIN - MODES 3, 4 AND 5 (Specification 3.1.1)

2.2.1 The SHUTDOWN MARGIN shall be greater than or equal to the limits shown in Figures 1 and 2.

2.3 Moderator Temperature Coefficient (Specification 3.1.3)

2.3.1 The Moderator Temperature Coefficient (MTC) limits are:

The BOL/ARO/HZP - MTC shall be less positive than $+0.7 \times 10^{-4}$ $\Delta k/k/^{o}F$ for power levels up to 70 percent RTP with a linear ramp to 0 $\Delta k/k/^{o}F$ at 100 percent RTP.

The EOL/ARO/RTP-MTC shall be less negative than -5.50×10^{-4} $\Delta k/k/^{o}F$.[1]

2.3.2 The MTC Surveillance limits are:

The 300 ppm/ARO/RTP-MTC should be less negative than or equal to -4.75×10^{-4} $\Delta k/k/^{o}F$.[1]

The 60 ppm/ARO/RTP-MTC should be less negative than -5.35×10^{-4} $\Delta k/k/^{o}F$.[1]

where: BOL stands for Beginning of Cycle Life
ARO stands for All Rods Out
HZP stands for Hot Zero THERMAL POWER
EOL stands for End of Cycle Life
RTP stands for RATED THERMAL POWER

2.4 Shutdown Bank Insertion Limits (Specification 3.1.5)

2.4.1 The shutdown banks shall be withdrawn to a position greater than or equal to 225 steps.

2.5 Control Bank Insertion Limits (Specification 3.1.6)

2.5.1 The control banks shall be limited in physical insertion as shown in Figure 3.

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[1]Applicable for full-power T-average of 586.4°F to 587.4°F.

3. 008AA2.22 001

The following conditions exist on Unit 1:

- A reactor trip and safety injection have occurred.
- Pressurizer pressure is 1300 psig and lowering.
- Pressurizer Code Safety Temperature Indicator, TI-0466, is rising.
- PRT level, temperature, and pressure are rising.
- All safety equipment has functioned as expected.

The operating crew has just entered 19010-C, E-1 Loss of Reactor or Secondary Coolant.

Which ONE of the following correctly states procedurally directed requirements for RCP operation and the reasons for the requirements?

A. Secure RCPs to reduce pressure in the vessel due to pressurized thermal shock concerns.

B.✓ Secure RCPs due to peak clad temperature concerns.

C. Do not secure RCPs because securing the pumps may raise the fuel rod peak clad temperature.

D. Do not secure the RCPs because securing the pumps may worsen the voiding in the reactor vessel head.

K/A
008 Pressurizer Vapor Space Accident
AA2.09 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Consequences of loss of pressure in RCS; methods for evaluating pressure loss.

K/A MATCH ANALYSIS
The stem provides indications for verifying the method of pressure loss is in the vapor space of the pressurizer. The question tests the consequences of this pressure loss - RCP tripping. Question is SRO level because it tests the reasons for securing RCPs.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. RCPs are not secured to reduce pressure.
B. Correct. Pumps must be secured iaw E-1 Step 1. Per referenced lesson plan, pumps are secured because PCT may be higher if pumps are secured at lower pressures.
C. Incorrect. Pumps must be secured iaw E-1 Step 1. Plausible because there is no break in the hot or cold legs when you may have an inventory loss concern in those areas. An applicant may think that it would make sense to keep the pumps running when the break is located in the upper portions of the system.
D. Incorrect. Pumps must be secured iaw E-1 Step 1. Plausible for same reasons as 'C' above. In addition, head voiding is a concern, and keeping the RCPs running would likely lessen the voiding.

REFERENCES
1. 19010, E-1 Loss of Reactor or Secondary Coolant, Rev. 28.2, 09/27/2002.
2. LO-LP-37111, Operator Response To Loss Primary Coolant, Rev. 15, 08/29/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B C C B D A D D C A Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	RCP VAPOR SPACE LOCA	Cog Level:	C/A 4.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

2. 008AA2.22 001

The following conditions exist on Unit 1:

- A reactor trip and safety injection have occurred.
- Pressurizer pressure is 1300 psig and lowering.
- Pressurizer Code Safety Temperature Indicator, TI-0466, is rising.
- PRT level, temperature, and pressure are rising.
- All safety equipment has functioned as expected.

The operating crew has just entered 19010-C, E-1 Loss of Reactor or Secondary Coolant.

Which ONE of the following correctly states procedurally directed requirements for RCP operation and the reasons for the requirements?

A. Secure RCPs to reduce pressure in the vessel due to pressurized thermal shock concerns.

B.✓ Secure RCPs due to peak clad temperature concerns.

C. Do not secure RCPs because securing the pumps may raise the fuel rod peak clad temperature.

D. Do not secure the RCPs because securing the pumps may worsen the voiding in the reactor vessel head.

K/A
008 Pressurizer Vapor Space Accident
AA2.09 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Consequences of loss of pressure in RCS; methods for evaluating pressure loss.

K/A MATCH ANALYSIS
The stem provides indications for verifying the method of pressure loss is in the vapor space of the pressurizer. The question tests the consequences of this pressure loss - RCP tripping. Question is SRO level because it tests the reasons for securing RCPs.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. RCPs are not secured to reduce pressure.
B. Correct. Pumps must be secured iaw E-1 Step 1. Per referenced lesson plan, pumps are secured because PCT may be higher if pumps are secured at lower pressures.
C. Incorrect. Pumps must be secured iaw E-1 Step 1. Plausible because there is no break in the hot or cold legs when you may have an inventory loss concern in those areas. An applicant may think that it would make sense to keep the pumps running when the break is located in the upper portions of the system.
D. Incorrect. Pumps must be secured iaw E-1 Step 1. Plausible for same reasons as 'C' above. In addition, head voiding is a concern, and keeping the RCPs running would likely lessen the voiding.

REFERENCES
1. 19010, E-1 Loss of Reactor or Secondary Coolant, Rev. 28.2, 09/27/2002.
2. LO-LP-37111, Operator Response To Loss Primary Coolant, Rev. 15, 08/29/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B C C B D A D D C A Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	RCP VAPOR SPACE LOCA	Cog Level:	C/A 4.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

3. 008AA2.22 001

The following conditions exist on Unit 1:

- A reactor trip and safety injection have occurred.
- Pressurizer pressure is 1300 psig and lowering.
- Pressurizer Code Safety Temperature Indicator, TI-0466, is rising.
- PRT level, temperature, and pressure are rising.
- All safety equipment has functioned as expected.

The operating crew has just entered 19010-C, E-1 Loss of Reactor or Secondary Coolant.

Which ONE of the following correctly states procedurally directed requirements for securing (or not securing) the RCPs and the reasons for the actions?

A. Secure RCPs to reduce pressure in the vessel due to pressurized thermal shock concerns.

B.✓ Secure RCPs due to peak clad temperature concerns.

C. Do not secure RCPs because securing the pumps may raise the fuel rod peak clad temperature.

D. Do not secure the RCPs because securing the pumps may worsen the voiding in the reactor vessel head.

K/A
008 Pressurizer Vapor Space Accident
AA2.09 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Consequences of loss of pressure in RCS; methods for evaluating pressure loss.

K/A MATCH ANALYSIS
The stem provides indications for verifying the method of pressure loss is in the vapor space of the pressurizer. The question tests the consequences of this pressure loss - RCP tripping. Question is SRO level because it tests the reasons for securing RCPs.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. RCPs are not secured to reduce pressure.
B. Correct. Pumps must be secured iaw E-1 Step 1. Per referenced lesson plan, pumps are secured because PCT may be higher if pumps are secured at lower pressures.
C. Incorrect. Pumps must be secured iaw E-1 Step 1. Plausible because there is no break in the hot or cold legs when you may have an inventory loss concern in those areas. An applicant may think that it would make sense to keep the pumps running when the break is located in the upper portions of the system.
D. Incorrect. Pumps must be secured iaw E-1 Step 1. Plausible for same reasons as 'C' above. In addition, head voiding is a concern, and keeping the RCPs running would likely lessen the voiding.

REFERENCES
1. 19010, E-1 Loss of Reactor or Secondary Coolant, Rev. 28.2, 09/27/2002.
2. LO-LP-37111, Operator Response To Loss Primary Coolant, Rev. 15, 08/29/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B C C B D A D D C A Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	RCP VAPOR SPACE LOCA	Cog Level:	C/A 4.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. 008AA2.22 001

The following conditions exist on Unit 1:

- A reactor trip and safety injection have occurred.
- Pressurizer pressure is 1300 psig and lowering.
- Pressurizer Code Safety Temperature Indicator, TI-0466, is rising.
- PRT level, temperature, and pressure are rising.
- All safety equipment has functioned as expected.

The operating crew has just entered 19010-C, E-1 Loss of Reactor or Secondary Coolant.

Which ONE of the following correctly states procedurally directed requirements for securing (or not securing) the RCPs and the reasons for the actions?

A. Secure RCPs to reduce pressure in the vessel due to pressurized thermal shock concerns.

B✓ Secure RCPs due to peak clad temperature concerns.

C. Do not secure RCPs because securing the pumps may raise the fuel rod peak clad temperature.

D. Do not secure the RCPs because securing the pumps may worsen the voiding in the reactor vessel head.

K/A
008 Pressurizer Vapor Space Accident
AA2.09 Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Consequences of loss of pressure in RCS; methods for evaluating pressure loss.

K/A MATCH ANALYSIS
The stem provides indications for verifying the method of pressure loss is in the vapor space of the pressurizer. The question tests the consequences of this pressure loss - RCP tripping. Question is SRO level because it tests the reasons for securing RCPs.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. RCPs are not secured to reduce pressure.
B. Correct. Pumps must be secured iaw E-1 Step 1. Per referenced lesson plan, pumps are secured because PCT may be higher if pumps are secured at lower pressures.
C. Incorrect. Pumps must be secured iaw E-1 Step 1. Plausible because there is no break in the hot or cold legs when you may have an inventory loss concern in those areas. An applicant may think that it would make sense to keep the pumps running when the break is located in the upper portions of the system.
D. Incorrect. Pumps must be secured iaw E-1 Step 1. Plausible for same reasons as 'C' above. In addition, head voiding is a concern, and keeping the RCPs running would likely lessen the voiding.

REFERENCES
1. 19010, E-1 Loss of Reactor or Secondary Coolant, Rev. 28.2, 09/27/2002.
2. LO-LP-37111, Operator Response To Loss Primary Coolant, Rev. 15, 08/29/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B C C B D A D D C A Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	RCP VAPOR SPACE LOCA	Cog Level:	C/A 4.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	Operator Response To A Loss Of Primary Coolant	NUMBER:	LO-LP-37111-15-C
PROGRAM:	Licensed Operator	REVISION:	15
SME:	Perry Tucker	DATE:	August 23, 2000
APPROVED:	***D. Scukanec***	DATE:	8/29/2000

INSTRUCTOR GUIDELINES:

I. FORMAT

A. Verbal lecture with visual aids

II. MATERIALS

A. Overhead projector

B. Transparencies

C. White board with markers

III. EVALUATION

A. Oral or written exam in conjunction with other lesson plans

IV. REMARKS

A. Ensure student has latest revision of EOP

2) Containment sump levels increasing

3) Containment humidity increasing

4) PRT level increasing

5) Containment radiation increasing

e. SGTR identified by use of secondary rad monitors — Covered in another lesson plan

2. Operator enters 19000-C (E-0) — LO-TP-37111-003

a. Actions

1) Performs immediate actions

2) Verify secondary status of plant systems for proper response to emergency

3) Diagnoses cause as LOCA and transitions to 19010-C — Objective 5

a) Known leakage from unisolable PORV

b) CNMT conditions

c) Failure to stabilize PZR pressure/level after ECCS flow reduced in E-0.

b. It does not matter if the cause of the emergency is known prior to entering procedures

1) E-0 must be entered first to verify proper ESF equipment actuation

3. Operator transitions to 19010-C, "Loss of Coolant Accident"

a. 19010-C is used to mitigate loss of primary or secondary coolant

b. Used to place the plant in a stable condition for further recovery action

C. Major Action Steps of 19010-C (general: not step-by-step) — Objective 8

1. Monitor plant equipment for optimal mode of operation — LO-TP-37111-002

a. Check if RCP's should be stopped — Note: Ensure students know how to perform each step of procedure

b. Verify readiness/operation of

1) Containment spray pumps

2) Diesels

3) SG feedwater source (AFW)

2. Check for equipment failure

a. Check for secondary break (requires transition) to 19020-C.

b. Check for SGTR (requires transition to 19030-C)

c. Check for pressurizer steam space leaks

3. Determine optimal method of long-term plant recovery

a. SI termination criteria met

1) Go to 19011-C, "SI termination"

b. SI termination criteria not met either

1) Go to 19012-C "Post LOCA CD and Depressurization" or,

2) Stay in 19010-C until directed to 19013-C, "Cold Leg Recirculation"

D. Key Utility Decisions

1. Upper head venting

a. May be required after plant placed in long-term cold recirculation mode

b. Performed to limit the buildup of hydrogen which could preclude core cooling

2. Once in long-term recirculation, plant personnel will develop and write subsequent recovery procedures

E. Special Considerations

1. Step 1 - RCP trip criteria

Objective 6

a. Tripping RCPs may enhance recovery for some special cases

Commitment per Procedures 60601, 60602 4.1.3.3b (1) (a), GL 85.012

1) Small break LOCA requires early tripping of RCP's

2) Reduces inventory loss caused by forced circulation

3) Results in lower peak clad temperature than if pumps were to trip later during accident

b. Criteria

1) At least one CCP or SIP operating

2) Less or equal to 1375 psig and decreasing and,

c. RCP's must not be stopped if ECCS pumps are not available to ensure cooling flow into the core

Ensure student understands INTENT of step (ECCS flow established)

1) RCP's force 2 phase flow through core.

2. ACCW pump verification (Step 3)

a. Ensure completion of the step. If an ACCW pump is already running, there is no need to start a second pump.

3. Steps 4 through 7

a. Verifies secondary system intact

b. Verifies primary to secondary boundary intact

c. Verifies PORV no steam space LOCA

1) Primary PORV or safety

2) Note step 7a RNO intent is to unisolate a PORV if it was isolated only for excessive seat leakage

d. Verifies PORVs available for pressure relief

1) COPS armed if WR CL temps < 350° F

Cold Overpressure protection

2) COPS <u>may</u> be disarmed if temp subsequently increases to > 350° F and the CSFST has remained Green throughout the event.

4. Reset SI

a. Reset SI means you block the signal, you haven't realigned any systems

4. 010G2.2.22 001

The following Unit 1 conditions exist:

- Pressurizer pressure is 2750 psig.
- Plant is in Mode 3.

Which ONE of the following correctly states the required Technical Specification action and its basis?

A. Restore pressurizer pressure within 1 hour because reactor coolant pressure boundary design conditions will not be exceeded since credit is taken in the Technical Specification Bases for pressurizer power operated relief valve automatic operation.

B. Restore pressurizer pressure within 1 hour because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

C. Restore pressurizer pressure within 5 minutes because reactor coolant pressure boundary design conditions will not be exceeded since credit is taken in the Technical Specification Bases for pressurizer power operated relief valve automatic operation.

D✓ Restore pressurizer pressure within 5 minutes because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

K/A
010 Pressurizer Pressure Control
G2.2.22 Knowledge of limiting conditions for operations and safety limits.

K/A MATCH ANALYSIS
The question tests knowledge related to the TS required actions associated with controlling pressure below the Safety Limit. The question is SRO-only knowledge because the TS Basis must be known in order to correctly answer the question.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Restore P w/i 5 min, not 1 hour. Plausible because 1 hour is correct if in Mode 1 or 2.
B. Incorrect. Restore P w/i 5 min, not 1 hour. Plausible because 1 hour is correct if in Mode 1 or 2.
C. Incorrect. PORVs are not credited in the Tech Spec Basis.
D. Correct. See TS Basis Page B2.1.2-1 and TS Section 2.0.

REFERENCES
1. Technical Specification Section 2.0, Safety Limits (SLs).
2. Technical Specification Basis Section 2.0, SLs.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D B B B A C C C D B	Scramble Range: A - D	
Tier:	2			Group:	1	
Key Word:	SAFETY LIMIT			Cog Level:	MEM 4.1	
Source:	N			Exam:	VG05301	
Test:	S			Author/Reviewer:	MAB/RSB	

4. 010G2.2.22 001

The following Unit 1 conditions exist:

- Pressurizer pressure is 2750 psig.
- Plant is in Mode 3.

Which ONE of the following correctly states the required Technical Specification action and its basis?

A. Restore pressurizer pressure within 1 hour because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and postulated accidents (PAs).

B. Restore pressurizer pressure within 1 hour because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

C. Restore pressurizer pressure within 5 minutes because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and postulated accidents (PAs).

D. ✓ Restore pressurizer pressure within 5 minutes because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

K/A
010 Pressurizer Pressure Control
G2.2.22 Knowledge of limiting conditions for operations and safety limits.

K/A MATCH ANALYSIS
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ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Restore P w/i 5 min, not 1 hour. Plausible because 1 hour is correct if in Mode 1 or 2 and a PA is another category of accident that must be analyzed.
B. Incorrect. Restore P w/i 5 min, not 1 hour. Plausible because 1 hour is correct if in Mode 1 or 2.
C. Incorrect. AOO is correct, not PA. Plausible because a PA is another category of accident that must be analyzed.
D. Correct. See TS Basis Page B2.1.2-1 and TS Section 2.0.

REFERENCES
1. Technical Specification Section 2.0, Safety Limits (SLs).
2. Technical Specification Basis Section 2.0, SLs.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D B B B A C C C D B	Scramble Range: A - D	
Tier:	2		Group:	1		
Key Word:	SAFETY LIMIT		Cog Level:	MEM 4.1		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

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B. Restore pressurizer pressure within 1 hour because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

C. Restore pressurizer pressure within 5 minutes because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and postulated accidents (PAs).

D✓ Restore pressurizer pressure within 5 minutes because reactor coolant pressure boundary design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs).

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010 Pressurizer Pressure Control
G2.2.22 Knowledge of limiting conditions for operations and safety limits.

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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D B B B A C C C D B	Scramble Range: A - D
Tier:	2		Group:	1	
Key Word:	SAFETY LIMIT		Cog Level:	MEM 4.1	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

2.2.3 Within 1 hour notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.4 Within 24 hours, notify the General Manager-Nuclear Plant and Vice President-Nuclear.

2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared and submitted to the NRC pursuant to 10 CFR 50.73.

2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

(continued)

5. 011EA2.08 001

The plant staff is recovering from a large break loss of coolant accident that occurred 5 hours ago. The break is thought to be on one of the cold legs. The control room staff is evaluating conditions for entry into 19014-C, ES-1.4 Transfer to Hot Leg Recirculation, in accordance with 19010-C, E-1 Loss of Reactor or Secondary Coolant. Plant conditions appear to be stable as the TSC and operating crew are making their evaluations of long term mitigation strategies.

Which ONE of the following correctly describes conditions needed to go to hot leg recirculation and the reason?

A. Enter 19014-C in 1.5 hours to flush high concentration boric acid in the reactor vessel out the break and back to the sump.

B.✓ Enter 19014-C in 2.5 hours to flush high concentration boric acid in the reactor vessel out the break and back to the sump.

C. Enter 19014-C in 1.5 hours to refill the reactor vessel and downcomer.

D. Enter 19014-C in 2.5 hours to refill the reactor vessel and downcomer.

K/A
011 Large Break LOCA
EA2.08 Ability to determine or interpret the following as they apply to a Large Break LOCA: Conditions necessary for recovery when accident reaches stable conditions.

K/A MATCH ANALYSIS
The plant is stable, several hours after a LBLOCA. The crew has a condition on time that they must meet in order to go to HL Recirc. The question is SRO level because the reason for performing the step is also tested.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. E-1 states that HL recirc is to be conducted at 7.5 hours after the accidient. The accident occurred 5 hours ago, so 5 + 1.5 is only 6.5 hours after the accident. Plausible because E-1, Step 20 states that the crew should prepare for HL Recirc at 6.5 hours after accident.
B. Correct. 5 hours + 2.5 hours = 7.5 hours (E-1, Step 21). LO-LP-37114-14, Page 4, provides the correct reason.
C. Incorrect. E-1 states that HL recirc is to be conducted at 7.5 hours after the accidient. The accident occurred 5 hours ago, so 5 + 1.5 is only 6.5 hours after the accident. Plausible because water is helping to refill the vessel.
D. Incorrect. Boron plate-out is correct, not filling the downcomer (which is on the cold leg side). Plausible because water is helping to refill the vessel.

REFERENCES
1. Farley 1995 Exam Question 009K3.21.
2. LO-LP-37114-14, Transfer To Hot Leg Recirculation, and Loss of Emergency Coolant Recirculation, Rev. 14, 11/08/2000 (OBJ #9).
3. 19010-C, E-1 Loss of Reactor or Secondary Coolant, Rev. 28.2.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B B C D D D A C B D	Scramble Range: A - D
Tier:	I		Group:	I	
Key Word:	HOT LEG RECIRC		Cog Level:	C/A 3.9	
Source:	M		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

5. 011EA2.08 001

The plant staff is recovering from a large break loss of coolant accident that occurred 5 hours ago. The break is thought to be on one of the cold legs. The control room staff is evaluating conditions for entry into 19014-C, ES-1.4 Transfer to Hot Leg Recirculation, in accordance with 19010-C, E-1 Loss of Reactor or Secondary Coolant. Plant conditions appear to be stable as the TSC and operating crew are making their evaluations of long term mitigation strategies.

Which ONE of the following correctly describes conditions needed to go to hot leg recirculation and the reason?

A. Enter ~~ES-1.4~~ 19014 in 1.5 hours to flush high concentration boric acid in the reactor vessel out the break and back to the sump.

B. ✓ Enter ES-1.4 in 2.5 hours to flush high concentration boric acid in the reactor vessel out the break and back to the sump.

C. Enter ES-1.4 in 1.5 hours to cool the upper portions of the core and reactor vessel.

D. Enter ES-1.4 in 2.5 hours to cool the upper portions of the core and reactor vessel.

→ Enter ES 1.4 in 2.5 hours to refill the Rx vessel & downcomer.

Incorrect: Downcomer is on the cold leg side. It should already be full.

K/A
011 Large Break LOCA
EA2.08 Ability to determine or interpret the following as they apply to a Large Break LOCA: Conditions necessary for recovery when accident reaches stable conditions.

K/A MATCH ANALYSIS
The plant is stable, several hours after a LBLOCA. The crew has a condition on time that they must meet in order to go to HL Recirc. The question is SRO level because the reason for performing the step is also tested.

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D. Incorrect. Boron plate-out is correct, not core or vessel cooling. Plausible because cooler water would be entering these areas while on HL Recirc.

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2. LO-LP-37114-14, Transfer To Hot Leg Recirculation, and Loss of Emergency Coolant Recirculation, Rev. 14, 11/08/2000 (OBJ #9).
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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B B C D D D A C B D	Scramble Range: A - D	
Tier:	1		Group:	1		
Key Word:	HOT LEG RECIRC		Cog Level:	C/A 3.9		
Source:	M		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

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The plant staff is recovering from a large break loss of coolant accident that occurred 5 hours ago. The break is thought to be on one of the cold legs. The control room staff is evaluating conditions for entry into 19014-C, ES-1.4 Transfer to Hot Leg Recirculation, in accordance with 19010-C, E-1 Loss of Reactor or Secondary Coolant. Plant conditions appear to be stable as the TSC and operating crew are making their evaluations of long term mitigation strategies.

Which ONE of the following correctly describes conditions needed to go to hot leg recirculation and the reason?

A. Enter ES-1.4 in 1.5 hours to flush high concentration boric acid in the reactor vessel out the break and back to the sump.

B✓ Enter ES-1.4 in 2.5 hours to flush high concentration boric acid in the reactor vessel out the break and back to the sump.

C. Enter ES-1.4 in 1.5 hours to cool the upper portions of the core and reactor vessel.

D. Enter ES-1.4 in 2.5 hours to cool the upper portions of the core and reactor vessel.

K/A
011 Large Break LOCA
EA2.08 Ability to determine or interpret the following as they apply to a Large Break LOCA: Conditions necessary for recovery when accident reaches stable conditions.

K/A MATCH ANALYSIS
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ANSWER / DISTRACTOR ANALYSIS
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3. 19010-C, E-1 Loss of Reactor or Secondary Coolant, Rev. 28.2.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B B C D D D A C B D Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	HOT LEG RECIRC	Cog Level:	C/A 3.9
Source:	M	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	Transferring To Hot Leg Recirc, And Loss Of Emergency Coolant Recirculation	NUMBER:	LO-LP-37114-14
PROGRAM:	Licensed Operator	REVISION:	14
SME:	Perry Tucker	DATE:	August 8, 2000
APPROVED:	***D. Scukanec***	DATE:	8/9/2000

INSTRUCTOR GUIDELINES:

I. FORMAT

A. Verbal lecture with visual aids

II. MATERIALS

A. Overhead projector
B. Transparencies
C. White board with markers

III. EVALUATION

A. Oral or written exam in conjunction with other lesson plans

IV. REMARKS

A. Ensure students have correct revision of EOP

III. LESSON OUTLINE	NOTES
I. INTRODUCTION	LO-TP-37114-001
A. 19014-C provides the necessary instructions for transferring the ECCS from cold leg recirculation to hot leg recirculation	Objective 6
B. Hot leg recirculation is used to:	Objective 9
1. Prevent exceeding the boric acid solubility limit during a cold leg break.	
a. Reactor vessel acts a concentrator for the boric acid in the ECCS water injecting into the core during a cold leg break	
1) ECCS in, steam out	
b. Dilute (from the containment sump) water is injected through the hot legs to flush some of the high concentration boric acid in the reactor vessel out the break back into the sump.	
2. If exceeded causes overheating by.	
a. Fouling heat transfer surfaces	
b. Blocking flow passages	
C. Hot leg recirc is entered 7.5 hours after event initiation	Objective 10
It is entered from 19010-C Step 20 when time requirements have been satisfied	
Based on TSC decision, it might also be entered, after transferring to cold-leg recirculation, from 19012-C, 19131-C, 19132-C	
D. System Lineup	LO-TP-37114-002 Objective 8
1. RHR discharges to RCS via common HL injection path (HV-8840) to hot legs 1 and 4	
2. SI trains realigned to all hot legs	
a. If both trains cannot be aligned, then the train that cannot will be left in cold leg recirc mode	
3. CCPs remain aligned to the cold legs	
II. PRESENTATION	
A. Major Action Steps of 19014-C	LO-TP-37114-003 Objective 7

EMERGENCY OPERATING PROCEDURE

E-1 LOSS OF REACTOR OR SECONDARY COOLANT

PURPOSE

PRB REVIEW REQUIRED

This procedure provides actions to recover from a loss of reactor or secondary coolant. (Applicable in Modes 1, 2, and 3)

MAJOR ACTIONS

- ◆ Monitor Plant Equipment for Optimal Mode of Operations
- ◆ Check for Subsequent Failure
- ◆ Determine Optimal Method of Long-Term Plant Recovery

ENTRY CONDITIONS

- 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION, Steps 22, 28, 33, and 38.
- 19005-C, ES-0.0 REDIAGNOSIS, Step 4
- 19011-C, ES-1.1 SI TERMINATION, Steps 10 and 27.
- 19020-C, E-2 FAULTED STEAM GENERATOR ISOLATION, Step 8.
- 19102-C, ECA-0.2 LOSS OF ALL AC POWER RECOVERY WITH SI REQUIRED, Step 15.
- 19112-C, ECA-1.2 LOCA OUTSIDE CONTAINMENT, Step 4.
- 19121-C, ECA-2.1 UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, Step 6.
- 19221-C, FR-C.1 RESPONSE TO INADEQUATE CORE COOLING, Steps 17 and 25.
- 19222-C, FR-C.2 RESPONSE TO DEGRADED CORE COOLING, Step 19.
- 19231-C, FR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK, Steps 27, 29, and 32.
- 19262-C, FR-I.2 RESPONSE TO LOW PRESSURIZER LEVEL, Step 4.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*20. At 6.5 Hours after event initiation, PREPARE for hot leg recirculation:	
a. Place the lockout selector switches for the following valves in the ON position and verify power to valves: • HV-8840 - RHR TO HL ISO VLV • HV-8809A - RHR PMP-A TO COLD LEG 1&2 ISO VLV • HV-8809B - RHR PMP-B TO COLD LEG 3&4 ISO VLV • HV-8802A - SI PMP-A TO HOT LEG 1&4 ISO VLV • HV-8802B - SI PMP-B TO HOT LEG 2&3 ISO VLV • HV-8835 - CL INJ FROM SIS	
b. Verify power to the following valves - VALVE POSITION INDICATORS LIT: • HV-8716A - RHR TRAIN A TO HOT LEG CROSSOVER ISO • HV-8716B - RHR TRAIN B TO HOT LEG CROSSOVER ISO • HV-8821A - SI PMP-A TO COLD LEG ISO VLV • HV-8821B - SI PMP-B TO COLD LEG ISO VLV	b. Close circuit breakers as necessary.
*21. At 7.5 Hours after event initiation, GO TO 19014-C, ES-1.4 TRANSFER TO HOT LEG RECIRCULATION.	

..000009.K3.21 | 10/23/1995

Farley 1 | **Exam Level** S

Mark Question | Print Recor | New Searc | Exit

Question

Which ONE of the following describes the basis for shifting the SI mode from cold leg recirculation to hot leg recirculation approximately 11 hours following a LOCA?.

Answer: Flush out the boron that may have plated out on the fuel rods at the top of the core so that heat transfer is improved.

Distracter 1 Equalize thermal stress on the reactor vessel, the head, core barrel, and steam generator tube sheet.

Distracter 2 Equalize flow through the reactor vessel to ensure adequate cooling of the core for any resultant core geometry.

Distracter 3 Cool the upper portions of the core and reactor vessel and equalize thermal stress on the steam generator.

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3:

6. 016A2.03 001

Pressurizer Water Level Indicator, LI-459A, becomes inoperable due to a signal interruption from the level transmitter (LT-459). No operator actions have been taken to address the failure.

Which ONE of the following correctly states the Reactor Trip System (RTS) Instrumentation operability requirements in accordance with Technical Specifications and the basis?

A. LCO 3.3.1 RTS Instrumentation is met because only two channels are required to be operable for the Reactor Trip System Instrumentation. The system is designed so that with the slow rate of charging available, pressure overshoot due to pressurizer level channel failure will cause the safety valve to lift before the reactor high pressure trip.

B. LCO 3.3.1 RTS Instrumentation is met because only two channels are required to be operable for the Reactor Trip System Instrumentation. The system is designed to prevent water relief through the pressurizer safety valves.

C. LCO 3.3.1 RTS Instrumentation is not met because three channels are required to be operable for the RTS Instrumentation. The system is designed so that with the slow rate of charging available, pressure overshoot due to pressurizer level channel failure will cause the safety valve to lift before the reactor high pressure trip.

D✓ LCO 3.3.1 RTS Instrumentation is not met because three channels are required to be operable for the RTS Instrumentation. The system is designed to prevent water relief through the pressurizer safety valves.

K/A
016 Non-nuclear Instrumentation
A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

K/A MATCH ANALYSIS
The pzr level instrumentation is NNI. The impacts of one channel being inoperable are that the TS LCO is not met for RTS Instrumentation. Knowing that the LCO is not met is the first step in applying TS in order to maintain the plant within its design basis. Therefore the question is testing knowledge necessary to mitigate the consequences of having the channel inoperable. The question is SRO level because it requires basis knowledge in order to arrive at the correct answer.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Three channels are required to be operable. Plausible because only two channels are required to be operable for PAM Instrumentation.
B. Incorrect. Three channels are required to be operable. Plausible because only two channels are required to be operable for PAM Instrumentation.
C. Incorrect. System is designed so that safety valves cannot lift before reactor trip. Plausible because three channels are required to be operable.
D. Correct. See TS 3.3.1 and its Basis.

REFERENCES
1. Technical Specification 3.3.1, Reactor Trip System Instrumentation.
2. Technical Specification 3.3.1 Basis.
3. Technical Specification 3.3.3, Post Accident Monitoring (PAM) Instrumentation.
4. V-LO-TX-28101, RPS-SSPS-AMSAC, Rev. 3.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D C B A D C B B A B Scramble Range: A - D

Tier:	2	Group:	2
Key Word:	PRESSURIZER LEVEL	Cog Level:	C/A 3.3
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

6. 016A2.03 001

Pressurizer Water Level Indicator, LI-459A, becomes inoperable due to a signal interruption from the level transmitter (LT-459). No operator actions have been taken to address the failure.

Which ONE of the following correctly states the Reactor Trip System (RTS) Instrumentation operability requirements in accordance with Technical Specifications and the basis?

A. LCO 3.3.1 RTS Instrumentation is met because only two channels are required to be operable for the Reactor Trip System Instrumentation. The system is designed so that with the slow rate of charging available, pressure overshoot due to pressurizer level channel failure will cause the safety valve to lift before the reactor high pressure trip.

B. LCO 3.3.1 RTS Instrumentation is met because only two channels are required to be operable for the Reactor Trip System Instrumentation. The system is designed to prevent water relief through the pressurizer safety valves.

C. LCO 3.3.1 RTS Instrumentation is not met because three channels are required to be operable for the RTS Instrumentation. The system is designed so that with the slow rate of charging available, pressure overshoot due to pressurizer level channel failure will cause the safety valve to lift before the reactor high pressure trip.

D✓ LCO 3.3.1 RTS Instrumentation is not met because three channels are required to be operable for the RTS Instrumentation. The system is designed to prevent water relief through the pressurizer safety valves.

K/A
016 Non-nuclear Instrumentation
A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

K/A MATCH ANALYSIS
The pzr level instrumentation is NNI. The impacts of one channel being inoperable are that the TS LCO is not met for RTS Instrumentation. Knowing that the LCO is not met is the first step in applying TS in order to maintain the plant within its design basis. Therefore the question is testing knowledge necessary to mitigate the consequences of having the channel inoperable. The question is SRO level because it requires basis knowledge in order to arrive at the correct answer.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Three channels are required to be operable. Plausible because only two channels are required to be operable for PAM Instrumentation.
B. Incorrect. Three channels are required to be operable. Plausible because only two channels are required to be operable for PAM Instrumentation.
C. Incorrect. System is designed so that safety valves cannot lift before reactor trip. Plausible because three channels are required to be operable.
D. Correct. See TS 3.3.1 and its Basis.

REFERENCES
1. Technical Specification 3.3.1, Reactor Trip System Instrumentation.
2. Technical Specification 3.3.1 Basis.
3. Technical Specification 3.3.3, Post Accident Monitoring (PAM) Instrumentation.
4. V-LO-TX-28101, RPS-SSPS-AMSAC, Rev. 3.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D C B A D C B B A B Scramble Range: A - D

Tier:	2	Group:	2
Key Word:	PRESSURIZER LEVEL	Cog Level:	C/A 3.3
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. 016A2.03 001

Pressurizer Water Level Indicator, LI-459A, becomes inoperable due to a signal interruption from the level transmitter (LT-459). No operator actions have been taken to address the failure.

Which ONE of the following correctly states the Reactor Trip System (RTS) Instrumentation operability requirements in accordance with Technical Specifications and the basis?

A. LCO 3.3.1 RTS Instrumentation is met because only two channels are required to be operable for the Reactor Trip System Instrumentation. The system is designed so that with the slow rate of charging available, pressure overshoot due to pressurizer level channel failure will cause the safety valve to lift before the reactor high pressure trip.

B. LCO 3.3.1 RTS Instrumentation is met because only two channels are required to be operable for the Reactor Trip System Instrumentation. The system is designed to prevent water relief through the pressurizer safety valves.

C. LCO 3.3.1 RTS Instrumentation is not met because three channels are required to be operable for the RTS Instrumentation. The system is designed so that with the slow rate of charging available, pressure overshoot due to pressurizer level channel failure will cause the safety valve to lift before the reactor high pressure trip.

D.✓ LCO 3.3.1 RTS Instrumentation is not met because three channels are required to be operable for the RTS Instrumentation. The system is designed to prevent water relief through the pressurizer safety valves.

K/A
016 Non-nuclear Instrumentation
A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.

K/A MATCH ANALYSIS
The pzr level instrumentation is NNI. The impacts of one channel being inoperable are that the TS LCO is not met for RTS Instrumentation. Knowing that the LCO is not met is the first step in applying TS in order to maintain the plant within its design basis. Therefore the question is testing knowledge necessary to mitigate the consequences of having the channel inoperable. The question is SRO level because it requires basis knowledge in order to arrive at the correct answer.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Three channels are required to be operable. Plausible because only two channels are required to be operable for PAM Instrumentation.
B. Incorrect. Three channels are required to be operable. Plausible because only two channels are required to be operable for PAM Instrumentation.
C. Incorrect. System is designed so that safety valves cannot lift before reactor trip. Plausible because three channels are required to be operable.
D. Correct. See TS 3.3.1 and its Basis.

REFERENCES
1. Technical Specification 3.3.1, Reactor Trip System Instrumentation.
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MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D C B A D C B B A B Scramble Range: A - D

Tier:	2	Group:	2
Key Word:	PRESSURIZER LEVEL	Cog Level:	C/A 3.3
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

K4 = Nominal set point of 109.5%

K5 = Modifier for Tavg change ≈ 2% RTP per °F (penalty above 588.4°F)/ (**no reward** given to set point below 588.4°F

K6 = Modifier for Temperature = 0.177% per psig (penalty below 2235 psig)/(reward given to set point above 2235 psig)

T = measured loop specific RCS average temperature in °F

T' = indicated loop specific RCS average temperature at RTP, ≤588.4°F

f2 (AFD) = modifier for Axial Flux Difference (**No penalty for AFD**)

A reactor trip will occur if 2 out of the 4 loop ΔTs increase to the variable OPΔT set point.

Bases: Provides assurance of fuel integrity under overpower conditions by ensuring the allowable kW/ft heat generation rate is not exceeded (particularly under conditions of steam line break)

9) **<u>High Pressurizer pressure trip</u>** — **2** out of **4** channels — **≥ 2385 psig**

Bases: Protects against over pressurization of the RCS in conjunction with PROVs and Code Safety Valves.

10) **<u>Low Pressurizer Pressure trip</u>** — **2** out of **4** channels — **≤ 1960 psig**
enabled above P-7; ≥ 10% reactor power

Bases: Provides protection against DNB.

11) **<u>High Pressurizer level trip</u>** — **2** out of **3** channels — **≥ 92% level**
enabled above P-7; ≥ 10% reactor power

Bases: Prevents water relief through the safeties.

12) **<u>Single loop loss of flow</u>** — **2** out of **3** channels — **≤ 90% flow**
1 out of **4** loops
enabled above P-8; ≥ 48% reactor power

Bases: Prevents DNB conditions resulting from loss of one or more reactor coolant pumps.

13) **<u>Two loop loss of flow</u>** — **2** out of **3** channels — **≤ 90% flow ****
2 out of **4** loops
enabled above P-7; ≥ 10% reactor power

Bases: Prevents DNB conditions resulting from loss of two or more reactor coolant pumps.

Table 3.3.3-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITIONS
1. Reactor Coolant System (RCS) Pressure (wide range)	2	B,G,H,J
2. RCS T_{hot} (wide range)	1/loop	C,G,H,J
3. RCS T_{cold} (wide range)	1/loop	D,G,H,J
4. Steam Generator (SG) Water Level (wide range)	1/SG	E,G,H,J
5. SG Water Level (narrow range)	2/SG	B,G,H,J
6. Pressurizer Level	2	B,G,H,J
7. Containment Pressure	2	B,G,H,J
8. Steam line Pressure	2/steam line	B,G,H,J
9. Refueling Water Storage Tank (RWST) Level	2	B,G,H,J
10. Containment Normal Sumps Level (narrow range)	2	B,G,H,J
11. Containment Water Level (wide range)	2	B,G,H,J
12. Condensate Storage Tank Level	2/tank[(a)]	B,G,H,J
13. Auxiliary Feedwater Flow	2/SG	B,G,H,J
14. Containment Radiation Level (high range)	2	B,G,H,K
15. Steam line Radiation Monitor	1/steam line	F,G,H,J
16. RCS Subcooling	2	B,G,H,J
17. Neutron Flux (extended range)	2	B,G,H,J
18. Reactor Vessel Water Level (RVLIS)	2	B,G,H,K
19. Hydrogen Monitors	2	B,G,I,J
20. Containment Pressure (extended range)	2	B,G,H,J
21. Containment Isolation Valve Position	2/penetration flow path[(b) (c)]	B,G,H,J
22. Core Exit Temperature - Quadrant 1	2[(d)]	B,G,H,J
23. Core Exit Temperature - Quadrant 2	2[(d)]	B,G,H,J
24. Core Exit Temperature - Quadrant 3	2[(d)]	B,G,H,J
25. Core Exit Temperature - Quadrant 4	2[(d)]	B,G,H,J

(a) Only required for the OPERABLE tank.

(b) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation phase A or containment ventilation isolation signals).

(c) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(d) A channel consists of two core exit thermocouples (CETs).

Table 3.3.1-1 (page 3 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT$^{(n)}$
8. Pressurizer Pressure						
a. Low	$1^{(f)}$	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1950 psig	$1960^{(g)}$ psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2395 psig	2385 psig
9. Pressurizer Water Level - High	$1^{(f)}$	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.9%	92%
10. Reactor Coolant Flow - Low						
a. Single Loop	$1^{(h)}$	3 per loop	N	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.4%	90%
b. Two Loops	$1^{(i)}$	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.4%	90%

(continued)

(f) Above the P-7 (Low Power Reactor Trips Block) interlock.

(g) Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 10 seconds for lead and 1 second for lag.

(h) Above the P-8 (Power Range Neutron Flux) interlock.

(i) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

(n) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is readjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Pressurizer Pressure — High (continued)

minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure — High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure — High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level — High

(LI-0459A, LI-460A, LI-0461A)

NOTE: Pressurizer Water Level channels are also required OPERABLE by the Post Accident Monitoring Technical Specification. Setpoints are given in percent of instrument span.

The Pressurizer Water Level — High trip Function provides a backup signal for the Pressurizer Pressure — High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level — High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Pressurizer Water Level — High (continued)

set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for over filling the pressurizer, the Pressurizer Water Level — High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow — Low

(LOOP 1	LOOP 2	LOOP 3	LOOP 4
FI-0414	FI-0424	FI-0434	FI-0444
FI-0415	FI-0425	FI-0435	FI-0445
FI-0416	FI-0426	FI-0436	FI-0446)

NOTE: The setpoints are given in percent of Loop flow.

a. Reactor Coolant Flow — Low (Single Loop)

The Reactor Coolant Flow — Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

(continued)

7. 022G2.1.12 001

The following Unit 1 conditions exist:

- Tavg = 300 °F
- Both CCPs have been declared inoperable due to a common cause

Which ONE of the following correctly states the required actions in accordance with Technical Specifications and the basis for the actions?

A.✓ Restore a CCP to operable status within 1 hour because the plant is not prepared to provide a high pressure response to Design Basis Events requiring Safety Injection.

B. Restore a CCP to operable status within 72 hours because at least 100% of ECCS flow equivalent to a single OPERABLE ECCS train is available.

C. Restore a CCP to operable status within 1 hour to regain protection from a single failure disabling the ECCS.

D. Enter LCO 3.0.3 due to no ECCS trains being operable to provide a high pressure response to Design Basis Events requiring Safety Injection.

K/A
022 Loss of Rx Coolant Makeup
G2.1.12 Ability to apply technical specifications for a system.

K/A MATCH ANALYSIS
The High Pressure SI, provided by the CCPs, falls into the category of Inventory Control. Therefore, RCS makeup is encompassed. The question tests Technical Specifications, which completes the K/A match. The question tests TS Basis knowledge, which makes it SRO level and it concerns a 1 hour Action Statement, which makes it required closed-book knowledge.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Retstoring a CCP to OPERABLE would allow the action statement to be completed. The reason given is taken out of the TS Basis Document.
B. Incorrect. 72 hours would only be applicable if 100% of ECCS flow were available. Plausible because 72 hours would be the case if 100% of ECCS flow were available.
C. Incorrect. Single failure criteria does not apply, per the TS Basis. Restoring a CCP would not place the plant in a situation where they would be single-failure proof. A single failure could still eliminate all high head SI. Plausible because single failure criteria applies in most other conditions where TS are used.
D. Incorrect. The TS cover the current situation. Plausible because no ECCS trains are operable.

REFERENCES
1. Technical Specifications 3.5.2, ECCS - Operaing.
2. Technical Specifications 3.5.2, Basis.
3. Technical Specifications 3.5.3, ECCS - Shutdown.
4. Technical Specifications 3.5.3, Basis.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D B C C D B B A B	Scramble Range: A - D
Tier:	1		Group:	1	
Key Word:	TS ECCS TECH SPEC		Cog Level:	C/A 4.0	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

7. 022G2.1.12 001

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3. Technical Specifications 3.5.3, ECCS - Shutdown.
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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A D B C C D B B A B	Scramble Range: A - D	
Tier:	1		Group:	1		
Key Word:	TS ECCS TECH SPEC		Cog Level:	C/A 4.0		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

1. 022G2.1.12 001

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4. Technical Specifications 3.5.3, Basis.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A D B C C D B B A B		Scramble Range: A - D
Tier:	1			Group:	1	
Key Word:	TS ECCS TECH SPEC			Cog Level:	C/A 4.0	
Source:	N			Exam:	VG05301	
Test:	S			Author/Reviewer:	MAB/RSB	

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 One ECCS train shall be OPERABLE.

APPLICABILITY: MODE 4. — Hot Shutdown 200 - 350 °F Tavg

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ECCS residual heat removal (RHR) subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately
B. Required ECCS centrifugal charging subsystem inoperable. <u>AND</u> At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	B.1 Restore required ECCS centrifugal charging subsystem to OPERABLE status.	72 hours
C. Required ECCS centrifugal charging subsystem inoperable.	C.1 Restore required ECCS centrifugal charging subsystem to OPERABLE status.	1 hour
D. Required Actions and associated Completion Times of Conditions B or C not met.	D.1 Be in MODE 5.	24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.3.1	------------------------NOTE------------------------ An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. --- The following SRs are applicable for all equipment required to be OPERABLE: SR 3.5.2.3 SR 3.5.2.7 SR 3.5.2.4	In accordance with applicable SRs

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS — Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS — Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) or containment emergency sump can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

(continued)

BASES

LCO (continued)

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor

(continued)

BASES

ACTIONS

A.1 (continued)

decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

With the required ECCS centrifugal charging subsystem inoperable, and at least 100% of the ECCS flow equivalent to a singe OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. Since the 72 hour Completion Time is acceptable when the unit is in MODES 1, 2, and 3 (Ref. 5) and MODE 4 represents less severe conditions for the initiation of a LOCA, the 72 hour Completion Time is also acceptable for MODE 4. This allows increased flexibility in plant operations under circumstances when components in the required train may be inoperable, but ECCS remains capable of delivering 100% of the required flow.

C.1

With no ECCS centrifugal charging subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS centrifugal charging subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

(continued)

BASES

ACTIONS (continued)

D.1

When the Required Actions of Conditions B or C cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 for SRs 3.5.2.3, 3.5.2.4 and 3.5.2.7 apply. Note that these Surveillance descriptions were written for a specification that is applicable in MODEs 1, 2, and 3, and SR 3.5.3.1 is applicable for MODE 4. However, the descriptions provided for SRs 3.5.2.3, 3.5.2.4, and 3.5.2.7 are applicable to MODE 4 as well. SR 3.5.3.1 is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4, if necessary.

REFERENCES

The applicable references from Bases 3.5.2 apply.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

-----NOTES-----

1. In MODE 3, either residual heat removal pump to cold legs injection flow path may be isolated by closing the isolation valve to perform pressure isolation valve testing per SR 3.4.14.1.

2. Operation in MODE 3 with ECCS pumps declared inoperable pursuant to LCO 3.4.12, "Cold Overpressure Protection System (COPS)," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds 375°F, whichever comes first.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. <u>AND</u> At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with the power lockout switches in the lockout position.		12 hours
Valve Number	Valve Function	Valve Position	
HV-8835 HV-8840 HV-8813 HV-8806 HV-8802A, B HV-8809A, B	SI Pump Cold Leg Inj. RHR Pump Hot Leg Inj. SI Pump Mini Flow Isol. SI Pump Suction from RWST SI Pump Hot Leg Inj. RHR Pump Cold Leg Inj.	OPEN CLOSED OPEN OPEN CLOSED OPEN	
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.		31 days
SR 3.5.2.3	Verify ECCS piping is full of water.		31 days
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.		In accordance with the Inservice Testing Program
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.		18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.7	Verify, by visual inspection, each ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	18 months

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS — Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system;

b. Rod ejection accident;

c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and

d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

There are three phases of ECCS operation: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the refueling water storage tank (RWST) and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment sumps have enough water to supply the required net positive suction head to the ECCS pumps, suction is switched to the containment sump for cold leg recirculation. After approximately 7.5 hours, the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the potential for boiling in the top of the core and ensure boron precipitation never occurs.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the RWST are also part of the

(continued)

BASES

BACKGROUND (continued)

ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, the RHR pumps, heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. Separate piping supplies each subsystem and each train within the subsystem. The discharge from the centrifugal charging pumps combines and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI pumps combines and divides into four supply lines, each of which feeds an injection line to one RCS cold leg. The discharge from each RHR pump feeds two injection lines each. Throttle valves for the CCP and SI injection lines are set to balance the flow to the RCS. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation is through the same paths as the injection phase. Subsequently, recirculation can be accomplished by injection into both the hot and cold legs.

(continued)

BASES

BACKGROUND (continued)

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start immediately in the programmed sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement for runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in a small break LOCA event. This event establishes the flow and discharge head at the design point for the centrifugal charging pumps. The SGTR and MSLB events also credit the centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

a. A large break LOCA event, with loss of offsite power and a single failure disabling one ECCS train; and

b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small LOCA to maintain core subcriticality. For smaller LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and automatically transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS — Shutdown."

As indicated in Note 1, either flow path may be isolated in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room.

As indicated in Note 2, operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Cold Overpressure Protection System (COPs)," is necessary since the arming temperature is the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the COPs arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level."

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8. 026G2.4.48 001

Unit 1 was operating at full power for 200 days and was preparing to shutdown for a refueling outage next week when an RCS leak developed. The plant shut down 20 hours ago and is currently cooling down to cold shutdown.

Current plant conditions are:

- The Unit RCS Temperature is 335 °F and RHR Train "A" shutdown cooling is in service.
- RHR train "B" is aligned for ECCS injection.
- RCPs 1, 2, and 3 are shut down
- ALB 008-D06, CCW TRAIN A RHR HX LO FLOW, is in alarm.
- Operators inform the SRO that the CCW Pump #1 has tripped and CCW Pump #5 cannot be started.
- The SRO declares CCW train "A" inoperable.

Which ONE of the following correctly describes actions that are required to be taken by the crew in response to the above conditions?

A. Do not align RHR Train B for shutdown cooling until RCS temperature is less than 250 °F. Continue the cooldown to 250 °F using the available steam generator(s). Immediately take action to restore one additional RCS loop to operation.

B. Immediately align RHR Train B for shutdown cooling and immediately take action to restore one additional RCS loop to operation.

C✓ Do not align RHR Train B for shutdown cooling until RCS temperature is less than 250 °F. Continue the cooldown to 250 °F using the available steam generator(s).

D. Immediately align RHR Train B for shutdown cooling, but action to restore one additional RCS loop to operation is not required.

K/A
026 Loss of Component Cooling Water
G2.4.48 Ability to interpret control room indications to verify the status and operation of the system, and understand how operator actions and directives affect plant and system conditions.

K/A MATCH ANALYSIS
The alarm is a control room indication (maybe the CCW flow also). The CCW system is in an abnormal configuration with flow at 4000 gpm to the RHR HX (partial loss of CCW). The Question also requires RHR/CCW system design knowledge for the applicant to know what actions are needed to accomplish the cooldown.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. TS 3.4.6 RCS loops Operable in Mode 4 is already met with 1 RCS loop operating and RHR B operable. Plausible because with RHR aligned for ECCS injection, the applicant may mistakenly believe it cannot be used to meet the minimum loops operable requirement in TS 3.4.6.
B. Incorrect. See C below for SR 3.5.3.1 and 12006-C guidance. While RHR in the shutdown cooling mode will remain operable for ECCS injection mode, 12006 provides more restrictive limits on single RHR Train operation in SDC mode above 250F that must be followed. Plausible because relying on TS 3.5.3 alone would allow the only remaining RHR train to be aligned for SDC in mode 4. See C below for TS 3.4.6 operability. Plausible because, with RHR aligned for ECCS injection, the candidate may mistakenly believe it cannot be used to meet the minimum loops operable requirement in TS 3.4.6.
C. Correct. While RHR train B is in SDC mode, it will remain operable for ECCS injection in this mode. 12006 provides more restrictive limits on single train RHR operation in the SDC mode above 250F that must be followed. 12006-C directs continuing the cooldown using SGs in lieu of the only available RHR train until temp is less than 250F, which meets the more restrictive limitations imposed by 12006-C to ensure operability as ECCS source is maintained. One RHR loop operable/operating and one RCS loop operating meet the requirements of TS 3.4.6, RCS loops - Mode 4.
D. Incorrect. While RHR Train B in the SDC mode will remain operable for ECCS injection in this mode, 12006 provides more restrictive limits on the single RHR train operation in SDC mode above 250F that must be followed. Plausible because relying on the TS 3.5.3 alone would allow only remaining RHR train to be aligned for SDC in mode 4.

REFERENCES
1. V-LO-TX-12101, Residual Heat Removal System, Rev. 1.0.
2. 1A1-D06, CCW TRAIN A RHR HX LO FLOW, Rev. 15, 03/26/2004.
3. 12006-C, Unit Cooldown To Cold Shutdown, Rev. 64, 05/12/2004.
4. 13011-1, Residual Heat Removal System, Rev. 56, 05/20/2004.
5. 13715-1, Component Cooling Water System, Rev. 18, 08/06/2003.

QUESTIONS REPORT
for Voglte 2005-301 Draft

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9

Answer: C A A B D C C B C A Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	CCW RHR COOLDOWN	Cog Level:	C/A 3.8
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

8. 026G2.4.48 001

Unit 1 was operating at full power for 200 days and was preparing to shutdown for a refueling outage next week when an RCS leak developed. The plant shut down 20 hours ago.

The following conditions exist following the plant shutdown:

- The Unit is at 335 °F and "A" Train Shutdown Cooling is in service.
- CCW temperature at the outlet of the CCW heat exchanger is 105 °F.
- Management wants the plant in cold shutdown within 3 hours.
- 1A1-D06, CCW TRAIN A RHR HX LO FLOW, is in alarm.
- Operators inform the SRO that the CCW flow through the "A" RHR Heat Exchanger is 4000 gpm due to a CCW pump trip and failure of the standby pump to start.

Which ONE of the following correctly states the ability of the plant to accomplish the desired cooldown?

A. The plant, in its current configuration, can get to cold shutdown within 3 hours.

B. The plant, in its current configuration, cannot accomplish the cooldown within 3 hours. Manually starting available CCW pumps and raising the CCW flow through the "A" RHR Heat Exchanger to 5000 gpm will allow cold shutdown to be reached within 3 hours.

C✓ The plant, in its current configuration, cannot accomplish the cooldown within 3 hours. Manually starting available CCW pumps and raising the CCW flow through the "A" RHR Heat Exchanger to 5000 gpm and placing a second train of RHR in service will allow cold shutdown to be reached within 3 hours.

D. The plant, even with starting additional CCW pumps and placing both trains of RHR in service, will take approximately 20 hours to get to cold shutdown.

K/A
026 Loss of Component Cooling Water
G2.4.48 Ability to interpret control room indications to verify the status and operation of the system, and understand how operator actions and directives affect plant and system conditions.

K/A MATCH ANALYSIS
The alarm is a control room indication (maybe the CCW flow also). The CCW system is in an abnormal configuration with flow at 4000 gpm to the RHR HX (partial loss of CCW). The Question also requires RHR/CCW system design knowledge for the applicant to know what actions are needed to accomplish the cooldown.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Is not possible. See referenced system description. Plausible because 3 hours is possible, but not under the current plant configuration.
B. Incorrect. Is not possible. See referenced system description. Plausible because this action would raise the cooldown capability, but it would still not be enough.
C. Correct. See referenced system description. Have utility verify that I am understanding the system description correctly because the description is not that clear.
D. Incorrect. Not true. See referenced system description. Plausible because 20 hours is correct for another scenario stated in system description.

REFERENCES
1. V-LO-TX-12101, Residual Heat Removal System, Rev. 1.0.
2. 1A1-D06, CCW TRAIN A RHR HX LO FLOW, Rev. 15, 03/26/2004.
3. 12006-C, Unit Cooldown To Cold Shutdown, Rev. 64, 05/12/2004.
4. 13011-1, Residual Heat Removal System, Rev. 56, 05/20/2004.
5. 13715-1, Component Cooling Water System, Rev. 18, 08/06/2003.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C A A B D C C B C A	Scramble Range: A - D	
Tier:	1		Group:	1		
Key Word:	CCW RHR COOLDOWN		Cog Level:	C/A 3.8		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

1. 026G2.4.48 001

Unit 1 was operating at full power for 200 days and was preparing to shutdown for a refueling outage next week when an RCS leak developed. The plant shut down 20 hours ago.

The following conditions exist following the plant shutdown:

- The Unit is at 335 °F and "A" Train Shutdown Cooling is in service.
- CCW temperature at the outlet of the CCW heat exchanger is 105 °F.
- Management wants the plant in cold shutdown within 3 hours.
- 1A1-D06, CCW TRAIN A RHR HX LO FLOW, is in alarm.
- Operators inform the SRO that the CCW flow through the "A" RHR Heat Exchanger is 4000 gpm due to a CCW pump trip and failure of the standby pump to start.

Which ONE of the following correctly states the ability of the plant to accomplish the desired cooldown?

A. The plant, in its current configuration, can get to cold shutdown within 3 hours.

B. The plant, in its current configuration, cannot accomplish the cooldown within 3 hours. Manually starting available CCW pumps and raising the CCW flow through the "A" RHR Heat Exchanger to 5000 gpm will allow cold shutdown to be reached within 3 hours.

C✓ The plant, in its current configuration, cannot accomplish the cooldown within 3 hours. Manually starting available CCW pumps and raising the CCW flow through the "A" RHR Heat Exchanger to 5000 gpm and placing a second train of RHR in service will allow cold shutdown to be reached within 3 hours.

D. The plant, even with starting additional CCW pumps and placing both trains of RHR in service, will take approximately 20 hours to get to cold shutdown.

K/A
026 Loss of Component Cooling Water
G2.4.48 Ability to interpret control room indications to verify the status and operation of the system, and understand how operator actions and directives affect plant and system conditions.

K/A MATCH ANALYSIS
The alarm is a control room indication (maybe the CCW flow also). The CCW system is in an abnormal configuration with flow at 4000 gpm to the RHR HX (partial loss of CCW). The Question also requires RHR/CCW system design knowledge for the applicant to know what actions are needed to accomplish the cooldown.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Is not possible. See referenced system description. Plausible because 3 hours is possible, but not under the current plant configuration.
B. Incorrect. Is not possible. See referenced system description. Plausible because this action would raise the cooldown capability, but it would still not be enough.
C. Correct. See referenced system description. Have utility verify that I am understanding the system description correctly because the description is not that clear.
D. Incorrect. Not true. See referenced system description. Plausible because 20 hours is correct for another scenario stated in system description.

REFERENCES
1. V-LO-TX-12101, Residual Heat Removal System, Rev. 1.0.
2. 1A1-D06, CCW TRAIN A RHR HX LO FLOW, Rev. 15, 03/26/2004.
3. 12006-C, Unit Cooldown To Cold Shutdown, Rev. 64, 05/12/2004.
4. 13011-1, Residual Heat Removal System, Rev. 56, 05/20/2004.
5. 13715-1, Component Cooling Water System, Rev. 18, 08/06/2003.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	C A A B D C C B C A	Scramble Range: A - D	
Tier:	1			Group:	1	
Key Word:	CCW RHR COOLDOWN			Cog Level:	C/A 3.8	
Source:	N			Exam:	VG05301	
Test:	S			Author/Reviewer:	MAB/RSB	

V-LO-TX-12101

6.3 DESIGN SUMMARY

The RHR system and support systems have been designed for an operational cooling rate of up to 100F/h not to exceed 100 Deg F in any 1 hour period.

The FSAR states:

" For purposes of designing the major RCS components, the plant heat up and cool down operations are conservatively represented by continuous 100F/h ramp temperature changes, between the shutdown temperature of 120F and the no-load temperature of 557F (for the pressurizer vessel, the design cool down rate is 200F/h). The number of plant heat up and cool down operations is defined as 200 each, which corresponds to 5 occurrences per year for the 40-year plant design life."

The RHRS is typically placed in operation approximately 2 to 4 hours after reactor shutdown and Mode 4 entry when the temperature and pressure of the RCS are less than **350F and 365 psig**, respectively. Assuming that two heat exchangers and two pumps are in service and that each heat exchanger is supplied with 5000 gal/min CCW initially at 105 Deg F, the RHRS is designed to reduce the temperature of the reactor coolant to 140 Deg F within 20 hours following reactor shutdown. Under these conditions, the time required to reduce the reactor coolant temperature from 350F to 200F is approximately 3 hours. The heat load handled by the RHRS during the cool down transient includes residual and decay heat from the core and RCP heat. The design heat load is based on the decay heat fraction that exists at 20 hours following reactor shutdown from an extended run at full power.

200°F is Cold S/D.

Assuming that only one heat exchanger and pump are in service and that the heat exchanger is supplied with CCW at 5000 gal/min and initially at 105DegF, the RHRS is capable of reducing the temperature of the reactor coolant from *350F to 200F within approximately 30 h.* The time required under these conditions to reduce reactor coolant temperature from 350F to 212F is approximately 20 h.

The RHRS has a design rating of **600 psig/400 Deg F**, well below the rating of the RCS. The RHRS is isolated from the RCS Hot Legs on the suction side by two motor-operated valves in series on each suction line. **Each MOV is interlocked with a pressure transmitter to prevent its opening if RCS pressure is greater than approximately 365 psig**. A control board annunciator will alert the operator if one or both of the loop suction valves are not fully closed and the RCS pressure exceeds 420 psig. The RHRS is isolated from the RCS cold legs on the discharge side by two series check valves and one MOV in each return line.

PRB REVIEW REQUIRED

UNIT COOLDOWN TO COLD SHUTDOWN

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	**ALL**
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	**NONE**
Information Use:	Available on plant site for reference as needed.	**NONE**

INITIALS

b. UNIT 2:

(1) Train A: 2HS-6506A in DISABLE (Aux Relay Panel 2ACPAR6) ____/____ IV

(2) Train B: 2HS-6506B in DISABLE (Aux Relay Panel 2BCPAR7) ____/____ IV

(3) Train C: OPEN Breaker 2CD1M05 (Steam Admission Valve, 2HV-5106) ____/____ IV

(4) Train C: OPEN 2CD1M05-K2 Link ____/____ IV

c. PLACE standby MDAFW Pumps hand switch in PULL-TO-LOCK: ________

(1) MDAFW pumps can be used to feed SGs as required by either starting the applicable pump or by opening the pump discharge cross connect valve.

(2) If pump(s) and/or the cross connect valve are used as described above, document status with CAUTION TAG.

d. If the TDAFW Pump is not in use, CLOSE HV-5122, 5125, 5127 and 5120. ________

CAUTION

If only one train of RHR is available, the only available RHR train should remain aligned for ECCS injection and the cool down to 250°F continued using the steam generators.

C4.2.12 If desired, when the RCS pressure is less than 365 psig (PI-0403 and/or PI-0405), and RCS temperature is less than 340°F, PLACE only one RHR Loop in operation per 13011, "Residual Heat Removal System". (CO 147, CO 1407, CO 1716, CO 3289, CO 3301, CO 3314, CO 44125) ________

a. OPERATE RHR HX Outlet Valve, HV-0606(0607), and Bypass Valve, FV-0618(0619), to control RCS temperature as necessary and RHR flow at a minimum total flow of 3000 gpm, ________

COMPONENT COOLING WATER SYSTEM

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	**ALL**
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	**NONE**
Information Use:	Available on plant site for reference as needed.	**NONE**

0 **PURPOSE**

This procedure provides instructions for the operation of the Component Cooling Water (CCW) System. Instructions are included in the following sections:

4.1.1 CCW Train A Startup To Standby

4.1.2 CCW Train B Startup To Standby

4.1.3 CCW Train A Startup From Standby

4.1.4 CCW Train B Startup From Standby

4.2.1 Shifting Train A CCW Pumps

4.2.2 Shifting Train B CCW Pumps

4.3.1 CCW Train A Shutdown To Standby

4.3.2 CCW Train B Shutdown To Standby

4.4.1 Alternate Makeup To Train A(B) CCW Surge Tank

4.4.2 CCW Chemical Addition

4.4.3 CCW Train A Fill And Vent

4.4.4 CCW Train B Fill And Vent

4.4.5 Train A CCW System Single Pump Operation

4.4.6 Train B CCW System Single Pump Operation

4.4.7 Lowering of CCW Surge Tank Level With Effluent Going To Clean Water Sump

4.4.8 Lowering of CCW Surge Tank Level With Effluent Going To CCW Drain Tank.

2.0 **PRECAUTIONS AND LIMITATIONS**

2.1 **PRECAUTIONS**

2.1.1 Thoroughly fill and vent all applicable CCW components prior to returning them to service after maintenance. This minimizes system performance degradation due to gas entrainment.

2.1.2 To prevent overheating of CCW when the Residual Heat Removal (RHR) System is placed in service, CCW flow through the RHR Heat Exchangers should not be throttled for temperature control.

WINDOW D06

ORIGIN	SETPOINT	
1-FSHL-1928	4500 gpm	CCW TRAIN A RHR HX LO FLOW

1.0 PROBABLE CAUSE

Component Cooling Water System pipe break or malfunction.

2.0 AUTOMATIC ACTIONS

NONE

3.0 INITIAL OPERATOR ACTIONS

GO to 18020-1, "Loss Of Component Cooling Water".

4.0 SUBSEQUENT OPERATOR ACTIONS

NONE

5.0 COMPENSATORY OPERATOR ACTIONS

NONE

END OF SUB-PROCEDURE

REFERENCES: 1X4DB137, 1X3D-BD-L01A, CX5DT101-93

9. 029EA2.05 001

Unit 1 has exceeded the High Pressure Reactor Trip Setpoint, but the reactor did not trip. The crew is performing 19211, FR-S.1 Response To Nuclear Power Generation / ATWT, to initiate emergency boration. The crew notes that the Pressurizer pressure is 2400 psig and control rods have not yet been inserted. Both Pressurizer PORV valve position indicating green lights are on and red lights are off.

Which ONE of the following procedural actions should the SRO direct and what are the reasons for those actions?

A. Do not open the Pressurizer PORVs due to DNB concerns.

B. Do not open the Pressurizer PORVs due to containment atmosphere concerns.

C. Open the Pressurizer PORVs and reduce pressure to avoid exceeding ASME pressure vessel code requirements.

D✓ Open the Pressurizer PORVs and reduce pressure to allow for adequate boration flow to the core.

K/A
029 ATWS
EA2.05 Ability to determine or interpret the following as they apply to an ATWS: System component valve position indications.

K/A MATCH ANALYSIS
PORVs are indicating closed and they should be open, and are procedurally required to be open, in order to ensure adequate boration flow. The question is SRO level because it tests the basis behind the step.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. PORVs are required to be open (19211-C, Step 5). Plausible because DNB is more likely as pressure lowers.
B. Incorrect. PORVs are required to be open (19211-C, Step 5). Plausible because opening PORVs will send RCS water to PRT which has a rupture disk.
C. Incorrect. Safety Valves will protect vessel and the procedure requires the PORVs to be opened to raise boration flow, not protect the vessel. Plausible because there are high pressure requirements for the vessel.
D. Correct. See 19211-C and Referenced lesson plan.

REFERENCES
1. 19211-C, FR-S.1 Response To Nuclear Power Generation / ATWT, Rev. 16, 02/25/2004.
2. LO-LP-37041, Anticipated Transient Without Trip, Rev. 12, 12/29/1999.
3. V-LO-TX-16001, Primary Systems, Rev. 3.0.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D A B D B A C D A D	Scramble Range: A - D	
Tier:	1		Group:		1	
Key Word:	ATWS ATWT PORV		Cog Level:		C/A 3.4	
Source:	N		Exam:		VG05301	
Test:	S		Author/Reviewer:		MAB/RSB	

9. 029EA2.05 001

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B. Do not open the Pressurizer PORVs due to containment atmosphere concerns.

C. Open the Pressurizer PORVs and reduce pressure to avoid exceeding ASME pressure vessel code requirements.

D✓ Open the Pressurizer PORVs and reduce pressure to allow for adequate boration flow to the core.

K/A
029 ATWS
EA2.05 Ability to determine or interpret the following as they apply to an ATWS: System component valve position indications.

K/A MATCH ANALYSIS
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C. Incorrect. Safety Valves will protect vessel and the procedure requires the PORVs to be opened to raise boration flow, not protect the vessel. Plausible because there are high pressure requirements for the vessel.
D. Correct. See 19211-C and Referenced lesson plan.

REFERENCES
1. 19211-C, FR-S.1 Response To Nuclear Power Generation / ATWT, Rev. 16, 02/25/2004.
2. LO-LP-37041, Anticipated Transient Without Trip, Rev. 12, 12/29/1999.
3. V-LO-TX-16001, Primary Systems, Rev. 3.0.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D A B D B A C D A D	Scramble Range: A - D
Tier:	1		Group:	1	
Key Word:	ATWS ATWT PORV		Cog Level:	C/A 3.4	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

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C. Open the Pressurizer PORVs and reduce pressure to avoid exceeding ASME pressure vessel code requirements.

D✓ Open the Pressurizer PORVs and reduce pressure to allow for adequate boration flow to the core.

K/A
029 ATWS
EA2.05 Ability to determine or interpret the following as they apply to an ATWS: System component valve position indications.

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C. Incorrect. Safety Valves will protect vessel and the procedure requires the PORVs to be opened to raise boration flow, not protect the vessel. Plausible because there are high pressure requirements for the vessel.
D. Correct. See 19211-C and Referenced lesson plan.

REFERENCES
1. 19211-C, FR-S.1 Response To Nuclear Power Generation / ATWT, Rev. 16, 02/25/2004.
2. LO-LP-37041, Anticipated Transient Without Trip, Rev. 12, 12/29/1999.
3. V-LO-TX-16001, Primary Systems, Rev. 3.0.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D A B D B A C D A D	Scramble Range: A - D	
Tier:	1			Group:	1	
Key Word:	ATWS ATWT PORV			Cog Level:	C/A 3.4	
Source:	N			Exam:	VG05301	
Test:	S			Author/Reviewer:	MAB/RSB	

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	Anticipated Transient Without Trip	NUMBER:	LO-LP-37041-12-C
PROGRAM:	Licensed Operator	REVISION:	12
SME:	Perry Tucker	DATE:	December 2, 1999
APPROVED:	***D. Scukanec***	DATE:	12/29/1999

INSTRUCTOR GUIDELINES:

I. LESSON FORMAT

A. Verbal lecture with visual aids

II. MATERIALS

A. Overhead projector

B. Transparencies

C. White board with markers

III. EVALUATION

A. Written or oral exam in conjunction with other lesson plans

IV. REMARKS

A. Ensure students have latest revision of EOP

B. Performance-based instructional units (IUs) are attached to the lesson plan as student handouts. After the lecture on Anticipated Transient Without Trip, the student should be given adequate self-study time for the IUs. The instructor should direct self-study activities and be available to answer questions that may arise concerning the IU material. After self-study, the student will perform, simulate, observe, or discuss (as identified on the cluster signoff criteria list) the task covered in the instructional unit in the presence of an evaluator.

makeup for lost coolant

6) Containment pressure remains within limits and minimum DNBR reached is 1.6

e. Uncontrolled rod withdrawal — LO-TP-37041-006

1) Primarily DNBR challenge

2) Result from rod control system malfunction

3) Increase nuclear power beyond what secondary is carrying away

4) This raises Tavg, pressurizer level, and pressure

5) Core power in the W analyzed transient went to 113% and was terminated by Doppler, moderator feedback, and rod motion Out limits

6) PORVs opened and maintained peak pressure below 2413#. However, the pressurizer did not go solid

7) Minimum calculated DNBR was 1.54

6. ATWT Transient Concerns

a. NRC ATWT performance criteria:

1) ASME pressure vessel code not exceeded, ie. does not exceed RCS pressure safety limit.

2) Fuel integrity maintained, ie. does not exceed minimum DNBR/Reactor Core safety limit.

3) No excessive radiation releases.

4) No containment failure.

5) Long term cooling and shutdown maintained.

6) Also resulted in AMSAC.

b. ATWT transient response varies depending on type of transient and operator actions, as well as the time in core life at which the ATWT occurs. (BOL has more severe overpressure transient.)

Note: If procedure not exited at step 4 it MUST be completed in its entirety (thru step 13)

4. Step 5 - Emergency borate the RCS.

   a.. Next to rods, most direct manner of adding negative reactivity to the core.

   b. RCS pressure checked to ensure adequate boration flow.

      1) PORV setpoint chosen as pressure at which flow is insufficient.

      2) 200 psi decrease adequate to allow increased injection flow.

5. Step 6 - CVI aligned to verify a barrier to radiation release in case the PRT rupture disc bursts.

6. Step 7 - Local Trip of Reactor and Turbine if not previously tripped

   a. Caution: Directions for verification of the first 16 steps of 19000-C if an SI signal or actuation occurs.

      1) This should be done by the operators without hindering the implementation of this procedure

   b. Note: 1. Time for EPIP implementation
      2. P-4 interlock problems with SI reset/ block, FWI & Steam Dumps

   c. Delayed until this step because they are more time consuming

   d. First try local tripping of RTB's and Bypass Brks and if this fails then trip the MG sets locally

   e. If turbine not tripped then dispatch operator to locally trip it.

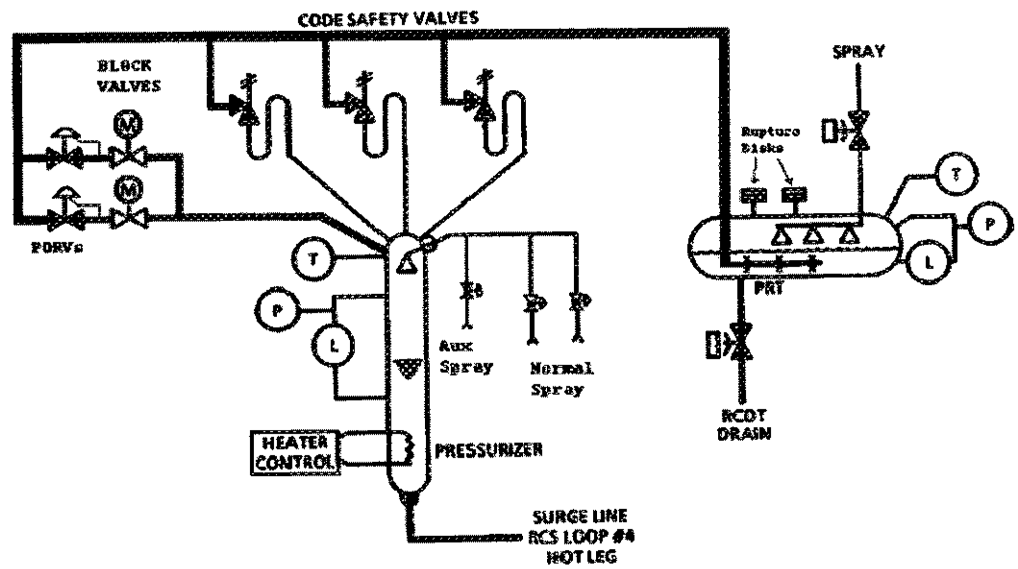
7. Step 8 - SG Inventory/AFW Flow Verification

   a. For return to criticality events, narrow range level may be maintained at lower flow rates, so the higher flow rate is not necessary if NR level is on scale at 10% in any one SG. If this procedure was entered due to an ATWT instead of a return to criticality then SG levels will be off scale low.

   b. Note that AFW flow should be greater than 1260 gpm, not 570 gpm as in other EOP's, if SG's <10% NR (32% adverse)

16-31 Safety Valves

The three pressurizer safety valves are spring loaded, enclosed, quick-acting type valves. They are set to open at a pressure of 2460 psig and have a combined capacity of 1.26 million pounds mass per hour. The capacity is equal to or greater than the maximum credible RCS surge rate into the pressurizer. The design basis for the relief capacity is the complete loss of load without a reactor trip. The safety valves discharge to the pressurizer relief tank. Steam or hydrogen leakage past the valve seats is inhibited by a loop seal in the piping from each safety valve. Condensation collects in the loop seal. The collection of water in the seals prevents the hydrogen from passing out of the pressurizer. The loop seals are provided with drain lines which are routed to the pressurizer relief tank.



16-32 POWER OPERATED RELIEF AND ISOLATION VALVES

A 6-inch relief line is attached to the upper head of the pressurizer. The line divides into two parallel 3-inch lines, each containing a power operated relief (PORV) and motor operated isolation valve. The two PORVs are solenoid operated steam ported valves controlled from the QMCB and remote shutdown panels. PORV PV-455 relief set point is 2345 psig and PORV PV-456 relief set point is set at 2335 psig. Each valve has a capacity of 210,000 lbs/hr. Actuation pressure is set to prevent operation of the pressurizer safety valves. A normally open motor operated isolation valve called "block valves" are located upstream of each relief valve. It is closed when necessary to isolate a PORV because of leakage. Isolation of the relief valves is allowed because the pressurizer safety valves provide the RCS with sufficient protection in the event of an accident. The PORV's were not taken credit for in the accident analysis.

Downstream of the power-operated relief valves, the two 3-inch lines combine into a common discharge line from the safety valves and dumps to the pressurizer relief tank.

In addition to the function of providing overpressure protection for the pressurizer and RCS when operating, the PORVs also protect the RCS when the plant is shut down and water solid. This is accomplished by varying the relief opening set point of the PORVs to open. Based on certain conditions, the set point will change. This is accomplished by the Cold Overpressure Protection System.

10. 035G2.1.20 001

The following Unit 1 conditions exist:

- The reactor is at 10% power.
- SG # 1 Primary to Secondary Leakage = 0.5 gpm
- SG # 2 Primary to Secondary Leakage = 0.1 gpm
- SG # 3 Primary to Secondary Leakage = 0.1 gpm
- SG # 4 Primary to Secondary Leakage = 0.2 gpm
- Pressurizer PORV-456A is leaking to the PRT at 1.2 gpm

Which ONE of the following correctly states the status of Technical Specification LCO 3.4.13, RCS Operational LEAKAGE, and the basis?

A.✓ Primary to Secondary Leakage limit is exceeded. The limit is based on the assumption that a single crack leaking at the maximum allowed rate would not propogate to a SGTR under the stress conditions of a LOCA.

B. Primary to Secondary Leakage limit is exceeded. The limit is to ensure that 10 CFR Part 20 - Standards For Protection Against Radiation, requirements are not violated.

C. Unidentified Leakage limit is exceeded. The limit is to prevent continued degradation of the Reactor Coolant Pressure Boundary.

D. Unidentified Leakage limit is exceeded. The limit is to ensure pressurizer level will be maintained in the event of a reactor trip.

K/A
035 Steam Generator
G2.1.20 Ability to execute procedure steps.

K/A MATCH ANALYSIS
Tech Specs can be considered a procedure that is used by the operators. The question tests the knowledge of whether or not a limit is violated and what the basis of the limit is. The applicant must have this knowledge in order to have the ability to execute the Tech Specs. Testing the basis info makes it SRO-only level.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. 0.5 gpm = 720 gpd, which is greater than TS limit.
B. Incorrect. Incorrect basis given. Plausible because it is partially correct.
C. Incorrect. All leakage listed in stem is identified leakage. Plausible because applicant may think that PORV leakage is unidentified and at 1.2 gpm, this would exceed the limit.
D. Incorrect. All leakage listed in stem is identified leakage. Plausible for same reason as C above.

REFERENCES
1. Technical Specification 3.4.13, Operational Leakage.
2. Technical Specification 3.4.13 Basis.
3. Vogtle Exam Bank Question LO-LP-16001-07-05.
4. Vogtle Exam Bank Question LO-LP-39202-01-21 (Just verifies that facility considers PORV leakage as identified.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A B D D B C B C A C	Scramble Range: A - D
Tier:	2		Group:	2	
Key Word:	TECH SPEC SG STEAM G		Cog Level:	C/A 4.2	
Source:	M		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

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K/A MATCH ANALYSIS
Tech Specs can be considered a procedure that is used by the operators. The question tests the knowledge of whether or not a limit is violated and what the basis of the limit is. The applicant must have this knowledge in order to have the ability to execute the Tech Specs. Testing the basis info makes it SRO-only level.

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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A B D D B C B C A C	Scramble Range: A - D
Tier:	2		Group:	2	
Key Word:	TECH SPEC SG STEAM G		Cog Level:	C/A 4.2	
Source:	M		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

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- SG # 1 Primary to Secondary Leakage = 0.5 gpm
- SG # 2 Primary to Secondary Leakage = 0.1 gpm
- SG # 3 Primary to Secondary Leakage = 0.1 gpm
- SG # 4 Primary to Secondary Leakage = 0.2 gpm
- Pressurizer PORV-456A is leaking to the PRT at 1.2 gpm

Which ONE of the following correctly states the status of Technical Specification LCO 3.4.13, RCS Operational LEAKAGE, and the basis?

A.✓ Primary to Secondary Leakage limits are exceeded. These limits are to ensure acceptable offsite doses in the event of a steam line break.

B. Primary to Secondary Leakage limits are exceeded. These limits are to ensure steam generator tube integrity is maintained.

C. Unidentified Leakage limits are exceeded. These limits are to prevent continued degradation of the Reactor Coolant Pressure Boundary.

D. Unidentified Leakage limits are exceeded. These limits are to ensure pressurizer level will be maintained in the event of a reactor trip.

K/A
035 Steam Generator
G2.1.20 Ability to execute procedure steps.

K/A MATCH ANALYSIS
Tech Specs can be considered a procedure that is used by the operators. The question tests the knowledge of whether or not a limit is violated and what the basis of the limit is. The applicant must have this knowledge in order to have the ability to execute the Tech Specs. Testing the basis info makes it SRO-only level.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. 0.5 gpm = 720 gpd, which is greater than TS limit.
B. Incorrect. Incorrect basis given. Plausible because it is partially correct.
C. Incorrect. All leakage listed in stem is identified leakage. Plausible because applicant may think that PORV leakage is unidentified and at 1.2 gpm, this would exceed the limit.
D. Incorrect. All leakage listed in stem is identified leakage. Plausible for same reason as C above.

REFERENCES
1. Technical Specification 3.4.13, Operational Leakage.
2. Technical Specification 3.4.13 Basis.
3. Vogtle Exam Bank Question LO-LP-16001-07-05.
4. Vogtle Exam Bank Question LO-LP-39202-01-21 (Just verifies that facility considers PORV leakage as identified.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A B D D B C B C A C	Scramble Range: A - D	
Tier:	2		Group:	2		
Key Word:	TECH SPEC SG STEAM G		Cog Level:	C/A 4.2		
Source:	M		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

1. 035G2.1.20 001

The following Unit 1 conditions exist:

- The reactor is at 10% power.
- SG # 1 Primary to Secondary Leakage = 0.5 gpm
- SG # 2 Primary to Secondary Leakage = 0.1 gpm
- SG # 3 Primary to Secondary Leakage = 0.1 gpm
- SG # 4 Primary to Secondary Leakage = 0.2 gpm
- Pressurizer PORV-456A is leaking to the PRT at 1.2 gpm

Which ONE of the following correctly states the status of Technical Specification LCO 3.4.13, RCS Operational LEAKAGE, and the basis?

A.✓ Primary to Secondary Leakage limits are exceeded. These limits are to ensure acceptable offsite doses in the event of a steam line break.

B. Primary to Secondary Leakage limits are exceeded. These limits are to ensure steam generator tube integrity is maintained.

C. Unidentified Leakage limits are exceeded. These limits are to prevent continued degradation of the Reactor Coolant Pressure Boundary.

D. Unidentified Leakage limits are exceeded. These limits are to ensure pressurizer level will be maintained in the event of a reactor trip.

K/A
035 Steam Generator
G2.1.20 Ability to execute procedure steps.

K/A MATCH ANALYSIS
Tech Specs can be considered a procedure that is used by the operators. The question tests the knowledge of whether or not a limit is violated and what the basis of the limit is. The applicant must have this knowledge in order to have the ability to execute the Tech Specs. Testing the basis info makes it SRO-only level.

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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A B D D B C B C A C	Scramble Range: A - D
Tier:	2		Group:	2	
Key Word:	TECH SPEC SG STEAM G		Cog Level:	C/A 4.2	
Source:	M		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and 0.9 gpm

e. 500 gallons per day primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for an event resulting in steam discharge to the atmosphere assumes primary to secondary LEAKAGE at the Technical Specification limit as an initial condition.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of an off-normal condition. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary

(continued)

BASES

LCO

c. Identified LEAKAGE (continued)

LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the accident analyses involving steam discharge to the atmosphere. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The 500 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

(continued)

LO-LP-39202-01-21
Pressurizer PORV-456A is leaking to the PRT at a rate of 0.5 gpm.
In accordance with Technical Specifications, this is:

A. Controlled leakage

B. Pressure boundry leakage

C. Identified leakage

D. Unidentified leakage.

LO-LP-39202-01
Define the following terms, as per Plant Vogtle Tech Specs:

a. action
b. axial flux difference
c. channel check
d. [deleted]
e. [deleted]
f. core alteration
g. dose equivalent I-131
h. E-Bar, average disintegration energy

i. identified leakage
j. operable/operability
k. mode
l. pressure boundary leakage
m. quadrant power tilt ratio
n. rated thermal power
o. shutdown margin
p. thermal power
q. unidentified leakage

LO-LP-16001-07-05
Which one of the following states the basis for the Technical Specification limit of 1 gpm steam generator tube leakage ?

a. **To ensure acceptable offsite doses in the event of a steamline break.**

b. To ensure that the steam generator tube integrity is maintained.

c. To ensure that pressurizer level will be maintained in the event of a reactor trip.

d. This is the minimum detectable leakage for the secondary radiation monitoring system.

LO-LP-16001-07
State the following for each RCS Tech Spec in Section 3.4 of Technical Specifications:

a. Limiting conditions for operation
b. Applicability
c. Action statements for actions of 1 hour or less

11. 038G2.4.46 001

Unit 1 tripped three minutes ago.

Currently plant conditions are as follows:

Go with first rewrite #1

- Steam Generator Pressures:
  - SG #1 = 600 psig and lowering.
  - SG #2 = 600 psig and lowering.
  - SG #3 = 600 psig and lowering.
  - SG #4 = 600 psig and lowering.

= B) stable Rx

- Steam Generator NR Levels are:
  - SG #1 = 0%.
  - SG #2 = 0%.
  - SG #3 = 5% and lowering.
  - SG #4 = 0%.

- Low Steam Line P has been blocked

- The Condenser Air Ejector Radiation Monitor is in alarm and the indication is rising.
- RCS Pressure is 1600 psig and lowering.

- Operators are at the point in 19000-C, E-0 Reactor Trip or Safety Injection, where they are evaluating if Main Steam Lines should be isolated.

Which ONE of the following correctly states the status of the Main Steam Line Isolation Valves, the ~~required actions~~, and reasons for those actions?

A.✓ Main Steam Isolation should have occurred, but did not. Attempt to close all Main Steam Line Isolation Valves to limit a radiation release to the atmosphere.

B. Main Steam Isolation should have occurred, but did not. Attempt to close all Main Steam Line Isolation Valves to prevent pressurized thermal shock conditions due to the increased tensile stress on the outside wall of the reactor vessel.

C. Main Steam Isolation did not occur because it was designed not to occur under the given conditions. Monitor steam generator pressures to ensure that Main Steam Isolation occurs when appropriate setpoints are reached to prevent pressurized thermal shock conditions.

D. Main Steam Isolation did not occur because it was designed not to occur under the given conditions. Manually close all Main Steam Isolation Valves to limit the radiation release to the public.

K/A
038 Steam Gen. Tube Rupture
G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

K/A MATCH ANALYSIS sp
All SGs are blowing down, which has caused a SGTR in #3 SG, as evidienced by its level being slightly higher than the other 3 SGs and the Air Ejector Rad alarm. The Air Ejector Rad alarm is indication that a main stm iso did not occur. The isolation should have occurred, therefore the air ejector rad monitor is not consistent with the plant conditions.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. The Main Steam Iso Valves should have closed because the SG presures lowered 350 psi over 3 minutes, which should have created an isolation signal. The air ejector rad monitor is inconsistent with the plant conditions and operators are required to take action to close the MSIVs, which will limit a radiation release to atmosphere.
B. Incorrect. The outside vessel wall will be under compressive stresses due to the cooldown. Plausible because main steam isolation should have occurred and main steam isolation is important because of pressurized thermal shock.
C. Incorrect. Main steam isolation should have occurred. Plausible because main steam isolation is important because of pressurized thermal shock.
D. Incorrect. Main steam isolation should have occurred. Plausible because closing the MSIVs is important to limit the release to the atmosphere.

REFERENCES
1. V-LO-TX-28101, RPS-SSPS-AMSAC, Rev. 3.
2. LO-LP-37121-17-C, Loss of Secondary Coolant / Faulted SG, Rev. 17, 09/13/2000.
3. 19000-C, E-0 Reactor Trip or Safety Injection, Rev. 29, 06/25/2004.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C C B A A D C D C Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	SGTR MSIS MSIV	Cog Level:	C/A 3.6
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

11. 038G2.4.46 001

Unit 1 tripped three minutes ago.

Currently plant conditions are as follows:

- Steam Generator Pressures:
  - SG #1 = 600 psig and lowering.
  - SG #2 = 600 psig and lowering.
  - SG #3 = 600 psig and lowering.
  - SG #4 = 600 psig and lowering.

- Steam Generator NR Levels are:
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- The Condenser Air Ejector Radiation Monitor is in alarm and the indication is rising.
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- Operators are at the point in 19000-C, E-0 Reactor Trip or Safety Injection, where they are evaluating if Main Steam Lines should be isolated.

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K/A
038 Steam Gen. Tube Rupture
G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

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Answer: A C C B A A D C D C Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	SGTR MSIS MSIV	Cog Level:	C/A 3.6
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

QUESTIONS REPORT
for Vogite 2005-301 Draft

1. 038G2.4.46 001

Unit 1 tripped three minutes ago.

Currently plant conditions are as follows:

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D. Main Steam Isolation did not occur, as designed. Manually close all Main Steam Isolation Valves to limit the radiation release to the public.

Revise wording for C & D

K/A
038 Steam Gen. Tube Rupture
G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

K/A MATCH ANALYSIS
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Answer: A C C B A A D C D C Scramble Range: A - D

Tier:	1	Group:	1
Key Word:	SGTR MSIS MSIV	Cog Level:	C/A 3.6
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	Loss Of Secondary Coolant/Faulted SG	NUMBER:	LO-LP-37121-17-C
PROGRAM:	Licensed Operator	REVISION:	17
SME:	Perry Tucker	DATE:	September 6, 2000
APPROVED:	***D. Scukanec***	DATE:	9/13/2000

INSTRUCTOR GUIDELINES:

I. FORMAT

A. Verbal lecture with visual aids

II. MATERIALS

A. Overhead projector

B. Transparencies

C. White board with markers

III. EVALUATION

A. Oral or written exam in conjunction with other lesson plans

IV. REMARKS

A. Ensure students have latest revision of EOP

III. LESSON OUTLINE:	NOTES

1. Check main steam line isolation and bypass valves shut — LO-TP-37121-004

   a. Verify that MSLI actuation has taken place or take action to close the valves as necessary — Note: Ensure students know how to perform each step

   b. This action will isolate most faults and reduce the cooldown rate of the RCS.

2. Check for at least one non-faulted SG

   a. Ensures secondary heat sink available for plant cooldown

   b. Inability to identify at least one non-faulted SG requires transition to ECA 2.1 (19121-C)

   c. Important diagnosis on identifying non-faulted SG's. Things to check:

      SG pressure Psat for RCS Tcold

      SG steam flow indications

      SG pressure indications

      SG level indications vs feed flow

3. Identify and isolate faulted SG(s)

   a. Identify which S/G are faulted and take action to isolate affected SG(s)

      1) Uncontrolled Depressurization or,

      2) Complete depressurization of a SG

   b. If no SG can be identified as faulted, then break must be downstream of MSIVs

      1) Break should be located and severity determined

      2) Isolate break if possible

4. Check for SGTR

   a. Determined by chemistry sample, rad monitors, or SG level increasing

   b. Intent is to help identify appropriate transition Check rad monitors, SG levels, etc.

5. Check if ECCS flow can be reduced

a. Determined by RCS subcooling , SG heat sink, RCS pressure, and Przr level parameters

b. Intent is to terminate ECCS flow quickly when termination criteria are met

E. Special Considerations

1. Cautions preceding Step 1 — Objective 3

a. Caution 1 - At least one SG must be maintained available for subsequent plant cooldown (controlled cooldown)

b. Caution 2 - Cautions operator against using faulted SG's for cooldown, but provides for using a faulted SG if it is the only means available

2. Step 3 and 4 - Faulted SG identification and isolation

a. The operator is not intended to "bog down" here if identification is difficult. Subsequent steps may be implemented while identification is underway. If I.D. is made, isolation may be performed

Be aware that the procedure should not be left until the condition of secondary integrity has been established

b. The operator should also be aware that faulted SG isolation steps may be commenced prior to reaching Step 4 of this procedure if:

1) The operator has determined that a secondary break indeed exists, and

2) The operator keeps the USS informed of intended actions

3) Stress importance of isolating faulted SG to prevent uncontrolled/excessive cooldown of RCS

a) Review consequences of excessive cooldown

c. Caution preceding Step 4 - If it is determined that the TDAFW pump is the only source of feedwater flow, and the faulted SG is the only supply of pump steam, full isolation should not be performed

MSLI Actuation Signals

	Coincidence	Set point
1) **Low Steam Pressure SI/SLI**	**2** out of **3** channels **1** out of **4** steam lines	≤ **585 psig***

*** Set point is rate sensitive**

Can be manually blocked below P-11

2) **Hi Steam Pressure negative Rate**	**2** out of **3** channels **1** out of **4** steam lines	≥ **100 psig** **50 sec**

Enabled below P-11 and Low Steam Pressure SLI manually blocked

3) **Hi-2 Containment Pressure**	**2** out of **3** channels	≥ **14.5 psig**
4) **Manual Steam Line Isolation**	**1** out of **2** Hand Switches	

Equipment affected by the Main Steam Isolation signal

1) Main Steam Isolation Valves 2 per steam line (train related)

2) Bypass Steam Isolation Valves 2 per steam line (train related)

Resetting Main Steam Line Isolation

To re-open steam line isolation valves if the SLI was due to HI-2 Containment Pressure, Low Steam Line Pressure, or Hi Steam Line Pressure negative rate the condition must be cleared and SLI must be reset by taken "MSLI Reset" to "Reset Position". Manual SLI actuation does not have to be reset to re-open steam line isolation valves.

Auxiliary Feed Water Actuation

The purpose of the AFW actuation is to ensure that the steam generators have adequate water level to provide continuous heat removal for the reactor core. There are two different types of AFW actuations, Motor Driven Auxiliary Feed Water Pumps and Turbine Driven Auxiliary Feed Water Pump start signals.

Motor Driven Auxiliary Feed Water Pump Start Signals

1) Safety Injection (Sequencer start)

2) Loss of or degraded voltage on **its associated** 1E 4160 V bus (Sequencer start)

3) S/G low-low level; **2** out of **4** narrow range channels on **1** out of **4** S/G.

4) Trip of both Main Feed Pumps

5) AMSAC actuation

LO-LP-37071-01-03
Following a large steam line break, monitoring of Critical Safety Function Status Trees per 19200-C, indicates a RED path for RCS Integrity.

Which one of the statements below correctly identifies the most limiting component and reason for concern that Pressurized Thermal Shock conditions may result in brittle failure of an existing flaw?

A. The pressurizer inside wall due to increased tensile stress resulting from the large temperature drop.

B. The pressurizer outside wall due to increased tensile stress resulting from lower internal pressure.

C. The reactor vessel inside wall due to the increased tensile stress resulting from the large temperature drop and neutron irradiation.

D. The reactor vessel outside wall due to the increased tensile stress resulting from the large pressure decrease and neutron irradiation.

LO-LP-37071-01
State the transients or accidents during which RCS coolant conditions will approach reactor vessel thermal shock limits.

Used to verify my thinking

12. 061G2.4.18 001

Unit 1 has recently tripped from 100% power. A planned refueling outage was scheduled to begin in two weeks.

The SRO is performing 19231-C, FR-H.1 (Response to Loss of Secondary Heat Sink). He reads the CAUTION prior to Step 12 "Steps 12 through 15 should be performed quickly in order to establish RCS heat removal by RCS bleed and feed."

Which ONE of the following correctly describes the time considerations for steam generator (SG) dryout?

A.✓ SGs will dryout approximately 30 minutes after the total loss of feedwater with all reactor coolant pumps running.

B. SGs will dryout approximately 30 minutes after the total loss of feedwater with no reactor coolant pumps running.

C. SGs will dryout approximately 50 minutes after the total loss of feedwater with all reactor coolant pumps running.

D. SGs will dryout approximately 50 minutes after the total loss of feedwater with no reactor coolant pumps running.

K/A
061 Auxiliary/Emergency Feedwater
G2.4.18 Knowledge of the specific bases for EOPs.

K/A MATCH ANALYSIS
K/A is met because the question tests the information located in the EOP lesson plan for the basis to a caution. It is SRO-only knowledge because it is basis information.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Per provided references.
B. / C. / D. Incorrect. Plausible because this is a memory item.

REFERENCES
1. Lesson Plan LO-LP-37051-17-C, Rev. 17, July 2, 2004.
2. 19231-C, FR-H.1, Response to Loss of Secondary Heat Sink, Rev. 26.5.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C C C D A D B A C Scramble Range: A - D

Tier:	2	Group:	1
Key Word:	SG DRYOUT HEAT SINK	Cog Level:	MEM 3.6
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. 061G2.4.18 001

Unit 1 has recently tripped from 100% power. A planned refueling outage was scheduled to begin in two weeks.

The SRO is performing 19231-C, FR-H.1 (Response to Loss of Secondary Heat Sink). He reads the CAUTION prior to Step 12 "Steps 12 through 15 should be performed quickly in order to establish RCS heat removal by RCS bleed and feed."

Which ONE of the following correctly describes the time considerations for steam generator (SG) dryout?

A.✓ SGs will dryout approximately 30 minutes after the total loss of feedwater with all reactor coolant pumps running.

B. SGs will dryout approximately 30 minutes after the total loss of feedwater with no reactor coolant pumps running.

C. SGs will dryout approximately 50 minutes after the total loss of feedwater with all reactor coolant pumps running.

D. SGs will dryout approximately 50 minutes after the total loss of feedwater with no reactor coolant pumps running.

K/A
061 Auxiliary/Emergency Feedwater
G2.4.18 Knowledge of the specific bases for EOPs.

K/A MATCH ANALYSIS
K/A is met because the question tests the information located in the EOP lesson plan for the basis to a caution. It is SRO-only knowledge because it is basis information.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Per provided references.
B. / C. / D. Incorrect. Plausible because this is a memory item.

REFERENCES
1. Lesson Plan LO-LP-37051-17-C, Rev. 17, July 2, 2004.
2. 19231-C, FR-H.1, Response to Loss of Secondary Heat Sink, Rev. 26.5.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A C C C D A D B A C	Scramble Range: A - D	
Tier:	2		Group:	1		
Key Word:	SG DRYOUT HEAT SINK		Cog Level:	MEM 3.6		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

1. 061G2.4.18 001

Unit 1 has recently tripped from 100% power two weeks prior to entering a planned refueling outage. The SRO is performing 19231-C, FR-H.1 (Response to Loss of Secondary Heat Sink). He reads the CAUTION prior to Step 12 "Establish bleed and feed quickly to minimize the chances of core uncovery." Which ONE of the following correctly describes the time considerations for steam generator (SG) dryout?

A.✓ SGs will dryout approximately 30 minutes after the total loss of feedwater when all reactor coolant pumps are running.

B. SGs will dryout approximately 30 minutes after the total loss of feedwater when no reactor coolant pumps are running.

C. SGs will dryout approximately 50 minutes after the total loss of feedwater when all reactor coolant pumps are running.

D. SGs will dryout approximately 50 minutes after the total loss of feedwater when no reactor coolant pumps are running.

K/A
061 Auxiliary/Emergency Feedwater
G2.4.18 Knowledge of the specific bases for EOPs.

K/A MATCH ANALYSIS
K/A is met because the question tests the information located in the EOP lesson plan for the basis to a caution. It is SRO-only knowledge because it is basis information.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Per provided references.
B. / C. / D. Incorrect. Plausible because this is a memory item.

REFERENCES
1. Lesson Plan LO-LP-37051-17-C, Rev. 17, July 2, 2004.
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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	A C C C D A D B A C	Scramble Range: A - D	
Tier:	2		Group:	1		
Key Word:	SG DRYOUT HEAT SINK		Cog Level:	MEM 3.6		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

III. SUMMARY

A. Review objectives

1. CITE POTENTIAL EVENTS THAT COULD LEAD TO A LOSS OF SECONDARY HEAT SINK. (COMMITMENT)

   Loss of MFW at power

   Loss of off-site power

   Transients resulting in loss of main and auxiliary feedwater when S/G are needed for RCS heat removal

2. STATE THE REASON FOR TRIPPING RCPS EARLY IN THE TRANSIENT.

   Tripped to eliminate their heat input

3. STATE RCS TEMPERATURE AND PRESSURE RESPONSE TO A LOSS OF SECONDARY HEAT SINK (LOSHS) WITH AND WITHOUT REQUIRED OPERATOR ACTIONS. (COMMITMENT)

   Reference Section II.B of lesson plan

4. STATE TIME CONSIDERATIONS ON INITIATION OF BLEED AND FEED. COMMITMENT)

   With RCP running, approximately 30 minutes

   Without RCP running, approximately 40 minutes

   Based on preventing or minimizing core uncovery

5. STATE THE PRECAUTIONS WHICH SHOULD BE TAKEN IN FEEDING A HOT, DRY STEAM GENERATOR FOLLOWING RECOVERY FROM A LOSS OF HEAT SINK ACCIDENT.

   Feed one S/G at a time

   During RCS feed and bleed, feed at fastest possible rate

   If RCS feed and bleed is reducing temps, then feed S/G at a rate of 30-100 gpm until level > 9% WR

   If feeding a S/G do not feed other dry S/G's until RCS temp is less than 550°F

6. INTERPRET THE CSFST FOR A CHALLENGE TO THE HEAT SINK SAFETY FUNCTION. (19200, F.0-3 - Heat Sink)

   19200-C, F.03, Heat Sink

- When performing you need to ensure this is below the present condensate pump discharge pressure

10. Step10 - Check SG levels

    a. If any SG is >10% NR level then exit procedure

    b. If not and condensate flow can be verified with at least one SG level trending to 10% NR then maintain flow and exit this procedure (unless bleed & feed in service)

    c. If flow can not be verified then go to step 10 to determine if bleed and feed required

11. Step 11 - Check for loss of secondary heat sink

    a. This step allows the operators to continue efforts to establish feed flow to the SG before taking more drastic action

12. Step 12 - 16 Establishing bleed and feed

    a. CAUTION: Establish bleed and feed quickly to minimize the chances of core uncovery.

    b. Time considerations to initiate — Objective 4

        1) With RCP's running - approximately 30 minutes from total loss of feedwater to SG dryout

        2) Without RCP's running - approximately 40 minutes from total loss of feedwater to SG dryout

    c. Basic procedure is to:

        1) Actuate SI and verify ECCS flow

        2) Open both PORV's and verify flow

            a) Arm COPs - this ensures the PORV's will not reclose at 2185 psig

            b) Pressure should equalize where PORV flow equals SI flow

        3) Two PORV's are required for adequate bleed path

            a) Head vents are used as alternate but do not satisfy the bleed path req

        4) Therefore if two PORV's are not available, we try to align at least one SG is to any possible — Attachment C

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. Check for loss of secondary heat sink: • WR level in any 3 SGs is less than 29% [44% ADVERSE]. -OR- • RCS pressure due to loss of secondary heat sink-GREATER THAN 2335 PSIG	11. Return to Step 1. OBSERVE NOTE AND CAUTIONS PRIOR TO STEP 1.

CAUTION: Steps 12 thru 15 should be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12. Actuate SI, if not previously actuated.	
13. Verify RCS feed path: a. Verify ECCS pump status: • CCPs - AT LEAST ONE RUNNING. -OR- • SI pumps - AT LEAST ONE RUNNING. b. Verify ECCS valve alignment - PROPER INJECTION LINEUP INDICATED ON MLBs.	13. Start pumps and align valves as necessary to establish a feed path using ATTACHMENT A or B. IF a feed path can NOT be established, THEN continue attempts to establish feed flow. Return to Step 7. OBSERVE CAUTION AND NOTES PRIOR TO STEP 7.

low pressure water source (i.e. fire water) and then depressurize that SG to atmospheric conditions

5) Verify the automatic Safety Injections actuations occur (first 16 steps of 19000-C).

13. Steps 26 - 31, Termination of ECCS Bleed & Feed path

a. ONLY commenced after at least one SG NR level >10% (32% adverse) AND RCS core exit TC's and WR hot leg temperatures are lowering.

b. ECCS pumps are sequentially stopped similar to SI termination in EOP 19012-C

c. PORV's are shut in a specified sequence with ECCS termination to prevent RCS overpressurization and to establish subcooling requirements — Place PORVs in auto

14. Step 36 - Completion of SI termination

a. The operator is directed to EOP 19011-C

b. 19011-C continues with realignment of plant systems

15. Establishing feed to a hot, dry SG following recovery from a loss of secondary heat sink and a feed and bleed of the primary — Objective 5

a. Definition of a hot, dry SG

1) A SG with primary side temperature >550°F and secondary has no liquid inventory (<9% WR, 31% adv cnmt) — 19235-C

2) Primary temp determined from hot leg RTD's

b. Restoration is performed to one SG at a time in case of failure due to excessive thermal stresses. Failures would be isolated to a particular SG

1) Check for SGTR or fault condition

a) If so, go to another SG, only after HL temps have been reduced below 550oF, and then start again

b) If no failure, feeding may continue

c. Feed rate considerations

1) If bleed and feed is initiated and RCS

13. 064A2.05 001

The following Unit 1 conditions exist:

- A loss of offsite power occurs
- Offsite power is projected to be unavailable for several hours and the SAT is unavailable
- Both Emergency Diesel Generators (EDGs) have a speed of 440 rpm
- A common voltage regulator malfunction has occurred in both EDGs resulting in:
  - The "A" EDG voltage reaching 3800 VAC
  - The "B" EDG voltage reaching 3700 VAC

Which ONE of the following procedural paths should the SRO implement to mitigate the electrical problems?

A. 19100-C, "ECA-0.0 Loss of All AC Power" and 13145-1, "Diesel Generators"

B. 19100-C, "ECA-0.0 Loss of All AC Power" and 13427-1, "4160V AC 1E Electrical Distribution System"

C✓ 18031-C, Loss of Class 1E Electrical Systems" and 13145-1, "Diesel Generators"

D. 18031-C, Loss of Class 1E Electrical Systems" and 13427-1, "4160V AC 1E Electrical Distribution System"

K/A
064 Emergency Diesel Generator
A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loading the ED/G.

K/A MATCH ANALYSIS
The "B" EDG does not load due to the voltage regulator problem. In order to get it to load, the guidance in the AOP will direct operators to an SOP which will address voltage regulator problems.

SRO only because ROs are required to know entry conditions, but SROs are also required to make transitions between procedures.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. A total loss of AC does not occur because the "A" EDG starts and loads. Plausible because if the applicant does not realize that the "A" EDG output breaker will close at 3800 Vac, then ECA-0.0 would be required.
B. Incorrect. No total loss of AC and 13427-1 will not address voltage regulator issues. Plausible because if the applicant does not realize that the "A" EDG output breaker will close at 3800 Vac, then ECA-0.0 would be required.
C. Correct. The AOP is appropriate for loss of a single safety bus. The AOP also contains guidance to direct operators to 13145-1 where voltage regulator problems are addressed.
D. Incorrect. No guidance exists in 13427-1 to address the voltage regulator problems. Plausible because the AOP does contain a transition to 13427-1 for energization of the safety bus.

REFERENCES
1. 18031-C, Loss of Class 1E Electrical Systems, Rev. 20.1, 12/19/2003.
2. 19100-C, ECA-0.0 Loss of All AC Power, Rev. 28, 12/19/2003.
3. 13145-1, Diesel Generators, Rev. 56, 07/22/2004.
4. 13427-1, 4160V AC 1E Electrical Distribution System, Rev. 34.1, 08/01/2003.
5. V-LO-TX-11101, Emergency Diesel Generator, Rev. 4.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C D A B C B A C B D Scramble Range: A - D

Tier:	2	Group:	1
Key Word:	DIESEL EDG	Cog Level:	C/A 3.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

13. 064A2.05 001

The following Unit 1 conditions exist:

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- Both Emergency Diesel Generators (EDGs) have a speed of 440 rpm
- A common voltage regulator malfunction has occurred in both EDGs resulting in:
  - The "A" EDG voltage reaching 3800 Vac
  - The "B" EDG voltage reaching 3700 Vac

Which ONE of the following procedural paths should the SRO implement to mitigate the electrical problems?

A. 19100-C, "ECA-0.0 Loss of All AC Power" ~~transitioning to~~ and ~~utilize~~ 13145-1, "Diesel Generators"

B. 19100-C, "ECA-0.0 Loss of All AC Power" ~~transitioning to~~ and 13427-1, "4160V AC 1E Electrical Distribution System"

C✓ 18031-C, Loss of Class 1E Electrical Systems" ~~transitioning to~~ and 13145-1, "Diesel Generators"

D. 18031-C, Loss of Class 1E Electrical Systems" ~~transitioning to~~ and 13427-1, "4160V AC 1E Electrical Distribution System"

They think answer should be "D."

K/A
064 Emergency Diesel Generator
A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loading the ED/G.

K/A MATCH ANALYSIS
The "B" EDG does not load due to the voltage regulator problem. In order to get it to load, the guidance in the AOP will direct operators to an SOP which will address voltage regulator problems.

SRO only because ROs are required to know entry conditions, but SROs are also required to make transitions between procedures.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. A total loss of AC does not occur because the "A" EDG starts and loads. Plausible because if the applicant does not realize that the "A" EDG output breaker will close at 3800 Vac, then ECA-0.0 would be required.
B. Incorrect. No total loss of AC and 13427-1 will not address voltage regulator issues. Plausible because if the applicant does not realize that the "A" EDG output breaker will close at 3800 Vac, then ECA-0.0 would be required.
C. Correct. The AOP is appropriate for loss of a single safety bus. The AOP also contains guidance to direct operators to 13145-1 where voltage regulator problems are addressed.
D. Incorrect. No guidance exists in 13427-1 to address the voltage regulator problems. Plausible because the AOP does contain a transition to 13427-1 for energization of the safety bus.

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1. 18031-C, Loss of Class 1E Electrical Systems, Rev. 20.1, 12/19/2003.
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3. 13145-1, Diesel Generators, Rev. 56, 07/22/2004.
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MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C D A B C B A C B D Scramble Range: A - D

Tier:	2	Group:	1
Key Word:	DIESEL EDG	Cog Level:	C/A 3.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. 064A2.05 001

The following Unit 1 conditions exist:

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- Offsite power is projected to be unavailable for several hours
- Both Emergency Diesel Generators (EDGs) have a speed of 440 rpm
- A common voltage regulator malfunction has occurred in both EDGs resulting in:
  - The "A" EDG voltage reaching 3800 Vac
  - The "B" EDG voltage reaching 3700 Vac

Which ONE of the following procedural paths should the SRO implement to mitigate the electrical problems?

A. 19100-C, "ECA-0.0 Loss of All AC Power" transitioning to 13145-1, "Diesel Generators"

B. 19100-C, "ECA-0.0 Loss of All AC Power" transitioning to 13427-1, "4160V AC 1E Electrical Distribution System"

C✓ 18031-C, Loss of Class 1E Electrical Systems" transitioning to 13145-1, "Diesel Generators"

D. 18031-C, Loss of Class 1E Electrical Systems" transitioning to 13427-1, "4160V AC 1E Electrical Distribution System"

K/A
064 Emergency Diesel Generator
A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loading the ED/G.

K/A MATCH ANALYSIS
The "B" EDG does not load due to the voltage regulator problem. In order to get it to load, the guidance in the AOP will direct operators to an SOP which will address voltage regulator problems.

SRO only because ROs are required to know entry conditions, but SROs are also required to make transitions between procedures.

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A. Incorrect. A total loss of AC does not occur because the "A" EDG starts and loads. Plausible because if the applicant does not realize that the "A" EDG output breaker will close at 3800 Vac, then ECA-0.0 would be required.
B. Incorrect. No total loss of AC and 13427-1 will not address voltage regulator issues. Plausible because if the applicant does not realize that the "A" EDG output breaker will close at 3800 Vac, then ECA-0.0 would be required.
C. Correct. The AOP is appropriate for loss of a single safety bus. The AOP also contains guidance to direct operators to 13145-1 where voltage regulator problems are addressed.
D. Incorrect. No guidance exists in 13427-1 to address the voltage regulator problems. Plausible because the AOP does contain a transition to 13427-1 for energization of the safety bus.

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MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C D A B C B A C B D Scramble Range: A - D

Tier:	2	Group:	1
Key Word:	DIESEL EDG	Cog Level:	C/A 3.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

NOTE: The Lo Lube Oil Pressure trip can be blocked if the engine has tripped on this condition with an emergency start present. Depressing the Emergency Stop Reset pushbutton prior to allowing the ASSD to reset will allow the engine to restart. If this is performed, an operator must be present to continuously monitor lube oil system parameters because the active trip will no longer trip the engine automatically. A manual emergency stop of the engine will be required to stop the engine due to an actual low lube oil pressure condition.

11.26 Engine Operation with Instrumentation and Controls

There are two types of engine start signals, NORMAL and EMERGENCY.

1. Normal Starts

The normal start signals can come from either the local control panel or from the QEAB. This type of start can either be a ***fast*** or ***slow*** start. The ***fast*** start allows the engine to reach rated speed within the T.S. limits. The ***slow*** start allows the engine to attain idle speed, and then ramp to rated speed in ≈ 25 seconds.
Auto Field flash circuits are disabled during slow starts. (Note: b, c, and d below also occur on a slow start)

For a *fast* start:

Momentarily push the start pushbutton. The following should occur next:

- **a.** After 1 sec, generator field is flashed.
- **b.** Signal sent to ASSD system to enable normal S/D signals.
- **c.** When engine speed reaches 200 RPM,
    1) Air start solenoids deenergize (isolates starting air)
    2) Field flash enabled
- **d.** After 5 seconds:
    1) Close signal sent to starting air solenoids valves
    2) Emergency start circuits enabled
    3) Normal start pushbutton seal-in removed

90% of 4160V = 3744 V

- **e.** When speed reaches 440 RPM and the generator is at 90% of rated voltage the DG output breaker can be closed (Ready to Load permissive).
- **f.** After ≈ 90 seconds, Group II lockout timing will be complete and all Group II trip sensors will be operable.

For a *slow* start:

The UNIT/PARALLEL switch must be in the PARALLEL position prior to start. This causes the BLUE unit mode light to be deenergized and enables the engine ramp circuit. Selecting PARALLEL will activate the DG governor slow start circuit. This circuit will ramp DG speed from idle speed (220 RPM) to rated speed (450 RPM) in approximately 25 seconds **(Notice this does not meet the T.S. requirements for engine starting time.)**

This slow start minimizes engine wear and is designed to lengthen the life of the engine components. Another aspect of the slow start is that the generator will not automatically flash after 200 RPM or 1 second after the receipt of the start signal. The field flash is prevented by selecting the parallel mode of operation on the Unit/Parallel switch and locally at the generator control panel selecting the slow start position on the Exciter Permissive switch. On a slow start the generator will be manually excited when the engine is stable at 450 RPM.

d. Governor and voltage regulator pre-position to 60 Hz, 4160 VAC and Generator control shifts to UNIT mode

e. at 1 sec, generator field is flashed

f. at 200 rpm (approx. 1-2 sec):

1) Air start solenoids close

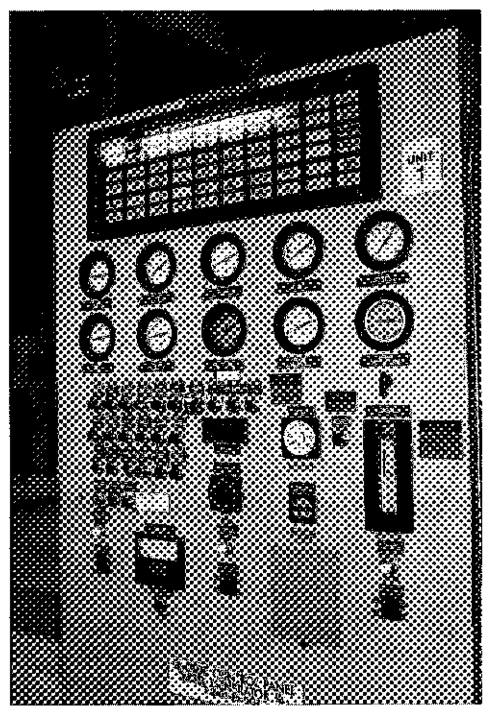
2) Generator field is again flashed

g. When speed reaches 440 RPM and the generator is at 90% of rated voltage, the DG output breaker closure permissive is made up. The DG output breaker does not close for a SIS or manual emergency start. For a Loss of Offsite Power the breaker is closed by the associated emergency bus sequencer.

h. After approx. 2 minutes, Group II lockout timing will be complete but only the Low Lube Oil pressure trip will be enabled.

If the DG does not start properly the Air start solenoids will remain energized until both air start receiver pressures < 150 psig. Then all emergency starts to the air start solenoids and field flash relay are de-activated. But, the normal manual start is available and the DG FAIL TO START alarm will be energized. Now the diesel can be manually started but only the emergency trips are enabled. Engineering evaluations have been performed that show each receiver holds enough air for at least 5 start attempts assuming initial receiver pressure was > 210 psig.

11.27 Controls on Engine Control Panel



Start Pushbutton - Normal local engine start

This is a momentary P/B. It does not need to be held for a minimum time to initiate an engine start.

Stop Pushbutton - Normal local engine stop

Operational Mode Pushbutton - Returns Engine from maintenance mode to operational mode.

Maintenance Mode Pushbutton - places engine in maintenance mode.

Emergency Stop Pushbutton - local emergency engine stop.

Emergency Start Selector Switch - EMERGENCY START position to start - NORMAL position to remove Emergency Start

Emergency Stop Reset Pushbutton - Pressed to reset a manual or automatic emergency trip signal.

Reset From LOCA/LOSP Pushbutton - Removes emergency start signal to allow normal engine shutdown and restores non-emergency trips to active status.

Engine Roll Pushbutton - In maintenance mode, air rolls engine.

2) Normally, the incoming source frequency should be slightly higher than the bus frequency to ensure some immediate load pickup by the incoming source when the breaker is closed. However, when paralleling the grid to the 1E bus being carried by a diesel, the grid frequency cannot be adjusted. The DG frequency must be adjusted. The DG frequency is adjusted to slightly higher than grid frequency to prevent motoring the DG when the incoming breaker is closed.

3) Normally, when paralleling two AC sources, the incoming breaker should be closed as close as possible to the 12 o'clock position to minimize the phase difference between the two sources. The slower the sync scope is rotating; the closer to the 12 o'clock position the sync scope needle should be before trying to close the breaker. In all cases, the breaker should be closed as close as possible to the 12 o'clock position on the 11 o'clock side of the sync scope.

4) Since grid frequency cannot be adjusted, to prevent motoring the Diesel Generator, its frequency is adjusted to slightly higher than grid frequency causing the sync scope to rotate counterclockwise. The breaker should still be closed as close to the 12 o'clock position as possible and still on the 11o'clock side.

In AUTO the DG output breaker automatically closes when the following are met:

- No 186 LO relays
- DG READY to LOAD Voltage and Frequency are met
- TRS-LR selected to Control Room
- Auto sync check relay senses proper phase rotation
- Auto Sync pushbutton is held until breaker closes

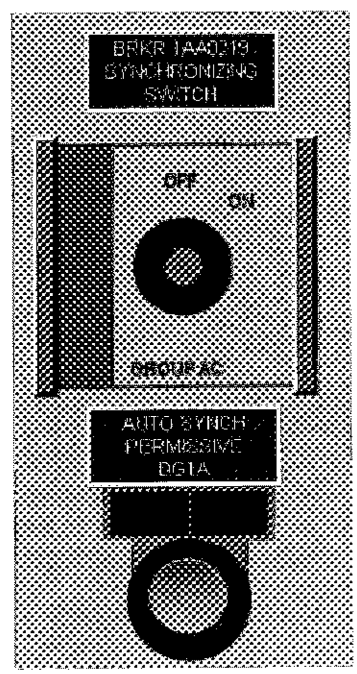


Figure 26

Synchronizing Lights are bright at the 6 o'clock position, and are dark at the 12 o'clock position. The Red AUTO SYNC PERMISSIVE LIGHT illuminates near the 12 o'clock position.

In MANUAL the operator manually closes breaker. (Always 11 o'clock side)

i. SYNCHRONIZATION SWITCH

There is a synchronizing interlock for the breakers that have the ability to cause different sources to be paralleled. This interlock helps ensure proper phase rotation between the two sources prior to closing the breakers. This switch also enables the synch scope and the sync lights.

j. UNIT PARALLEL SWITCH (Figure 23)

This is a 3 Position Spring return to center position switch which is used to select voltage regulator and speed control operating modes, (if LRS in LOCAL). There is no procedure guidance for parallel operations from the LOCAL panel.

Abnormal Operating Procedures

LOSS OF CLASS 1E ELECTRICAL SYSTEMS

PURPOSE

PRB REVIEW REQUIRED

SECTION A

Section A of this procedure addresses the loss of one train of either 4160V AC or 480V AC Class 1E Electrical System with the Diesel Generator failing to tie to the same train.

SECTION B

Section B of this procedure addresses the loss of one train of either 4160V AC or 480v AC Class 1E Electrical System with the Diesel Generator tying to the same train.

SYMPTOMS

- Loss of offsite power to one train of 1E Electrical System (1AA02, 1BA03, 2AA02, 2BA03) concurrent with diesel failure to tie on same train.
- Electric fault on Unit 1 Switchgear 1AA02 or 1BA03 or on Unit 2 Switchgear 2AA02 or 2BA03.
- Loss of one train of 480V Class 1E power.
- Loss of offsite power to one train of 1E Electrical System (1AA02, 1BA03, 2AA02, 2BA03) concurrent with diesel tying on same train.

A. LOSS OF POWER WITH DIESEL GENERATOR FAILING TO TIE TO SAME TRAIN

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
A20. Initiate 13145, DIESEL GENERATORS and align the Train A(B) Diesel Generator for automatic starting.	
A21. Restore Train A(B) components, MFP(s), and AFW pumps as required by current plant conditions.	
A22. Return to the Unit Operating Procedure currently in effect.	

END OF SUB-PROCEDURE TEXT

B. LOSS OF POWER WITH DIESEL GENERATOR TYING TO SAME TRAIN

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
B7. Initiate applicable Technical Specification requirements: • AC electrical power sources - LCO 3.8.1 or LCO 3.8.2. • AFWS - LCO 3.7.5. • RCS Specific Activity - SR 3.4.16.2 if >15% power transient has occurred.	
* B8. Try to restore power to normal power by initiating 13427, 4160V AC ELECTRICAL DISTRIBUTION.	* B8. <u>WHEN</u> the cause for the loss of normal power is corrected, <u>THEN</u> initiate 13427, 4160V AC ELECTRICAL DISTRIBUTION to restore the associated 4160V AC to its normal alignment. Restore power to 4160V AC 1E bus from an alternate source by initiating 13427, 4160V AC ELECTRICAL DISTRIBUTION, with SS authorization.
B9. Initiate 13145, DIESEL GENERATORS and align the Train A(B) DG for automatic starting.	
B10. Return to the UOP currently in effect.	

END OF SUB-PROCEDURE TEXT

EMERGENCY OPERATING PROCEDURE

ECA-0.0 LOSS OF ALL AC POWER

PURPOSE

PRB REVIEW REQUIRED

This procedure provides actions to respond to a loss of all AC power. (Applicable in Modes 1,2,3,4)

MAJOR ACTIONS

- Check Plant Conditions
- Restore AC Power
- Maintain Plant Conditions for Optimal Recovery
- Evaluate Energized AC Emergency Bus
- Select Recovery Guideline After AC Power Restoration

SYMPTOMS/ENTRY CONDITIONS

The symptoms are:

- Both emergency AC buses are de-energized.

The entry conditions are:

- 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION, Step 3.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:

- 91001-C EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS should be implemented at this time.
- If the Diesel Generator output breaker did not close on a bus loss of power and power has been restored to an AC emergency bus from a RAT, it will be necessary to reset the sequencer using the DG BRKR FAILED TO CLOSE SEQUENCER UV RESET PUSHBUTTON to enable normal stopping of AC emergency loads.
- The battery capacity for DG field flashing may only support 5 Diesel Generator start attempts.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6. Try to restore power to any AC emergency bus:	
a. Start DG.	a. Dispatch operator to locally start DG by initiating 13145, DIESEL GENERATORS. • IF starting air pressure is greater than 150 psig, THEN emergency start DG at the DG panel using Emergency Start Switch • IF starting air pressure is less than 150 psig, THEN manually start DG (locally or remotely).

DIESEL GENERATORS

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	**ALL**
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	**NONE**
Information Use:	Available on plant site for reference as needed.	**NONE**

4.8 **Generator Failure During Loss Of Offsite Power**

CAUTION

This section provides instructions for restoring the generator if the engine starts but the generator fails to develop adequate output voltage during startup due to a voltage regulator malfunction. These instructions should only be used to try to restore the generator during a loss of offsite power incident and are not to be used during routine testing or operation.

4.4.8.1 CHECK for any tripped relays at the Diesel Generator Control Panel PDG1 (PDG3).

4.4.8.2 If any relays are tripped, INITIATE maintenance to correct the problem.

4.4.8.3 If no relays are tripped DEPRESS the Field Flash Pushbutton 1-HS-4459 (4460) for 3-5 seconds.

4.4.8.4 CHECK that Generator volts rises to 4025-4330 volts.

4.4.8.5 If generator voltage goes up but does not stabilize between 4025 and 4330 volts, TRANSFER to the Redundant Rectifier Bridge per Step 4.4.9 or TRANSFER to the Alternate Voltage Regulator per Step 4.4.10.

4160V AC 1E ELECTRICAL DISTRIBUTION SYSTEM

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	**ALL**
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	**NONE**
Information Use:	Available on plant site for reference as needed.	**NONE**

14. 067G2.4.28 001

The following sequence of events occurred on Unit 2:

- At 1000 hours a fire broke out in the Protected Area and burned for two hours, but no safety-related equipment was affected.
- At 1010 hours the Shift Manager made an emergency classification.
- At 1030 hours the plant received a credible security threat that stated the fire was the result of sabotage.

Which ONE of the following correctly states when the NRC was required to be notified in accordance with 00152-C, Federal and State Reporting Requirements, and/or 91001-C, Emergency Classification and Implementing Instructions?

A.✓ 1100 hours

B. 1110 hours

C. 1115 hours

D. 1130 hours

K/A
067 Plant Fire On-site
G2.4.28 Knowledge of procedures relating to emergency response to sabotage.

K/A MATCH ANALYSIS
The fire was a result of sabotage and the reportability requirements are contained in procedures.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. See 00152-C, Page 55 of 68. Fire of unknown origin - 1 hour reportable.
B. Incorrect. Plausible because it is 1 hour after the classification.
C. Incorrect. Plausible because it is 1 hour after the fire has burned for 15 minutes. An emergency classification exists for a fire that burns for 15 minutes.
D. Incorrect. Plausible because it is 1 hour after the phone call.

REFERENCES
1. 00152-C, Federal and State Reporting Requirements, Rev. 33, 02/20/2004.
2. 91001-C, Emergency Classification and Implementing Instructions, Rev. 20.1, 09/12/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A B D B A A A D D C Scramble Range: A - D

Tier:	1	Group:	2
Key Word:	FIRE SABOTAGE REPORT	Cog Level:	C/A 3.3
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

14. 067G2.4.28 001

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Key Word:	FIRE SABOTAGE REPORT	Cog Level:	C/A 3.3
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Test:	S	Author/Reviewer:	MAB/RSB

QUESTIONS REPORT
for Voglte 2005-301 Draft

1. 067G2.4.28 001

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Key Word:	FIRE SABOTAGE REPORT	Cog Level:	C/A 3.3
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

PRB REVIEW REQUIRED

FEDERAL AND STATE REPORTING REQUIREMENTS

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	
Information Use:	Available on plant site for reference as needed.	**ALL**

TABLE 3 (CONT'D)

SECURITY REPORTABILITY MATRIX

Event Description		System/Component Status	Compensatory Measures Taken Within 10 Minutes of the Event Discovery	Reportability
G.	**INTERRUPTION OF NORMAL OPERATIONS**			
1.	Fire or Explosion of suspicious or unknown origin within the PA, including the Isolation Zone	N/A	N/A	One Hour
2.	Suspect explosive device discovered	N/A	N/A	One Hour
3.	Unauthorized use of, vandalism, or tampering, with machinery, components or controls, including the security system/components	N/A	N/A	One Hour
4.	Credible Bomb or Extortion Threat	N/A	N/A	One Hour
5.	Results of Bomb Search	N/A	N/A	One Hour
6.	Unsubstantiated Bomb or Extortion Threat	N/A	N/A	Log

PRB REVIEW REQUIRED

EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS

PROCEDURE USAGE REQUIREMENTS-	SECTIONS
Continuous Use: Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	• Data Sheet 1
Reference Use: Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	
Information Use: Available on plant site for reference as needed.	**Remainder of Procedure**

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	ELECTRICAL POWER	RADIOACTIVITY	PLANT SYSTEMS	NATURAL PHENOMENON	HAZARDS	SECURITY	BARRIERS	SHUTDOWN SYSTEMS	OTHER
GENERAL	Modes 1-4: Loss of AC power to BOTH AA02 AND BA03 AND EITHER: Restoration of at least one emergency bus NOT likely within 4 hrs of time of loss. OR Loss or Potential Loss of FUEL CLAD BARRIER determined from Fission Product Barrier EAL's (Figure 1 Cont. sheet 1)	Valid reading on RE-12444 that is > 30E+0 µci/cc for > 15 min. OR this value is expected to be exceeded for > 15 min. (Note 1) - - - - - Valid site boundary dose assessment of release indicates > 1000 mrem TEDE OR > 5000 mrem CDE thyroid OR field survey results indicate > 1000 mrem/hr that is expected to continue for > 1 hr OR field survey samples indicate thyroid dose of 5000 mr CDE for 1 hr of inhalation				Loss of physical control of the control room due to a security event OR loss of physical control of remote shutdown capability due to a security event	Loss of THREE barriers - - - - - Loss of ANY TWO barriers AND Potential Loss of the THIRD barrier - - - - - Modes 1-2: Subcriticality CSFST is RED AND EITHER Core Cooling OR Heat Sink CSFST is RED		Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate potential for uncontrolled radionuclide releases that can reasonably be expected to exceed 1000 mrem TEDE or 5000 mrem CDE thyroid dose levels outside the site boundary OR Actual or imminent substantial core degradation with potential for loss of containment
SITE AREA	Modes 1-4: Loss of AC power on BOTH AA02 AND BA03 for > 15 min. - - - - - Modes 1-4: Unplanned loss of voltage on ALL Vital DC buses (AD1, BD1, CD1, and DD1) for > 15 min.	Valid reading on RE-12444 that is > 3E+0 µci/cc for > 15 min. OR on RE-12839 that is > 5E+2 µci/cc for > 15 min. OR these values are expected to be exceeded for > 15 min. (Note 1) - - - - - Valid site boundary dose assessment of release indicates > 100 mrem TEDE OR > 500 mrem CDE thyroid OR field survey results indicate > 100 mrem/hr that is expected to continue for > 1 hr OR field survey samples indicate thyroid dose of 500 mrem CDE for 1 hr of inhalation	Mode 1-2: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip did NOT occur from the control room. - - - - - Modes 1-4: Significant transient is in progress and ESF annunciators needed to monitor the transient are not available AND ESF control board AND compensatory non-alarming indications are NOT available		Control room evacuation has been initiated AND control of the plant CANNOT be established from remote shutdown panels within 15 min.	Intrusion into a plant vital area by a hostile force (e.g., confirmed bomb device discovered within Vital area)	Loss of TWO barriers - - - - - Loss of ONE barrier AND Potential Loss of a SECOND barrier - - - - - Potential Loss of BOTH Fuel Clad AND RCS barriers - - - - - Modes 1-4: The reactor is tripped AND Subcriticality CSFST is RED	Modes 5-6: Loss of reactor vessel water level as indicated by loss of RHR cooling as determined by AOP 18019-C AND any ONE of the following: Vessel level < 183 ft (< 62% RVLIS), OR Containment area rad monitors RE-002/003 valid high alarm (15 mr/hr) with vessel head removed, OR Core exit temperatures > 711° F with vessel head installed - - - - - Modes 1-4: Complete loss of function needed to achieve or maintain Hot Shutdown (EITHER Core Cooling OR Heat Sink CSFST is RED) AND no other Heat Sink available	Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate actual or likely major failures of plant functions needed for protection of the public.
ALERT	Modes 1-4: Loss of voltage on EITHER AA02 OR BA03 for > 15 min. AND the remaining energized 1-E bus does not have a backup power supply available. - - - - - Modes 5, 6, or Defueled: Loss of AC power on BOTH AA02 AND BA03 for > 15 min.	Valid reading on RE-12444 that is > 2E-1 µci/cc for > 15 min. OR on RE-12839 that is > 4E+0 µci/cc for > 15 min. OR on RE-018 that is > 8E-1 µci/cc for > 15 min. (Note 1) - - - - - Confirmed sample analysis for gaseous OR liquid release indicates concentrations or release rates > 200X ODCM limits for > 15 min. - - - - - Radiation levels higher than normal in Fuel Handling or Containment Bldg AND irradiated fuel is uncovered, OR < 184 feet water level in refueling cavity, spent fuel pool, or fuel transfer canal that results in uncovering irradiated fuel. (Note 2) - - - - - Damage to irradiated fuel causing a valid high alarm on one or more of the following: RE-008 (2.5 mr/hr), RE-2532/2533 A/B (3E-4 µci/cc), or during Mode 6 fuel transfer RE-0002/003 (15 mr/hr) - - - - - Valid rad levels > 15 mr/hr in the Control room OR > 100 mr/hr in areas requiring infrequent access to maintain plant safety functions (e.g., local charging station)	Modes 1-3: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip occurred from the control room. - - - - - Modes 1-4: Unplanned loss of most or all annunciators or indicators in the control room for plant safety systems AND EITHER a significant transient is in progress or compensatory non-alarming indications are NOT available	Tornado striking plant vital area. OR sustained hurricane force winds of 100 mph or greater on site as verified by meteorological instrumentation (15 minute average). - - - - - Seismic monitoring system confirms seismic event > 0.12 g.	Aircraft crash confirmed to affect plant vital areas - - - - - Report or detection of toxic, flammable, or asphyxiant gases within a facility structure in concentrations that requires evacuation of a room or area needed for safe operation of the plant - - - - - Major fire or explosion in a vital area AND affected safety system parameters show degraded performance or there is visible damage to permanent safety related structures or safety related equipment within the specified area - - - - - Report of visible structural damage caused by natural or destructive phenomena which threatens the ability of the structure to perform its safety related function to ANY of the following plant structures: Containment, Auxiliary Building, Control Building, Fuel Handling Building, Diesel Generator Building, Condensate Storage Tank, AFW Pumphouse, NSCW Cooling Tower, RWST - - - - - Control Room evacuation has been initiated	Intrusion into the Protected area by a hostile force (e.g., unauthorized vehicle penetrates protected area in a hostile manner)	Loss or Potential Loss of Fuel Clad barrier - - - - - Loss or Potential Loss of RCS barrier	Modes 5-6: Unplanned loss of RHR cooling AND EITHER RCS temperature is > 200° F OR RCS temperature is increasing uncontrolled toward 200° F	Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate plant safety systems may be substantially degraded and that increased monitoring of plant functions is warranted.
NOUE	Loss of Off-Site power to BOTH AA02 AND BA03 for > 15 min (neither has a connected to an energized Off-Site source) AND BOTH AA02 AND BA03 are powered by Diesel Generators - - - - - Modes 5-6: Unplanned loss of voltage on ALL Vital DC buses (AD1, BD1, CD1, and DD1) for > 15 min.	Valid reading on RE-12444/12442 that is > 2E-3 µci/cc for > 60 min. OR on RE-12839 that is > 6E-2 µci/cc for > 60 min. OR on RE-018 that is > 8E-3 µci/cc for > 60 min. (Note 1) - - - - - Confirmed sample analysis for gaseous OR liquid release indicates concentrations or release rates > 2X ODCM limits for > 60 min. - - - - - Radiation levels higher than normal in Fuel Handling or Containment Bldg AND uncontrolled water level decrease in refueling cavity, spent fuel pool, or fuel transfer canal, BUT all irradiated fuel assemblies remain covered with water. (Note 2) - - - - - Valid area rad monitor readings increase 1000X over normal levels. (Note 2)	Modes 4-5: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip occurred from the control room. - - - - - Modes 1-4: Unplanned loss of most or all annunciators or indicators in the control room for plant safety systems for > 15 min. AND compensating non-alarming indicators are available - - - - - Loss of ALL of the following On-Site communications systems: In-plant telephone, Gaitronics, Sound powered phone, and Plant radio communications - - - - - Loss of ALL of the following Off-Site communications systems: ENN and Telephone capability to Off-Site network	Report of tornado striking within protected area. - - - - - Hurricane force winds of 74 mph or greater forecast by the National Weather Service (NWS- Columbia S.C. office) to be at the plant site in the next four hours. - - - - - Seismic monitoring system indicates seismic event OR plant operators report an earthquake was "felt"	Aircraft crash causes damage to safety related plant structures or safety related systems within the protected area - - - - - Report or detection of toxic, flammable, or asphyxiant gases that could enter the site area in amounts > life-threatening or flammable concentrations that could affect normal operation of the plant. - - - - - Report of turbine failure resulting in casing penetration or damage to turbine or generator seals - - - - - Fire in an area contiguous or adjacent to a vital area that is not extinguished within 15 min. of control room notification by fire alarm or personnel report - - - - - Unanticipated explosion within protected area resulting in visible damage to permanent structures or equipment	Confirmed Security Event which indicates a potential degradation in the level of safety of the plant. (i.e.,) 1. Bomb device in Protected Area but outside Vital area. 2. Hostage situation inside Protected Area. 3. Civil disturbance on plant site outside Protected Area. 4. Credible Bomb threat (Note 3) 5. Credible Attack threat (Note 3) - - - - - Notification by Local, County, or State officials of potential for evacuation of site personnel based on an Off Site event.	Loss or Potential Loss of Containment barrier - - - - - RCS Chemistry analysis indicates Dose equivalent I-131 > 1 µci/gm for > 48 hrs. or in excess of Tech Spec figure 3.4.16-1 OR RCS specific activity > 100/Ē µci/gm gross radioactivity. - - - - - RCS Unidentified leakage > 10 gpm OR RCS Pressure boundary leakage > 10 gpm OR RCS identified leakage > 25 gpm		Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate potential degradation of the level of safety of the plant - - - - - Modes 1-4: Plant NOT brought to required operating mode within Tech Spec LCO OR TRM Technical Requirement required action completion time limit. - - - - - Modes 1-4: Uncontrolled depressurization of one or more steam generators.

NOTE 1: Classification should be based on ODCM or Off site Dose Calculation computer program results; however, if the monitor reading(s) is sustained for longer than the period indicated and release assessment has NOT or CANNOT be completed within this period, then declaration MUST be made based on the valid reading.

NOTE 2: "Normal levels" are the highest reading in the last 24 hours prior to the emergency, excluding the current peak value.

NOTE 3: See section 2.0 of this procedure.

Figure 4 - EMERGENCY CLASSIFICATION LEVEL DETERMINATION

15. 069G2.1.33 001

Which ONE of the following states a condition that would require entry into the Containment Pressure Technical Specification (LCO 3.6.4) action statement and the Technical Specification Basis for that requirement? (Assume Unit 1 is in Mode 1.)

A. Containment pressure is + 1.8 psig (positive 1.8 psig). The high containment pressure requirement is in place as an Accident Analysis input for the Loss Of Coolant Accident (LOCA), which is more limiting than the Steam Line Break (SLB) Accident Analysis with respect to peak containment pressure. Therefore, the LOCA analysis bounds the SLB analysis with respect to peak containment pressure.

B. Containment pressure is + 1.9 psig (positive 1.9 psig). The high containment pressure requirement is in place as an Accident Analysis input for the Steam Line Break (SLB) Accident Analysis, which is more limiting than the Loss Of Coolant Accident (LOCA) with respect to peak containment pressure. Therefore, the SLB analysis bounds the LOCA analysis with respect to peak containment pressure.

C. Containment pressure is - 0.3 psig (negative 0.3 psig). The low containment pressure requirement protects against a containment design negative pressure limit of - 2.0 psig (negative 2.0 psig) in the event of an inadvertant containment spray actuation.

D✓ Containment pressure is - 0.4 psig (negative 0.4 psig). The low containment pressure requirement protects against a containment design negative pressure limit of - 3.0 psig (negative 3.0 psig) in the event of an inadvertant containment spray actuation.

K/A
069 Loss of CTMT Integrity
G2.1.33 High Containment Pressure - Ability to recognize indications for system operational parameters which are entry-level conditions for technical specifications.

K/A MATCH ANALYSIS
The question meets the K/A because the applicant must recognize the entry condition for the TS. The question is SRO-only because the applicant must then use TS Basis knowledge to decipher between the two answers that display data that is outside of that allowed by the LCO. The question is closed book because all applicants are required to know 1-hr TS without reference and basis info is closed book required knowledge for SRO applicants.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. P not high enough to exceed TS.
B. Incorrect. SLB bounds LOCA. Plausible because P is above TS limit.
C. Incorrect. P is within allowable TS LCO requirement and negative P limit is -0.3 psig. Plausible because P is at the limit, but not is not exceeding the limit.
D. Correct. P is below TS low limit and basis is correct according to Ref. 2 below.

REFERENCES
1. Technical Specification LCO 3.6.4
2. Technical Specfication Bases for LCO 3.6.4.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	D B A C D D C C B A	Scramble Range: A - D
Tier:	1		Group:	2	
Key Word:	CONTAINMENT PRESSURE		Cog Level:	MEM 4.0	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

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QUESTIONS REPORT
for Voglte 2005-301 Draft

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			Answer:	D B A C D D C C B A	Scramble Range: A - D
Tier:	1		Group:	2	
Key Word:	CONTAINMENT PRESSURE		Cog Level:	NEW C/A 4.0	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.3 psig and ≤ +1.8 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	12 hours

BASES

APPLICABLE SAFETY ANALYSES (continued)

The containment was also designed for an external pressure load equivalent to -3 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure. The initial pressure condition used in this analysis was 14.093 psia. This resulted in a minimum pressure inside containment of 11.77 psia, which is less than the design load.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of 10 CFR 50.36 (c)(2)(ii).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature

(continued)

BASES

APPLICABILITY (continued)

limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1
(PI-0934, PI-0935, PI-0936, PI-0937, P-9871, PI-10945)

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 6.2.
2. 10 CFR 50, Appendix K.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4A Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 17.7 psia (3.0 psig). This resulted in a maximum peak pressure from a LOCA of 36.5 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, 36.5 psig, does not exceed the containment design pressure, 52 psig.

(continued)

16. 072A2.02 001

The following radiation monitors were declared inoperable yesterday.

RE-2565A, Containment Particulate Monitor
RE-2565B, Containment Iodine Monitor
RE-2565C, Containment Gaseous Monitor

Now Unit 1 Tavg is 300 °F and heating up when RE-003, Containment Area Low Range Monitor, is declared inoperable due to a detector failure.

Which ONE of the following correctly describes the compliance status of LCO 3.3.6, Containment Ventilation Isolation Instrumentation?

A. LCO 3.3.6 is met, therefore no action statements are required to be entered.

B. LCO 3.3.6 is not met. Performing maintenance and declaring RE-2565A operable would meet the LCO allowing all action statements to be exited.

C. LCO 3.3.6 is not met. Performing maintenance and declaring RE-2565B operable would meet the LCO allowing all action statements to be exited.

D✓ LCO 3.3.6 is not met. Performing maintenance and declaring RE-003 operable would meet the LCO allowing all action statements to be exited.

QUESTIONS REPORT
for Voglte 2005-301 Draft

K/A
072 Area Radiation Monitoring
A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

K/A MATCH ANALYSIS
RE-003, an area monitor, is inoperable due to a detector failure. Tech Specs are impacted by this failure and must be complied with to correct, control, or mitigate the consequences. The question is SRO-only level because it requires basis knowledge to arrive at the answer. The question is closed book because if RE-003 were also inoperable, there would be an immediate action for Action Statement B (< 1 hr TS are memory items).

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Two operable channels do not exist. Plausible because applicant may not know what configuration makes two operable channels.
B. Incorrect. RE-2565B or C would also need to be declared operable for this channel to be operable.
C. Incorrect. RE-2565A or C would also need to be declared operable for this channel to be operable.
D. Correct. TS Basis states that RE-002 and RE-003 are separate channels, thus making RE-003 operable would allow the LCO to be met.

REFERENCES
1. Technical Specification 3.3.6, Containment Ventilation Isolation Instrumentation.
2. Technical Specification 3.3.6 Basis.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D A A A C A D D D A		Scramble Range: A - D
Tier:	2		Group:		2	
Key Word:	AREA RADIATION ARM		Cog Level:		C/A 2.9	
Source:	N		Exam:		VG05301	
Test:	S		Author/Reviewer:		MAB/RSB	

16. 072A2.02 001

The following radiation monitors were declared inoperable yesterday.

RE-2565A, Containment Particulate Monitor
RE-2565B, Containment Iodine Monitor
RE-2565C, Containment Gaseous Monitor

Now Unit 1 Tavg is 300 °F and heating up when RE-003, Containment Area Low Range Monitor, is declared inoperable due to a detector failure.

Which ONE of the following correctly describes the compliance status of LCO 3.3.6, Containment Ventilation Isolation Instrumentation?

A. LCO 3.3.6 is met, therefore no action statements are required to be entered.

B. LCO 3.3.6 is not met. Performing maintenance and declaring RE-2565A operable would meet the LCO allowing all action statements to be exited.

C. LCO 3.3.6 is not met. Performing maintenance and declaring RE-2565B operable would meet the LCO allowing all action statements to be exited.

D✓ LCO 3.3.6 is not met. Performing maintenance and declaring RE-003 operable would meet the LCO allowing all action statements to be exited.

K/A
072 Area Radiation Monitoring
A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

K/A MATCH ANALYSIS
RE-003, an area monitor, is inoperable due to a detector failure. Tech Specs are impacted by this failure and must be complied with to correct, control, or mitigate the consequences. The question is SRO-only level because it requires basis knowledge to arrive at the answer. The question is closed book because if RE-003 were also inoperable, there would be an immediate action for Action Statement B (< 1 hr TS are memory items).

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Two operable channels do not exist. Plausible because applicant may not know what configuration makes two operable channels.
B. Incorrect. RE-2565B or C would also need to be declared operable for this channel to be operable.
C. Incorrect. RE-2565A or C would also need to be declared operable for this channel to be operable.
D. Correct. TS Basis states that RE-002 and RE-003 are separate channels, thus making RE-003 operable would allow the LCO to be met.

REFERENCES
1. Technical Specification 3.3.6, Containment Ventilation Isolation Instrumentation.
2. Technical Specification 3.3.6 Basis.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D A A A C A D D D A	Scramble Range: A - D	
Tier:	2		Group:	2		
Key Word:	AREA RADIATION ARM		Cog Level:	C/A 2.9		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

1. 072A2.02 001

The following radiation monitors were declared inoperable yesterday.

RE-2565A, Containment Particulate Monitor
RE-2565B, Containment Iodine Monitor
RE-2565C, Containment Gaseous Monitor

Now Unit 1 Tavg is 300 °F and heating up when RE-003, Containment Area Low Range Monitor, is declared inoperable due to a detector failure.

Which ONE of the following correctly describes the compliance status of LCO 3.3.6, Containment Ventilation Isolation Instrumentation?

A. LCO 3.3.6 is met, therefore no action statements are required to be entered.

B. LCO 3.3.6 is not met. Performing maintenance and declaring RE-2565A operable would meet the LCO allowing all action statements to be exited.

C. LCO 3.3.6 is not met. Performing maintenance and declaring RE-2565B operable would meet the LCO allowing all action statements to be exited.

D.✓ LCO 3.3.6 is not met. Performing maintenance and declaring RE-003 operable would meet the LCO allowing all action statements to be exited.

QUESTIONS REPORT
for Voglte 2005-301 Draft

K/A
072 Area Radiation Monitoring
A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure.

K/A MATCH ANALYSIS
RE-002, an area monitor, is inoperable due to a detector failure. Tech Specs are impacted by this failure and must be complied with to correct, control, or mitigate the consequences. The question is SRO-only level because it requires basis knowledge to arrive at the answer. The question is closed book because if RE-002 were also inoperable, there would be an immediate action for Action Statement B (< 1 hr TS are memory items).

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Two operable channels do not exist. Plausible because applicant may not know what configuration makes two operable channels.
B. Incorrect. RE-2565B or C would also need to be declared operable for this channel to be operable.
C. Incorrect. RE-2565A or C would also need to be declared operable for this channel to be operable.
D. Correct. TS Basis states that RE-002 and RE-003 are separate channels, thus making RE-003 operable would allow the LCO to be met.

REFERENCES
1. Technical Specification 3.3.6, Containment Ventilation Isolation Instrumentation.
2. Technical Specification 3.3.6 Basis.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:		Scramble Range: A - D
Tier:	2		Group:	2	
Key Word:	AREA RADIATION ARM		Cog Level:	C/A 2.9	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

3.3 INSTRUMENTATION

3.3.6 Containment Ventilation Isolation Instrumentation

LCO 3.3.6 The Containment Ventilation Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Only one radiation monitoring channel OPERABLE.	A.1 Restore at least two channels to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. ------------NOTE------------ Only applicable in MODE 1, 2, 3, or 4. ------------------------------------ One or more Functions with one or more manual or automatic actuation channels inoperable. <u>OR</u> No radiation monitoring channels OPERABLE. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1	Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge supply and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ------------NOTE------------ Only applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. ------------------------------	C.1	Place and maintain containment purge and exhaust valves in closed position.	Immediately
	<u>OR</u>		
No radiation monitoring channels OPERABLE. <u>OR</u> Required Action and associated Completion Time for Condition A not met.	C.2	Enter applicable Conditions and Required Actions of LCO 3.9.4, "Containment Penetrations," for containment purge supply and exhaust isolation penetrations not in required status.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Purge and Exhaust Isolation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.6.3	Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.6.4	Perform COT.	92 days
SR 3.3.6.5	Perform SLAVE RELAY TEST.	18 months
SR 3.3.6.6	----------NOTE---------- Verification of setpoint not required. ------------------------------ Perform TADOT.	18 months
SR 3.3.6.7	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.8	Verify RESPONSE TIMES are within limits.	18 months on a STAGGERED TEST BASIS

Table 3.3.6-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1,2,3,4	2	SR 3.3.6.6	NA
2.	Automatic Actuation Logic and Actuation Relays	1,2,3,4	2	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3.	Containment Radiation	1,2,3,4,6[(c)]	2[(a)]	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	
	a. Gaseous (RE-2565C)				(b)
	b. Particulate (RE-2565A)				(b)
	c. Iodine (RE-2565B)				(b)
	d. Area Low Range (RE-0002, RE-0003)				≤ 15 mr/h[(c)] ≤ 50x background[(d)]
4.	Safety Injection[(d)]	1,2,3,4	Refer to LCO 3.3.2, "ESFAS Instrumentation,"Function 1, for all initiation functions and requirements.		

(a) Containment ventilation radiation (RE-2565) is treated as one channel and is considered OPERABLE if the particulate (RE-2565A) and iodine monitors (RE-2565B) are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE.

(b) Setpoints will not exceed the limits of Specifications 5.5.4.h and 5.5.4.i of the Radioactive Effluent Controls Program.

(c) During CORE ALTERATIONS and movement of irradiated fuel assemblies within containment.

(d) During MODES 1, 2, 3, and 4.

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Ventilation Isolation Instrumentation

BASES

BACKGROUND

Containment ventilation isolation instrumentation closes the containment isolation valves in the Mini Purge System and the Shutdown Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini Purge System may be in use during reactor operation and the Shutdown Purge System will be in use with the reactor shutdown.

Containment ventilation isolation initiates on automatic safety injection (SI) signal or by manual actuation of Phase A Isolation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss this mode of initiation.

Two radiation monitoring channels are required as input to the Containment Ventilation Isolation System. The containment purge exhaust monitors measure radiation in an air sample from the containment purge exhaust line. The purge exhaust radiation monitors consist of three different type detectors: gaseous, particulate, and iodine. The purge exhaust radiation detectors are treated as one channel which is considered OPERABLE if the particulate and iodine monitors are OPERABLE or the noble gas monitor is OPERABLE. In addition, two individual channels of containment area low range gamma monitors are provided. The two required radiation monitoring channels may be made up of any combination of the above described channels. Since the purge exhaust monitors constitute a sampling system, various components such as sample line valves, sample pumps, and filter motors are required to support monitor OPERABILITY.

Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from any one of the detectors initiates containment ventilation isolation, which closes both inner and outer containment isolation valves in the Mini Purge System and the Shutdown Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event, within approximately 60 seconds. The isolation of the purge supply and exhaust valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment purge supply and exhaust isolation radiation monitors act as backup to the SI signal to ensure closing of the purge supply and exhaust valves for events occurring in MODES 1 through 4. Manual isolation (using individual valve handswitches) following a radiation alarm is the assumed means for isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. <u>Manual Initiation</u>

   The LCO requires two channels OPERABLE. The operator can initiate Containment ventilation isolation at any time by using either of two switches in the control room (containment isolation Phase A switches). Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

   The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

   Each channel consists of one CIA handswitch and the interconnecting wiring to the actuation logic cabinet.

(continued)

BASES

LCO (continued)

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two channels of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable MODES and specified conditions for the Containment ventilation isolation portion of these Functions are different and less restrictive than those for their SI roles. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment ventilation isolation remains OPERABLE. During CORE ALTERATIONS or movement of irradiated fuel assemblies in containment, the required channels provide input to control room alarms to ensure prompt operator action to manually close the containment purge and exhaust valves. It is also acceptable during CORE ALTERATIONS or movement of irradiated fuel to meet the requirements of this LCO by maintaining the radiation monitoring instrumentation necessary to initiate containment ventilation isolation OPERABLE, in accordance with the requirements stated for MODES 1, 2, 3, and 4 operability. The purge exhaust radiation detectors (RE-2565A, B&C) are treated as one channel which is considered OPERABLE if the particulate (RE-2565A) and iodine (RE-2565B) monitors are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE. In addition, two individual channels of containment area low range gamma monitors (RE-0002 & RE-0003) are provided. The two required radiation monitoring channels may be made up of any combination of the above described channels.

(continued)

BASES

LCO

3. Containment Radiation (continued)

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

(continued)

BASES

LCO (continued)

4. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements. The safety injection initiation function is applicable in MODES 1, 2, 3, and 4 only.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Containment Radiation, and Safety Injection Functions are required OPERABLE in MODES 1, 2, 3, and 4. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the Containment ventilation isolation instrumentation must be OPERABLE in these MODES.

During CORE ALTERATIONS or movement of irradiated fuel assemblies in containment, the air locks may be open provided they are isolable per LCO 3.9.4. Since the air locks can only be closed manually, it is assumed that containment ventilation isolation is accomplished by manually closing the purge and exhaust ventilation valves. Therefore, only OPERABLE radiation monitors are required to alert the operators of the need for containment ventilation isolation.

While in MODES 5 and 6 without fuel handling in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of

(continued)

BASES

ACTIONS (continued)

the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one required containment ventilation isolation radiation monitor channel. The failed channel must be restored to OPERABLE status. Four hours are allowed to restore the affected channel based on the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a manual or automatic actuation channel is inoperable, no radiation monitoring channels operable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

C.1 and C.2

Condition C addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Required Action A.1. If no radiation monitoring channels are operable or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge supply and exhaust isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each penetration not in the required status. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.6.3

SR 3.3.6.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.4

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). For MODES 1, 2, 3, and 4, this test verifies the capability of the instrumentation to provide the containment purge and exhaust system isolation. During CORE ALTERATIONS and movement of irradiated fuel in containment, this test verifies the capability of the required channels to generate the signals required for input to the control room alarm. The setpoint shall be left consistent with the current unit specific calibration procedure tolerance.

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay.

For slave relays and associated auxiliary relays in the CVI actuation system circuit that are Potter and Brumfield (P&B) type Motor Driven Relays (MDR), the SLAVE RELAY TEST is performed on an 18-month frequency. This test frequency is based on relay reliability assessments presented in WCAP-13878, "Reliability Assessment of Potter and Brumfield MDR Series Relays." The reliability assessments are relay specific and apply only to Potter and Brumfield MDR series relays. Quarterly testing of the slave relays associated with non-P&B MDR auxiliary relays will be administratively controlled until an alternate method of testing the auxiliary relays is developed or until they are replaced by P&B MDR series relays.

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.6 (continued)

The test also includes trip devices that provide actuation signals directly to the SSPS, bypassing the analog process control equipment. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them. The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.6.7

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

SR 3.3.6.8

This SR ensures the individual channel RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in the FSAR. Individual component response times are not modeled in the analyses. The analyses model the overall or elapsed time, from the point at which the parameter exceeds the Trip Setpoint Valve at the sensor, to the point at which the equipment in both trains reaches the required functional state.

RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

(continued)

BASES

REFERENCES

1. 10 CFR 100.11.
2. NUREG-1366.
3. WCAP-13878-P-A, Rev. 2, August 2000.
4. WCAP-13900, Rev. 0, April 1994.
5. WCAP-14129, Rev. 1, January 1999.

17. 074G2.4.6 001

In 19221-C, Response to Inadequate Core Cooling, contains the following two steps:

(a) If attempts to establish high head safety injection using the high head portion of safety injection are ineffective, then depreesurize the intact steam generators (SG) to atmospheric pressure.

(b) If SG level is above a minimum value, then start the respective reactor coolant pump (RCP).

Which ONE of the following correctly describes the basis for these two steps?

A✓ (a) Intact SG are depressurized to lower the pressure in the RCS to raise accumulator flow and low head ECCS flow.

(b) The steam generator level must be above a minimum value to ensure that SG tubes are kept cool to prevent creep rupture failure of the SG tubes after starting the RCP.

B. (a) Intact SG are depressurized to lower the pressure in the RCS to raise accumulator flow and low head ECCS flow.

(b) The steam generator level must be above a minimum value to ensure a heat sink is available for cooling the core once the RCP is started.

C. (a) Intact SG are depressurized to lower the pressure in the RCS to prevent pressurized thermal shock conditions in the reactor vessel.

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D. (a) Intact SG are depressurized to lower the pressure in the RCS to prevent pressurized thermal shock conditions in the reactor vessel.

(b) The steam generator level must be above a minimum value to ensure that SG tubes are kept cool to prevent creep rupture failure of the SG tubes after starting the RCP.

K/A
074 Inad. Core Cooling
G2.4.6 Knowledge of symptom based EOP mitigation strategies.

K/A MATCH ANALYSIS
The question tests knowledge of the symptoms based EOP mitigation strategy, namely the basis behind EOP steps. Because the basis is being tested, the question is considered SRO-only level.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. Per referenced lesson plan and cooborated with two Vogtle Exam Bank Questions.
B. Incorrect. See referenced lesson plan. Plausible because it is desirable to have SG level for core cooling also, but the basis for the step is as stated in the lesson plan.
C. Incorrect. See referenced lesson plan. Plausible because it is desirable to have SG level for core cooling also, but the basis for the step is as stated in the lesson plan. Also plausible because lowering RCS pressure is considered a good thing when trying to prevent PTS (but cooling down is not).
D. Incorrect. See referenced lesson plan. Plausible because it is partially correct.

REFERENCES
1. LO-LP-37061-10, Response to Inadequate Core Cooling, Rev. 10, 01/09/2002.
2. Vogtle Exam Bank Question LO-LP-37061-02-04.
3. Vogtle Exam Bank Question LO-LP-37061-02-06

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	A D A D A A B C A D	Scramble Range: A - D
Tier:	1		Group:	2	
Key Word:	INADEQUATE CORE COOL		Cog Level:	MEM 4.0	
Source:	M		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

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			Answer:	A D A D A A B C A D	Scramble Range: A - D

Tier:	1	Group:	2
Key Word:	INADEQUATE CORE COOL	Cog Level:	MEM 4.0
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QUESTIONS REPORT
for Voglte 2005-301 Draft

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Key Word:	INADEQUATE CORE COOL		Cog Level:	MEM 4.0		
Source:	M		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	Response To Inadequate Core Cooling	NUMBER:	LO-LP-37061-10
PROGRAM:	Licensed Operator	REVISION:	10
SME:	Perry Tucker	DATE:	January 9, 2002
APPROVED:	***D. Scukanec***	DATE:	1/21/2002

Instructor Guidelines:

I. FORMAT

A. Verbal lecture with visual aids

II. MATERIALS

A. Overhead projector

B. Transparencies

C. White board with markers

III. EVALUATION

A. Oral or written exam in conjunction with other lesson plans

IV. REMARKS

A. Ensure students have latest revision of EOP

B. Performance-based instructional units (IUs) are attached to the lesson plan as student handouts. After the lecture on Inadequate Core Cooling, the student should be given adequate self-study time for the IUs. The instructor should direct self-study activities and be available to answer questions that may arise concerning the IU material. After self-study, the student will perform, simulate, observe, or discuss (as identified on the cluster signoff criteria list) the task covered in the instructional unit in the presence of an evaluator.

2) With CETCs above 711°F superheat at the core exit is indicated

3) At this coolant level and with CETCs greater than 711°F an inadequate core cooling condition has been reached

3. If either core cooling red path conditions exist, the operators are directed to Procedure 19221, "Response to Inadequate Core Cooling". The use of these actions is intended to be minimized since they are extraordinary and beyond the original design basis of the equipment or could lead to jeopardizing other Critical Safety Functions.

B. Vogtle Procedure 19221 was developed from Westinghouse Function Guideline (FRG) FR-C.1, "Response to Inadequate Core Cooling"

1. The purpose of this procedure is to provide actions to restore core cooling — Objective 1

2. The major action steps to be performed in 19221 are: — LO-TP-37061-005

a. Reinitiate high pressure SI

b. Rapid depressurization of the secondary

c. Restart RCPs and/or open PORVs and vents

3. These actions are done in order and the effectiveness of each is evaluated prior to performing the next

4. The preferred means of establishing core cooling is with high head safety injection.

a. If this is ineffective or unavailable, the secondary side of the plant is rapidly depressurized.

1. These actions will cool and depressurize the RCS, allowing the accumulators to dump and low head SI to inject.

2. Considered more effective in reducing RCS pressure than opening PORVs if the RCS is highly voided, and also does not result in additional RCS inventory losses.

b. The third action, starting RCPs and opening PORVs, is done only if the first two are ineffective in restoring core cooling.

1. Starting RCPs will provide temporary core

d. Continued RCS depressurization will cause SI accumulator injection and temporary core recovery. Accumulators are isolated after injection to prevent nitrogen injection which would reduce effectiveness of secondary heat sink.

e. Check the RCS hot leg temperature trends to determine effectiveness of SG depressurization in reducing RCS pressure.

f. CETC and hot leg temperatures may initially increase as superheated steam is forced out of core by the advancing froth but should quickly decrease to saturation and continue to decrease as RCS depressurizes.

g. Continued SG depressurization to atmospheric conditions will enhance low-head SI injection flow

h. If temperature and inventory are satisfied, exit procedure

3. RCP restart and/or opening pressurizer PORVs — *Steps 18-24*

a. Starting RCPs will provide forced two-phase flow through core and temporarily improve core cooling. CETCs should rapidly decrease and RVLIS dynamic range should rapidly increase as a steam-water mixture is forced through the core by the RCPs. — *Normal conditions not required for starting RCPs*

b. RCPs are only started in this step if there is sufficient water level in their associated SG to protect the steam generator tubes from creep rupture failure — *SAMG Phenomena*

c. RCPs will maintain core cooling as long as they continue to run with a secondary heat sink available. However, degraded core cooling conditions still exist.

d. RCPs will not run indefinitely under highly voided RCS conditions. Therefore, still required to:

1) Reduce RCS pressure

2) Inject SI accumulators

3) Increase SI flow - low head SI pumps

e. If unable to reduce RCS pressure via S/G's only option is to enlarge hole in RCS to reduce pressure

LO-LP-37061-02-04
In 19221-C, "Response to Inadequate Core Cooling", if attempts to establish adequate core cooling using the high head portion of ECCS are ineffective, the intact SGs are depressurized to atmospheric pressure. Which one of the following describes what the depressurization of the SGs is intended to accomplish?

A. To prevent the SGs from becoming faulted or suffering a tube rupture

B. To depressurize the RCS to increase accumulator and low head ECCS flow

C. To depressurize the RCS to prevent PTS of the reactor vessel

D. To minimize the length of time that the low head SI pumps will run on mini-flow, thereby decreasing the likelihood that they will trip due to overheating before they can be used for ECCS injection

LO-LP-37061-02
Using EOP 19221 as a guide, briefly describe how each step is accomplished.

LO-LP-37061-02-06

During an Inadequate Core Cooling event, 19221-C, a step in "Response To Inadequate Core Cooling" directs the RCPs to be started. There is however, a requirement that the associated SG NR level be at a minimum value before the its respective RCP can started. What is the purpose for the level requirement?

A. It ensures a heat sink is available for cooling the core once the RCP is started.

B. The level requirement ensures a "thermal layer" of water is available to maintain SG pressure as high as possible in the event of a SG tube failure.

C. It provides cooling for the high temperature gases that will be circulating through the loop. Cooling of the gases will lower RCS pressure, allowing low head ECCS injection to provide additional core cooling.

D. Ensures the SG tubes are kept cool to prevent Creep Rupture Failure of the SG tubes.

LO-LP-37061-02

Using EOP 19221 as a guide, briefly describe how each step is accomplished.

1. 076G2.4.11 001

"A" train nuclear service cooling water is completely lost while running a surveillance test on the "A" diesel generator (DG). The SRO has directed the RO to emergency trip the "A" DG.

Which ONE of the following describes the correct actions to be taken by the SRO while performing 18021-C, Loss of Nuclear Service Cooling Water?

A. Direct the operator to depress and hold the "A" Run / Stop Push Button ("A" Pull-To-Run / Push-To-Stop Button), located at the DG. Due to Technical Specification 3.0.6, which describes the rules of operability for supported equipment, the "A" DG does not need to be declared inoperable.

B. Direct the operator to depress and release the "A" Run / Stop Push Button, located at the Engine Control Panel. Declare the "A" DG inoperable and enter Technical Specification 3.8.1, "AC Sources - Operating".

C. Direct the operator to transfer the "A" DG to local control and place in maintenance mode. Due to Technical Specification 3.0.6, which describes the rules of operability for supported equipment, the "A" DG does not need to be declared inoperable.

D. Direct the operator to transfer the "A" DG to local control and place in maintenance mode. Declare the "A" DG inoperable and enter Technical Specification 3.8.1, "AC Sources - Operating".

K/A
076 Service Water
G2.4.11 Knowledge of abnormal condition procedures.

K/A MATCH ANALYSIS
The question tests knowledge of operability and shutting down the DG to preclude inadvertant start when NSCW is not available (guidance in AOP). The question is SRO-only knowledge because of the operability determination.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. The DG is inoperable with no NSCW to support operation. Plausible because TS 3.0.6 does state that in most cases you do not need to cascade TS.
B. Incorrect. The Run / Stop PB is located at the DG, not the Engine Control Panel and it would also need to be held in the depressed position. Plausible because the not all PBs need to be held.
C. Incorrect. See analysis for "A" above.
D. Correct. Refer to 18021-C, Page 6 and Tech Spec 3.7.8.

REFERENCES
1. 13145-1, Diesel Generators, Rev. 56, Page 28 and 29.
2. Tech Spec 3.8.1
3. Tech Spec 3.7.8
4. Tech Spec 3.0.6
5. 18021-C, Loss of Nuclear Service Cooling Water System, Rev. 13

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D C C A C C C D C D		Scramble Range: A - D
Tier:	2		Group:	1		
Key Word:	EDG NCSW		Cog Level:	MEM 3.6		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

18

1. 076G2.4.11 001

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C. Direct the operator to transfer the "A" DG to local control and place in maintenance mode. Due to Technical Specification 3.0.6, which describes the rules of operability for supported equipment, the "A" DG does not need to be declared inoperable.

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Pull-to / to

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4. Tech Spec 3.0.6
5. 18021-C, Loss of Nuclear Service Cooling Water System, Rev. 13

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D C C A C C C D C D	Scramble Range: A - D	
Tier:	2		Group:	1		
Key Word:	EDG NCSW		Cog Level:	MEM 3.6		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8. Perform the following (see ATTACHMENT A for handswitch listing): a. Start fans in UNAFFECTED train: • CTB Coolers in high speed • CTB Aux Air Cooler • Reactor Cavity Cooler b. Place fans in AFFECTED train in PULL-TO-LOCK or STOP as required: • CTB Coolers high speed • CTB Coolers low speed • CTB Aux Air Cooler • Reactor Cavity Cooler	
9. IF either diesel generator cannot be operated due to the loss of NSCW, THEN place it in maintenance mode by initiating 13145, DIESEL GENERATORS.	
10. Perform a Safety Function Determination per Tech Spec 3.0.6 to determine the Tech Spec impact of the loss of NSCW.	
11. Restore the NSCW train to operation within 72 hours.	11. Comply with Tech Spec LCO 3.7.8 or LCO 3.7.9.

III. LESSON OUTLINE:	NOTES:

F. If Loss is Sustained, Shutdown the Affected Train DG and implement Associated Technical Specifications.

1. The DG requires NSCW cooling to be Operable.

   a. A fully loaded DG could overheat in as little as three minutes following the loss of cooling. — Objective 1

   b. If the DG is running and not required, or the other DG is available, the DG loads should be transferred (or secured) and the DG shutdown as soon as possible.

2. A shutdown DG should be placed in a condition that would prevent an emergency start.

   a. Transfer the DG to LOCAL Control and place in the MAINTENANCE MODE. — 13145

   b. Easiest method is to press the Emergency Stop Push buttons on the QEAB.

3. Rendering the DG inoperable requires that the associated TS actions be initiated (T.S. 3.8.1 or 3.8.2)

G. Implement Affected Tech Spec Actions

1. There are two T.S. associated with NSCW

   a. T.S. 3.7.8 - Deals with operability of the NSCW pumping system.

      Requires restoration of an inoperable system w/in 72 hrs or a shutdown is required.

   b. T.S. 3.7.9 - Deals with the Ultimate heat Sink

      1) Immediate problems with capacity or heat removal capability require restoration w/in 72 hours, or a subsequent plant shutdown

      2) Loss of Makeup Capability allows up to 8 days for restoration before a shutdown must be considered.

2. Check NSCW return Temperatures <95°F

   a. Start Fans as necessary

   b. Or shift loads to other train as necessary

3.7 PLANT SYSTEMS

3.7.8 Nuclear Service Cooling Water (NSCW) System

LCO 3.7.8 Two NSCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One NSCW train inoperable.	-------------------NOTES------------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by NSCW system. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by NSCW system. ---	
	A.1 Restore NSCW system to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.

LCO 3.0.4 is only applicable for entry into a MODE or other specified Condition in the Applicability in MODES 1, 2, 3, and 4.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s).

Automatic load sequencers for Train A and Train B ESF buses shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Perform SR 3.8.1.1 for required OPERABLE offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		(continued)

NOTES

a. Due to oiling of the cylinders, some oil is expected to be discharged from the cylinder head indicator cocks while rolling the engine.

b. A small amount of moisture mist is expected to be discharged from the indicator cocks while rolling the engine.

4.4.1.15 DEPRESS the Engine Roll Pushbutton, and ROLL the engine on starting air for at least two revolutions.

4.4.1.16 CHECK all cylinder cocks for evidence of moisture.

4.4.1.17 CLOSE all cylinder cocks.

4.4.1.18 DEPRESS the OPERATIONAL mode pushbutton 1-HS-4575 (4576).

4.4.1.19 OBSERVE the blue UNIT AVAILABLE light comes ON and the red STOPPING light goes OFF.

4.4.1.20 PLACE the LOCAL/REMOTE Switch 1-HS-4516 (4517) in REMOTE.

4.4.1.21 COMPLETE Checklist 3 for Diesel Generator 1A (Checklist 4 for Diesel Generator 1B), "Cylinder Moisture Check Independent Verification".

4.4.2 **Emergency Stopping Train A (B) Diesel Generator.**

CAUTION

An Emergency Stop signal will trip the Diesel Generator under all conditions and will prevent re-starting the engine until manually reset.

4.4.2.1 To initiate an Emergency Stop from the Electrical Auxiliary Board:

a. DEPRESS both Emergency Stop Pushbuttons 1-HS-4567B (4568B) and 1-HS-4567C (4568C),

b. VERIFY that generator voltage drops to zero.

4.4.2.2 To initiate an Emergency Stop from the Engine Control Panel:

a. At the Engine Control Panel, DEPRESS Emergency Stop Pushbutton 1-HS-4567A (4568A),

b. VERIFY that red EMERGENCY STOP lamp comes ON.

CAUTION

If a Diesel Generator start signal is present the engine will restart when the RUN/STOP Pushbutton is released.

Deleted: Valve

4.4.2.3 To initiate an emergency stop from the engine:

a. At the southwest end of the engine, PUSH and HOLD the RUN/STOP Pushbutton, 1-HS-4688 (4689).

Deleted: Valve

b. VERIFY the engine comes to a complete stop.

c. RELEASE the RUN/STOP Pushbutton, 1-HS-4688 (4689).

Deleted: Valve

NOTE

An Emergency Stop signal can only be reset from the Engine Control Panel.

4.4.2.4 After the engine has stopped, DEPRESS the RESET FROM LOCA/LOSP Pushbutton 1-HS-4583 (4584) at the Engine Control Panel.

4.4.2.5 VERIFY the green LOSS OF OFFSITE PWR OR SAFETY INJ SIGNAL lamp goes OFF.

4.4.2.6 DEPRESS Emergency Stop Reset Pushbutton 1-HS-4581 (4582) at the Engine Control Panel.

4.4.2.7 VERIFY that the red EMERGENCY STOP lamp goes OFF.

4.4.2.8 NOTIFY the Diesel Generator System Engineer of the Diesel Generator operation by dispatching the following:

a. A completed copy of Completion Sheet 1,

b. a copy of every completed 11885-C, "Diesel Generator Operating Log" if taken.

19. G2.1.14 001

While performing an emergency downpower, all annunciators in the Unit 1 Control Room are unexpectedly lost at 0900 hours and the SS makes an emergency classification at 0910 hours.

Which ONE of the following describes the emergency classification and required notifications?

A. The SS was required to declare a NOUE and was expected to notify plant personnel by 0905.

B. The SS was required to declare a NOUE and was expected to notify plant personnel by 0915.

C. The SS was required to declare an Alert and was expected to notify plant personnel by 0905.

D✓ The SS was required to declare an Alert and was expected to notify plant personnel by 0915.

K/A

G2.1.14

Knowledge of system status criteria which require the notification of plant personnel.

K/A MATCH ANALYSIS

ED is expected to notify plant personnel within 5 minutes of declaring an Alert or higher. The system status portion of the K/A is met by giving them a total loss of annunciators.

ANSWER / DISTRACTOR ANALYSIS

A. Incorrect. Unplanned loss of annunciators places plant in automatic Alert.
B. Incorrect. Unplanned loss of annunciators places plant in automatic Alert.
C. Incorrect. Expectation is within 5 minutes of declaring Alert or higher.
D. Correct. Alert declared at 0910 and expectation is within 5 minutes of declaring Alert.

All distractors are plausible based on memory nature of items.

REFERENCES

1. LO-LP-40101-39-C, EPIP Overview, Rev. 39, 05/03/2004.
2. 91001-C, Emergency Classification and Implementing Instructions, Rev. 20.1, 09/12/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D C C B B C D C C Scramble Range: A - D

Tier:	3	Group:	
Key Word:	NOTIFICATIONS ALARMS	Cog Level:	C/A 3.3
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

19. G2.1.14 001

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D✓ The SS was required to declare an Alert and was expected to notify plant personnel by 0915.

K/A
G2.1.14
Knowledge of system status criteria which require the notification of plant personnel.

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MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D D C C B B C D C C Scramble Range: A - D

QUESTIONS REPORT
for Voglte 2005-301 Draft

Tier:	3	Group:	
Key Word:	NOTIFICATIONS ALARMS	Cog Level:	C/A 3.3
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

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D✓ The SS was required to declare an Alert and was expected to notify plant personnel by 0915.

K/A
G2.1.14
Knowledge of system status criteria which require the notification of plant personnel.

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All distractors are plausible based on memory nature of items.

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1. LO-LP-40101-39-C, EPIP Overview, Rev. 39, 05/03/2004.
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QUESTIONS REPORT
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Tier:	3	Group:	
Key Word:	NOTIFICATIONS ALARMS	Cog Level:	C/A 3.3
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

	ELECTRICAL POWER	RADIOACTIVITY	PLANT SYSTEMS	NATURAL PHENOMENON	HAZARDS	SECURITY	BARRIERS	SHUTDOWN SYSTEMS	OTHER
GENERAL	Modes 1-4: Loss of AC power to BOTH AA02 AND BA03 AND EITHER Restoration of at least one emergency bus NOT likely within 4 hrs of time of loss. OR Loss or Potential Loss of FUEL CLAD BARRIER determined from Fission Product Barrier EAL's (Figure 4 Cont. sheet 1)	Valid reading on RE-12444 that is > 30E+0 µci/cc for > 15 min. OR this value is expected to be exceeded for > 15 min. (Note 1) Valid site boundary dose assessment of release indicates > 1000 mrem TEDE OR > 5000 mrem CDE thyroid OR field survey results indicate > 1000 mrem/hr that is expected to continue for > 1 hr OR field survey samples indicate thyroid dose of 5000 mr CDE for 1 hr of inhalation				Loss of physical control of the control room due to a security event OR loss of physical control of remote shutdown capability due to a security event	Loss of THREE barriers Loss of ANY TWO barriers AND Potential Loss of the THIRD barrier Modes 1-2: Subcriticality CSFST is RED AND EITHER Core Cooling OR Heat Sink CSFST is RED		Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate potential for uncontrolled radionuclide releases that can reasonably be expected to exceed 1000 mrem TEDE or 5000 mrem CDE thyroid dose levels outside the site boundary OR Actual or imminent substantial core degradation with potential for loss of containment
SITE AREA	Modes 1-4: Loss of AC power on BOTH AA02 AND BA03 for > 15 min. Modes 1-4: Unplanned loss of voltage on ALL Vital DC buses (AD1, BD1, CD1, and DD1) for > 15 min.	Valid reading on RE-12444 that is > 3E+0 µci/cc for > 15 min. OR on RE-12839 that is > 5E+2 µci/cc for > 15 min. OR these values are expected to be exceeded for > 15 min. (Note 1) Valid site boundary dose assessment of release indicates > 100 mrem TEDE OR > 500 mrem CDE thyroid OR field survey results indicate > 100 mrem/hr that is expected to continue for > 1 hr OR field survey samples indicate thyroid dose of 500 mrem CDE for 1 hr of inhalation	Mode 1-2: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip did NOT occur from the control room Modes 1-4: Significant transient is in progress and ESF annunciators needed to monitor the transient are not available AND ESF control board AND compensatory non-alarming indications are NOT available		Control room evacuation has been initiated AND control of the plant CANNOT be established from remote shutdown panels within 15 min.	Intrusion into a plant vital area by a hostile force (e.g., confirmed bomb device discovered within Vital area)	Loss of TWO barriers Loss of ONE barrier AND Potential Loss of a SECOND barrier Potential Loss of BOTH Fuel Clad AND RCS barriers Modes 1-4: The reactor is tripped AND Subcriticality CSFST is RED	Modes 5-6: Loss of reactor vessel water level as indicated by loss of RHR cooling as determined by AOP 18019-C AND any ONE of the following: Vessel level < 183.5 (< 62% RVLIS) OR Containment area rad monitors RE-0002/0003 valid high alarm (15 mr/hr) with vessel head removed, OR Core exit temperatures > 711° F with vessel head installed. Modes 1-4: Complete loss of function needed to achieve or maintain Hot Shutdown (EITHER Core Cooling OR Heat Sink CSFST is RED) AND no other Heat Sink available	Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate actual or likely major failures of plant functions needed for protection of the public.
ALERT	Modes 1-4: Loss of voltage on EITHER AA02 OR BA03 for > 15 min. AND the remaining energized 1-E bus does not have a backup power supply available. Modes 5, 6, or Defueled: Loss of AC power on BOTH AA02 AND BA03 for > 15 min.	Valid reading on RE-12444 that is > 2E-1 µci/cc for > 15 min. OR on RE-12839 that is > 6E+0 µci/cc for > 15 min. OR on RE-018 that is > 8E-1 µci/cc for > 15 min. (Note 1) Confirmed sample analysis for gaseous OR liquid release indicates concentrations or release rates > 200X ODCM limits for > 15 min. Radiation levels higher than normal in Fuel Handling or Containment Bldg AND irradiated fuel is uncovered, OR < 104 feet water level in refueling cavity, spent fuel pool, or fuel transfer canal that results in uncovering irradiated fuel. (Note 2) Damage to irradiated fuel causing a valid high alarm on one or more of the following: RE-000 (2.5 mr/hr), RE-2532/2533 A/B (5E-4 µci/cc), or during Mode 6 fuel transfer RE-0?/?8 (15 mr/hr) Valid rad levels > 10 mr/hr in the Control room OR > 100 mr/hr in areas requiring infrequent access to maintain plant safety functions (e.g., local charging station)	Modes 1-3: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip occurred from the control room. Modes 1-4: Unplanned loss of most or all annunciators or indicators in the control room for plant safety systems AND EITHER a significant transient is in progress or compensatory non-alarming indications are NOT available	Tornado striking plant vital area OR sustained hurricane force winds of 100 mph or greater on site as verified by meteorological instrumentation (15 minute average) Seismic monitoring system confirms seismic event > 0.12 g.	Aircraft crash confirmed to affect plant vital areas. Report or detection of toxic, flammable, or asphyxiant gasses within a facility structure in concentrations that requires evacuation of a room or area needed for safe operation of the plant. Major fire or explosion in a vital area AND affected safety system parameters show degraded performance or there is visible damage to permanent safety related structures or safety related equipment within the specified area Report of visible structural damage (caused by natural or destructive phenomena), which threatens the ability of the structure to perform its safety related function to ANY of the following plant structures: Containment, Auxiliary Building, Control Building, Fuel Handling Building, Diesel Generator Building, Condensate Storage Tank, AFW Pumphouse, NSCW Cooling Tower, RWST Control Room evacuation has been initiated	Intrusion into the Protected area by a hostile force (e.g., unauthorized vehicle penetrates protected area in a hostile manner)	Loss or Potential Loss of Fuel Clad barrier Loss or Potential Loss of RCS barrier	Modes 5-6: Unplanned loss of RHR cooling AND EITHER RCS temperature is > 200° F OR RCS temperature is increasing uncontrolled toward 200° F.	Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate plant safety systems may be substantially degraded and that increased monitoring of plant functions is warranted.
NOUE	Loss of Off-Site power to BOTH AA02 AND BA03 for > 15 min. (neither bus is connected to an energized Off-Site source) AND BOTH AA02 AND BA03 are powered by Diesel Generators Modes 5-6: Unplanned loss of voltage on ALL Vital DC buses (AD1, BD1, CD1, and DD1) for > 15 min.	Valid reading on RE-12444/12442 that is > 2E-3 µci/cc for > 60 min. OR on RE-12839 that is > 6E-2 µci/cc for > 60 min. OR on RE-018 that is > 8E-3 µci/cc for > 60 min. (Note 1) Confirmed sample analysis for gaseous OR liquid release indicates concentrations or release rates > 2X ODCM limits for > 60 min. Radiation levels higher than normal in Fuel Handling or Containment Bldg AND uncontrolled water level decrease in refueling cavity, spent fuel pool, or fuel transfer canal, BUT all irradiated fuel assemblies remain covered with water. (Note 2) Valid area rad monitor readings increase 1000X over normal levels (Note 2)	Modes 4-5: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip occurred from the control room Modes 1-4: Unplanned loss of most or all annunciators or indicators in the control room for plant safety systems for > 15 min. AND compensating non-alarming indicators are available Loss of ALL of the following On-Site communications systems: In-plant telephone, Gaitronics, Sound powered phone, and Plant radio communications Loss of ALL of the following Off-Site communications systems: ENN and Telephone capability to Off-Site network	Report of tornado striking within protected area. Hurricane force winds of 74 mph or greater forecast by the National Weather Service (NWS-Columbia S.C. office) to be at the plant site in the next four hours. Seismic monitoring system indicates seismic event OR plant operators report an earthquake was "felt"	Aircraft crash causes damage to safety related plant structures or safety related systems within the protected area. Report or detection of toxic, flammable or asphyxiant gasses that could enter the site area in amounts > life-threatening or flammable concentrations that could affect normal operation of the plant. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. Fire in an area contiguous or adjacent to a vital area that is not extinguished within 15 min. of control room notification by fire alarm or personnel report. Unanticipated explosion within protected area resulting in visible damage to permanent structures or equipment.	Confirmed Security Event which indicates a potential degradation in the level of safety of the plant. (i.e.,) 1. Bomb device in Protected Area but outside Vital Area. 2. Hostage situation inside Protected Area. 3. Civil disturbance on plant site outside Protected Area. 4. Credible Bomb threat (Note 3) 5. Credible Attack threat (Note 3) Notification by Local, County, or State officials of potential for evacuation of site personnel based on an Off-Site event	Loss or Potential Loss of Containment barrier RCS Chemistry analysis indicates Dose equivalent I-131 > 1 µci/gm for > 48 hrs. or in excess of Tech Spec. figure 3.4.16-1 OR RCS specific activity > 100/Ē µci/gm gross radioactivity. RCS Unidentified leakage > 10 gpm OR RCS Pressure boundary leakage > 10 gpm OR RCS identified leakage > 25 gpm		Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate potential degradation of the level of safety of the plant. Modes 1-4: Plant NOT brought to required operating mode within Tech Spec LCO OR TRM Technical Requirement required action completion time limit. Modes 1-4: Uncontrolled depressurization of one or more steam generators.

NOTE 1: Classification should be based on ODCM or Off-site Dose Calculation computer program results, however, if the monitor reading(s) is sustained for longer than the period indicated and release assessment has NOT or CANNOT be completed within this period, then declaration MUST be made based on the valid reading.

NOTE 2: "Normal levels" are the highest reading in the last 24 hours prior to the emergency excluding the current peak value.
NOTE 3: See section 2.0 of this procedure.

Figure 4 - EMERGENCY CLASSIFICATION LEVEL DETERMINATION

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	EPIP Overview	NUMBER:	LO-LP-40101-39-C
PROGRAM:	Licensed Operator Training	REVISION:	39
SME:	Perry Tucker	DATE:	April 27, 2004
APPROVED:	D. Scukanec	DATE:	May 3, 2004

INSTRUCTOR GUIDELINES:

I. LESSON FORMAT

A. Lecture with Visual Aids.
B. Initial Training: Two four hour classes.
C. Requal MRE (RE-080) Training: Requal objectives along with Section III (I.B) criteria.

II. MATERIALS NEEDED

A. White Board with Markers.
B. Overhead Projector.
C. Transparencies.
D. Series 91000-C Emergency Plan Implementation Procedures.
E. Emergency Planning Zone Maps.

III. EVALUATION

A. Written or Oral Exam in conjunction with other Lesson Plans for Initial training.
B. Written exam for Requalification (Management Of Radiological Emergencies) training. Passing criteria requires a final grade of at least 80% for MRE.

IV. REMARKS

A. Training in Emergency Plan responsibilities and core damage assessment for SRO's is required per 60601-C. Training for RO's in Emergency Plan responsibilities is required per 60602-C. This training also satisfies the annual training requirements for Emergency Directors as specified in 91601-C and Section O of the Vogtle Emergency Plan.

2) Event Notification Worksheet (Form 361)

d. Directions for ENN Communicators (Checklist 4)

1) ERO Recall Instructions

2) Message Notification Instructions

3) Back-up Communications

4. Plant Page Announcement Checklist — Checklist 1

a. It is important that this announcement be made expeditiously since this alerts emergency response personnel to begin their appropriate actions as applicable — ED delegates

1) Expected to be completed within 5 minutes of the declaration of an Alert or higher emergency

2) ASAP for any subsequent upgrades

b. Use Sheet 1 for most emergency classifications in a menu / "fill in the blank" format.

c. Use Sheet 2 for Security emergencies

1) Warble tone is not activated

5. Emergency Recall System Instructions — Checklist 4

a. Will be completed by a control room communicator

b. ERO personnel should be recalled only if an Alert, SAE or GE has been declared or when Directed by the ED.

1) VEGP Management should be notified of an NOUE regardless of day or time of day

c. Primary and Backup System/Method

1) Envelopes located in the CR ED black binder

a) Tape sealed until used

2) Manila colored

a) Primary

3) Red colored

1. G2.2.21 001

The "A" Train RHR Pump impeller was replaced and a significant portion of its motor control circuit wiring was replaced.

Which ONE of the following correctly describes post maintenance testing requirements for the work that was performed?

A. Perform flow capacity performance tests (Inservice Test) at the Tech Spec Runout flow or maximum Accident Analysis flow, but mini-flow conditions are not required to be tested. An Auto-start Functional Test must also be performed, but a Manual Start/Stop Test is not required to be performed.

B.✓ Perform flow capacity performance tests (Inservice Test) at the Tech Spec Runout flow or maximum Accident Analysis flow AND at mini-flow conditions. Test signals shall be used to verify the auto-start functions and Manual Start/Stop Test must also be performed.

C. Perform flow capacity performance tests (Inservice Test) at the Tech Spec Runout flow or maximum Accident Analysis flow, but mini-flow conditions are not required to be tested. Test signals shall be used to verify the auto-start functions and Manual Start/Stop Test must also be performed.

D. Perform flow capacity performance tests (Inservice Test) at the Tech Spec Runout flow or maximum Accident Analysis flow AND at mini-flow conditions. An Auto-start Functional Test must also be performed, but a Manual Start/Stop Test is not required to be performed.

K/A
G2.2.21
Knowledge of pre- and post- maintenance operability requirements.

K/A MATCH ANALYSIS
The proper post-maintenance tests must be performed as a basis for declaring the pump operable. The question tests knowledge of PMT and operability, which are SRO-only required knowledge.

ANSWER / DISTRACTOR ANALYSIS
B. Correct. See referenced procedure, Pages 22 - 25.
A & C Incorrect. Plausible because applicant may not know the requirements for testing at mini-flow conditions because the more important concern may be flow to the core.
D. Incorrect. A Manual start/stop test is required. Plausible because the first part of the distractor is correct.

REFERENCES
1. 29401-C, Work Order Functional Tests, Rev. 20, 07/03/2003.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9	
			Answer:	B D D C C B D D C C	Scramble Range: A - D
Tier:	3		Group:		
Key Word:	RHR PMT OPERABILITY		Cog Level:	MEM 3.5	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

20. G2.2.21 001

The "A" Train RHR Pump impeller was replaced and a significant portion of its motor control circuit wiring was replaced.

Which ONE of the following correctly describes post maintenance testing requirements for the work that was performed?

A. Perform flow capacity performance tests (Inservice Test) at the Tech Spec Runout flow or maximum Accident Analysis flow, but mini-flow conditions are not required to be tested. An Auto-start Functional Test must also be performed, but a Manual Start/Stop Test is not required to be performed.

B. Perform flow capacity performance tests (Inservice Test) at the Tech Spec Runout flow or maximum Accident Analysis flow AND at mini-flow conditions. Test signals shall be used to verify the auto-start functions and Manual Start/Stop Test must also be performed.

C. Perform flow capacity performance tests (Inservice Test) at the Tech Spec Runout flow or maximum Accident Analysis flow, but mini-flow conditions are not required to be tested. Test signals shall be used to verify the auto-start functions and Manual Start/Stop Test must also be performed.

D. Perform flow capacity performance tests (Inservice Test) at the Tech Spec Runout flow or maximum Accident Analysis flow AND at mini-flow conditions. An Auto-start Functional Test must also be performed, but a Manual Start/Stop Test is not required to be performed.

K/A
G2.2.21
Knowledge of pre- and post- maintenance operability requirements.

K/A MATCH ANALYSIS
The proper post-maintenance tests must be performed as a basis for declaring the pump operable. The question tests knowledge of PMT and operability, which are SRO-only required knowledge.

ANSWER / DISTRACTOR ANALYSIS
B. Correct. See referenced procedure, Pages 22 - 25.
A & C Incorrect. Plausible because applicant may not know the requirements for testing at mini-flow conditions because the more important concern may be flow to the core.
D. Incorrect. A Manual start/stop test is required. Plausible because the first part of the distractor is correct.

REFERENCES
1. 29401-C, Work Order Functional Tests, Rev. 20, 07/03/2003.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	B D D C C B D D C C		Scramble Range: A - D
Tier:	3			Group:		
Key Word:	RHR PMT OPERABILITY			Cog Level:	MEM 3.5	
Source:	N			Exam:	VG05301	
Test:	S			Author/Reviewer:	MAB/RSB	

1. G2.2.21 001

The "A" Train RHR Pump impeller was replaced and a portion of its motor control circuit wiring was replaced.

Which ONE of the following correctly describes post maintenance testing requirements for the work that was performed? (IST or Inservice Test)

A. Perform flow capacity performance tests at the Tech Spec Runout flow or maximum Accident Analysis flow, but mini-flow conditions are not required to be tested. An Auto-start Functional Test must also be performed, but a Manual Start/Stop Test is not required to be performed.

B. Perform flow capacity performance tests at the Tech Spec Runout flow or maximum Accident Analysis flow AND at mini-flow conditions. Test signals shall be used to verify the auto-start functions and Manual Start/Stop Test must also be performed.

C. Perform flow capacity performance tests at the Tech Spec Runout flow or maximum Accident Analysis flow, but mini-flow conditions are not required to be tested. Test signals shall be used to verify the auto-start functions and Manual Start/Stop Test must also be performed.

D. Perform flow capacity performance tests at the Tech Spec Runout flow or maximum Accident Analysis flow AND at mini-flow conditions. An Auto-start Functional Test must also be performed, but a Manual Start/Stop Test is not required to be performed.

K/A
G2.2.21
Knowledge of pre- and post- maintenance operability requirements.

K/A MATCH ANALYSIS
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Tier:	3		Group:			
Key Word:	RHR PMT OPERABILITY		Cog Level:	MEM 3.5		
Source:	N		Exam:	VG05301		
Test:	S		Author/Reviewer:	MAB/RSB		

QUESTIONS REPORT
for Voglte 2005-301 Draft

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D. Perform flow capacity performance tests at the Tech Spec Runout flow or maximum Accident Analysis flow AND at mini-flow conditions. An Auto-start Functional Test must also be performed, but a Manual Start/Stop Test is not required to be performed.

K/A
G2.2.21
Knowledge of pre- and post- maintenance operability requirements.

K/A MATCH ANALYSIS
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			Answer:	B D D C C B D D C C	Scramble Range: A - D
Tier:	3		Group:		
Key Word:	RHR PMT OPERABILITY		Cog Level:	MEM ~~C/A~~ 3.5	
Source:	N		Exam:	VG05301	
Test:	S		Author/Reviewer:	MAB/RSB	

WORK ORDER FUNCTIONAL TESTS

PROCEDURE USAGE REQUIREMENTS	SECTIONS
Continuous Use: Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed by the procedure.	**None**
Reference Use: Procedure or applicable section(s) available at the Work location for ready reference by person performing steps.	**None**
Information Use: Available on plant site for reference as needed.	**All**

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ATTACHMENT A (cont'd.)

PUMPS/PUMP MOTORS

Pump test activities consist of the following:

1. Pump auto start functional test.
2. Pump manual start operational test. (Verify pump discharge pressure during 15 minute operation).
3. Pump attendant system readiness verification (lube oil level/flow, cooling water flow, seal water flow).
4. Pump flow capacity performance test.
5. Inservice Test (IST) for pumps in the IST Program (see Notes at the end of this section).
6. Response time testing may be required for pumps in the IST program (see notes at end of this section).

Pump Motor Maintenance/Replacement

Following replacement of the pump motor or coupling device the pump shall be demonstrated operable by performance of pump attendant system readiness verification, and a pump flow capacity performance test.

If motor maintenance required disconnection of cooling water supply line(s), adequate cooling water flow shall be verified following reassembly.

If maintenance is performed on the pump motor then the pump shall be demonstrated operable by performing a pump attendant system readiness and by performance of a pump manual start operational test (check pump and motor bearing temperature/vibration during pump run period). Measure motor starting/running current. Verify heater is operational if plicable.

Pump Maintenance /Replacement

If maintenance is performed on pump internals (shaft, bearings, impeller, wearing rings, pump shaft seals, casing, etc.), the pump shall be demonstrated operable by performance of pump attendant system readiness checks, (NPSH checks if required), and a pump flow capacity performance test. Check pump bearing temperature/vibration during the pump run period.

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ATTACHMENT A (cont'd.)

PUMPS/PUMP MOTORS (cont'd.)

If maintenance is performed on safety-related pumps which requires the replacement of the rotating elements or changing running position of the rotating elements, the minimum requirements relative to flow capacity data range during a "Pump Flow Capacity Performance Test" are as follows:

1. Test at miniflow.

2. Test at Technical Specification runout flow or maximum accident analysis flow.

3. Test at a minimum of 3 data points between the above two points. The points chosen should correspond to other identified accident or normal operating points. In the absence of either of these, the data points should be approximately equally spaced over the operating curve.

Deviation from the flow capacity data range requirements of step 3 requires approval from Engineering Support department. (30102)

If maintenance is performed on pump packing gland, pump seal water tank or line, pump lubrication system (oil change, filter, heat exchanger, or pressure control device), or pump cooling water system (heat exchanger, strainer, etc.) then the pump shall be demonstrated operable by performing a pump attendant system readiness check and a pump manual start operational test to ensure that pump will start and operate satisfactorily.

Pump Control Circuit Maintenance

0-99 Horsepower

If maintenance is performed on the pump motor control circuit, the pump motor shall be demonstrated operable by ensuring that the pump motor starts and stops on demand by performing a manual start/stop test from all of the control locations. If the pump motor performs an auto start function, test signals shall be used to verify auto start capability. If an instrument which provides an interlock or auto start function undergoes maintenance, then the pump shall be demonstrated operable by performance of an auto start functional test via appropriate test signals. If the motor provides an interlock or auto close/open function in the control circuit of another component and it undergoes maintenance, the other component shall be demonstrated operable by performance of an automatic function test of all the open/close functions/interlocks via appropriate simulated test signals.

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ATTACHMENT A (cont'd.)

PUMPS/PUMP MOTORS (cont'd.)

100 Horsepower

If maintenance is performed on the motor control circuit of a pump, the pump motor shall be demonstrated operable by ensuring that the breaker/starter (in test position) closes and opens on demand by performing a manual start/stop test from all the control locations. If the pump motor performs an auto start function, then test signals shall be used to verify auto start capability. If an instrument which provides an interlock or auto start function undergoes maintenance, the pump shall be demonstrated operable by performance of an auto start functional test via appropriate test signals. If the motor provides an interlock or auto close/open function in the control circuit of another component and it undergoes maintenance, the other component shall be demonstrated operable by performance of an automatic function test of all the open/close functions/interlocks via appropriate simulated test signals.

Pump Motor Power Circuit Maintenance

Same as Pump Motor maintenance.

NOTES

IST surveillance procedure retesting is required for all pumps in the IST Program upon completion of the following maintenance activities:

- Replacement of rotating parts (including the motor or turbine) (*)
- Machining, welding, grinding or rebalancing of rotating parts (including the motor or turbine) (*)
- Disassembly and reassembly of rotating parts
- Bearing replacement or disassembly and reassembly with the same parts
- Mechanical seal maintenance
- Adjustment of pipe supports within 10 feet of the pump suction or discharge
- Realignment of the pump with its driver
- Packing adjustments, additions of packing rings or repacking of the stuffing box (gland)
- Any other work which might adversely affect pump performance (differential pressure, flow or vibration) (*)
- Response time test required, for any of the above that are indicated with an (*).

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ATTACHMENT A (cont'd.)

PUMPS/PUMP MOTORS (cont'd.)

IST surveillance procedure retesting is not required upon completion of the following maintenance activities:

- Routine oil changes, if the oil has not degraded to the point that mechanical damage is suspected
- Addition of oil
- Flushing of cooling systems (e.g., lube oil coolers)

Pumps that are included in the IST Program are flagged in the NPMIS Equipment file with the special indicator "INSP" as well as a note in the comments section.

Following painting, cleaning, or other cosmetic maintenance on the pump, or motor, no test is required.

During all test operations, prior to pump start, visually observe the pump attendant system, to ensure proper oil levels, seal water level/flow, cooling water flow, etc. During operation check pump discharge pressure, flowrate, bearing status (temperature and vibration) to ensure proper operation and ensure no excessive packing/seal leakage or other external pump leakage.

NSCW SUPPLIED MOTOR COOLERS

NSCW supplied motor cooler that have been removed, disassembled or drained for any type intenance activities which include orifice insepctions, must have the following performed:

- A review of procedure 83308-C should be performed to ensure a flow test is performed when required.

- If required, a flow verification test will be performed per procedure 83308-C " Testing Of Safety-Related NSCW System Coolers" to ensure proper water flow through each cooler.

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21. G2.2.7 001

The plant manager has assigned you to develop a test which is not described in the FSAR and requires a change to the Plant's Technical Specifications.

Which ONE of the following correctly describes the approval process with respect to 10 CFR 50.59, "Changes, Tests, and Experiments," in accordance with 00056-C, 10 CFR 50.59 Screening and Evaluations?

A. A 10 CFR 50.59 Screening is required, but no Evaluation is needed because the test will be classified as "Screened Out."

B. A 10 CFR 50.59 Evaluation is required with Plant Review Board approval. NRC approval is not required.

C✓ A 10 CFR 50.59 Evaluation and a License Amendment are required regardless of the effect on the risk to the public.

D. A 10 CFR 50.59 Evaluation and a License Amendment are required only if there is an increased risk to the pubic.

K/A
G2.2.7
Knowledge of the process for conducting tests or experiments not described in the safety analysis report.

K/A MATCH ANALYSIS
Question tests knowledge of conducting a test that is not described in the FSAR.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. A Safety Evaluation is required.
B. Incorrect. NRC approval is required because the TS change is a change to the Op License.
C. Correct. A License Amendment is required because of the TS Change.
D. Incorrect. The Safety Evaluation is required to determine if there is an increased risk to the public, therefore it is required even if there is no increased risk.

All distractors are plausible because they all pertain to parts of the process that must be followed when making changes to the plant or the procedures that guide operation of the plant.

REFERENCES
1. 00056-C, 10 CFR 50.59 Screenings and Evaluations, Rev. 21.2, 10/03/2001.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C B B B B A B B D B Scramble Range: A - D

Tier:	3	Group:	
Key Word:	50.59 TS FSAR TECH	Cog Level:	C/A 3.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

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K/A

G2.2.7

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Tier:	3	Group:	
Key Word:	50.59 TS FSAR TECH	Cog Level:	C/A 3.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

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C✓ A 10 CFR 50.59 Evaluation and a License Amendment are required regardless of the effect on the risk to the public.

D. A 10 CFR 50.59 Evaluation and a License Amendment are required only if there is an increased risk to the pubic.

K/A
G2.2.7
Knowledge of the process for conducting tests or experiments not described in the safety analysis report.

K/A MATCH ANALYSIS
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REFERENCES
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Answer: C B B B B A B B D B Scramble Range: A - D

QUESTIONS REPORT
for Voglte 2005-301 Draft

Tier:	3	Group:	
Key Word:	50.59 TS FSAR TECH	Cog Level:	C/A 3.2
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

PRB REVIEW REQUIRED

10CFR50.59 SCREENINGS AND EVALUATIONS

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	
Information Use:	Available on plant site for reference as needed.	ALL

e. Temporary shielding.

f. Changes to computer software. (CO0038694)

g. Removal of each non-functional annunciator. The evaluation must be completed prior to physical removal of the card.

h. Changes to plans/programs such as the Process Control Program, the Inservice Inspection Program, and the Spill Prevention Control and Countermeasures Plan.

i. Changes to commitments with the keyword "ITS NRC COMMITMENT." Changes to these commitments, other than editorial changes, require a 50.59 evaluation.

j. Condition Reports dispositioned "Use-as-is" or "Repair." (CO0038462)

2.0 DEFINITIONS

2.1 LICENSING DOCUMENT

Those documents that govern the operation of the Vogtle Electric Generating Plant. Licensing documents include the following: Quality Assurance (QA) Program, Emergency Plan, Final Safety Analysis Report (FSAR) (as updated), Fire Hazards Analysis (FHA) (as updated), Operating License, Technical Specifications, Technical Specification Bases, Environmental Protection Plan, Offsite Dose Calculation Manual (ODCM), Process Control Program (PCP), Core Operating Limits Report (COLR), Technical Requirements Manual (TRM), Pressure Temperature Limits Report (PTLR), Inservice Inspection and Testing Programs, and the Security Plan, Contingency Plan, and Security Training and Qualification Plan.

2.2 SCREENING

A process for determining whether a proposed activity requires a 10CFR50.59 evaluation to be performed. Those activities determined to require a 10CFR50.59 evaluation are classified as "Screened In." If the activity may be implemented without performing a 10CFR50.59 evaluation it is classified as "Screened Out."

2.3 10CFR50.59 EVALUATION

The documented evaluation of a proposed change, test, or experiment against the eight criteria in 10CFR50.59(c)(2). The evaluation determines if a proposed change, test or experiment requires a license amendment approved by the NRC prior to implementation.

CHANGE

A modification or addition to, or removal from, the facility or procedures that affects: (1) a design function, (2) the method of performing or controlling the function, or (3) an evaluation that demonstrates that intended function will be accomplished.

3.0 RESPONSIBILITIES

3.1 DEPARTMENT MANAGERS

Department managers have the following responsibilities:

3.1.1 Assure 10CFR50.59 screenings and evaluations prepared within their department receive adequate reviews.

3.1.2 Assure personnel within their department who prepare or review 10CFR50.59 screenings and evaluations, performed in accordance with this procedure, are qualified to support such activities.

3.2 SUPERVISORY PERSONNEL

Supervisory personnel will ensure Applicability Determinations are performed on proposed changes, tests, or experiments. Supervisory personnel may include managers, superintendents, supervisors, team leaders, and assistant team leaders. The term supervisor is used generically and may refer to any of these positions.

3.3 PERFORMANCE ANALYSIS MANAGER

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The Performance Analysis Manager is responsible for establishing a log for assigning a unique number for 10CFR50.59 evaluations and for ensuring transmittal of copies of completed evaluations to the Safety Review Board (SRB).

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3.4 PLANT REVIEW BOARD (PRB)

3.4.1 Reviews proposed changes to design, procedures, and licensing documents in accordance with Procedure 00002-C, "Plant Review Board-Duties And Responsibilities." (CO0038088) (CO0038163)

3.4.2 Provides determination whether the proposed activity can be implemented in accordance with applicable regulatory requirements.

2 10CFR50.59 CONSIDERATIONS

2.1 Quotation from 10CFR50.59

NOTE

For the purposes of these guidelines, Updated FSAR refers to the current FSAR as updated per 10 CFR 50.71(e), approved changes to the Updated FSAR which have not yet been submitted to the NRC by amendment, and documents incorporated into the Updated FSAR by reference.

(c)(1) A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to § 50.90 only if:

(i) A change to the technical specifications incorporated in the license is not required, and

(ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section.

(c)(2) A licensee shall obtain a license amendment pursuant to §50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

(i) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report (as updated);

(ii) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report (as updated);

(iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated);

(iv) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report (as updated);

(v) Create a possibility for an accident of a different type than any previously evaluated in the final safety analysis report (as updated);

(vi) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated);

4.3 **10CFR50.59 SCREENING CRITERIA**

4.3.1 Structure for Performing 10CFR50.59 Screenings

4.3.1.1 For the purposes of performing 10CFR50.59 screenings, the 10CFR50.59(c)(1) criteria identified in subsection 4.2.1 must be addressed. Using the guidance contained in NEI 96-07 these criteria have been rephrased into the following five questions.

1. Does the activity involve a modification, addition to, or removal of a structure, system, or component (SSC) such that a design function as described in the Updated FSAR is adversely affected?

2. Does the activity involve a change to procedures that adversely affects the performance or method of control of a design function as described in the Updated FSAR?

3. Does the activity involve an adverse change to a method of evaluation or use of an alternate method of evaluation from that described in the Updated FSAR that is used in establishing the design bases or in the safety analyses?

4. Does the activity involve a test or experiment not described in the Updated FSAR which is outside the reference bounds of the design bases as described in the Updated FSAR or is inconsistent with the analyses or descriptions described in the Updated FSAR?

5. Does the activity involve a change to the Technical Specifications and/or Environmental Protection Plan. Changes to the Technical Specifications or Environmental Protection Plan shall be addressed in plant-specific procedures. (See Sections 4.5 and 5.2.1.4.)?

4.3.1.2 The 10CFR50.59 screening/evaluation form (similar to that shown in Figure 1) may be structured by listing each of the questions with a YES and NO block before each question. A NOT APPLICABLE (N/A) block is also included before the 10CFR50.59 evaluation questions which are discussed in Section 4.4. An explanation of either a YES or NO response is required. The form should also provide ample space to provide the justification for each answer. For an activity not to require a 10CFR50.59 evaluation, the answer to all of the 10CFR50.59 screening questions must be NO.

a. If 10CFR50.59 screening questions 1, 2, or 4 are answered YES, then 10CFR50.59 evaluation questions 1-7 must be answered; 10CFR50.59 evaluation question 8 is answered N/A.

22. G2.3.6 001

The USS has received a completed release permit for the following tanks:

- Waste Monitor Tank 009 (Unit 1)
- Waste Monitor Tank 010 (Unit 2)

Due to the plant schedule, Operations Management would like both tanks to be realeased at the same time in accordance with 13216-1 and 13216-2, Liquid Waste Release.

Which ONE of the following correctly states the procedure requirements given the above conditions?

A. Two tanks may never be released at the same time under any conditions.

B. Two tanks may be released without additional authorization because they are on different Units.

C✓ The two tanks may be released simultaneously as long as the USS receives authorization from the Chemistry Superintendent.

D. The two tanks may be released simultaneously as long as the USS receives authorization from the HP Supervisor.

K/A
G2.3.6
Knowledge of the requirements for reviewing and approving release permits.

K/A MATCH ANALYSIS
The USS has the responsibility for approving release permits.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Two tanks may be released at the same time with Chem Superintendent permission. Plausible because it is logical that part of a controlled release would be one tank at a time.
B. Incorrect. Two tanks may be released at the same time with Chem Superintendent permission. Plausible because applicant may think that as long as each tank meets specifications for release, that it would be OK to authorize both at the same time.
C. Correct. See Step 2.1.6 of both referenced procedures.
D. Incorrect. Chem Superintendent permission must be received. Plausible because applicant may think that HP Supv has authority to approve releases that may contain certain levels of radioactivity.

REFERENCES
1. 13216-1, Liquid Waste Release, Rev. 32, 08/05/2004.
2. 13216-2, Liquid Waste Release, Rev. 19, 08/05/2004.

MCS	Time:	1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9	
					Answer:	C B B D D B D A C A	Scramble Range: A - D

Tier:	3	Group:	
Key Word:	LIQUID WASTE RELEASE	Cog Level:	MEM 3.1
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

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D. The two tanks may be released simultaneously as long as the USS receives authorization from the HP Supervisor.

K/A
G2.3.6
Knowledge of the requirements for reviewing and approving release permits.

K/A MATCH ANALYSIS
The USS has the responsibility for approving release permits.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Two tanks may be released at the same time with Chem Superintendent permission. Plausible because it is logical that part of a controlled release would be one tank at a time.
B. Incorrect. Two tanks may be released at the same time with Chem Superintendent permission. Plausible because applicant may think that as long as each tank meets specifications for release, that it would be OK to authorize both at the same time.
C. Correct. See Step 2.1.6 of both referenced procedures.
D. Incorrect. Chem Superintendent permission must be received. Plausible because applicant may think that HP Supv has authority to approve releases that may contain certain levels of radioactivity.

REFERENCES
1. 13216-1, Liquid Waste Release, Rev. 32, 08/05/2004.
2. 13216-2, Liquid Waste Release, Rev. 19, 08/05/2004.

QUESTIONS REPORT

for Voglte 2005-301 Draft

MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	C B B D D B D A C A	Scramble Range: A - D	

Tier:	3	Group:	
Key Word:	LIQUID WASTE RELEASE	Cog Level:	MEM 3.1
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. G2.3.6 001

The USS has received a completed release permit for the following tanks:

- Waste Monitor Tank 009 (Unit 1)
- Waste Monitor Tank 010 (Unit 2)

Due to the plant schedule, Operations Management would like both tanks to be realeased at the same time.

Which ONE of the following correctly states the procedure requirements given the above conditions?

A. Two tanks may never be released at the same time under any conditions.

B. Two tanks may be released without additional authorization because they are on different Units.

C✓ The two tanks may be released simultaneously as long as the USS receives authorization from the Chemistry Superintendent.

D. The two tanks may be released simultaneously as long as the USS receives authorization from the HP Supervisor.

K/A
G2.3.6
Knowledge of the requirements for reviewing and approving release permits.

K/A MATCH ANALYSIS
The USS has the responsibility for approving release permits.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Two tanks may be released at the same time with Chem Superintendent permission. Plausible because it is logical that part of a controlled release would be one tank at a time.
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C. Correct. See Step 2.1.6 of both referenced procedures.
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REFERENCES
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2. 13216-2, Liquid Waste Release, Rev. 19, 08/05/2004.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: C B B D D B D A C A Scramble Range: A - D

Tier:	3	Group:	
Key Word:	LIQUID WASTE RELEASE	Cog Level:	MEM 3.1
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

PRB REVIEW REQUIRED

LIQUID WASTE RELEASE

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	**ALL**
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	**NONE**
Information Use:	Available on plant site for reference as needed.	**NONE**

0 **PRECAUTIONS AND LIMITATIONS**

1 **PRECAUTIONS**

2.1.1 The Liquid Waste Processing System is potentially radioactive. Caution should be exercised to avoid spillage and to minimize exposure.

2.1.2 Once a Waste Monitor Tank (WMT) has been placed on recirculation for sampling, the tank shall remain isolated to prevent introduction of liquids that could alter the concentration of the contained volume.

2.1.3 Radiation Monitor 2-RE-0018 reading should be observed at least once every 2 hours during the release to assure that the activity does not exceed the setpoint on the "Batch Liquid Release Permit".

2.1.4 If a high alarm is received from 2-RE-0018 while releasing a tank, the release shall be stopped immediately and the Unit Shift Supervisor and Chemistry notified.

2.1.5 If 2-RE-0018 reads less than expected, release can continue provided Chemistry is notified and 2-RX-0018 does not show a trouble condition.

2.1.6 DO NOT release more than one Waste Monitor Tank per plant site at the same time, unless authorized by the Chemistry Superintendent.

2.1.7 If a high alarm is received from 2-RE-0018 while flushing with tank water, flush with demin water per Section 4.8.

2.2 **LIMITATIONS**

.1 Refer to the ODCM, Chapter 2, Table 2-1, for 2-RE-0018 or 2-FT-0018 and 2-FI-1085A(B) inoperability.

Deleted: Section 1.5.4

2.2.2 CONCURRENT verification is required when valving in a Waste Monitor Tank for release. Independent verification is required for restoration of the release path.

PRB REVIEW REQUIRED

LIQUID WASTE RELEASE

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	**ALL**
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	**NONE**
Information Use:	Available on plant site for reference as needed.	**NONE**

0 **PRECAUTIONS AND LIMITATIONS**

.1 **PRECAUTIONS**

2.1.1 The Liquid Waste Processing System is potentially radioactive. Caution should be exercised to avoid spillage and to minimize exposure.

2.1.2 Once a Waste Monitor Tank (WMT) has been placed on recirculation for sampling, the tank shall remain isolated to prevent introduction of liquids that could alter the concentration of the contained volume.

2.1.3 Radiation Monitor 1-RE-0018 reading should be observed at least once every 2 hours during the release to assure that the activity does not exceed the setpoint on the "Batch Liquid Release Permit".

2.1.4 If a high alarm is received from 1-RE-0018 while releasing a tank, the release shall be stopped immediately and the Unit Shift Supervisor and Chemistry notified.

2.1.5 If 1-RE-0018 reads less than expected, release can continue provided Chemistry is notified and 1-RX-0018 does not show a trouble condition.

2.1.6 DO NOT release more than one Waste Monitor Tank per plant site at the same time, unless authorized by the Chemistry Superintendent.

2.1.7 If a high alarm is received from 1-RE-0018 while flushing with tank water, flush with demin water per Section 4.8.

Deleted: 9

2.2 **LIMITATIONS**

?.1 Refer to the ODCM, Chapter 2, Table 2-1, for 1-RE-0018 or 1-FT-0018 and 1-FI-1085A(B) inoperability.

Deleted: Section 1.5.4

2.2.2 CONCURRENT verification is required when valving in a Waste Monitor Tank for release. Independent verification is required for restoration of the release path.

1. GEN2.3.6 001

Waste Monitor tank 009 is full and needs to be released. Radiation Monitor 1-RE-0018, Waste Disposal Liquid Effluent Monitor, is inoperable. Estimated return-to-service time is 36 hours.

Which of the following describes the actions necessary per 13216-1, "Liquid Waste Release" to make a radioactive liquid release under these circumstances?

A. No release can be made until Radiation Monitor 1-RE-0018 is returned to service.

B. Sample results must show that the specific activity of the liquid to be released is less than 1E4 microcuries per milliliter, and the radiochemists approval obtained.

C. One sample per tank must be analyzed and one qualified member of the plant staff verifies the release rate calculations and discharge line valving.

D✓ At least two separate samples must be analyzed by Chemistry and Chemistry "Action statement 37 Sheet" attached to the release permit. Document verification and method in the ABO logbook.

D

Ref: Liquid Waste Release Procedure 13216-1, LO-LP-47110-18-C

CFR 43.4

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9

Answer: D A B C D A B C D A Scramble Range: A - D

RO Tier:		SRO Tier:	T3
K/A Value:	2.1/3.1	Cog. Level:	C/A
Source:	MODBK	Exam:	VG01301
Test:	SRO	Misc:	LRM09

Previous Vogtle Exam.
I tried to stay away from this and test something a little different.

23. G2.4.16 001

Unit 1 was operating at 100% rated thermal power when a loss of offsite power occurred. The reactor failed to trip and the crew entered 19211-C, FR-S.1, Response to Nuclear Power Generation/ATWT. The "A" Train Diesel Generator energized its bus, and the "B" Train Safety Bus was grounded and de-energized. As the crew began performing steps in FR-S.1, the "A" Train Diesel Generator tripped and could not be restarted from the control room.

Which ONE of the following correctly describes the procedure transitions that the Unit Supervisor must direct?

A. Complete all the actions of 19211-C and then go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, after completion of 19100.

B. Complete all the actions of 19211-C and then go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, while completing 19100.

C. Stop performance of 19211-C and immediately go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, after completion of 19100.

D✓ Stop performance of 19211-C and immediately go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, while completing 19100.

K/A
G2.4.16
Knowldge of EOP implementation hierarchy and coordination with other support procedures.

K/A MATCH ANALYSIS
The question addresses the heirarchy of EOPs and ECAs, which is an EOP support procedure. The question is SRO level because it requires the applicant to know that the FRPs are developed with the assumption that at least one safety train of power is available.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Operator will not exit ECA-0.0 prior to restoring power.
B. Incorrect. FR-S.1 assumes power to one safety bus.
C. Incorrect. Operator will not exit ECA-0.0 prior to restoring power.
D. Correct. FR-S.1 assumes power to one safety bus. ECA-0.0 uses an "initiating" step to send operators to an SOP to try to get the EDG started.

Distractors are plausible because applicants may not know the assumptions behind the FRPs and may not know that they need to stop performing an FRP before it is completed because normally FRPs must be completed prior to exiting them (particularly the highest ranked safety function).

REFERENCES
1. Turkey Point 2002-301 SRO Exam Question.
2. LO-LP-37031-15-C, Loss of All AC Power, Rev. 15, 12/14/200.
3. LO-LP-37002, Format and Use of EOP's, Rev. 14, 12/31/2002.
4. 19100-C, ECA-0.0 Loss of All AC Power, Rev. 28, 12/19/2003.

MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D C C D B C A C B B	Scramble Range: A - D	
Tier:	3			Group:		
Key Word:	ATWS LOSS OF ALL AC			Cog Level:	C/A 4.0	
Source:	B			Exam:	VG05301	
Test:	S			Author/Reviewer:	MAB/RSB	

ATWT ✓

1. G2.4.16 001

Unit 1 was operating at 100% rated thermal power when a loss of offsite power occurred. The reactor failed to trip and the crew entered 19211-C, FR-S.1, Response to Nuclear Power Generation/ATWS. The "A" Train Diesel Generator energized its bus, and the "B" Train Safety Bus was grounded and de-energized. As the crew began performing steps in FR-S.1, the "A" Train Diesel Generator tripped and could not be restarted from the control room.

Which ONE of the following correctly describes the procedure transitions that the Unit Supervisor must direct?

19211-C ✓

A. Complete all the actions of FR-S.1 and then go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, after completion of ECA-0.0. 19100 ✓

B. Complete all the actions of FR-S.1 and then go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, while completing ECA-0.0. ✓

C. Stop performance of FR-S.1 and immediately go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, after completion of ECA-0.0. ✓

D.✓ Stop performance of FR-S.1 and immediately go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, while completing ECA-0.0. 19100 ✓

K/A
G2.4.16
Knowldge of EOP implementation hierarchy and coordination with other support procedures.

K/A MATCH ANALYSIS
The question addresses the heirarchy of EOPs and ECAs, which is an EOP support procedure. The question is SRO level because it requires the applicant to know that the FRPs are developed with the assumption that at least one safety train of power is available.

ANSWER / DISTRACTOR ANALYSIS
A. Incorrect. Operator will not exit ECA-0.0 prior to restoring power.
B. Incorrect. FR-S.1 assumes power to one safety bus.
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MCS	Time: 1	Points: 1.00	Version:	0 1 2 3 4 5 6 7 8 9		
			Answer:	D C C D B C A C B B		Scramble Range: A - D
Tier:	3			Group:		
Key Word:	ATWS LOSS OF ALL AC			Cog Level:	C/A 4.0	
Source:	B			Exam:	VG05301	
Test:	S			Author/Reviewer:	MAB/RSB	

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Which ONE of the following correctly describes the procedure transitions that the Unit Supervisor must direct?

A. Complete all the actions of FR-S.1 and then go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, after completion of ECA-0.0.

B. Complete all the actions of FR-S.1 and then go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, while completing ECA-0.0.

C. Halt performance of FR-S.1 and immediately go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, after completion of ECA-0.0.

D✓ Halt performance of FR-S.1 and immediately go to 19100, ECA-0.0 Loss of All AC Power. Power will be restored to a safety bus using 13145, Diesel Generators, while completing ECA-0.0.

K/A
G2.4.16
Knowldge of EOP implementation hierarchy and coordination with other support procedures.

K/A MATCH ANALYSIS
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MCS	Time: 1	Points:	1.00	Version:	0 1 2 3 4 5 6 7 8 9		
				Answer:	D C C D B C A C B B	Scramble Range: A - D	
Tier:	3			Group:			
Key Word:	ATWS LOSS OF ALL AC			Cog Level:	C/A 4.0		
Source:	B			Exam:	VG05301		
Test:	S			Author/Reviewer:	MAB/RSB		

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	Loss Of All AC Power	NUMBER:	LO-LP-37031-15-C
PROGRAM:	Licensed Operator	REVISION:	15
SME:	Perry Tucker	DATE:	December 11, 2000
APPROVED:	***D. Scukanec***	DATE:	12/14/2000

INSTRUCTOR GUIDELINES:

I. LESSON FORMAT

A. Verbal lecture with visual aids

II. MATERIALS

A. Overhead projector

B. Transparencies

C. White board with markers

III. EVALUATION

A. Written or oral exam in conjunction with other lesson plans

IV. REMARKS

A. This lesson plan meets the training commitments made in FSAR Question 730.1 and NUMARC Position on Station Blackout.

B. Ensure students have latest revision of EOP

III. LESSON OUTLINE:	NOTES
I. INTRODUCTION	
A. This lesson will give the student a general knowledge of ECA 0.0 "Loss of All AC Power"	
B. Present Lesson Objectives	LO-TP-37031-001
II. PRESENTATION	
A. General Information	
1. Definition of loss of all AC power	
a. Loss of grid power (RATS) and some combination of events preventing the emergency diesels from energizing the emergency AC buses	Objective 1
1) Immediate consequences to plant equipment if <u>not</u> accompanied by some other event (LOCA, S/G TR) are not severe	
2) If power cannot be restored, consequences to plant and public safety can potentially be extreme	
a) Core uncovering	
2. Basis	
a. The object of ECA 0.0 is to provide guidance to respond to a loss of all AC power in order to mitigate deterioration of RCS conditions while AC power is not available	Objective 7
3. ECA usage	
a. Guidance provided in other EOP's not applicable following loss of all AC	
1) Other EOP's written on the premise that at least one AC emergency bus energized	
b. ECA 0.0 has priority over all other guidelines	
1) Steps include actions that monitor and maintain critical safety functions	
B. Plant Response to Loss of Power	
1. Chief concern to loss of RCS Fluid through RCPs seals with no makeup capability	Objective 2
2. Without cooling seal situation degrades over	

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	Format And Use Of EOP's	NUMBER:	LO-LP-37002-14-C
PROGRAM:	Licensed Operator	REVISION:	14
SME:	Perry Tucker	DATE:	September 4, 2002
APPROVED:	D. Scukanec	DATE:	12/31/2002

INSTRUCTOR GUIDELINES:

I. LESSON FORMAT

A. Verbal lecture with visual aids

II. MATERIALS

A. Overhead projector

B. Transparencies

C. White board with markers

III. EVALUATION

A. Oral or written exam in conjunction with other lesson plans

IV. REMARKS

A. Ensure students have latest revision of EOP

B. Performance-based instructional units (IUs) are attached to the lesson plan as student handouts. After the lecture on Format and Use of EOP's, the student should be given adequate self-study time for the IUs. The instructor should direct self-study activities and be available to answer questions that may arise concerning the IU material. After self-study, the student will perform, simulate, observe, or discuss (as identified on the cluster signoff criteria list) the task covered in the instructional unit in the presence of an evaluator.

Only three EOP's with immediate actions

1) 19000-C "Reactor trip or Safety Injection"

2) 19100-C "Loss of All AC"

3) 19211-C "ATWT"

c. Unless words "dispatch operator" or "locally" appear in an EOP step, the action can be done in the Main Control Room

d. When operator is instructed to "try" to perform an action

1) Generally, the effort is made concurrent with other procedure steps

2) The effort should continue until the intent of the step has been accomplished

3) Examples

a) "Try" to restore off-site power

b) "Try" to start a RCP

e. During performance of EOP's in response to accident it may be necessary to step outside limits of Tech. Specs.

1) This is recognized by the NRC as being necessary on occasion to optimize recovery actions in response to accident situations and thus protect the general public

2) Example: SGTR requires rapid RCS cool- down and depressurization in excess of Tech. Spec. limits to minimize radiation release to public

f. Other procedures such as AOPs or ARPs may be performed in parallel with EOPs as long as their actions do not conflict with the EOP steps. EOP actions take priority.

g. Other terms and definitions

1) "Faulted" is a term generally reserved for secondary systems (i.e., steam continuity in a secondary pressure generators). The term describes a dis- GSU-1204 Item 6

boundary, allowing either feedwater or steam to leak out

2)"Ruptured" is a term used to describe a break in the RCS to S/G pressure boundary. "Ruptured" is used particularly in reference to steam generator tubes

3)"Intact" is a term used to describe a steam generator which has neither a tube rupture nor a secondary side break

4)"Affected" is a term used to describe a component which has undergone a change due to some initiating event (i.e., affected steam generator being ruptured or faulted

5)"Controlled cooldown" is a term used to describe a gradual reduction in temperature of plant systems (usually the primary system cooldown is of most concern). The "controlled cooldown" is performed by staying within the T.S. pressure-temperature limits. A control-led cooldown is performed at less than 100o per hour. (50oF for natural circulation cooldown)

6)"Maximum rate" is a term used to describe the method in which an operator performs a particular task. For example, if the procedure step instructs an operator to "dump steam to the condenser at maximum rate", then the operator would use the steam dump system at its fullest capacity to achieve the desired goal of the procedure step. (Not following Tech. Specs. cooldown limits)

C. Control Room Usage of the Status Trees — Objective 6

1. Trees ask series of questions about plant conditions — Ensure students have a copy of 19200-C

   a. Question asked is dependent upon answer to previous question

   b. Only one entry, several exit points of which only one can result from a pass through the tree

2. Trees are monitored in order of priority — Objective 7

   a. Subcriticality first; inventory last

1) 18004 directs entry into 19013 to swap to cold leg recirc if RWST level lowers to 39%

2) 18004 directs entry into 19111-C to respond to an unisolable leak outside containment

2. Operators transition from one EOP to another by observing "Go To" orders within the A/ER or the RNO column, or from the fold out page, or from a terminus of a CSFST.

a. Example: If any S/G pressure lowering in an uncontrolled manner GO TO 19020.

3. Operators perform concurrent System Operating Procedures by observing "by initiating" orders

a. Example: Place letdown in service by initiating 13006.

4. Entry into 19200-C

a. Two possible outcomes — Objective 11

1) Operator remains in 19000-C and is directed by an action step to start monitoring the status trees, or

2) Operator transfers to some other EOP, at which point he begins to monitor the CSFST's

b. CSFST monitoring takes place in parallel with recovery actions being performed by the operator — Note: Not while in 19000 Occurs at Step 28

1) The EOP-ORP actions in progress are suspended if either a RED or ORANGE condition is detected on a status tree

2) ORP's are not to be performed while a CSF is being restored unless required by the FRP in effect

3) After restoration of a CSF from a RED or ORANGE, recovery actions may continue when the FRP is completed — Note: Red path still exist even when FRP completed (eg. Integrity)

c. Upon continuation of recovery actions, some judgement is required by the operator to avoid inadvertent reinstatement of a RED or ORANGE condition

5. Entry in 19100-C from 19000-C — Objective 10 (cont'd)

a. If a loss of all AC is originating event, direct entry into 19100 is allowed

b. Entry is expected to be a rare occurrence

c. Special considerations

1) FRG's cannot be implemented

2) CSFST's monitored for info only

d. Once in 19100-C, cannot transition to any other procedure unless power is restored to at least one ESF bus and a specific step within the procedure directs the transition

6. Entry into 19005-C "REDIAGNOSIS" — Objective 12

a. Has no entry symptoms or transitions

1) Used purely as operator aid

2) Entered only on operator judgement after leaving 19000 if SI was initiated or required

3) Should not be used if a function restoration procedure is in effect

b. Bases - provides a means of redirecting an operating crew to the correct "series" procedures following misdiagnosis of an event or confirming that the crew has correctly diagnosed plant symptoms and are in the correct series of procedures.

1) Steps 3 and 4 provide a list of procedures applicable to given conditions

2) If you have come from one of the procedures listed, return to that procedure and step in effect

3) If you determine you have misdiagnosed and have determined the correct series to be in, you should go to the first procedure in that series (or return to the one you left if the series is the same).

4) If you did not come from any of the procedures listed, go to Step 1 of the first procedure in the list

a) Basically from 19005-C you should be going to 19121-C, 19020-C, 19030-C or 19010-C unless it is a "return to" situation.

EMERGENCY OPERATING PROCEDURE

ECA-0.0 LOSS OF ALL AC POWER

PURPOSE

PRB REVIEW REQUIRED

This procedure provides actions to respond to a loss of all AC power. (Applicable in Modes 1,2,3,4)

MAJOR ACTIONS

- ◆ Check Plant Conditions
- ◆ Restore AC Power
- ◆ Maintain Plant Conditions for Optimal Recovery
- ◆ Evaluate Energized AC Emergency Bus
- ◆ Select Recovery Guideline After AC Power Restoration

SYMPTOMS/ENTRY CONDITIONS

The symptoms are:

- Both emergency AC buses are de-energized.

The entry conditions are:

- 19000-C, E-0 REACTOR TRIP OR SAFETY INJECTION, Step 3.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:

- 91001-C EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS should be implemented at this time.
- If the Diesel Generator output breaker did not close on a bus loss of power and power has been restored to an AC emergency bus from a RAT, it will be necessary to reset the sequencer using the DG BRKR FAILED TO CLOSE SEQUENCER UV RESET PUSHBUTTON to enable normal stopping of AC emergency loads.
- The battery capacity for DG field flashing may only support 5 Diesel Generator start attempts.

* 6. Try to restore power to any AC emergency bus:

a. Start DG.

RESPONSE NOT OBTAINED:

a. Dispatch operator to locally start DG by initiating 13145, DIESEL GENERATORS.

- IF starting air pressure is greater than 150 psig, THEN emergency start DG at the DG panel using Emergency Start Switch
- IF starting air pressure is less than 150 psig, THEN manually start DG (locally or remotely).

QUESTIONS REPORT
for Westinghouse 3 Loop Questions

1. G2.4.16 (S) 001

Unit 4 is operating at 100% power with the 4A EDG out of service when the following sequence of events occur:

- The operators respond to an ATWS using FR-S.1, "Response to Nuclear Power Generation/ATWS."
- The reactor trips due to a loss of off-site power.
- The 4B EDG locks out and cannot be restarted.

Which ONE of the following describes the correct operator response?

A. Complete the actions of FR-S.1 and then go to ECA-0.0, "Loss of All AC Power." Power will be restored to a 4KV bus using the appropriate ONOP upon completion of ECA-0.0.

B. Complete the actions of FR-S.1 and then go to ECA-0.0. Power will be restored to a 4KV bus using the appropriate ONOP while performing the actions of ECA-0.0.

C. Stop performance of FR-S.1 and immediately go to ECA-0.0. Power will be restored to a 4KV bus using the appropriate ONOP upon completion of ECA-0.0.

D. Stop performance of FR-S.1 and immediately go to ECA-0.0. Power will be restored to a 4KV bus using the appropriate ONOP while performing the actions of ECA-0.0.

Question Source: 1999 NRC Exam
Enabling Objective EO5 of ADM-211 LP#6902320

Distractor Analysis:
A: Incorrect, FR-S.1 assumes at least one emergency 4KV bus has power. When both busses are deenergized, ECA-0.0 takes precedence over FR-S.1. Operators should not wait until completion of ECA-0.0 to repower a bus.
B: Incorrect, FR-S.1 assumes at least one emergency 4KV bus has power. When both busses are deenergized, ECA-0.0 takes precedence over FR-S.1.
C: Incorrect, Operators should not wait until completion of ECA-0.0 to repower a bus.
D: Correct, ADM-211, step 5.1.2 and step 5.13.2 first example; and ECA-0.0, step 10 RNO.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: D C A A B C A A C B Scramble Range: A - D

RO Tier:	T3	SRO Tier:	T3
K/A Value:	ATWS EOP IMP	Cog. Level:	C/A (3.0/4.0)
Source:	B	Exam:	TP02301
Test:	S	Misc:	SDR

24. G2.4.44 001

Which ONE of the following correctly states the minimum protective action recommendations following declaration of a General Emergency?

A✓ Evacuate all people within a 2 mile radius and all people within 5 miles in the downwind affected zones. Shelter all remaining people within the 10 mile EPZ.

B. Evacuate all people within a 5 mile radius and all people within 10 miles in the downwind affected zones. Shelter all remaining people within the 10 mile EPZ.

C. Evacuate all people within a 2 mile radius and shelter all remaining people within a 5 mile radius.

D. Evacuate all people within a 5 mile radius and shelter all remaining people within a 10 mile radius.

K/A
G2.4.44
Knowledge of emergency plan protective action recommendations.

K/A MATCH ANALYSIS
Question tests knowledge of PARs.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. See referenced lesson plan and procedure.
B, C, D. Incorrect. See referenced lesson plan and procedure. Plausible because applicants may not correctly remember the minimum PARs as stated in the LP and Procedure.

REFERENCES
1. LO-LP-40101, EPIP Overview, Rev. 39, 05/03/2004.
2. 91035-C, Protective Action Guidelines, Rev. 19, 03/12/2004.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C B D D D B B C C Scramble Range: A - D

Tier:	3	Group:	
Key Word:	PARS PROTECTIVE ACTI	Cog Level:	MEM 4.0
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. G2.4.44 001

Which ONE of the following correctly states the minimum protective action recommendations following declaration of a General Emergency?

A.✓ Evacuate all people within a 2 mile radius and all people within 5 miles in the downwind affected zones. Shelter all remaining people within the 10 mile EPZ.

B. Evacuate all people within a 5 mile radius and all people within 10 miles in the downwind affected zones. Shelter all remaining people within the 10 mile EPZ.

C. Evacuate all people within a 2 mile radius and shelter all remaining people within a 5 mile radius.

D. Evacuate all people within a 5 mile radius and shelter all remaining people within a 10 mile radius.

K/A
G2.4.44
Knowledge of emergency plan protective action recommendations.

K/A MATCH ANALYSIS
Question tests knowledge of PARs.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. See referenced lesson plan and procedure.
B, C, D. Incorrect. See referenced lesson plan and procedure. Plausible because applicants may not correctly remember the minimum PARs as stated in the LP and Procedure.

REFERENCES
1. LO-LP-40101, EPIP Overview, Rev. 39, 05/03/2004.
2. 91035-C, Protective Action Guidelines, Rev. 19, 03/12/2004.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C B D D D B B C C Scramble Range: A - D

Tier:	3	Group:	
Key Word:	PARS PROTECTIVE ACTI	Cog Level:	MEM 4.0
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. G2.4.44 001

Which ONE of the following correctly states the minimum protective action recommendations following declaration of a General Emergency?

A.✓ Evacuate all people within a 2 mile radius and all people within 5 miles in the downwind affected zones. Shelter all remaining people within the 10 mile EPZ.

B. Evacuate all people within a 5 mile radius and all people within 10 miles in the downwind affected zones. Shelter all remaining people within the 10 mile EPZ.

C. Evacuate all people within a 2 mile radius and shelter all remaining people within a 5 mile radius.

D. Evacuate all people within a 5 mile radius and shelter all remaining people within a 10 mile radius.

K/A
G2.4.44
Knowledge of emergency plan protective action recommendations.

K/A MATCH ANALYSIS
Question tests knowledge of PARs.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. See referenced lesson plan and procedure.
B, C, D. Incorrect. See referenced lesson plan and procedure. Plausible because applicants may not correctly remember the minimum PARs as stated in the LP and Procedure.

REFERENCES
1. LO-LP-40101, EPIP Overview, Rev. 39, 05/03/2004.
2. 91035-C, Protective Action Guidelines, Rev. 19, 03/12/2004.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A C B D D D B B C C Scramble Range: A - D

Tier:	3	Group:	
Key Word:	PARS PROTECTIVE ACTI	Cog Level:	MEM 4.0
Source:	N	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

VOGTLE ELECTRIC GENERATING PLANT

TRAINING LESSON PLAN

TITLE:	EPIP Overview	NUMBER:	LO-LP-40101-39-C
PROGRAM:	Licensed Operator Training	REVISION:	39
SME:	Perry Tucker	DATE:	April 27, 2004
APPROVED:	D. Scukanec	DATE:	May 3, 2004

INSTRUCTOR GUIDELINES:

I. LESSON FORMAT

A. Lecture with Visual Aids.
B. Initial Training: Two four hour classes.
C. Requal MRE (RE-080) Training: Requal objectives along with Section III (I.B) criteria.

II. MATERIALS NEEDED

A. White Board with Markers.
B. Overhead Projector.
C. Transparencies.
D. Series 91000-C Emergency Plan Implementation Procedures.
E. Emergency Planning Zone Maps.

III. EVALUATION

A. Written or Oral Exam in conjunction with other Lesson Plans for Initial training.
B. Written exam for Requalification (Management Of Radiological Emergencies) training. Passing criteria requires a final grade of at least 80% for MRE.

IV. REMARKS

A. Training in Emergency Plan responsibilities and core damage assessment for SRO's is required per 60601-C. Training for RO's in Emergency Plan responsibilities is required per 60602-C. This training also satisfies the annual training requirements for Emergency Directors as specified in 91601-C and Section O of the Vogtle Emergency Plan.

facility data information transfer

9. Emergency Response Data System (ERDS)

   a. Provides plant data from the IPC to the NRC Operations Center

   b. Must be activated within 1 hour of an Alert or higher emergency

   c. Connected in the TSC — Chemistry Supv

      1) TSC manager responsible to ensure this is done

G. Protective Action Guidelines — **91305-C** Objective 36 *(SRO)*

1. Protective actions for on-site and off-site personnel are planned provisions to ensure their safety in the event of an emergency

2. The Emergency Director is responsible for implementing on-site protective action

3. The ED is responsible for <u>recommending</u> off-site protective actions, only the states are authorized to <u>implement</u> off-site protective actions

4. Protective action recommendations shall be made on the basis of:

   a. Current or projected plant conditions, or — PAR's 1 & 2

   b. Calculated or measured dose rates — PAR 3

      1) Used as soon as available

   c. If a General Emergency has been declared, protective action recommendations <u>SHOULD</u> be made during initial notifications whether or not dose calculations have been completed

      1) This **minimum** protective action is the evacuation of a 2 mile radius and 5 miles in the downwind affected zones.

      2) A 15-minute time limit is required to notify the offsite agencies whenever a PAR is <u>given</u> or has been <u>upgraded</u> or <u>changed</u>.

5. The on-site protective actions are:

   a. Non-essential personnel site dismissal — PAR's 4 & 5

   b. Distributing potassium iodine (KI) for plant workers — Checklist 1

c. Use of additional dosimetry, respiratory protection, and protective clothing

6. Off-site Protective Actions are:

a. Evacuation of off-site personnel — PAR's 1 thru 3

b. Recommend personnel to seek shelter, close windows, secure ventilation systems, etc.

c. Appropriate Protective Actions are determined utilizing Table 1 of 91305-C

1) Affected zones determined using Data Sheets 1 or 2 and Figure 1 EPZ map.

H. Estimating Offsite Dose — **91304-C**

1. The purpose of this procedure is to provide instructions for personnel to estimate radiological releases resulting in off-site doses which would require the implementation of protective action recommendations.

2. The HP/CH Shared Foreman shall initially be responsible for determining release rates and carrying out initial off-site dose calculations. The SS (ED) shall be responsible to ensure that these calculations are performed — Objective 41 *(SRO)*

a. HP/CH Shared Foreman is the designated On-Shift Dose Analyst

b. HP Supervisor shall assume responsibility when the TSC is activated

c. Dose Assessment Supervisor shall assume responsibility when the EOF Dose Assessment is activated

3. The primary method of calculation of off-site doses is the computer dose calculation — MIDAS

4. The calculation uses the following input data:

a. Windspeed and direction (primarily from the Plant computer)

b. Plant radiological and effluent data (primarily from Plant computer)

c. Direct measurement of effluent path with a portable instrument

5. The Output of the calculation is:

Declared pregnant women shall be excluded from receiving emergency exposure

36. SUMMARIZE THE EVALUATION AND IMPLEMENTATION OF PROTECTIVE ACTION GUIDELINES AS SPECIFIED IN THE EPIP'S *(SRO ONLY)*

    a. The Emergency Director is responsible for implementing on-site protective actions

    b. The ED is responsible for recommending off-site protective actions, only the states are authorized to implement off-site protective actions

    c. Protective action recommendations shall be made on the basis of:
        1) Current or projected plant condition, and/or
        2) Calculated or measured dose rates
        3) If a General Emergency has been declared, protective action recommendations SHOULD be made during initial notifications, whether or not dose calculations have been completed

    d. A 15-minute time limit is required to notify the offsite agencies whenever a PAR is given or has been upgraded or changed.

37. DELETED

38. DELETED

39. DESCRIBE THE FIVE PREREQUISITES AS LISTED IN THE PROCEDURE THAT MUST BE FACTORED INTO THE DECISION TO TERMINATE AN EMERGENCY *(SRO ONLY)*

    a. Plant radiation levels are stable or decreasing with time
    b. The affected reactor is in stable condition and can be maintained indefinitely in a stable condition
    c. Fire or other similar emergency conditions no longer constitute a hazard to safety-related systems, equipment, or personnel
    d. Radioactive releases to the environment have ceased or been controlled to within permissible license limits
    e. Discussions with plant management, VEGP Emergency Response Organization, and off-site authorities do not result in identification of any valid reason for not terminating the emergency

40. STATE THE POSITION RESPONSIBLE FOR RECOVERY OPERATIONS AND THE PRIMARY FOR THAT POSITION (91501-C) *(SRO ONLY)*

    Recovery Manager

    Primary: Nuclear Plant General Manager
    Alternate: Plant Operations Assistant General Manager

41. STATE WHO IS RESPONSIBLE FOR INITIALLY PERFORMING OFF-SITE DOSE CALCULATIONS *(SRO ONLY)*

OSERO:	HP/CH Shared Foreman
TSC:	HP Supervisor
EOF:	Dose Assessment Supervisor (upon EOF Dose Assessment Activation)

PRB REVIEW REQUIRED

PROTECTIVE ACTION GUIDELINES

PROCEDURE USAGE REQUIREMENTS-		SECTIONS
Continuous Use:	Procedure must be open and readily available at the work location. Follow procedure step by step unless otherwise directed.	• Table 1 • Data Sheet 1 • Data Sheet 2 • Checklist 1
Reference Use:	Procedure or applicable section(s) available at the work location for ready reference by person performing steps.	
Information Use:	Available on plant site for reference as needed.	**Remainder of Procedure**

Information Use

1.0 PURPOSE

The purpose of this procedure is to provide instruction(s) for protective action(s) and the factors to be considered in selection of an appropriate measure.

2.0 RESPONSIBILITY

2.1 The Emergency Director (ED) shall be responsible for implementing onsite protective actions throughout the emergency. He shall also be responsible for offsite protective action recommendations (PAR) to the States. This responsibility shall not be delegated.

2.2 The Health Physics (HP) Supervisor (HP Foreman if TSC is not activated) shall be responsible for evaluating the radiological situation onsite and for recommending onsite protective actions to the ED. The HP Supervisor shall also have responsibility for making offsite dose estimates and recommending offsite protective actions to the ED until relieved of that responsibility by the Dose Assessment Supervisor.

2.3 The Dose Assessment Supervisor shall be responsible for making offsite dose estimates and recommending offsite protective actions to the ED.

3.0 PREREQUISITES

A Notification of Unusual Event, Alert, Site Area Emergency, or General Emergency has been declared in accordance with Procedure 91001-C, "Emergency Classification and Implementing Instructions".

PRECAUTIONS

4.1 The offsite protective actions determined by this procedure shall be presented to appropriate State personnel as recommendations. Only the States are authorized to implement offsite protective actions.

4.2 Protective action recommendations shall be made on the basis of plant conditions and/or dose assessment results. If a General Emergency is declared, protective action recommendations shall be made during initial notification based on plant conditions and in accordance with guidelines of Table 1. A 15-minute time limit is required to notify the offsite agencies whenever a PAR is given or has been upgraded or changed. Dose assessment results should be used to refine protective action recommendations (but not reduce) after adequate data becomes available.

Protective action guidelines shall not imply an acceptable dose under any circumstances.

4.4 Weather conditions and weather forecasts will be considered by the States and Counties in their decisions regarding implementation of Vogtle Electric Generating Plant (VEGP) recommended protective actions. Weather should therefore not influence VEGP protective action recommendation for the public except for changes in plume trajectory.

4.4.1 Utilize forecast changes in wind direction in the determination of expected changes in plume trajectory. Areas for which protective actions are recommended may be expanded using simple manual rotation of the plume footprint.

4.5 At times, selection of protective actions must be considered on the basis of an expected degradation of plant systems and equipment prior to the release of radioactivity.

5.0 PROCEDURE

5.1 ONSITE PROTECTIVE ACTIONS

NOTE

Direct radiation monitoring, contamination control, personnel dosimetry and other onsite protective measures shall be conducted in accordance with Health Physics Procedure 00930-C, "Radiation And Contamination Control", unless directed otherwise by the HP Supervisor.

5.1.1 The HP Supervisor shall prepare appropriate radiological assessments. Based on these assessments, he shall recommend onsite protective actions to the ED using the following criteria:

5.1.1.1 Site dismissal with or without monitoring of non-involved personnel shall be mandatory for a Site Area Emergency or a General Emergency. For an Alert classification, a Site Dismissal without monitoring of all non- involved personnel is normally conducted. A Site Dismissal with monitoring may be ordered at the Alert classification if the monitoring and possible decontamination of evacuees is required.

.1.2 Additionally the HP Supervisor shall:

5.1.1.2.1 Make thyroid dose estimates for workers entering airborne radioactivity areas and shall recommend the use of potassium iodide (KI) as a thyroid blocking agent to the ED when thyroid doses are projected, or have been estimated, to be equal to or above 25 rem.

5.1.1.2.2 Direct radiological survey personnel (In-Plant Monitoring Teams for CR, TSC, PESB, and OSC, habitability personnel performing habitability surveys for EOF) to issue KI to those individuals who are candidates for KI based on criteria in 5.1.1.2.1. Ensure those personnel actually exposed to 25 rem or greater receive 130 mg daily of KI for at least 3 days.

5.1.1.2.3 Obtain completed KI Distribution Checklist (Checklist 1) from the radiological survey personnel.

5.1.1.2.4 Consult with candidate's supervisor for replacement of candidates who have reported KI sensitivity, or who have received the maximum (10) dosages allowed.

5.1.1.3 The use of additional dosimetry, respiratory protection and protective clothing shall be recommended by the HP Supervisor on the basis of criteria in Procedures 91301-C, "Emergency Exposure Guidelines" and 00920-C, "Radiation Exposure Limits And Administrative Guidelines".

5.1.2 The ED shall be responsible for implementing onsite protective actions per Procedure 91102-C, "Duties Of The Emergency Director", and Procedure 91403-C, "Site Dismissal", after consultation with the HP Supervisor.

1.3 Radiological survey personnel directed to issue KI shall:

1.3.1 Obtain KI from the CR/TSC, OSC, or EOF emergency kits.

5.1.3.2 Obtain the name(s) or location(s) of personnel who are to be issued KI.

5.1.3.3 Obtain sufficient copies of the KI Distribution Checklist (Checklist 1).

5.1.3.4 Follow the instructions on the KI Distribution Checklist (Checklist 1).

5.1.3.5 Report to the HP Supervisor or his designee after completion of KI distribution.

OFFSITE PROTECTIVE ACTIONS

5.2.1 If a General Emergency has been declared, offsite protective action recommendations shall be made based on plant conditions (see Table 1). Dose assessment results may increase the recommended protective action, but should not decrease the initial General Emergency protective action recommendations.

5.2.2 The Dose Assessment Supervisor shall prepare appropriate radiological assessments as described in Procedure 91304-C, "Estimating Offsite Dose".

5.2.3 The Dose Assessment Supervisor, in consultation with the HP Supervisor, shall review plant status and estimate the potential for a release or, if a release is occurring, for changes in the release rate.

5.2.4 The Dose Assessment Supervisor shall update and refine dose estimates for critical receptor site locations per Procedure 91304-C, "Estimating Offsite Dose" approximately every 15 to 30 minutes, or upon significant changes, in one or more of the following parameters:

5.2.4.1 Release rates.

5.2.4.2 Duration of the releases.

5.2.4.3 Isotopic mixture of the release (varies as a function of effective age).

5.2.4.4 Meteorological conditions.

~~2.5 In addition, he shall make dose projections for potential releases or potential increases in release rates.

5.2.6 The Dose Assessment Supervisor shall compare the plant condition and dose estimates with the Guidelines for Recommended Protective Actions for Gaseous Plume Exposure in Table 1.

5.2.7 If offsite doses exceed the action thresholds, then evacuation shall be recommended. Evacuations will require approximately 2.5 to 3 h for completion.

.8 If offsite doses do not exceed the action threshold, or if local constraints prevent evacuation, sheltering may be prescribed by the states or counties together with the following actions:

5.2.8.1 Remain indoors.

5.2.8.2 Close windows.

5.2.8.3 Turn off ventilation system.

5.2.8.4 Seal cracks in doors, windows, or walls with wet material (paper, cloth, etc.).

5.2.9 Protective action recommendations shall be made in accordance with Table 1. Zones should be used when making recommendations. Savannah River Site (SRS) is one big zone. Use miles when referencing the SRS zone. (i.e., evacuate out to 5 miles in SRS).

5.2.10 After the plume has passed, it may still be advisable to consider the possibility of evacuation if high dose rates due to ground deposition are possible. Dose rates due to deposited radioactivity shall be determined using sampling information obtained per Procedure 91302-C, "In-Plant Sampling And Surveys" and Procedure 91303-C, "Field Sampling And Surveys".

5.2.11 The ED shall make offsite protective action recommendations to offsite authorities in accordance with Procedure 91102-C, "Duties Of The Emergency Director", after consultation with the Dose Assessment Supervisor.

6.0 REFERENCES

6.1 VEGP EMERGENCY PLAN

6.2 PROCEDURES

6.2.1 00920-C, "Radiation Exposure Limits And Administrative Guidelines"

6.2.2 91001-C, "Emergency Classification And Implementing Instructions"

6.2.3 91102-C, "Duties Of The Emergency Director"

6.2.4 91301-C, "Emergency Exposure Guidelines"

6.2.5 91302-C, "In-Plant Sampling And Surveys"

6.2.6 91303-C, "Field Sampling And Surveys"

6.2.7 91304-C, "Estimating Offsite Dose"

6.2.8 91403-C, "Site Dismissal"

6.3 NUREG-0654, FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants".

6.4 Manual of Protective Action Guides and Protective Actions For Nuclear Incidents, EPA-400-R-92-001, Environmental Protection Agency, Office of Radiation Programs, October 1991.

6.5 10 CFR 50.47 (b) (10).

END OF PROCEDURE TEXT

Continuous Use

TABLE 1

GUIDELINES FOR RECOMMENDED PROTECTIVE ACTIONS FOR GASEOUS PLUME EXPOSURE

NON-INVOLVED STATION PERSONNEL AND GENERAL POPULATION

NOTE

a. Affected zones (Data Sheets 1 and 2) are the direct downwind zone and each adjacent zone.

b. Once a PAR has been issued to the offsite agencies, it will not be retracted or reduced to a less restrictive PAR.

CONDITION	RECOMMENDED ACTION
PAR 1. A General Emergency has been declared.	• Use Data Sheet 1 to identify affected zones for Georgia and South Carolina. Savannah River Site is a zone. (Figure 1 – 10 mile EPZ map may be used as a reference)
PAR 2. General Emergency has been declared with: - large amounts of fission products or noble gases are in the containment atmosphere (RE-0005/0006 reading >1.0E+8 mrem/hr) OR - possible fuel over temperature damage has occurred (or is likely) and containment failure has occurred (or is judged imminent).	• Use Data Sheet 2 to identify affected zones for Georgia and South Carolina. Savannah River Site is a zone. (Figure 1 - 10 mile EPZ map may be used as a reference).

A, E-5, F-5, SRS to 5 Miles

Deleted: severe core

TABLE 1 (Cont'd.)

GUIDELINES FOR RECOMMENDED PROTECTIVE ACTIONS FOR GASEOUS PLUME EXPOSURE

NON-INVOLVED STATION PERSONNEL AND GENERAL POPULATION

CONDITION	RECOMMENDED ACTION
PAR 3. An actual release has occurred or is imminent and the projected dose to individuals in the population (outside the site boundary) is calculated to be: a. Total Effective Dose Equivalent Equal to or Greater than 1 rem OR b. Committed Dose Equivalent for Thyroid Equal to or Greater than 5 rem	• General Emergency should be declared in accordance with 91001-C, "Emergency Classification And Implementing Instructions". • Use Data Sheet 2 to identify affected zones for Georgia and South Carolina. Savannah River Site is a zone. (Figure 1 - 10 mile EPZ map may be used as a reference).

A, B-5, C-5, D-5, E-5, F-5, F-10, G-10, SRS to 10 miles.

TABLE 1 (Cont'd.)

GUIDELINES FOR RECOMMENDED PROTECTIVE ACTIONS FOR GASEOUS PLUME EXPOSURE

EMERGENCY TEAM PERSONNEL

CONDITION	RECOMMENDED ACTION
PAR 4. An actual release has occurred or is imminent and the projected dose to Emergency Team workers is calculated to be: a. Committed Dose Equivalent of 25 rem to the thyroid	Issue potassium iodide.
PAR 5. An actual release has occurred or is imminent and the projected dose to Emergency Team workers is calculated to be: a. Total Effective Dose Equivalent of 5 rem OR b. Committed Dose Equivalent of 50 rem to the thyroid or other organs OR c. Shallow Dose Equivalent of 50 rem to the skin	Evacuate personnel unless emergency exposure is authorized per Procedure 91301-C "Emergency Exposure Guidelines".

Continuous Use

DATA SHEET 1
AFFECTED ZONES FOR PROTECTIVE ACTION RECOMMENDATIONS

7 miles

PAR 1

WIND DIRECTION FROM	EVACUATE ZONES	SHELTER ZONES
11.25 - 33.75	A, B-5, C-5, SRS to 2 Miles	Remainder of 10 mile EPZ
33.75 - 56.25	A, B-5, C-5, D-5, SRS to 2 Miles	Remainder of 10 mile EPZ
56.25 - 78.75	A, C-5, D-5, E-5, SRS to 2 Miles	Remainder of 10 mile EPZ
78.75 - 101.25	A, C-5, D-5, E-5, F-5, SRS to 2 Miles	Remainder of 10 mile EPZ
101.25 - 123.75	A, D-5, E-5, F-5, SRS to 2 Miles	Remainder of 10 mile EPZ
123.75 - 146.25	A, D-5, E-5, F-5, SRS to 2 Miles	Remainder of 10 mile EPZ
146.25 - 168.75	A, E-5, F-5, SRS to 5 Miles	Remainder of 10 mile EPZ
168.75 - 191.25	A, F-5, SRS to 5 Miles	Remainder of 10 mile EPZ
191.25 - 213.75	A, F-5, SRS to 5 Miles	Remainder of 10 mile EPZ
213.75 - 236.25	A, SRS to 5 Miles	Remainder of 10 mile EPZ
236.25 - 258.75	A, SRS to 5 Miles	Remainder of 10 mile EPZ
258.75 - 281.25	A, B-5, SRS to 5 Miles	Remainder of 10 mile EPZ
281.25 - 303.75	A, B-5, SRS to 5 Miles	Remainder of 10 mile EPZ
303.75 - 326.25	A, B-5, SRS to 5 Miles	Remainder of 10 mile EPZ
326.25 - 348.75	A, B-5, SRS to 2 Miles	Remainder of 10 mile EPZ
348.75 - 11.25	A, B-5, C-5, SRS to 2 Miles	Remainder of 10 mile EPZ

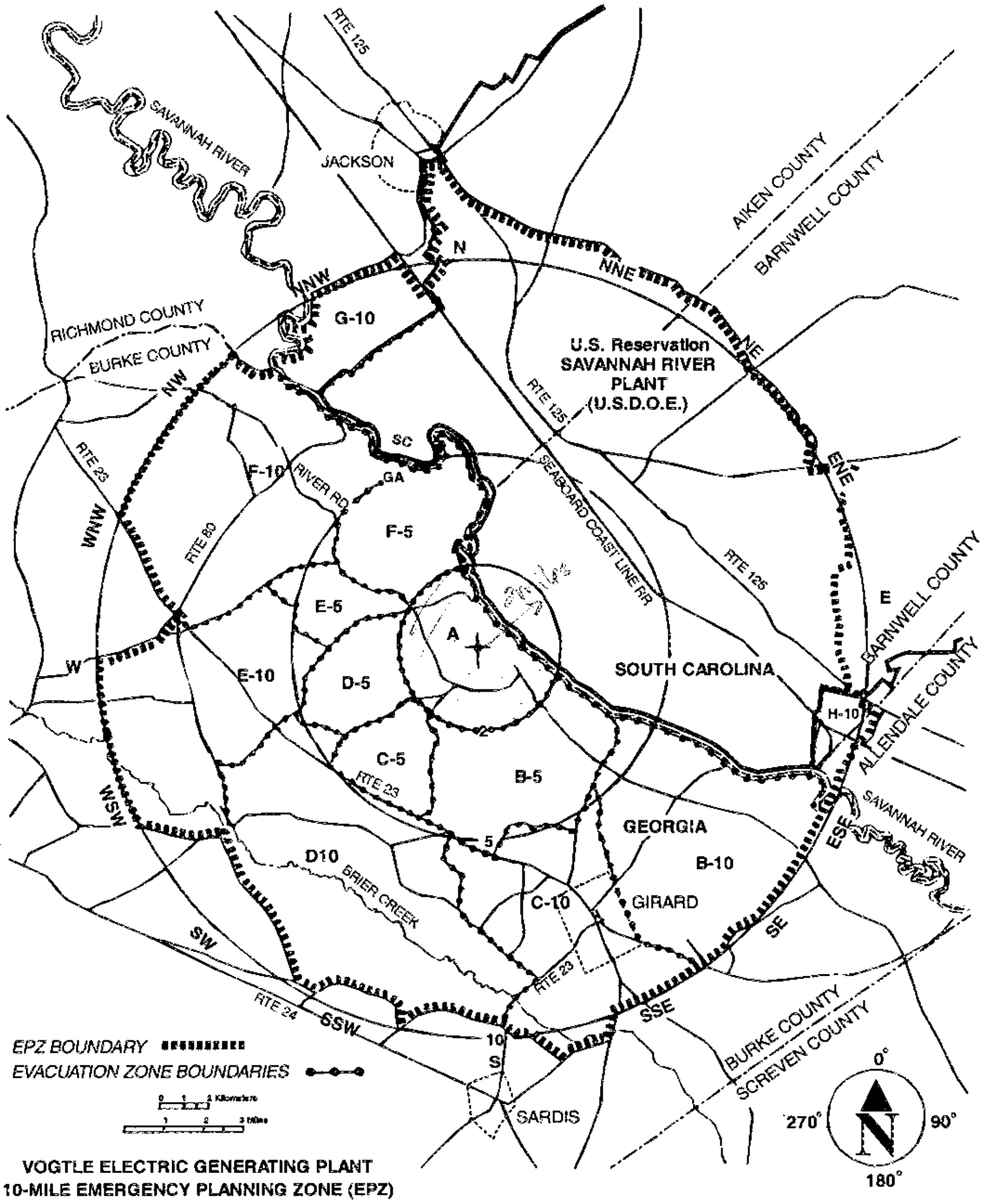
Continuous Use

DATA SHEET 2

AFFECTED ZONES FOR PROTECTIVE ACTION RECOMMENDATIONS

PAR 2 & 3

WIND DIRECTION FROM	EVACUATE ZONES	SHELTER ZONES
11.25 - 33.75	A, B-5, C-5, D-5, E-5, F-5, C-10, D-10, SRS to 5 Miles	Remainder of 10 mile EPZ
33.75 - 56.25	A, B-5, C-5, D-5, E-5, F-5, C-10, D-10, E-10, SRS to 5 Miles	Remainder of 10 mile EPZ
56.25 - 78.75	A, B-5, C-5, D-5, E-5, F-5, D-10, E-10, F-10, SRS to 5 Miles	Remainder of 10 mile EPZ
78.75 - 101.25	A, B-5, C-5, D-5, E-5, F-5, D-10, E-10, F-10, SRS to 5 Miles	Remainder of 10 mile EPZ
101.25 - 123.75	A, B-5, C-5, D-5, E-5, F-5, E-10, F-10, G-10, SRS to 5 Miles	Remainder of 10 mile EPZ
123.75 - 146.25	A, B-5, C-5, D-5, E-5, F-5, E-10, F-10, G-10, SRS to 10 Miles	Remainder of 10 mile EPZ
146.25 - 168.75	A, B-5, C-5, D-5, E-5, F-5, F-10, G-10, SRS to 10 Miles	Remainder of 10 mile EPZ
168.75 - 191.25	A, B-5, C-5, D-5, E-5, F-5, F-10, G-10, SRS to 10 Miles	Remainder of 10 mile EPZ
191.25 - 213.75	A, B-5, C-5, D-5, E-5, F-5, G-10, SRS to 10 Miles	Remainder of 10 mile EPZ
213.75 - 236.25	A, B-5, C-5, D-5, E-5, F-5, SRS to 10 Miles	Remainder of 10 mile EPZ
236.25 - 258.75	A, B-5, C-5, D-5, E-5, F-5, H-10, SRS to 10 Miles	Remainder of 10 mile EPZ
258.75 - 281.25	A, B-5, C-5, D-5, E-5, F-5, B-10, H-10, SRS to 10 Miles	Remainder of 10 mile EPZ
281.25 - 303.75	A, B-5, C-5, D-5, E-5, F-5, B-10, C-10, H-10, SRS to 10 Miles	Remainder of 10 mile EPZ
303.75 - 326.25	A, B-5, C-5, D-5, E-5, F-5, B-10, C-10, H-10, SRS to 10 Miles	Remainder of 10 mile EPZ
326.25 - 348.75	A, B-5, C-5, D-5, E-5, F-5, B-10, C-10, D-10, SRS to 5 Miles	Remainder of 10 mile EPZ
348.75 - 11.25	A, B-5, C-5,D -5, E-5, F-5, B-10, C-10, D-10, SRS to 5 Miles	Remainder of 10 mile EPZ



SAVANNAH RIVER
RTE 125
JACKSON
AIKEN COUNTY
BARNWELL COUNTY
RICHMOND COUNTY
BURKE COUNTY
G-10
U.S. Reservation
SAVANNAH RIVER
PLANT
(U.S.D.O.E.)
N
NNE
NE
ENE
E
NW
NNW
WNW
W
WSW
SW
SSW
S
SSE
SE
ESE
RTE 23
RTE 80
F-10
RIVER RD
SC
GA
F-5
SEABOARD COAST LINE RR
E-5
A
D-5
E-10
SOUTH CAROLINA
BARNWELL COUNTY
ALLENDALE COUNTY
H-10
C-5
B-5
GEORGIA
D10
BRIER CREEK
B-10
C-10
GIRARD
RTE 24
10
5
SARDIS
BURKE COUNTY
SCREVEN COUNTY
EPZ BOUNDARY
EVACUATION ZONE BOUNDARIES
0°
270°
90°
180°
N
VOGTLE ELECTRIC GENERATING PLANT
10-MILE EMERGENCY PLANNING ZONE (EPZ)

FIGURE 1

Continuous Use

CHECKLIST 1
KI DISTRIBUTION CHECKLIST

Prior to issuing KI, ask the candidate the following:

1. Name ____ ☐

   SS No. ______________ or TLD No. ______________ ☐

CAUTION

a. Allergic reactions to seafood, shellfish or bananas are indications that an individual will have an allergic reaction to KI.

b. DO NOT ISSUE KI to candidate if response to questions 2, 3, or 4 is "yes". Call HP Supervisor and report candidate's name, situation, and await further instructions.

2. Do you have a known allergic reaction or sensitivity to KI? ☐

   Yes ______ Go to Step 6.
   No ______ Go to Step 3.

3. Have you received KI in the past 24 hours? ☐

   Yes ______ Go to Step 6.
   No ______ Go to Step 4.

4. Have you received KI for 10 or more days? ☐

   Yes ______ Go to Step 6.
   No ______ Go to Step 5.

NOTE

The Field Monitoring Team Kits contain bottles of KI tablets. The FMT members are to take the prescribed amount issued by this checklist, from the kits, when directed by the Emergency Director.

5. Issue <u>ONE</u> 130 mg dose of KI and have candidate sign Step 7. ☐

Sheet 2 of 2

CHECKLIST 1
KI DISTRIBUTION CHECKLIST CON'T

6. KI not to be issued because of: ☐

   _______ Allergy/sensitivity to KI.
   _______ Received KI within last 24 hours.
   _______ Received KI for 10 or more days.

7. I have been issued ONE 130 mg dose of KI and will take it when directed by the Emergency Director or his designee. ☐

   ______________________ ________ ________
   Candidate's Signature Date Time

8. Issuing individual: ☐

   ______________________ ________ ________
   Name Date Time

9. Submit completed checklist to HP Supervisor or designee. ☐

25. G2.4.45 001

Unit 1 Operators are moving irradiated fuel in the reactor and spent fuel pool when a fuel handling accident occurs.

The following radiation monitors are in alarm:

- RE-005 (Containment High Range)
- RE-2562A (Containment Atmosphere)
- RE-2565A (Containment Vent)
- RE-008 (FHB)

Which ONE of the following radiation monitor alarms both will be used by the Emergency Director in assessing the Emergency Action Level?

A.✓ RE-005 and RE-008

B. RE-008 and RE-2562A

C. RE-008 and RE-2565A

D. RE-005 and RE-2565A

K/A
G2.4.45
Ability to prioritize the significance of each annunciator or alarm.

K/A MATCH ANALYSIS
Question addresses alarm prioritization from the perspective of the Emergency Director. This allows for the K/A match at the SRO-only level because this is knowledge that the ED must have to effectively prioritize alarms to make an EAL Classification.

ANSWER / DISTRACTOR ANALYSIS
A. Correct. See Figure 4 of Reference 3 for correct answer.
B, C, D. Reference 2 provided for plausible distractors. Memory level nature of the question makes the distractors plausible.

REFERENCES
1. Byron SRO Exam Question G2.4.45, 06/29/2000.
2. V-LO-TX-32101 Digital Radiation Monitoring System (DRMS), Rev. 0.
3. 91001-C, Emergency Classification and Implementation Instructions, 09/12/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: A A A D D A C A D C Scramble Range: A - D

Tier:	3	Group:	
Key Word:	EAL RAD MONITORS	Cog Level:	MEM 3.6
Source:	M	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. G2.4.45 001

Unit 1 Operators are moving irradiated fuel in the reactor and spent fuel pool when a fuel handling accident occurs.

The following radiation monitors are in alarm.
- RE-005 (Containment High Range)
- RE-2562A (Containment Atmosphere)
- RE-2565A (Containment Vent)
- RE-008 (FHB)

Which ONE of the following radiation monitor alarms will Both be used by the Emergency Director in assessing the Emergency Action Level?

A. RE-005 & 8

B✓ RE-008 & 2562A

C. RE-2562A & ~~8~~ 5 or 8

D. RE-2565A & 5

K/A
G2.4.45
Ability to prioritize the significance of each annunciator or alarm.

K/A MATCH ANALYSIS
Question addresses alarm prioritization from the perspective of the Emergency Director. This allows for the K/A match at the SRO-only level because this is knowledge that the ED must have to effectively prioritize alarms to make an EAL Classification.

ANSWER / DISTRACTOR ANALYSIS
B. Correct. See Figure 4 of Reference 3 for correct answer.
A, C, D. Reference 2 provided for plausible distractors. Memory level nature of the question makes the distractors plausible.

REFERENCES
1. Byron SRO Exam Question G2.4.45, 06/29/2000.
2. V-LO-TX-32101 Digital Radiation Monitoring System (DRMS), Rev. 0.
3. 91001-C, Emergency Classification and Implementation Instructions, 09/12/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B D A D B B D A D D Scramble Range: A - D

QUESTIONS REPORT
for Voglte 2005-301 Draft

Tier:	3	Group:	
Key Word:	EAL RAD MONITORS	Cog Level:	MEM 3.6
Source:	M	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. G2.4.45 001

Unit 1 Operators are moving irradiated fuel in the reactor and spent fuel pool when a fuel handling accident occurs.

The following radiation monitors are in alarm.
- RE-005 (Containment High Range)
- RE-2562A (Containment Atmosphere)
- RE-2565A (Containment Vent)
- RE-008 (FHB)

Which ONE of the following radiation monitor alarms will be used by the Emergency Director in assessing the Emergency Action Level?

A. RE-005

B.✓ RE-008

C. RE-2562A

D. RE-2565A

K/A
G2.4.45
Ability to prioritize the significance of each annunciator or alarm.

K/A MATCH ANALYSIS
Question addresses alarm prioritization from the perspective of the Emergency Director. This allows for the K/A match at the SRO-only level because this is knowledge that the ED must have to effectively prioritize alarms to make an EAL Classification.

ANSWER / DISTRACTOR ANALYSIS
B. Correct. See Figure 4 of Reference 3 for correct answer.
A, C, D. Reference 2 provided for plausible distractors. Memory level nature of the question makes the distractors plausible.

REFERENCES
1. Byron SRO Exam Question G2.4.45, 06/29/2000.
2. V-LO-TX-32101 Digital Radiation Monitoring System (DRMS), Rev. 0.
3. 91001-C, Emergency Classification and Implementation Instructions, 09/12/2000.

MCS Time: 1 Points: 1.00 Version: 0 1 2 3 4 5 6 7 8 9
Answer: B D A D B B D A D D Scramble Range: A - D

Tier:	3	Group:	
Key Word:	EAL RAD MONITORS	Cog Level:	MEM 3.6
Source:	M	Exam:	VG05301
Test:	S	Author/Reviewer:	MAB/RSB

1. RCS Leak in progress ***AND*** RCS Subcooling is Less Than 24°F [38°F ADVERSE] ***OR***
2. Indication that a SG is ruptured ***AND*** it has a NON-Isolable Secondary Line Break Outside Containment ***OR***
3. Indication of a SGTR ***AND*** a Prolonged Release of Secondary Coolant is occurring from the AFFECTED SG to the Environment ***OR***
4. Containment Radiation Monitors ~~RE-005~~ / 006 > 2.0E+4 mr/hr ***OR***

→ **LOSS of RCS Barrier**

OR

HEAT SINK CSFST RED	
	Y ***OR***
	N

RCS INTEGRITY CSFST RED	
	Y ***OR***
	N

1. NON-Isolable RCS leak (including SG tube leakage) GREATER THAN the Capacity of One Charging Pump in the normal charging mode ***OR***

→ **Potential Loss of RCS Barrier**

JUDGMENT: Opinion of the ED that the RCS Barrier is Lost or Potentially Lost ***OR*** the inability to determine the status of the RCS Barrier

FIGURE 2 - REACTOR COOLANT SYSTEM (RCS) INTEGRITY

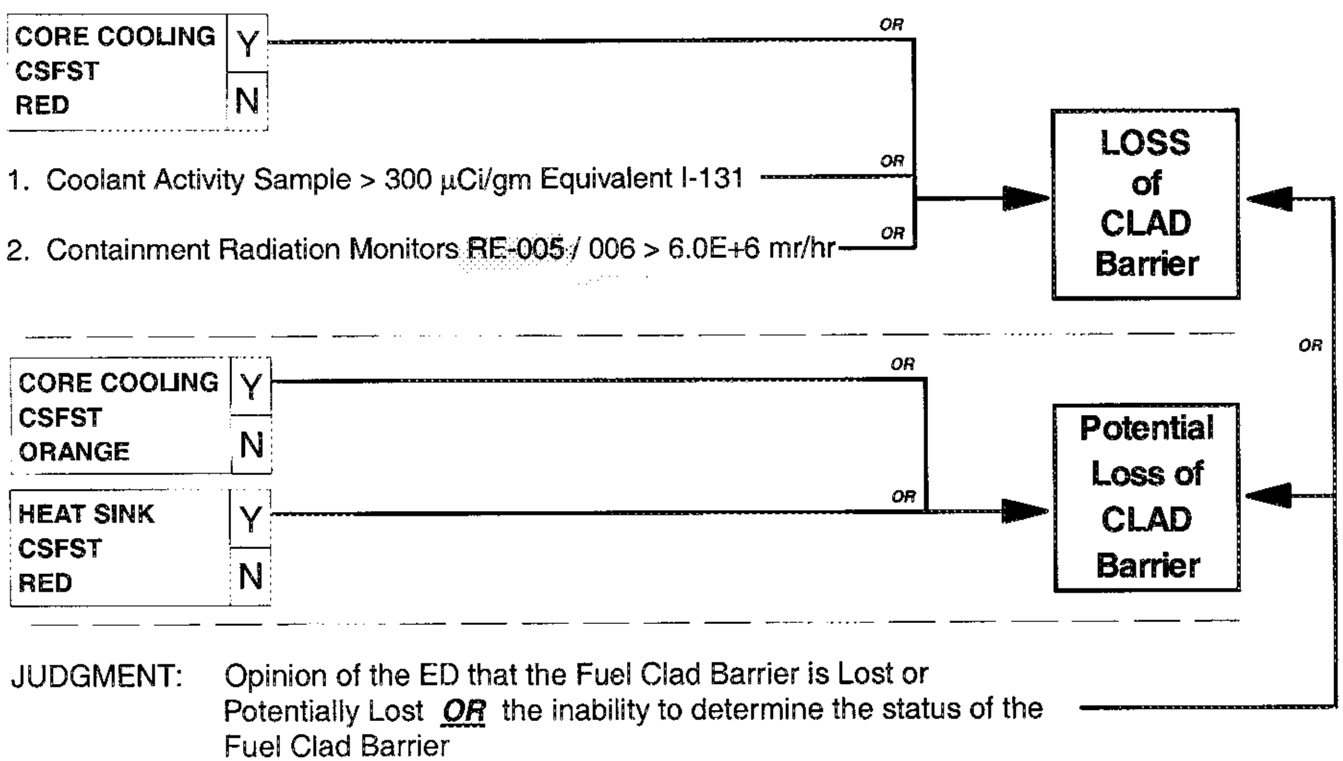


FIGURE 1 - FUEL CLADDING INTEGRITY

1. Rapid Unexplained Containment Pressure Decrease Following Initial Pressure Increase — OR
2. Containment Pressure OR Sump Level Response Not Consistent With Loss of Primary or Secondary Coolant Conditions — OR
3. Containment Isolation Valve(s) or Damper(s) are NOT Closed ***AND*** a Downstream Pathway to the Environment Exists Following Containment Isolation — OR
4. Primary-to-Secondary Leakage > Tech Spec Limits ***AND*** a Prolonged Release of Contaminated Secondary Coolant is Occurring to the Environment — OR

→ **LOSS of CNTMT Barrier**

OR

CONTAINMENT CSFST RED	
Y	— OR
N	

CORE COOLING CSFST RED	
Y	— OR
N	

1. Containment Hydrogen Concentration > 6% — OR
2. Containment Pressure > 21.5 psig ***AND*** Less Than the Following MINIMUM Depressurization Equipment Operable or Operating: — OR
   - Four Containment Coolers (either speed) ***AND*** One Train of Containment Spray
3. Containment Radiation Monitors RE-005 / 006 > 2.4E+8 mr/hr — OR

→ **Potential Loss of CNTMT Barrier**

JUDGMENT: Opinion of the ED that the CNTMT Barrier is Lost or Potentially Lost ***OR*** the inability to determine the status of the CNTMT Barrier → LOSS of CNTMT Barrier / Potential Loss of CNTMT Barrier

FIGURE 3 - CONTAINMENT INTEGRITY

	ELECTRICAL POWER	RADIOACTIVITY	PLANT SYSTEMS	NATURAL PHENOMENON	HAZARDS	SECURITY	BARRIERS	SHUTDOWN SYSTEMS	OTHER
GENERAL	Modes 1-4: Loss of AC power to BOTH AA02 AND BA03 AND EITHER Restoration of at least one emergency bus NOT likely within 4 hrs of time of loss. OR Loss or Potential Loss of FUEL CLAD BARRIER determined from Fission Product Barrier EAL's (Figure 1 Data sheet 1)	Valid reading on RE-12444 that is > 30E+0 µci/cc for > 15 min. OR this value is expected to be exceeded for > 15 min. (Note 1) - - - Valid site boundary dose assessment of release indicates > 1000 mrem TEDE OR > 5000 mrem CDE thyroid OR field survey results indicate > 1000 mrem/hr that is expected to continue for > 1 hr OR field survey samples indicate thyroid dose of 5000 mr CDE for 1 hr of inhalation				Loss of physical control of the control room due to a security event OR loss of physical control of remote shutdown capability due to a security event	Loss of THREE barriers - - - Loss of ANY TWO barriers AND Potential Loss of the THIRD barrier - - - Modes 1-2: Subcriticality CSFST is RED AND EITHER Core Cooling OR Heat Sink CSFST is RED		Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate potential for uncontrolled radionuclide releases that can reasonably be expected to exceed 1000 mrem TEDE or 5000 mrem CDE thyroid dose levels outside the site boundary OR Actual or imminent substantial core degradation with potential for loss of containment
SITE AREA	Modes 1-4: Loss of AC power on BOTH AA02 AND BA03 for > 15 min. - - - Modes 1-4: Unplanned loss of voltage on ALL Vital DC buses (AD1, BD1, CD1, and DD1) for > 15 min.	Valid reading on RE-12444 that is > 2E+0 µci/cc for > 15 min. OR on RE-12839 that is > 6E+2 µci/cc for > 15 min. OR these values are expected to be exceeded for > 15 min. (Note 1) - - - Valid site boundary dose assessment of release indicates > 100 mrem TEDE OR > 500 mrem CDE thyroid OR field survey results indicate > 100 mrem/hr that is expected to continue for > 1 hr OR field survey samples indicate thyroid dose of 500 mrem CDE for 1 hr of inhalation	Mode 1-2: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip did NOT occur from the control room. - - - Modes 1-4: Significant transient is in progress and ESF annunciators needed to monitor the transient are not available AND ESF control board AND compensatory non-alarming indications are NOT available		Control room evacuation has been initiated AND control of the plant CANNOT be established from remote shutdown panels within 15 min.	Intrusion into a plant vital area by a hostile force (e.g., confirmed bomb device discovered within Vital area)	Loss of TWO barriers - - - Loss of ONE barrier AND Potential Loss of a SECOND barrier - - - Potential Loss of BOTH Fuel Clad AND RCS barriers - - - Modes 1-4: The reactor is tripped AND Subcriticality CSFST is RED	Modes 5-6: Loss of reactor vessel water level as indicated by loss of RHR cooling as determined by AOP 18019-C AND any ONE of the following: Vessel level < 183 ft (≤ 62% RVLIS), OR Containment area rad monitors RE-002/003 valid high alarm (15 mr/hr) with vessel head removed, OR Core exit temperatures > 711° F with vessel head installed. - - - Modes 1-4: Complete loss of function needed to achieve or maintain Hot Shutdown (EITHER Core Cooling OR Heat Sink CSFST is RED) AND no other Heat Sink available.	Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate actual or likely major failures of plant functions needed for protection of the public.
ALERT	Modes 1-4: Loss of voltage on EITHER AA02 OR BA03 for > 15 min. AND the remaining energized 1-E bus does not have a backup power supply available. - - - Modes 5, 6, or Defueled: Loss of AC power on BOTH AA02 AND BA03 for > 15 min.	Valid reading on RE-12444 that is > 2E-1 µci/cc for > 15 min. OR on RE-12839 that is > 6E+0 µci/cc for > 15 min. OR on RE-018 that is > 6E-1 µci/cc for > 15 min. (Note 1) - - - Confirmed sample analysis for gaseous OR liquid release indicates concentrations or release rates > 200X ODCM limits for > 15 min. - - - Radiation levels higher than normal in Fuel Handling or Containment Bldg AND irradiated fuel is uncovered, OR < 154 feet water level in refueling cavity, spent fuel pool, or fuel transfer canal that results in uncovering irradiated fuel. (Note 2) - - - Damage to irradiated fuel causing a valid high alarm on one or more of the following: RE-008 (25 mr/hr), RE-2532/2533 A/B (2E-4 µci/cc), or during Mode 6 fuel transfer RE-002/003 (15 mr/hr) - - - Valid rad levels >15 mr/hr in the Control room OR >100 mr/hr in areas requiring infrequent access to maintain plant safety functions (e.g., local charging station)	Modes 1-3: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip occurred from the control room. - - - Modes 1-4: Unplanned loss of most or all annunciators or indicators in the control room for plant safety systems AND EITHER a significant transient is in progress or compensatory non-alarming indications are NOT available	Tornado striking plant vital area OR sustained hurricane force winds of 100 mph or greater on site as verified by meteorological instrumentation (15 minute average). - - - Seismic monitoring system confirms seismic event > 0.12 g.	Aircraft crash confirmed to affect plant vital areas. - - - Report or detection of toxic, flammable, or asphyxiant gasses within a facility structure in concentrations that requires evacuation of a room or area needed for safe operation of the plant. - - - Major fire or explosion in a vital area AND affected safety system parameters show degraded performance or there is visible damage to permanent safety related structures or safety related equipment within the specified area. - - - Report of visible structural damage (caused by natural or destructive phenomena) which threatens the ability of the structure to perform its safety related function to ANY of the following plant structures: Containment Auxiliary Building Control Building Fuel Handling Building Diesel Generator Building Condensate Storage Tank AFW Pumphouse NSCW Cooling Tower RWST - - - Control Room evacuation has been initiated	Intrusion into the Protected area by a hostile force (e.g., unauthorized vehicle penetrates protected area in a hostile manner)	Loss or Potential Loss of Fuel Clad barrier - - - Loss or Potential Loss of RCS barrier	Modes 5-6: Unplanned loss of RHR cooling AND EITHER RCS temperature is > 200° F OR RCS temperature is increasing uncontrolled toward 200° F	Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate plant safety systems may be substantively degraded and that increased monitoring of plant functions is warranted.
NOUE	Loss of Off-Site power to BOTH AA02 AND BA03 for > 15 min (neither bus is connected to an energized Off-Site source) AND BOTH AA02 AND BA03 are powered by Diesel Generators - - - Modes 5-6: Unplanned loss of voltage on ALL Vital DC buses (AD1, BD1, CD1, and DD1) for > 15 min.	Valid reading on RE-12444/12442 that is > 2E-3 µci/cc for > 60 min. OR on RE-12839 that is > 6E-2 µci/cc for > 60 min. OR on RE-018 that is > 6E-3 µci/cc for > 60 min. (Note 1) - - - Confirmed sample analysis for gaseous OR liquid release indicates concentrations or release rates > 2X ODCM limits for > 60 min. - - - Radiation levels higher than normal in Fuel Handling or Containment Bldg AND uncontrolled water level decrease in refueling cavity, spent fuel pool, or fuel transfer canal, BUT all irradiated fuel assemblies remain covered with water. (Note 2) - - - Valid area rad monitor readings increase 1000X over normal levels (Note 2)	Modes 4-6: Automatic reactor trip setpoint was exceeded AND an automatic reactor trip did NOT occur AND a successful manual trip occurred from the control room. - - - Modes 1-4: Unplanned loss of most or all annunciators or indicators in the control room for plant safety systems for > 15 min. AND compensating non-alarming indicators are available - - - Loss of ALL of the following On-Site communications systems: In-plant telephone, Gaitronics, Sound powered phone, and Plant radio communications - - - Loss of ALL of the following Off-Site communications systems: ENN and Telephone capability to Off-Site network	Report of tornado striking within protected area. - - - Hurricane force winds of 74 mph or greater forecast by the National Weather Service (NWS- Columbia, S.C. office) to be at the plant site in the next four hours. - - - Seismic monitoring system indicates seismic event OR plant operators report an earthquake was "felt"	Aircraft crash causes damage to safety related plant structures or safety related systems within the protected area. - - - Report or detection of toxic, flammable, or asphyxiant gases that could enter the site area in amounts > life-threatening or flammable concentrations that could affect normal operation of the plant. - - - Report of turbine failure resulting in casing penetration or damage to turbine or generator seals. - - - Fire in an area contiguous or adjacent to a vital area that is not extinguished within 15 min. of control room notification by fire alarm or personnel report. - - - Unanticipated explosion within protected area resulting in visible damage to permanent structures or equipment.	Confirmed Security Event which indicates a potential degradation in the level of safety of the plant. (i.e.,) 1. Bomb device in Protected Area but outside Vital area. 2. Hostage situation inside Protected Area. 3. Civil disturbance on plant site outside Protected Area. 4. Credible Bomb threat (Note 3) 5. Credible Attack threat (Note 3) - - - Notification by Local, County, or State officials of potential for evacuation of site personnel based on an Off-Site event	Loss or Potential Loss of Containment barrier - - - RCS Chemistry analysis indicates Dose equivalent I-131 > 1 µci/gm for > 48 hrs. or in excess of Tech Spec. figure 3.4.16-1 OR RCS specific activity > 100/Ē µci/gm gross radioactivity. - - - RCS Unidentified leakage > 10 gpm OR RCS Pressure boundary leakage > 10 gpm OR RCS identified leakage > 25 gpm		Other conditions exist which in the judgment of the EMERGENCY DIRECTOR indicate potential degradation of the level of safety of the plant - - - Modes 1-4: Plant NOT brought to required operating mode within Tech Spec LCO OR TRM Technical Requirement required action completion time limit - - - Modes 1-4: Uncontrolled depressurization of one or more steam generators

NOTE 1: Classification should be based on ODCM or Off-site Dose Calculation computer program results, however, if the monitor reading(s) is sustained for longer than the period indicated and release assessment has NOT or CANNOT be completed within this period, then declaration MUST be made based on the valid reading.

NOTE 2: "Normal levels" are the highest reading in the last 24 hours prior to the emergency, excluding the current peak value

NOTE 3: See section 2.0 of this procedure

Figure 4 - EMERGENCY CLASSIFICATION LEVEL DETERMINATION

PERMS MONITORS

Gas

Monitor	Type	Effluent Monitor	Auto Actions
Plant Vent RE-12442A	BC-400 Airborne Particulate	X	
Plant Vent RE-12442B	Na I Airborne Iodine	X	
Plant Vent RE-12442C	BC-400 Airborne Gas	X	
Cnmt Vent RE-2565A	BC-400 Airborne Particulate	X	CVI
Cnmt Vent RE-2565B	Na I Airborne Iodine	X	CVI
Cnmt Vent RE-2565C	BC-400 Airborne Gas	X	CVI
Cnmt Atmos RE-2562A	BC-400 Airborne Particulate		
Cnmt Atmos RE-2562B	Passive (filter)		
Cnmt Atmos RE-2562C	BC-400 Airborne Gas		
CR Intake RE-12116	GM Inline Vent		CRI
CR Intake RE-12117	GM Inline Vent		CRI
Waste Gas ARE-013	GM Inline Gas		
Waste Gas ARE-014	GM Inline Gas	X	Auto closes RV-14 to isolate Waste Gas release
WG Decay Tank exh RE-39A/B	GM Inline Vent Gas	X	
FHB Vent RE-2532A/B	GM Inline Vent Gas	X	FHB Isolation
FHB Vent RE-2533A/B	GM Inline Vent Gas	X	FHB Isolation
RPF Vent ARE-16980	Beta Scint.	X	
Steamline RE-0724	N16		
SJAE RE-0810	Noble Gas		

Liquid

Monitor	Type	Effluent Monitor	Auto Actions
CCW RE-017A/B	Na I Liquid		
Liquid Waste RE-0018	Na I Liquid	X	Auto close RV-018 to isolate liquid waste release
SGBD RE-0019	Na I Liquid		
NSCW RE-0020A/B	Na I Liquid	X	
SGBD RE-021	Na I Liquid	X	Auto close RV-021 to isolate SGBD effluent to WWRB
TB Drain RE-0848	Na I Liquid	X	Re-aligns TB effluent to the TB Drain Tank
ACCW RE-1950	Na I Liquid		
CVCS RE-48000	Na I Liquid		

PAMS MONITORS

Monitor	Type	Effluent Monitor	Auto Actions
Plant Vent RE-12444A	***Passive (filter)***	X	
Plant Vent RE-12444B	***Passive (filter)***	X	
Plant Vent RE-12444C	BC-400 Wide Range - Low	X	
Plant Vent RE-12444D	BC-400 Wide Range - High	X	
Plant Vent RE-12444E	BC-400 Wide Range - High	X	
SJAE / SPE RE-12839A	Passive	X	
SJAE / SPE RE-12839B	Passive	X	
SJAE / SPE RE-12839C	BC-400 Wide Range - Low	X	Aligns exhaust through a HEPA filter
SJAE / SPE RE-12839D	BC-400 Wide Range - Mid	X	Aligns exhaust through a HEPA filter
SJAE / SPE RE-12839E	BC-400 Wide Range - High	X	Aligns exhaust through a HEPA filter
Main Steam RE-13119 RE-13120 RE-13121 RE-13122	Strap on GM One per main steam line		

ARMS MONITORS

Monitor	Type	Safety Related	Auto Actions
Control Room RE-0001	GM - Area		
Cnmt - Low Range RE-002	GM - Area	X	CVI
Cnmt - Low Range RE-003	GM - Area	X	CVI
Cnmt Access Hatch RE-004	GM - Area		
Cnmt - High Range RE-005	Ion Chamber	X	
Cnmt - High Range RE-006	Ion Chamber	X	
Sampling Room ARE-007	GM - Area		
FHB RE-008	GM - Area		
Seal Table RE-0011	GM - Area		
RPF HIC ARE-16971	GM - Area		
RPF Demin ARE-16972	GM - Area		
RPF Dress-out ARE-16973	GM - Area		

..G2.4.45 06/29/2000

Byron 1 **Exam Level** S

Mark Question | Print Recor | New Searc | Exit

Question

Unit 2 was operating at 100% power when a large break LOCA occurred. All safeguards equipment responded as designed. The crew has transitioned to 2BEP-1, "Loss of Reactor or Secondary Coolant."

Which of the following radiation monitor alarms will be used by the Station Director in assessing the emergency action levels?

Answer: 2RT-AR020 (Containment High Range)

Distracter 1 2RT-AR001 (Containment Area)

Distracter 2 2RT-AR011 (Containment Fuel Handling Incident)

Distracter 3 2RT-PR011 (Containment Atmosphere)

Distracter Analysis:

Answer:

Distracter 1:

Distracter 2:

Distracter 3: