Draft Submittal

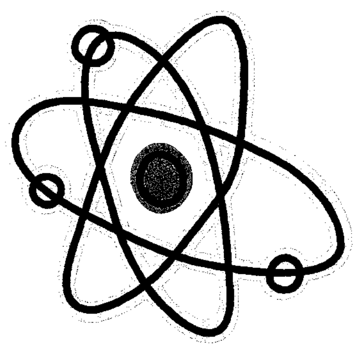
(Pink Paper)

Senior Reactor Operator Written Exam

MCGUIRE MAY 2008 EXAM - 50-369, 370/2008-301
DRAFT SRO WRITTEN EXAM

NRC Examination

Written Exam
(SRO)



McGuire Nuclear Station
45-day Submittal
March 17, 2008

Site-Specific Written Examination
McGuire Units 1 and 2
Senior Reactor Operator
Answer Key

1.	B	26.	B	51.	B	76.	A
2.	D	27.	C	52.	A	77.	D
3.	C	28.	B	53.	C	78.	C
4.	A	29.	A	54.	A	79.	D
5.	D	30.	C	55.	B	80.	D
6.	D	31.	C	56.	A	81.	B
7.	C	32.	A	57.	A	82.	A
8.	A	33.	B	58.	A	83.	A
9.	C	34.	B	59.	D	84.	A
10.	B	35.	C	60.	B	85.	D
11.	A	36.	B	61.	C	86.	A
12.	C	37.	D	62.	C	87.	A
13.	C	38.	A	63.	A	88.	A
14.	A	39.	D	64.	A	89.	B
15.	B	40.	C	65.	C	90.	A
16.	C	41.	B	66.	B	91.	D
17.	A	42.	C	67.	A	92.	B
18.	C	43.	B	68.	C	93.	C
19.	C	44.	A	69.	A	94.	B
20.	A	45.	C	70.	C	95.	A
21.	D	46.	A	71.	D	96.	B
22.	A	47.	D	72.	B	97.	D
23.	B	48.	C	73.	B	98.	A
24.	A	49.	A	74.	A	99.	A
25.	B	50.	C	75.	B	100.	D

Facility:	McGuire 2008 NRC Exam	Date of Exam:	5/12/2008

Tier	Group	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	SRO-Only Points A2	SRO-Only Points G*	SRO-Only Points Total
1. Emergency & Plant Evolutions	1	3	3	3				3	3			3	18	3	3	6
	2	1	2	1				2	2			1	9	2	2	4
	Tier Totals	4	5	4				5	5			4	27	5	5	10
2. Plant Systems	1	2	2	3	3	2	3	2	2	3	3	3	28	3	2	5
	2	1	1	1	1	1	1	1	1	1	1	0	10	0	2	1	3
	Tier Totals	3	3	4	4	3	4	3	3	4	4	3	38	5	3	8

3. Generic Knowledge & Abilities Categories	1	2	3	4	Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
	3	3	2	2	10	1	2	2	2	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; operationally important, site-specific systems that are not included on the outline should be added. Refer to section D.1.b of ES-401, for guidance regarding elimination of inappropriate K/A statements.

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

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

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

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

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

9. For Tier 3, select topics from Section 2 of the K/A Catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10CFR55.43

McGuire 2008 NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
009 / Small Break LOCA / 3						X	2.1.27 - Conduct of Operations: Knowledge of system purpose and / or function.	4.0	76
026 / Loss of Component Cooling Water / 8						X	2.1.25 - Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.	4.2	77
029 / Anticipated Transient Without Scram (ATWS) / 1					X		EA2.01 - Ability to determine or interpret the following as they apply to a ATWS: Reactor nuclear instrumentation	4.7	78
055 / Station Blackout / 6					X		EA2.03 - Ability to determine or interpret the following as they apply to a Station Blackout: Actions necessary to restore power	4.7	79
056 / Loss of Off-site Power / 6					X		AA2.18 - Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Reactor coolant temperature, pressure, and PZR level recorders	4.0	80
058 / Loss of DC Power / 6						X	2.2.37 Equipment Control: Ability to determine operability and/or availability of safety related equipment	4.6	81
007 / Reactor Trip / 1						X	2.4.46 - Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions.	4.2	39
008 / Pressurizer Vapor Space Accident / 3				X			AA1.03 - Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: Turbine bypass in manual control to maintain header pressure	2.8	40
009 / Small Break LOCA / 3		X					EK2.03 - Knowledge of the interrelations between the small break LOCA and the following: S/Gs	3.0	41
011 / Large Break LOCA / 3			X				EK3.06 - Knowledge of the reasons for the following responses as the apply to the Large Break LOCA: Actuation of Phase A and B during LOCA initiation	4.3	42
015 / 17 / Reactor Coolant Pump Malfunctions / 4				X			AA1.16 - Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Low power reactor trip block status lights	3.2	43
022 / Loss of Reactor Coolant Makeup / 2					X		AA2.04 - Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Pump Makeup: How long PZR level can be maintained within limits	2.9	44
025 / Loss of Residual Heat Removal System / 4		X					AK2.03 - Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Service water or closed cooling water pumps	2.7	45
027 / Pressurizer Pressure Control System Malfunction / 3	X						AK1.03 - Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Latent heat of vaporization/condensation	2.6	46
029 / Anticipated Transient Without Scram (ATWS) / 1		X					EK2.06 - Knowledge of the interrelations between the and the following an ATWS: Breakers, relays, and disconnects	2.9	47
055 / Station Blackout / 6			X				EK3.02 - Knowledge of the reasons for the following responses as the apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power	4.3	48

McGuire 2008 NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
056 / Loss of Off-site Power / 6				X			AA1.12 - Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: Reactor building cooling unit	3.2	49
057 / Loss of Vital AC Electrical Instrument Bus / 6			X				AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus	4.1	50
058 / Loss of DC Power / 6	X						AK1.01 - Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation	2.8	51
062 / Loss of Nuclear Service. Water / 4					X		AA2.06 - Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The length of time after the loss of CCW flow to a component before that component may be damaged	2.8	52
065 / Loss of Instrument Air / 8					X		AA2.08 - Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment	2.9	53
E04 / LOCA Outside Containment / 3	X						EK1.2 - Knowledge of the operational implications of the following concepts as they apply to the (LOCA Outside Containment): Normal, abnormal and emergency operating procedures associated with (LOCA Outside Containment).	3.5	54
E11 / Loss of Emergency Coolant Recirculation / 4						X	2.1.23 - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.3	55
E12 / Uncontrolled Depressurization of all Steam Generators / 4						X	2.1.20 - Conduct of Operations: Ability to interpret and execute procedure steps.	4.6	56
K/A Category Totals:	3	3	3	3	6	6	Group Point Total:		18/6

McGuire 2008 NRC Exam
Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

EAPE # / Name Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Q#
003 / Dropped Control Rod / 1					X		AA2.02 - Ability to determine and interpret the following as they apply to the Dropped Control Rod: Signal inputs to rod control system	2.8	82
033 / Loss of Intermediate Range Nuclear Instrumentation / 7					X		AA2.05 - Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Nature of abnormality, from rapid survey of control room data	3.1	83
059 / Accidental Liquid RadWaste Release / 9						X	2.2.38 - Equipment Control: Knowledge of conditions and limitations in the facility license.	4.5	84
E06 / Degraded Core Cooling / 4						X	2.1.20 - Conduct of Operations: Ability to interpret and execute procedure steps.	4.6	85
005 / Inoperable/Stuck Control Rod / 1				X			AA1.02 - Ability to operate and / or monitor the following as they apply to the Inoperable / Stuck Control Rod: Rod selection switches	3.7	57
024 / Emergency Boration / 1					X		AA2.03 - Ability to determine and interpret the following as they apply to the Emergency Boration: Correlation between boric acid controller setpoint and boric acid flow	2.9	58
032 / Loss of Source Range Nuclear Instrumentation / 7	X						AK1.01 - Knowledge of the operational implications of the following concepts as Effects of voltage changes on performance	2.5	59
033 / Loss of Intermediate Range Nuclear Instrumentation / 7				X			AA1.02 - Ability to operate and / or monitor the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Level trip bypass	3.0	60
036 / Fuel Handling Incidents / 8		X					AK2.01 - Knowledge of the interrelations between the Fuel Handling Incidents and the following: Fuel handling equipment	2.9	61
067 / Plant Fire On-site / 8					X		AA2.12 - Ability to determine and interpret the following as they apply to the Plant Fire on Site: Location of vital equipment within fire zone	2.9	62
069 / Loss of Containment Integrity / 5			X				AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Containment Integrity: Guidance contained in EOP for loss of containment integrity	3.8	63
076 / High Reactor Coolant Activity / 9		X					AK2.01 - Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors	2.6	64
E02 / SI Termination / 3						X	2.4.8 - Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOP's.	3.8	65
K/A Category Totals:	1	2	1	2	4	3	Group Point Total:		9/4

McGuire 2008 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G		Imp.	Q #
010 Pressurizer Pressure Control								X				A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures	3.6	86
026 Containment Spray								X				A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of ESF	4.4	87
061 Auxillary/Emergency Feedwater											X	2.4.4 - Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.7	88
073 Process Radiation Monitoring											X	2.2.22 - Equipment Control: Knowledge of limiting conditions for operations and safety limits	4.7	89
076 Service Water								X				A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Service water header pressure	3.1	90
003 Reactor Coolant Pump	X											K1.08 - Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems: Containment isolation	2.7	1
004 Chemical and Volume Control					X							K5.46 - Knowledge of the operational implications of the following concepts as they apply to the CVCS: Reason for going solid in PZR (collapsing steam bubble): make sure no steam is in PRT when PORV is opened to drain RCS	2.5	2
005 Residual Heat Removal		X										K2.03 - Knowledge of bus power supplies to the following: RCS pressure boundary motor-operated valves	2.7	3
006 Emergency Core Cooling										X		A4.01 - Ability to manually operate and/or monitor in the control room: Pumps	4.1	4
007 Pressurizer Relief/Quench Tank									X			A3.01 - Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT	2.7	5
008 Component Cooling Water		X										K2.02 - Knowledge of bus power supplies to the following: CCW pump, including emergency backup	3.0	6
008 Component Cooling Water				X								K4.01 - Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Automatic start of standby pump	3.1	7

McGuire 2008 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G		Imp.	Q #
010 Pressurizer Pressure Control	X											K1.07 - Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: Containment	2.9	8
012 Reactor Protection									X			A3.03 - Ability to monitor automatic operation of the RPS, including: Power supply	3.4	9
012 Reactor Protection						X						K6.03 - Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Trip logic circuits	3.3	10
013 Engineered Safety Features Actuation				X								K4.08 - Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following Redundancy	3.1	11
022 Containment Cooling								X				A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Major leak in CCS	3.1	12
025 Ice Condenser						X						K6.01 - Knowledge of the effect of a loss or malfunction of the following will have on the ice condenser system: Upper and lower doors of the ice condenser	3.4	13
025 Ice Condenser					X							K5 01 - Knowledge of operational implications of the following concepts as they apply to the ice condenser system: Relationships between pressure and temperature	3.0	14
026 Containment Spray											X	2.2.22 - Equipment Control: Knowledge of limiting conditions for operations and safety limits.	4.0	15
026 Containment Spray			X									K3.02 - Knowledge of the effect that a loss or malfunction of the CSS will have on the following: Recirculation spray system	4.2	16
039 Main and Reheat Steam										X		A4.04 - Ability to manually operate and/or monitor in the control room: Emergency feedwater pump turbines	3.8	17
039 Main and Reheat Steam				X								K4.07 - Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Reactor building isolation	3.4	18
059 Main Feedwater							X					A1.03 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves.	2.7	19
061 Auxillary/Emergency Feedwater			X									K3.01 - Knowledge of the effect that a loss or malfunction of the AFW will have on the following: RCS	4.4	20
062 AC Electrical Distribution									X			A3.05 - Ability to monitor automatic operation of the ac distribution system, including: Safety-related indicators and controls	3.5	21
063 DC Electrical Distribution							X					A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the dc electrical system controls including: Battery capacity as it is affected by discharge rate	2.5	22

McGuire 2008 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 1

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G		Imp.	Q #
064 Emergency Diesel Generator						X						K6.08 - Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks	3.2	23
073 Process Radiation Monitoring								X				A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Erratic or failed power supply	2.5	24
076 Service Water											X	2.4.30 - Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.	2.7	25
076 Service Water			X									K3.07 - Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads	3.7	26
078 Instrument Air											X	2.4.35 - Emergency Procedures / Plan: Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects.	3.8	27
103 Containment										X		A4.01 - Ability to manually operate and/or monitor in the control room: Flow control, pressure control, and temperature control valves, including pneumatic valve controller	3.2	28
K/A Category Totals:	2	2	3	3	2	3	2	5	3	3	5	Group Point Total:		28/5

McGuire 2008 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q#
015 Nuclear Instrumentation											X	2.1.7 - Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.7	91
028 Hydrogen Recombiner and Purge Control								X				A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the HRPS; and (b) based on those predictions, use Procedures to correct, control, or mitigate the consequences of those malfunctions or operations: The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment dam-age in containment	4.0	92
079 Station Air								X				A2.01 - 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	3.2	93
014 Rod Position Indication				X								K4.03 - Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following: Rod Bottom lights	3.2	29
017 In-core Temperature Monitor						X						K6.01 - Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors	2.7	30
015 Nuclear Instrumentation System		X										K2.01 - Knowledge of bus power supplies to the following: NIS channels, components, and interconnections	3.3	31
028 Hydrogen Recombiner and Purge Control	X											K1.01 - Knowledge of the physical connections and/or cause-effect relationships between the HRPS and the following systems: Containment annulus ventilation system (including pressure limits)	2.5	32
029 Containment Purge										X		A4.04 - Ability to manually operate and/or monitor in the control room: Containment Evacuation signal	3.5	33
033 Spent Fuel Pool Cooling			X									K3.01 - Knowledge of the effect that a loss or malfunction of the Spent Fuel Pool Cooling System will have on the following: Area ventilation systems	2.6	34
035 Steam Generator								X				A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the S/GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Unbalanced flows to the S/Gs	3.2	35
041 Steam Dump/Turbine Bypass Control									X			A3.03 - Ability to monitor automatic operation of the SDS, including: Steam flow	2.7	36

McGuire 2008 NRC Exam
Written Examination Outline
Plant Systems – Tier 2 Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G		Imp.	Q #
071 Waste Gas Disposal							X					A1.06 - Ability to predict and/or monitor changes in parameters(to prevent exceeding design limits) associated with Waste Gas Disposal System operating the controls including: Ventilation system	2.5	37
086 Fire Protection					X							K5.03 - Knowledge of the operational implication of the following concepts as they apply to the Fire Protection System: Effect of water spray on electrical components	3.1	38
K/A Category Totals:	1	1	1	1	1	1	1	3	1	1	1	Group Point Total:		12/3

Facility:	McGuire 2008 NRC Exam	Date: 5/12/2008				
Category	K/A #	Topic	RO		SRO-Only	
			IR	Q#	IR	Q#
1. Conduct of Operations	2.1.35	Knowledge of the fuel-handling responsibilities of SRO's.			3.9	94
	2.1.18	Ability to make accurate, clear and concise logs, records, status boards, and reports.	3.6	66		
	2.1.13	Knowledge of facility requirements for controlling vital / controlled access.	2.5	67		
	2.1.8	Ability to coordinate personnel activities outside the control room.	3.4	68		
	Subtotal			3		1
2. Equipment Control	2.2.7	Knowledge of the process for conducting special or infrequent tests.			3.6	95
	2.2.22	Knowledge of limiting conditions for operations and safety limits.			4.7	96
	2.2.40	Ability to apply technical specifications for a system.	3.4	69		
	2.2.13	Knowledge of tagging and clearance procedures.	4.1	70		
	2.2.6	Knowledge of the process for making changes to procedures.	3.0	71		
	Subtotal			3		2
3. Radiation Control	2.3.14	Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.			3.8	97
	2.3.6	Ability to approve release permits			3.8	98
	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.	3.2	72		
	2.3.11	Ability to control radiation releases.	3.8	73		
	Subtotal			2		2
4. Emergency Procedures / Plan	2.4.46	Ability to verify that the alarms are consistent with the plant conditions.			4.2	99
	2.4.8	Knowledge of how abnormal operating procedures are used in conjunction with EOP's.			4.5	100
	2.4.17	Knowledge of EOP terms and definitions.	3.9	74		
	2.4.14	Knowledge of general guidelines for EOP usage.	3.8	75		
	Subtotal			2		2
Tier 3 Point Total				10		7

Tier / Group	Randomly Selected K/A	Reason for Rejection
2 / 2	015 G2.1.34	Q 91 Generic topic selected provided no relationship with system selected. Randomly reselected G2.1.7
2 / 1	026 A2.01	Q 87 Procedures have no relationship to, and no guidance for, phenomenon related to topic. Randomly reselected A2.03
1 / 1	015 AA1.04	Q 43 Facility does not have selected component or indication. Randomly reselected AA1.16
2 / 2	029 A4.01	Q 33 Facility cannot operate or monitor purge flow rate from control room; can only start or stop fans. Randomly reselected A4.04
2 / 2	027 K2.01	Q 31 Facility does not have system or fans provided specifically for iodine removal. Kept K2 category and randomly reselected system 015
2 / 2	014 K4.02	Q 29 Lower Electrical Limit is CE, not applicable to WEC design. Randomly reselected K4.03
2 / 1	012 K6.07	Q 10 Facility does not have Core Protection Calculators, CE design. Randomly reselected K6.03
2 / 1	008 K4.07	Q 7 Facility does not have swing pump breaker. Randomly reselected K4.01
1 / 1	055 EA2.05	Q 79 Excessive topic overlap with RO examination related to DC distribution. References did not support an SRO level test item without excessive overlap. Randomly reselected EA2.03 from 055 topic area.
1 / 1	058 G2.4.21	Q 81 Facility and generic references did not support any test item directly related to KA topic. Topic selected could not be tested at SRO level in either closed or open reference format. Randomly reselected Generic topic 2.2.37 from required Tier 1 and Tier 2 generic topics for 058 topic area.
2 / 1	073 G2.2.4	Q 89 Facility has no difference in system between units that could be developed into a test item, either at RO or SRO level. Randomly reselected generic 2.2.22 from required Tier 1 and Tier 2 generic topics for 073 topic area

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	LOCA 009 G2.1.27	
	Importance Rating		4.0

Conduct of Operations: Knowledge of system purpose and / or function.

Proposed Question: SRO 76

Given the following conditions:

- A reactor trip has occurred.
- Safety Injection is actuated. All equipment has actuated as designed.
- The crew is performing EP/1/A/5000/E-0, Reactor Trip or Safety Injection.
- NC System pressure is 1700 psig and lowering slowly.
- Pressurizer level is off-scale low.
- Containment pressure is 1.7 psig and rising slowly.
- FWST level is 300 inches and dropping at 2 inches per minute.
- SG pressures are 1050 psig and stable.
- CA flow is 600 gpm.
- The operators are performing E-1, Loss of Reactor or Secondary Coolant.

Which ONE (1) of the following describes the procedure that will be used next, to mitigate the event in progress, and the technical specification basis of FWST parameters for this event?

A. ES-1.2, Post LOCA Cooldown and depressurization; FWST minimum volume ensures a sufficient volume of water in the containment sump after ECCS injection to initiate Cold Leg Recirculation.

B. ES-1.3, Transfer to Cold Leg Recirculation; FWST minimum volume ensures a sufficient volume of water in the containment sump after ECCS injection to initiate Cold Leg Recirculation.

C. ES-1.2, Post LOCA Cooldown and depressurization; FWST minimum volume ensures that post LOCA core cooling requirements are met for the ECCS injection phase even with an anticipated loss of Cold Leg Recirculation.

D. ES-1.3, Transfer to Cold Leg Recirculation; FWST minimum volume ensures that post LOCA core cooling requirements are met for the ECCS injection phase even with an anticipated loss of Cold Leg Recirculation.

Proposed Answer: A

Explanation (Optional):

A. Correct. Correct Procedure and FWST basis. ES-1.2 will be entered because the rate of change on FWST level will result in conditions NOT being met for ES-1.3 for another 40-60 minutes. ES-1.2 transition will come significantly sooner

B. Incorrect. Procedure is incorrect because ES-1.3 will not be performed next, it will take too long to reach conditions

C. Incorrect. Basis is incorrect, because a loss of cold leg recirculation is beyond design basis for FWST operability.

D. Incorrect. Basis and procedure are incorrect, as described in B and C above

Technical Reference(s): E-1, Rev 11; ES-1.2 Rev 11; ES-1.3 Rev 23 (Attach if not previously provided)

TS 3.5.4 basis Rev 70

EP-E1 p11, 15, 59 Rev 17; FH-FW p 23, 67 Rev 40

Proposed references to be provided to applicants during examination: None

Learning Objective: FH-FW Obj 5; EP-E1 Obj 2 (As available)

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____

Question Cognitive Memory or Fundamental Knowledge ____

Level:

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:
KA matched because item evaluates function of FWST (RWST)

SRO level because the applicant must determine procedure entry based on plant conditiuons and also know TS basis for operability of FWST

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14. **Check if NC System cooldown and depressurization is required:**	
__ a. NC pressure - GREATER THAN 286 PSIG.	a. Perform the following: __ 1) **IF** containment pressure has remained less than 3 PSIG, **THEN GO TO** EP/1/A/5000/ES-1.2 (Post LOCA Cooldown And Depressurization). __ 2) **IF** ND flow to cold legs greater than 500 GPM, **THEN GO TO** Step 15.
__ b. **GO TO** EP/1/A/5000/ES-1.2 (Post LOCA Cooldown And Depressurization).	
15. **Check transfer to Cold Leg Recirc criteria:**	
__ a. FWST level - LESS THAN 180 INCHES ("FWST LEVEL LO" ALARM).	__ a. **RETURN TO** Step 13.
__ b. Check S/I systems - ALIGNED FOR COLD LEG RECIRC.	__ b. **GO TO** EP/1/A/5000/ES-1.3 (Transfer To Cold Leg Recirc).

2.0 PROCEDURE SERIES BACKGROUND

2.1. E-1, Loss of Reactor or Secondary Coolant

2.1.1 Loss of Reactor Coolant

In order to describe the various phenomena that can occur during a LOCA, it is convenient to define five categories of accidents based on the size of the break and number of S/I trains. This section describes four break sizes and Safeguard equipment status as follows:

1. Breaks between 3/8" ($\approx$0.1 in^2) and 1" ($\approx$0.8 in^2) diameter with minimum safety injection. NC pressure will stabilize above steam generator pressure.
2. Breaks between 3/8" ($\approx$0.1 in^2) and 1" ($\approx$0.8 in^2) diameter with maximum safety injection. The NC will repressurize.
3. Breaks between 1" ($\approx$0.8 in^2) and 13.5" ($\approx$1 ft^2) diameter. NC pressure goes below steam generator pressure.
4. Breaks greater than 1 ft^2. The NC will rapidly depressurize to close to the containment atmospheric pressure.

Breaks smaller than 3/8" ($\approx$0.1 in^2) with normal charging are considered to be leaks rather than small LOCAs since NC pressure and Pzr level do not go down. If charging flow is not available, the transient would be similar to the response described below for small LOCAs.

SMALL LOCAs

The flowpath through the E-1 series is dependent upon the break size, the break location, and operator/Station Management decisions. For a break size of up to 1 inch diameter, the amount of S/I flow determines the flow path in the E-1 series. If minimum S/I flow is assumed, the E-1 S/I-termination criteria would not be met, repressurization of the reactor coolant may not occur, and S/I flow equals the break flow. This constitutes a safe and stable condition for the long term provided the heat sink is maintained. As long as S/I and Auxiliary Feedwater are available, the reactor will reach equilibrium conditions for the steam generator pressures. Long-term cooling may require depressurizing to cold shutdown while stepping down S/I flow, so ES-1.2, Post LOCA Cooldown and Depressurization would be used.

If maximum S/I flow is assumed such that S/I flow is greater than break flow, the reactor will rapidly repressurize, and may in fact end up with the pressurizer filled solid. At this point, the NC system will rapidly repressurize and the S/I termination criteria will be met, and S/I may be terminated using ES-1.1, S/I Termination. However, if S/I is not terminated, or more realistically, if S/I termination is delayed, the core will remain cooled and in a safe and long term stable condition. The NC system will remain in an acceptable, although possibly not desirable, condition.

2.2. ES-1.1, Safety Injection Termination

S/I TERMINATION is entered based on the following criteria:

1. The NC is subcooled,
2. An adequate secondary heat sink exists,
3. NC pressure is either stable or going up, and
4. Pressurizer level is indicating greater than 11% (29% ACC).

These conditions combined indicate that the NC is in a safe state with adequate core cooling and that S/I flow can be reduced without jeopardizing the safety of the plant. S/I pumps are stopped in a prescribed sequence as long as control is maintained, until makeup is only from normal charging pump lineup. Appropriate transitions are provided in case all S/I pumps cannot be stopped or must be restarted. If S/I pumps are stopped and control is maintained, then the plant configuration is essentially realigned to a pre-S/I condition at no-load or some lower stable temperature.

2.3. ES-1.2, Post LOCA Cooldown and Depressurization

For a LOCA, plant design is to use makeup water from the FWST until it is drained. Recirculation from the containment sump to the NC is then used for long-term cooling and makeup. The time to switch over to recirculation depends on the size of the break, the FWST water volume, and whether containment spray is initiated. For some smaller breaks, it is possible to cool down and depressurize the NC to a cold shutdown condition before the FWST is drained. When doing this, it is important to maintain adequate core cooling by maintaining inventory while also trying to minimize FWST depletion.

For any loss of NC inventory, the NC pressure will be dependent on the size of the break, the NC fluid shrink due to cooldown, and the S/I flow rate. For smaller breaks the NC pressure will remain stabilized for a long period of time at high NC pressures (greater than 400 psig). For these breaks, transfer to cold leg recirculation may be necessary while NC pressure remains high.

Procedure ES-1.2 provides actions to reduce the NC temperature and pressure to or below 200°F and 400 psig. This is done by establishing a S/G cooldown and selectively reducing S/I flow by stopping S/I pumps or establishing normal charging flow if minimum subcooling and Pzr level can be established. From there, the plant staff can determine how to completely depressurize the plant to stop NC inventory loss and effect repairs.

3.6. Final Plant Status

E-1 provides the actions to recover from a loss of reactor or secondary coolant.

The following table summarizes the exit guidance from E-1. The left column lists each step that provides a potential exit point from E-1. The right column lists the transition procedure(s). If an exit transition is necessary, the operator should transition to Step 1 unless otherwise directed.

E-1 STEP NUMBER	TRANSITION PROCEDURE(S)
Step 3	E-2, Faulted Steam Generator Isolation
Step 4, 5	E-3, Steam Generator Tube Rupture
Step 7	ES-1.1, Safety Injection Termination
Step 13	ECA-1.1, Loss of Emergency Coolant Recirc
	ECA-1.2, LOCA Outside Containment
Step 14	ES-1.2, Post LOCA Cooldown and Depressurization
Step 15	ES-1.3, Transfer to Cold Leg Recirc
Step 20	ES-1.4, Transfer to Hot Leg Recirc

Other transitions may be made as a result of the Foldout Page directives. These are summarized in the following table.

E-1 FOLDOUT PAGE STEP	TRANSITION PROCEDURE(S)
Secondary Integrity Criteria	E-2, Faulted Steam Generator Isolation
SGTR Transition Criteria	E-3, Steam Generator Tube Rupture
CA Suction Sources	G-1, Enclosure 20, CA Suction Source Realignment
Cold Leg Recirc Switchover Criteria	ES-1.3, Transfer To Cold Leg Recirculation

A. Purpose

This procedure provides actions to cool down and depressurize the NC System to Cold Shutdown conditions following a loss of reactor coolant inventory.

B. Symptoms or Entry Conditions

This procedure is entered from:

- EP/1/A/5000/E-0 (Reactor Trip Or Safety Injection), Step 26, when NC System pressure goes down after stopping all but one NV pump.
- EP/1/A/5000/E-0 (Reactor Trip Or Safety Injection), Step 29, when Pzr level can not be maintained using normal charging.
- EP/1/A/5000/E-1 (Loss Of Reactor Or Secondary Coolant), Step 14, if symptoms of a small break LOCA exist.
- EP/1/A/5000/ES-1.1 (Safety Injection Termination), Step 6, when NC System pressure goes down after stopping all but one NV pump.
- EP/1/A/5000/ES-1.1 (Safety Injection Termination), Step 9, when Pzr level can not be maintained using normal charging.
- EP/1/A/5000/ES-1.1 (Safety Injection Termination), Step 10, when NC System pressure is less than shutoff head pressure of the NI pumps.

therefore are not susceptible to reference leg problems like losing level due to small leaks or evaporation. Problem over the years with FWST level instrumentation has initiated design studies and level instrumentation redesign to make the level indication more reliable. Site Plan MG-97-0035 addresses the FWST Level Instrumentation Improvement Project. It addresses potential common mode failures such as submergence, impulse line freezing and reference line blockage. As a result, modifications are in progress under NSM-12496 and 22496. ***{If a transmitter problem occurs, the operators would first notice it on cross-channel comparison, performed periodically (SOER 97-01 Review), unless it were a gross problem which would cause a level alarm from one of the transmitters.}***

FWST Pressure

Atmospheric pressure exists in the FWST since it is normally vented to atmosphere, and therefore pressure is not monitored in the FWST.

FWST Level

Four channels provide Control Room level indication alarms and protection logic utilized in Normal Operation and ECCS/NS pumps switch-over from the FWST to the Containment Sump following a LOCA. There are three Safety Related Level Channels required by Tech Specs under ESFAS Instrumentation and two Non-Safety Related level instruments used to monitor FWST Level during normal operations.

Each of the three Safety Related level instruments (FWP5000 Channel 4, FWP5010 Channel 1, and FWP5020 Channel 2) has a completely separate reference and variable leg tap, and are located 120 degrees in circumference from each other. Therefore a single failure will not affect more than one channel. Their range is from 0 - 500" WC.

NOTE: **The setpoints at which the Low Level Auto-Switch-over to Cold Leg Recirculation is 180" H_2O. The setpoint at which the Control Room Crew will manually swap Containment Spray Pump Suction to the Containment Sump is 33" H_20. The Non-Safety Related Upper Narrow Range Level Instrument has a range of 405" - 530" WC.**

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling and makeup operations, and to the ECCS and the Containment Spray System during accident conditions.

The RWST supplies both trains of the ECCS and the Containment Spray System through separate supply headers during the injection phase of a loss of coolant accident (LOCA) recovery. A motor operated isolation valve is provided in each header to isolate the RWST once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST—Low Level signal. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and since injection phase passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves.

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

The ECCS pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.

When the suction for the ECCS and Containment Spray System pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

BASES

BACKGROUND (continued)

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS—Operating"; B 3.5.3, "ECCS—Shutdown"; and B 3.6.6, "Containment Spray Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained due to the location of the piping connection. The ECCS water boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. This assumption is important in ensuring the required shutdown capability. Although the maximum temperature is a conservative assumption in the feedwater line break analysis, SI termination occurs very quickly in this analysis and long before significant RCS heatup occurs. The minimum temperature is an assumption in the MSLB actuation analyses.

For a large break LOCA analysis, the RWST level setpoint equivalent to the minimum water volume limit of 372,100 gallons and the lower boron concentration limits listed in the COLR are used to compute the post

BASES

ACTIONS (continued)

restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated 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 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA and that the boron content assumed for the injection water in the MSLB analysis is available. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

Objective 6

2.6 Refueling Water Storage Tank Design Basis

The FWST has (as a minimum) a usable capacity of 372,100 gallons of borated water. The water in the FWST is electrically heated when water temperature decreases to < 75°F. The tank capacity provides an adequate amount of borated water to insure:

- A sufficient volume of borated refueling water needed to increase the boron concentration of the initially spilled water to a point that assures no return to criticality with the reactor at cold shutdown and all control rods fully inserted in the core with the exception of the most reactive rod cluster control assembly.
- A sufficient volume to refill the reactor vessel above the nozzles after a LOCA.
- A sufficient volume of water in the lower compartment of the containment following ECCS Injection to permit the initiation of Cold and Hot Leg Recirculation.
- A sufficient volume of borated water to insure that the radiation dose at the surface of the refueling cavity is limited to 2.5 milli rem per hour during the period when a fuel assembly is transferred over the reactor vessel flange.

The FWST is surrounded by a seismic wall. The basis of the seismic wall is that in the event a Tornado induced missile ruptures the FWST, the wall is high enough to retain a sufficient volume of FWST water to provide NPSH to the Centrifugal Charging Pumps and the Safety Injection Pumps. The Missile induced rupture assumes that there is a Main Steamline Break in conjunction with an FWST rupture. There is no concern for the ND Pumps because it is assumed that the Steam Break Outside Containment Event will not cause primary pressure to be reduced below the Shut-off Head of the pumps. The FWST overflows to the Spent Fuel Pool and to the FWST trench. The following parameters are associated with the FWST:

• Minimum Volume modes 1-4	372,100 gallons
• Minimum Volume modes 5-6	Cycle Dependent (See COLR)
• Minimum Boron Concentration	Cycle Dependent (See COLR)
• Minimum Temperature	70°F
• Maximum Temperature	100°F

A. <u>Purpose</u>

This procedure provides the necessary instructions for transferring the Safety Injection System and Containment Spray System to the recirculation mode.

B. <u>Symptoms or Entry Conditions</u>

This procedure is entered from:

- EP/1/A/5000/E-1 (Loss Of Reactor Or Secondary Coolant), Step 15, on low FWST level.
- EP/1/A/5000/ECA-2.1 (Uncontrolled Depressurization Of All Steam Generators), Step 12, on low FWST level.
- Other procedures whenever FWST level reaches the switchover setpoint.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
N/A	N/A	5.0	5.0	4.0

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose for each procedure in the E-1 series. EPE1001			X	X	
2	Discuss the entry and exit guidance for each procedure in the E-1 series. EPE1002			X	X	
3	Discuss the mitigating strategy (major actions) of each procedure in the E-1 series. EPE1003			X	X	X
4	Discuss the basis for any note, caution or step for each procedure in the E-1 series. EPE1004			X	X	X
5	Given the Foldout page discuss the actions included and the basis for these actions. EPE1005			X	X	X
6	Given the appropriate procedure, evaluate a given scenario describing accident events and plant conditions to determine any required action and its basis. EPE1006			X	X	X
7	Discuss the time critical task(s) associated with the E-1 series procedures including the time requirements and the basis for these requirements. EPE1007			X	X	X

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
2	2	2	2	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the Refueling Water System.	X	X	X	X	
2	Provided with the FW training drawing and OP/1 or 2/A/6200/14 and OP/1 or 2/A/6200/13, discuss the various lineups that can be utilized to transfer water, provide makeup, or purify the refueling water.	X	X	X	X	
3	Identify the valves/pumps/instrumentation that can be operated or monitored from the Control Room.			X	X	X
4	Given a Limit and Precaution associated with the FW System, discuss its basis and when it applies.	X	X	X	X	X
5	Concerning the Technical Specifications related to the Refueling Water System;					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is(are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech Specs, determine required action(s).			X	X	X
	• Discuss the basis for a given Tech Spec LCO or Safety Limit.				X	*
	*** SRO Only**					

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	Loss CCW 026 G2.1.25	
	Importance Rating		4.2

Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: SRO 77

Initial conditions:

Time = 0 minutes

- Unit 1 is at 100% power.
- "A" Train KC pumps are running.
- Operators have been dispatched to initiate YM makeup to the KC Surge Tank.
- "A" KC Surge Tank level is 6.5 ft.
- "B" KC Surge Tank level is 6.5 ft.

Current conditions:

Time = 5 minutes

- "A" KC Surge Tank level is 5.6 feet
- "B" KC Surge Tank level is 6.4 feet.

Which ONE (1) of the following describes the approximate KC system net leak rate, and the action and procedure use required in AP/21, Loss of KC or KC System Leakage?

(Reference Provided)

A. 50 GPM; Isolate KC Non-Essential Headers in accordance with Enclosure 2.

B. 50 GPM; Isolate "A" KC train from "B" KC train.

C. 100 GPM; Isolate KC Non-Essential Headers in accordance with Enclosure 2.

D. 100 GPM; Isolate "A" KC train from "B" KC train.

Proposed Answer: D

Explanation (Optional):

D is correct per conditions. Applicant must interpret curve and determine the leak rate indications based on level decreases

A incorrect because leak rate is wrong (1/2 of actual, as interpreted by curve.) Also, action is incorrect, as procedure will direct splitting trains for indication shown

B incorrect because leak rate is incorrect. Plausible because action is correct

C incorrect because procedure use is incorrect. Approximately 0.1 feet/minute, perform step 20 to split trains

Technical Reference(s): AP/21 Step 20 Rev 9 (Attach if not previously provided)

AP/21 Basis Document Rev 3

Proposed references to be provided to applicants during examination: OP/1/A/6100/22 Enclosure 4.3 Curve 7.31

Learning Objective: None (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam Modified from 2007 NRC exam 78

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

KA matched because use of a curve is required and interpretation of that curve is required to determine KC (CCW) leak rate. SRO level because assessment of conditions based on available indications, and selection of procedures (attachments) is required

The following conditions exist:

- Unit 1 is in Mode 1, 100% power.
- "A" and "B" Train KC pumps are running.
- All available makeup has been established to the KC surge tanks.
- "A" KC Surge tank level is decreasing 0.04 ft/min.
- "B" KC Surge Tank level is decreasing at 0.03 ft/min.
- "A" KC Surge Tank level is presently 3.2 ft.
- "B" KC Surge Tank level is presently 3.4 ft.
- NCP bearing temperatures are approximately 180 degrees F and rising slowly.

Which ONE (1) of the following describes the action and procedure use required?

A. Enter AP/08, NC Pump Malfunctions, and trip NC Pumps.

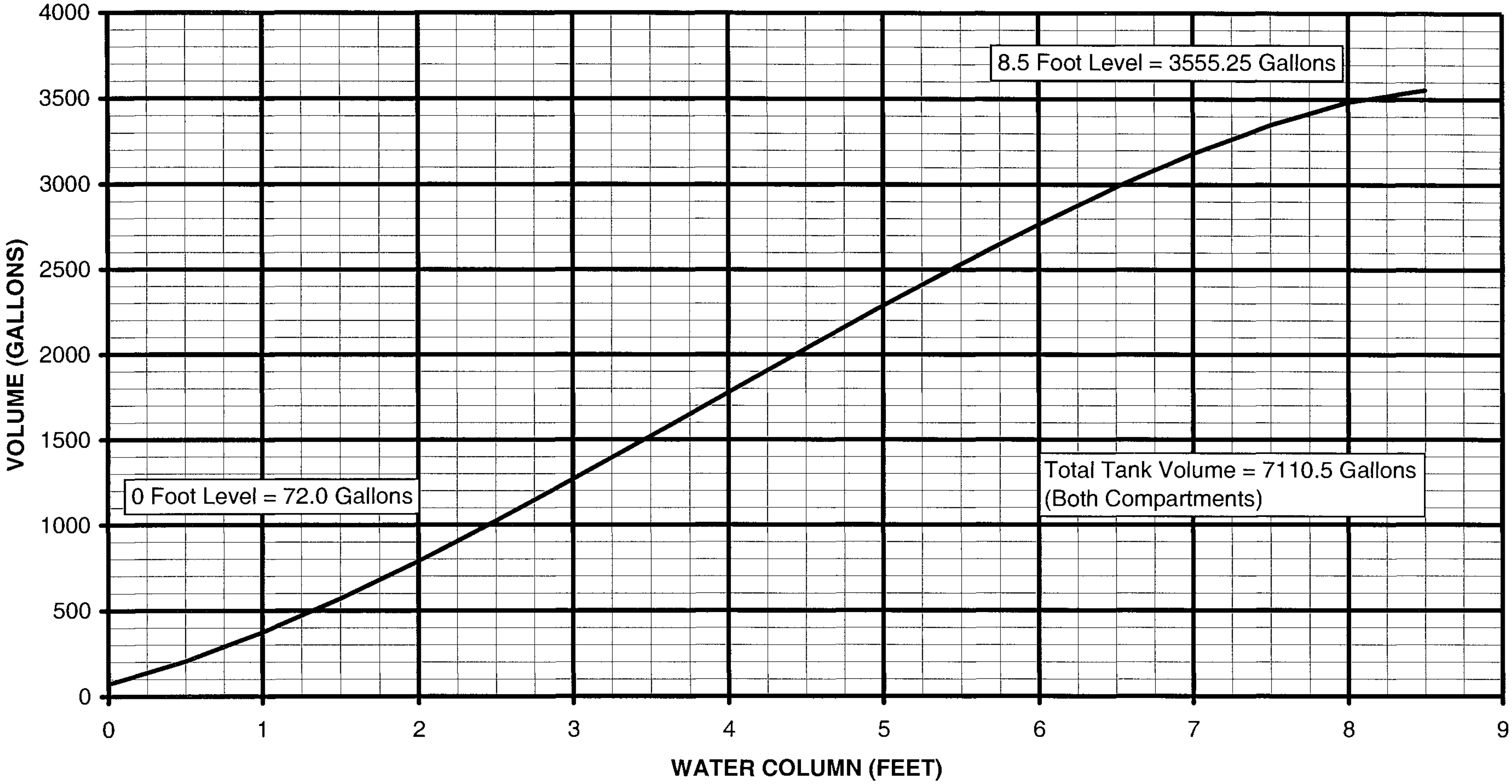
B. Trip the reactor; enter E-0, Reactor Trip or Safety Injection. Trip NCPs and trip "A" KC Pumps.

C. Enter AP/21, Loss of KC or KC System Leakage, and Isolate KC Non-Essential Headers in accordance with Enclosure 2.

D. Enter AP/21, Loss of KC or KC System Leakage, and isolate "A" KC train from "B" KC train.

Ans. D

ENCLOSURE 4.3
CURVE 7.31
COMPONENT COOLING SURGE TANK
(VOLUME vs. COMPARTMENT LEVEL)



This data is also provided on the OAC.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. Symptoms

- **"LO KC HX A INLET FLOW" computer alarm**
- **"LO KC HX B INLET FLOW" computer alarm**
- **Low flow alarms on components supplied by KC**
- **High temperature alarms on components supplied by KC**
- **Low level or level going down in KC Surge Tank**
- **Abnormal KC pump Flow**
- **"LO KC SURGE TANK COMPARTMENT A LEVEL" computer alarm**
- **"LO KC SURGE TANK COMPARTMENT B LEVEL" computer alarm**
- **"KC SURGE TANK ABNORMAL LEVEL" alarm.**

C. Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Check any KC pump - ON.**	**Perform the following:** a. Isolate: __ • Normal letdown __ • Excess letdown __ • ND letdown. __ b. Close all NM valves located on 1MC-8 (vertical board).
__ 2. **Monitor Foldout page.**	
__ 3. **Secure any dilution in progress.**	
__ 4. **Check ND - IN RHR MODE.**	__ **GO TO Step 7.**

1. **KC header isolation criteria:**

   - **IF** KC surge tank level goes below 2 ft due to KC system leak, **THEN** immediately isolate affected train **PER** Enclosure 2 (Isolation of KC Non-essential Headers).

2. **NC pump trip criteria:**

   - **IF** NC pump motor bearing temperature reaches 195°F, **THEN** perform the following:

     a. Trip the reactor.
     b. **WHEN** reactor is tripped, **THEN** trip all NC pumps.
     c. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection), while continuing in this procedure as time and conditions allow.

3. **ND pump trip and flow isolation criteria (Applies if ND aligned for RHR):**

   - **IF** KC cooling lost to either ND train's HX, **AND** NC temperature is greater than 150°F, **THEN** perform the following on train of ND that lost KC flow to its ND HX:

     a. Stop associated ND pump.

     b. **IF** 1A ND HX lost KC flow, **THEN** close:

        - 1ND-33 (A ND Hx Bypass)
        - 1ND-32 (A ND Hx To Letdown Hx).

     c. **IF** 1B ND HX lost KC flow, **THEN** close:

        - 1ND-18 (B ND Hx Bypass)
        - 1ND-17 (B ND Hx To Letdown Hx).

     d. **IF** both ND pumps off **THEN REFER TO** AP/1/A/5500/19 (Loss of ND or ND System Leak).

4. **KC pump trip criteria:**

   - **IF** KC surge tank level goes below .5 ft and valid, **THEN:**

     a. Trip affected pumps.

     b. Isolate affected train **PER** Enclosure 2 (Isolation of KC Non-essential Headers).

5. **VCT high temperature:**

   - **IF** "VCT HI TEMP" alarm (1AD-7, D-1) is received, **THEN REFER TO** Enclosure 6 (VCT High Temperature Actions).

STEP DESCRIPTION FOR AP

STEP 1:

PURPOSE:

Ensure letdown and all NM is isolated if KC pumps are off.

DISCUSSION:

Since KC cools the letdown Hx, letdown is isolated if KC is lost. Note subsequent steps that trip KC pumps or isolate cooling to the aux bldg non-ess header will also isolate letdown. Engineering has calculated that if KC flow through the NM Hx's is lost that it would take less than 3 minutes to flash.

REFERENCES
OEDB 98-18676

STEP 2:

PURPOSE:

Cue the operator to monitor the foldout page.

DISCUSSION:

A foldout page was chosen for this AP as a human-factors' consideration. Maintaining critical items on a separate page ensures they are performed in a timely manner. The foldout page contains actions that apply throughout the AP as described in items below:

1) "KC header isolation criteria" ensures the non-essential headers are isolated from the KC pump and essential header prior to emptying the surge tank. If a leak occurs, the non-essential headers should be isolated prior to air binding the KC pumps. If the leak is on the operating train KC essential header, isolating the non-essential headers will prevent them from draining (so they can be restored in a timely manner using other train). If the leak is on one of the non-essential headers, this isolation protects the essential header. Adequate protection of equipment cooled by the non-essential headers is provided by isolating letdown and by other foldout page items. Note efforts to makeup to the surge tank and isolate leaks will be initiated for smaller leaks prior to having to isolate entire headers. This foldout partially addresses some concerns raised by the NRC in OEDB 98-017559, Loss of inventory from KC. A major concern was not getting a leaking non-essential header isolated in time to prevent the inoperability of the safety-related headers.
2) "NC pump trip criteria". Isolation of the reactor bldg non-essential header or loss of KC pumps may lead to NC pump trip criteria due to loss of cooling to motor bearings. Since

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 15. **Check both train's KC surge tank level - GREATER THAN 3 FT.**	__ **GO TO Step 20.**

NOTE

- The following OAC points may be used to determine level drop in next step. These points are also displayed on the KC system graphic:
  - M1P1317 (1A Train KC surge tank level rate)
  - M1P1318 (1B Train KC surge tank level rate).
- A 0.10 ft/min level drop in one train's surge tank equals approximately 50 GPM leak.

__ 16. **Check sum of both trains' KC surge tank level drops - LESS THAN OR EQUAL TO 0.10 FT/MIN.**	__ **IF level is dropping faster than 0.10 ft/min, THEN GO TO Step 20.**

NOTE The next step allows maintaining current KC system alignment for small leaks that should be within the capacity of normal makeup. Allowing level to drop to 2 ft allows more time for operators to locally align makeup, prior to taking action to isolate KC headers.

17. **Do not continue until at least one of the following occurs:**

__ • KC makeup has been locally opened from RN.

OR

__ • Either train's KC Surge Tank level is less than or equal to 2 ft.

OR

__ • Both KC surge tank levels are stable or going up.

If KC surge tank level is greater than 3 ft, time is given for the operators to attempt to initiate makeup and check results to see if the surge tank level can be maintained. Per engineering, YM makeup should be sufficient to keep up with the FSAR design basis leak of 50 GPM. Operators should be able to initiate makeup prior to reaching 2 ft in the surge tanks (assuming makeup is initiated when KC lo level alarms at 4.5 ft). If the trains are cross-tied, allowing leaving the cross-ties open doubles the volume (and time) to initiate makeup. Note that for larger leaks, or if level reaches 2 ft, the cross-ties will be closed to protect the other train. If makeup is initiated and level stabilizes, operator actions are greatly simplified.

STEP 16 NOTES:

PURPOSE:

Give operator information for determining leak rate.

DISCUSSION:

0.1ft/min level drop is equivalent to the design basis leak (50 gpm) that YM makeup should be able to keep up with.

STEP 16:

PURPOSE:

Procedure flow path controlling step.

DISCUSSION:

If leak is greater than design basis leak, operators need to find the leak and isolate it.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 18. **Check KC surge tank level on both train(s) - STABLE OR GOING UP.**	**IF KC surge tank level is still going down in an uncontrolled manner, THEN:** __ a. **IF** level goes below 2 ft, **THEN** ensure Foldout page item 1 is implemented. __ b. **GO TO** Step 20.
__ 19. **GO TO Step 38.**	
20. **Isolate 1A KC Train from 1B KC Train as follows:**	
__ a. Check any 1A KC Train pump - RUNNING.	__ a. **GO TO** Step 20.f.
b. Check the following valves - OPEN: __ • 1KC-3A (Trn A Rx Bldg Non Ess Ret Isol) __ • 1KC-230A (Trn A Rx Bldg Non Ess Sup Isol).	__ b. **GO TO** Step 20.f.
c. Close the following valves: __ 1) 1KC-53B (Trn B Aux Bldg Non Ess Sup Isol). __ 2) 1KC-2B (Trn B Aux Bldg Non Ess Ret Isol). __ 3) 1KC-228B (Trn B Rx Bldg Non Ess Sup Isol). __ 4) 1KC-18B (Trn B Rx Bldg Non Ess Ret Isol).	
__ d. **WHEN** valves in Step 20.c are closed, **THEN** check 1A KC Surge Tank level - GOING DOWN.	__ d. **IF** 1A KC Surge Tank level stabilizes, **AND** 1B KC Surge tank level continues to go down, **THEN** leak is on 1B Essential header.
__ e. **GO TO** Step 21.	

STEP 17 NOTE:

PURPOSE:

Inform operators why they're waiting.

DISCUSSION:

STEP 17:

PURPOSE:

Establish a hold point in the AP until one of the listed items is met.

DISCUSSION:

The basis for this step is to allow a chance for makeup to be established to compensate for the leak. If it does, or if RN has to be established to keep up, or if the surge tank gets less than 2 ft, the hold point is released.

STEPS 18 & 19:

PURPOSE:

Flow path controlling steps.

DISCUSSION:

If makeup is keeping up with the leak, actions to isolated entire headers are bypassed. If level is going down, it's time to begin isolating headers to stop the leak.

STEP 20:

PURPOSE:

Begin the process of isolating KC headers so the leaking header can be identified. The first step involves closing the non-operating trains' 4 cross-ties to split the two essential headers.

DISCUSSION:

If the A train pumps are running and supplying the Rx non-ess header, then the B train cross-ties (Aux non-ess supply&return, and Rx non-ess supply&return) are closed. Otherwise, if A train is not operating, then the A train cross-ties are closed. After the cross-ties are closed, the surge tank levels are checked. If the operating trains' level stabilizes, and the non-operating

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	029 EA2.01	
	Importance Rating		4.7

Ability to determine or interpret the following as they apply to a ATWS: Reactor nuclear instrumentation

Proposed Question: SRO 78

Given the following conditions:

- An ATWS has occurred on Unit 1.
- The crew is performing FR-S.1, Response to Nuclear Power Generation/ATWS.
- NC Boration is in progress.
- SI has actuated.
- All SG pressures are approximately 800 psig and trending down.
- NC Temperature is approximately 490 degrees F and trending down.
- Enclosure 2 (Faulted SG Isolation) has been initiated.
- Reactor Power indicates approximately 4% and trending down slowly.

Which ONE (1) of the following describes the mitigation strategy for the event in progress?

A. Remain in FR-S.1 and perform Enclosure 2. Transition to E-0, Reactor Trip or Safety Injection ONLY after all steps of Enclosure 2 are complete and NC system temperature is stable.

B. Remain in FR-S.1 and perform Enclosure 2. Transition to E-0, Reactor Trip or Safety Injection when Intermediate Range amps are going down.

C. Conditions exist that allow exit from FR-S.1. When directed, exit FR-S.1 while continuing performance of Enclosure 2. Transition to E-0, Reactor Trip or Safety Injection, prior to transition to ES-1.1, SI Termination.

D. Conditions exist that allow exit from FR-S.1. When directed, exit FR-S.1 and terminate performance of Enclosure 2. Transition to E-2, Faulted Steam Generator Isolation, prior to transition to ES-1.1, SI Termination.

Proposed Answer: C

Explanation (Optional):

A is incorrect. FR-S.1 has guidance to isolate a faulted SG, but cannot transition until power is below 5%

B is incorrect. Would go to E-0 after FR-S.1 is complete and directed by FR-S.1 (Power <5%)

C is Correct. Power less than 5%, transition may occur. Enclosure 2 is still completed if in progress. If conditions are present, some steps of FR-S.1 may have to be performed prior to exit, because the crew may not be at step 17

D is incorrect. Credible because a fault exists and procedure flowpath is correct, but E-0 is performed first

Technical Reference(s): FR-S.1, Rev 10 (Attach if not previously provided)

OMP 4-3 p14, 18

Proposed references to be provided to applicants during examination: None

Learning Objective: FR-S.1 Obj 4 (As available)

Question Source:

Bank # ____

Modified Bank # McGuire 2006 NRC 80 (Note changes or attach parent)

New ____

Question History: Last NRC Exam ____

Question Cognitive Level:

Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content:

55.41 ____

55.43 5

Comments:

KA is matched because transition is made based upon PR NI indications.

SRO level because the item addresses FR-S.1 strategy and compliance with EOPs. The applicant must determine exit conditions available and interpret use of EOP attachments while performing other procedures

Given the following conditions:

- An ATWS has occurred on Unit 1.
- The crew is performing FR-S.1, Response to Nuclear Power Generation/ATWS.
- NC Boration is in progress.
- SI has actuated.
- All SG pressures are approximately 800 psig and trending down.
- NC Temperature is approximately 490 degrees F and trending down.
- Reactor Power indicates approximately 7% and trending down slowly.

Which ONE (1) of the following describes the mitigation strategy for the event in progress?

A. Remain in FR-S.1 and perform Enclosure 2 (Faulted SG Isolation). Transition to E-0, Reactor Trip or Safety Injection when Enclosure 2 is complete.

B. Remain in FR-S.1 and perform Enclosure 2 (Faulted SG Isolation). Transition to E-0, Reactor Trip or Safety Injection when reactor power is less than 5%.

C. Exit FR-S.1; Transition to E-0, Reactor Trip or Safety Injection to ensure actuated components are in their correct alignments.

D. Exit FR-S.1; Transition to E-0, Reactor Trip or Safety Injection and ONLY perform steps of subsequent EOPs that do not contradict the actions taken in FR-S.1.

Ans. B

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13. **Check steamlines intact:** __ • All S/G pressures - STABLE OR GOING UP __ • All S/Gs - PRESSURIZED.	**<u>IF</u> any S/G depressurized <u>OR</u> pressure going down in an uncontrolled manner, <u>THEN</u>:** a. Ensure the following valves closed: __ • All MSIVs __ • All MSIV bypass valves. __ b. **<u>IF</u>** any S/G depressurized **<u>OR</u>** pressure still going down in an uncontrolled manner, **<u>THEN</u>** isolate any faulted S/G(s) **<u>PER</u>** Enclosure 2 (Faulted S/G Isolation).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16. **Check reactor subcritical:** __ • P/R channels - LESS THAN 5% __ • W/R Neutron Flux - LESS THAN 5% __ • I/R SUR - NEGATIVE.	**Perform the following:** __ a. Continue to borate. __ b. **IF** boration is not available, **THEN** allow NC System to heat up. __ c. Perform actions of any other Critical Safety Function procedures that apply or are in effect that do not cool down NC System or add positive reactivity to the core. __ d. **RETURN TO** Step 5.
17. **Ensure adequate shutdown margin:** __ a. Obtain current NC boron concentration from Primary Chemistry. __ b. **WHEN** current NC boron concentration is obtained, **THEN** perform shutdown margin calculation **PER** OP/0/A/6100/006 (Reactivity Balance Calculation). c. **WHEN** following conditions satisfied, **THEN** NC System boration may be stopped: __ • Adequate shutdown margin is obtained __ • Uncontrolled cooldown has been stopped.	
__ 18. **REFER TO RP/0/A/5700/000 (Classification of Emergency).**	
__ 19. **RETURN TO procedure and step in effect.**	

END

7.15.1.5 Orange Path

IF any valid orange path is encountered, the operator is expected to scan all of the remaining trees, and then, if no valid red is encountered, promptly implement the corresponding EP. **IF** during the performance of an orange path procedure, any valid red condition or higher priority valid orange condition arises, the red or higher priority orange condition is to be addressed first, and the original orange path procedure suspended.

7.15.1.6 Completion of Red or Orange Path Procedure

Once procedure is entered due to a red or orange condition, that procedure should be performed to completion, unless preempted by some higher priority condition. It is expected that the actions in the procedure will clear the red or orange condition before all the operator actions are complete. However, these procedures should be performed to the point of the defined transition to a specific procedure or to the "procedure and step in effect" to ensure the condition remains clear. At this point any lower priority red or orange paths currently indicating or previously started but **NOT** completed shall be addressed.

FR-S.1, P.1 and Z.1 can be entered from either an orange or red path status. **IF** the color changes from orange to red while you are in one of these EPs, the crew should continue and complete the EP from where they are. Crew does **NOT** have to backup and restart the EP. **IF** the orange path is exited, and it subsequently turns red, the EP must be re-entered at Step 1.

Upon continuation of recovery actions in Optimal Recovery procedure, some judgment may be required by the operator to avoid inadvertent reinstatement of a Red or Orange condition by undoing some critical step in the Function Recovery procedure. The Optimal Recovery procedures are optimal assuming that safety equipment is available. The appearance of a Red or Orange condition in most cases implies that some equipment or function required for safety is **NOT** available, and by implication some adjustment may be required in the Optimal Recovery procedure.

7.10.5 Use of Enclosures

The decision on whether to read or hand-off an enclosure will be based on SRO judgment depending on the event. The following are some general guidelines to help the SRO make this decision.

- It is usually preferable for SRO to read enclosure if:
  - The crew must wait for the enclosure to be completed in order to continue in the EP/AP.
  - No more ROs are available to continue in the EP/AP, unless RO can perform enclosure concurrent with performing other steps.
  - There are no more time critical actions to be performed.

- It is usually preferable for SRO to hand-off the enclosure if:
  - It is critical for the crew to continue in the body of the procedure in a timely manner.
  - It is a valve checklist.
  - Actions are outside the horseshoe.

Additionally, an enclosure will be handed off if procedure specifies to hand-off the enclosure or if it is the foldout page.

7.11 Place Keeping Aids

EPs and APs contain a single line to the left and adjacent to the step number. The line is provided as a placekeeping aid. Check-off the place keeping line after step is completed. For a "check" step that requires no action, step can be checked after it is read. For steps that require action, step should be checked when action has been completed. For example, if a valve must be closed, place keeping line should be checked when operator states that valve is closed on second three way communication. For slow moving valves or situations where procedure reader must move on while waiting for step completion, circle the place keeping line until step is completed; check-off the place keeping line when performer later feeds back that step is completed. ONE EXCEPTION to this is performance of ES-1.3 (Transfer to Cold Leg Recirc). While performing multiple valve manipulations in ES-1.3, operator should proceed in EP in a timely manner and just check-off steps as they are read. This avoids excessive delays when performing this time critical evolution. (This exception is implied by ES-1.3 note that states that double three-way communication is **NOT** required.)

IF the step is a diagnostic step that requires transition to RNO, place right arrow (→) next to step in lieu of, or in addition to check mark.

Note that if you read an **IF/THEN** step that does **NOT** require performing its substeps ("**IF**"condition **NOT** met), do **NOT** check the substeps. The substeps will **NOT** be read or performed, and you only need to check steps if you READ them. Check next to **IF**/THEN step if it is all that is read, whether it has a place keeping line or not.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	055 EA2.03	
	Importance Rating		4.7

Ability to determine or interpret the following as they apply to a Station Blackout Actions necessary to restore power

Proposed Question: SRO 79

Given the following:

- A LOOP has occurred on Unit 1.
- Unit 2 is unaffected.
- The Unit 1 crew is performing ECA-0.0, Loss of All AC Power.
- The Standby Makeup Pump is ON.
- NCS subcooling is 8°F.
- Pressurizer level is 4% and lowering slowly.
- The crew was unable to start EITHER Diesel Generator.

Which ONE of the following describes the procedure that will be required for restoring power to Bus ETA, and the subsequent recovery procedure that will be performed upon transition from ECA-0.0?

A. AP/7, Loss of Electrical Power; ECA-0.1, Loss of All AC Power Recovery Without SI Required

B. Enclosure 9, Energizing Unit 1 4160 V Bus from Unit 2 – SATA or SATB; ECA-0.1, Loss of All AC Power Recovery Without SI Required

C. AP/7, Loss of Electrical Power; ECA-0.2, Loss of All AC Power Recovery With SI Required

D. Enclosure 9, Energizing Unit 1 4160 V Bus from Unit 2 – SATA or SATB; ECA-0.2, Loss of All AC Power Recovery With SI Required

Proposed Answer: D

Explanation (Optional):

A. Incorrect. AP/7 is plausible because it is the procedure normally used for any electrical restoration. In this condition, Enclosure 9 will be used. Even though Auto SI conditions do not exist, the crew will perform ECA-0.2 based on RCS subcooling and PZR level values requiring SI when power restored

B. Incorrect. Enclosure 9 is correct. Plausible because even though Auto SI conditions do not exist, the crew will perform ECA-0.2 based on RCS subcooling and PZR level values requiring SI when power restored

C. Incorrect. Incorrect restoration, but correct recovery procedure for these plant conditions

D. Correct

Technical Reference(s): ECA-0.0, Encl 9 Rev 24 (Attach if not previously provided)

EP-E0 Rev 24

EP-ECA0 Rev 12

EP-E0 Rev 12

OMP 4-3 p 22 Rev 26

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 2, 5

Comments:

KA is matched because the applicant must identify where the actions are contained for restoration of power.(title also identifies actions) and SRO level

because assessment of conditions and selection of procedures is required
(Requires knowledge of strategy)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
40. **Select recovery procedure as follows:**	
__ a. Check Standby Makeup pump - ON.	__ a. **IF** all NC pump seal cooling is lost, **THEN** notify station management that NC pump seal cooldown will occur as the entire NC system is cooled via natural circ cooldown in subsequent EPs.
__ b. Check NC subcooling based on core exit T/Cs - GREATER THAN 0°F.	b. Perform the following: __ 1) Align additional RN valves **PER** Enclosure 24 (RN S/I Valves). __ 2) **GO TO** EP/1/A/5000/ECA-0.2 (Loss Of All AC Power Recovery With S/I Required).
__ c. Check Pzr level - GREATER THAN 11% (29% ACC).	c. Perform the following: __ 1) Align additional RN valves **PER** Enclosure 24 (RN S/I Valves). __ 2) **GO TO** EP/1/A/5000/ECA-0.2 (Loss Of All AC Power Recovery With S/I Required).
d. Check the following valves - CLOSED: __ • 1NI-9A (NC Cold Leg Inj From NV) __ • 1NI-10B (NC Cold Leg Inj From NV).	d. **IF** any NV pump on, **THEN** perform the following: __ 1) Align additional RN valves **PER** Enclosure 24 (RN S/I Valves). __ 2) **GO TO** EP/1/A/5000/ECA-0.2 (Loss Of All AC Power Recovery With S/I Required).
__ e. **GO TO** EP/1/A/5000/ECA-0.1 (Loss Of All AC Power Recovery Without S/I Required).	

END

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
C. Operator Actions	
__ 1. **Monitor Foldout page.**	
2. **Check Reactor Trip:**	**Perform the following:**
__ • All rod bottom lights - LIT	__ a. Trip reactor.
__ • Reactor trip and bypass breakers - OPEN	b. **IF** reactor will not trip, **THEN**:
__ • I/R amps - GOING DOWN.	__ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees).
	__ • **GO TO** EP/1/A/5000/FR-S.1 (Response To Nuclear Power Generation/ATWS).
3. **Check Turbine Trip:**	**Perform the following:**
__ • All throttle valves - CLOSED.	__ a. Trip turbine.
	b. **IF** turbine will not trip, **THEN**:
	__ 1) Place turbine in manual.
	__ 2) Close governor valves in fast action.
	3) **IF** governor valves will not close, **THEN** close:
	__ • All MSIVs
	__ • All MSIV bypass valves.
__ 4. **Check 1ETA and 1ETB - ENERGIZED.**	**Perform the following:**
	__ a. **IF** both busses de-energized, **THEN GO TO** EP/1/A/5000/ECA-0.0 (Loss Of All AC Power).
	__ b. **WHEN** time allows, **THEN** try to restore power to de-energized bus **PER** AP/1/A/5500/07 (Loss of Electrical Power) while continuing with this procedure.

STEP 40 Select recovery procedure:

PURPOSE: To select the appropriate loss of all AC power recovery procedure.

BASIS: This step provides the criteria by which the operator determines which recovery procedure actions to implement. The criteria are:

1. The existence of NC subcooling
2. The existence of pressurizer level
3. The confirmation that S/I equipment is not operating (NI-9 and NI-10 closed)

Two recovery procedures are provided based on these criteria. These are procedures ECA-0.1 and ECA-0.2.

If the operator determines all criteria are satisfied, ECA-0.1 should be implemented to attempt plant recovery utilizing normal operational systems.

If any criterion is not satisfied, ECA-0.2 should be implemented to recover the plant utilizing safeguards systems.

To ensure S/I has not actuated upon power restoration, the positions of the cold leg injection isolation valves (NI-9 and NI-10) are checked. These valves do not "seal in" the S/I signal and do not receive a signal through the D/G load sequencer that is deenergized. If an S/I signal was generated prior to power restoration, the procedure would reset the signal after the time delay and no equipment would reposition (NI-9 and NI-10 would remain closed). If either valve were open at this point in the procedure, it would indicate that an S/I signal was generated with power restored and certain valves may have repositioned; specifically valves that receive direct S/I signals. In this case, ECA-0.2 would direct the operator to the correct procedure to handle the accident or to terminate the spurious S/I.

3.5. ECA-0.0 Enclosures

Enclosure 1, Unit 1(2) SSF Actions – ECA-0.0 Actions

This enclosure provides actions to be taken upon manning the SSF. These actions, if necessary, include starting the SSF D/G, loading equipment on the bus (standby makeup pump, battery chargers) and monitoring D/G operation.

Enclosure 2, Unit 1(2) EMXA-4 ECA-0.0 Actions

This one step enclosure provides the instructions necessary to transfer EMXA-4 to the SSF. A caution provides guidance for operating Kirk-key interlocked breakers. A note provides the fastest pathway from the Control Room to ETA room.

STEP 2 & 3 Check Reactor and Turbine Trip: (IMMEDIATE ACTIONS)

PURPOSE: To ensure the reactor and turbine are tripped.

BASIS: Reactor trip must be checked to ensure the only heat being added to the NC system is from decay heat and NC pump heat. The safeguards systems protecting the plant during accidents are designed assuming only decay heat and pump heat are being added to the NC.

If the reactor is not tripped, the RNO directs us to trip it manually. If the reactor cannot be tripped F-0, CSF Status Trees, is implemented and a transition is made to FR-S.1, Response to Nuclear Power Generation/ATWS, to deal with the ATWS conditions.

The turbine is tripped to prevent an uncontrolled cooldown of the NC due to steam flow that the turbine would require.

If the turbine is not tripped, the RNO directs us to trip it manually. If the turbine will not trip, steam is isolated to it by first attempting to close the turbine governor valves. If the turbine will not runback, steam is isolated to it by closing the MSIVs and bypass valves.

STEP 4 Check 1ETA and 1ETB - ENERGIZED. (IMMEDIATE ACTION)

PURPOSE: To ensure electrical power to at least one emergency bus.

BASIS: AC power must be checked from either offsite sources or the diesel generators to ensure adequate power sources to operate safeguards equipment. At least one train of safeguards equipment is required to deal with emergency conditions.

If both AC emergency busses are deenergized, the RNO directs a transition to ECA-0.0, Loss of All AC Power.

7.18 Multiple Use of EPs and APs.

The Control Room SRO will determine how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress. More than one EP shall **NOT** be run concurrently unless directed by the procedure. Generally the use of APs in conjunction with EPs should be avoided. In some instances it would be proper to use an AP concurrently during a major accident which is being addressed by the EPs. An example of this is upon loss of all Nuclear Service Water in the middle of an accident, the operators would need to utilize the AP for Loss of RN also. **IF** an AP is used during an S/I event, USE CAUTION. APs are generally written assuming an S/I has **NOT** occurred (exception - AP/35, ECCS Actuation During Plant Shutdown). Evaluate any AP steps in post S/I events to ensure the steps do **NOT** conflict with any EP in effect. **NOT** all AP actions would be appropriate if an S/I occurred. (Enclosures in EP/G-1 (Generic Enclosures) may be used when reference by EPs or APs.)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	055 EA2.05	
	Importance Rating		3.7

Ability to determine or interpret the following as they apply to a Station Blackout: When battery is approaching fully discharged

Proposed Question: SRO 79

EA2.0

Fix or replace

Given the following:

- A Blackout has occurred on Unit 1.
- Unit 2 is unaffected.
- The Unit 1 crew is performing ECA-0.0, Loss of All AC Power.

Which ONE of the following describes the technical specification design basis for the operability of Battery EVCA, and the action required to extend the life of Battery EVCA during the blackout?

The battery has adequate storage capacity to supply the duty cycle output for…

A. 1 hour; evaluate shutting down associated inverter and aligning vital AC Panelboards to KRP.

B. 1 hour; evaluate removing the OAC from service.

C. 4 hours; evaluate shutting down associated inverter and aligning vital AC Panelboards to KRP

D. 4 hours; evaluate removing the OAC from service.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. OAC is supplied from Aux Control Power (DCA/DCB) Plausible because it is an action performed if power cannot be restored

C. Incorrect. Incorrect time, though standard time for design basis battery life, And also time for TS LCO action

D. Incorrect. Incorrect time, though standard time for design basis battery life, And also time for TS LCO action. OAC is supplied from Aux Control Power (DCA/DCB) Plausible because it is an action performed if power cannot be restored

Technical Reference(s): ECA-0.0, AP/7 Enclosure 7 (Attach if not previously provided)

TS Basis 3.8.4

Proposed references to be provided to applicants during examination: None

Learning Objective: ____ (As available)

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____

55.43 2, 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	056 AA2.18	
	Importance Rating		4.0

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Reactor coolant temperature, pressure, and PZR level recorders

Proposed Question: SRO 80

Given the following:

- A loss of off-site power has occurred.
- Both Units have tripped.
- Unit 1 SRO has been directed to initiate cooldown to Mode 5.
- The following conditions exist on Unit 1 upon transition to ES-0.1, Reactor Trip Response.
    - All control rods are inserted.
    - NC SYSTEM Tcold temperature.
        - Loop 1A 535°F
        - Loop 1B 532°F
        - Loop 1C 533°F
        - Loop 1D 533°F

Which ONE of the following choices describes (1) actions that will be required for the above conditions, and (2) the procedure required for NC System Cooldown?

A. (1) Close MSIVs ONLY;
(2) OP/1/A/6100/002, Controlling Procedure for Unit Shutdown.

B. (1) Close MSIVs ONLY;
(2) ES-0.2, Natural Circulation Cooldown.

C. (1) Close MSIVs AND Initiate Emergency Boration in accordance with AP/38, Emergency Boration;
(2) OP/1/A/6100/002, Controlling Procedure for Unit Shutdown.

D. (1) Close MSIVs AND Initiate Emergency Boration in accordance with AP/38, Emergency Boration;
(2) ES-0.2, Natural Circulation Cooldown.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. MSIVs are closed, but if a cooldown is required with a LOOP, then ES-0.2 would be performed instead of the Controlling Procedure. Also, due to Loop 1D temperature, emergency boration is required
B. Incorrect. Due to Loop 1D temperature, emergency boration is required
C. Incorrect. Actions are correct but procedure is incorrect as in A
D. Correct.

Technical Reference(s): ES-0.1, Rev 27; ES-0.2 Rev 10 (Attach if not previously provided)
EP-E0 Rev 12

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:

KA is met because item evaluates interpretation of RCS temperature trends. SRO level because the assessment requires interpretation of indications to take action within selected EOPs/AOPs

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Check NC temperatures:** __ • **IF** any NC pump on, **THEN** check NC T-Avg - STABLE OR TRENDING TO 557°F. OR __ • **IF** all NC pumps off, **THEN** check NC T-Colds - STABLE OR TRENDING TO 557°F.	**Perform the following based on plant conditions:** a. **IF** temperature less than 557°F **AND** going down, **THEN** perform the following: __ 1) Ensure all steam dump valves closed. 2) **IF** MSR "RESET" light is dark, **THEN** perform the following: __ a) Depress "SYSTEM MANUAL". __ b) Depress "RESET". __ 3) Ensure all SM PORVs closed. 4) **IF** any SM PORV can not be closed, **THEN** perform the following: __ a) Close SM PORV isolation valve. __ b) **IF** SM PORV isolation valve can not be closed, **THEN** dispatch operator to close SM PORV isolation valve. __ 5) Ensure S/G blowdown is isolated. 6) **IF** cooldown continues, **THEN** control feed flow as follows: a) **IF** S/G N/R level is less than 11% in all S/Gs, **THEN** throttle feed flow to achieve the following: __ • Minimize cooldown __ • Maintain total feed flow greater than 450 GPM. b) **WHEN** N/R level is greater than 11% in at least one S/G, **THEN** throttle feed flow further to: __ • Minimize cooldown __ • Maintain at least one S/G N/R level greater than 11%. (RNO continued on next page)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5. (Continued)

7) **IF** cooldown continues, **THEN** perform the following:

__ a) Close all MSIVs.

__ b) Close all MSIV bypass valves.

__ c) Close 1AS-12 (U1 SM To AS Hdr Control Inlet Isol).

d) **IF** the MSIVs will not close, **THEN** perform the following:

__ (1) Initiate Main Steam Isolation signal.

(2) **IF** all S/G pressures are above 775 PSIG, **THEN** reset the following to allow automatic SM PORV operation:

__ 1. Main Steamline Isolation.

__ 2. SM PORVs.

__ 8) **IF** cooldown continues **AND** faulted S/G exists, **THEN** stop feeding faulted S/G.

9) **IF** cooldown continues, **THEN** select "CLOSE" on the following switches:

__ • 1SM-83 (A SM Line Drain Isol)

__ • 1SM-89 (B SM Line Drain Isol)

__ • 1SM-95 (C SM Line Drain Isol)

__ • 1SM-101 (D SM Line Drain Isol).

(RNO continued on next page)

STEP 5 Check NC temperatures:

PURPOSE: To ensure that NC heat is being properly removed through the secondary side.

BASIS: NC average temperature stable or trending to the no-load value of 557°F with any NC pump running indicates that the secondary steam dump system is operating as designed. If temperature is stable, even if not at 557°F, you can continue in the left-hand column.

If no NC pump is running, then the NC average temperature will be higher than the no-load value as natural circulation conditions are established. However, if the steam dump system is working properly, the cold leg temperatures will stabilize at the no-load value.

If the cooldown is excessive, it can be controlled by:

1. Stopping all steam from being dumped,
2. Controlling feed flow, or
3. Closing the MSIVs.

Steam dump should be stopped by assuring that steam dump valves are closed, S/G PORVs are closed, and SM Line drains are closed.

Excessive feed to the S/Gs can also result in cooling down the NC and it may be necessary to reduce feed flow to the minimum for decay heat removal until S/G level is in the narrow range.

If the cooldown continues, the main steamlines are isolated to stop any steam leakage downstream of the MSIV's, such as a stuck open condenser steam dump valve. Also, AS-12 is isolated to ensure flow to the Aux Steam system is secured.

If NC temperature is greater than no-load and going up, then steam dump from the secondary must be raised for decay heat removal.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **Check feedwater status:**	**Establish total feed flow to S/Gs greater than 450 GPM or maintain at least one S/G N/R level greater than 11% using one of the following:**
__ • Check any CA pump - ON.	
__ • Check total feed flow to S/Gs - GREATER THAN 450 GPM.	__ • Start CA pumps. OR __ • Use main feedwater **PER** Enclosure 4 (Reestablishing CF Flow).
12. **Check if shutdown margin adequate:**	
__ a. All control rods - FULLY INSERTED.	a. Perform the following: __ 1) **IF** all rod position indication is lost, **OR** greater than 5 rods not fully inserted, **THEN** emergency borate total of 13,200 gallons of 7000 PPM boron solution **PER** AP/1/A/5500/38 (Emergency Boration). __ 2) **IF** 2 to 5 rods not fully inserted, **THEN** emergency borate 2100 gallons of 7000 PPM boron solution for each rod not fully inserted **PER** AP/1/A/5500/38 (Emergency Boration).
__ b. Stop any boron dilutions in progress.	
__ c. Check all NC T-Colds - GREATER THAN 534°F.	c. Borate as follows: __ 1) Set boric acid flow control pot at 6.5. __ 2) Initiate emergency boration **PER** AP/1/A/5500/38 (Emergency Boration). __ 3) **WHEN** all NC T-Colds are above 534°F, **THEN** emergency boration may be secured. __ 4) **GO TO** Step 13.
__ d. **IF AT ANY TIME** any NC T-Cold goes below 534°F, **THEN** perform Step 12.c.	

STEP 10 Check NC T-Ave - GREATER THAN 553 ºF

STEP 11 Check feedwater status:

PURPOSE: To ensure the proper feedwater alignment following a reactor trip.

BASIS: T-Ave is not expected to fall to the feedwater isolation setpoint of 553°F, so feedwater isolation should not have occurred. If T-Ave is less than 553°F, then by checking the status lights lit, all feedwater isolation valves can be assured closed for a S/G as required.

Establishing minimum feed flow to the steam generators or minimum S/G levels ensures a secondary heat sink for decay heat removal. The feedwater source may be from either the CA pumps or main feedwater on the bypass lines.

STEP 12 Check if shutdown margin adequate: (Continuous Action Step)

PURPOSE: To ensure that the shutdown margin is adequate.

BASIS: A subcritical core is confirmed if all rods are at the bottom according to the rod bottom lights and the rod position indicators. If these indications reveal that one rod is not inserted, no immediate action is required since the core is designed for adequate shutdown margin with one rod stuck out.

Any boron dilutions in progress should be stopped to ensure shutdown margin is not challenged.

If more than one rod fails to insert fully, the shutdown reactivity margin must be made up through emergency boration to account for the reactivity worth of the stuck rods. If two to five rods do not fully insert, then emergency borate 2100 gallons of 7000 ppm boron solution. If all rod position indication is lost or more than five rods are not fully inserted, emergency borate 13,200 gallons of 7000 ppm boron solution.

Also, if NC T-Colds are less than the cycle specific value at which SDM is calculated to be challenged (typically near 534°F), the loss in shutdown margin due to low NC temperatures must be made up by emergency boration per AP/38 until all NC T-colds are above the specified temperature.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
43. **Check if Reactor Trip was performed as part of normal shutdown as follows:**	
__ a. Check if OP/1/A/6100/003 (Controlling Procedure For Unit Operation), Enclosure 4.10 (Shutdown Via Reactor Trip) - IN EFFECT PRIOR TO TRIP.	__ a. **GO TO** Step 44.
__ b. Check any NC pump - ON.	__ b. **GO TO** Step 44.
__ c. **RETURN TO** step in effect in OP/1/A/6100/003 (Controlling Procedure For Unit Operation), Enclosure 4.10 (Shutdown Via Reactor Trip).	
__ 44. **REFER TO OP/1/A/6100/003 (Controlling Procedure For Unit Operation), Enclosure 4.10 (Shutdown Via Reactor Trip) and perform applicable steps.**	
45. **Determine if Natural Circulation cooldown is required:**	
__ a. Check if plant cooldown - REQUIRED.	__ a. **GO TO** OP/1/A/6100/003 (Controlling Procedure For Unit Operation), Enclosure 4.1 (Power Increase).
__ b. Check if all NC pumps - OFF.	__ b. **GO TO** OP/1/A/6100/002 (Controlling Procedure For Unit Shutdown).
__ c. **GO TO** EP/1/A/5000/ES-0.2 (Natural Circulation Cooldown).	

END

STEPS 22 – 43 These steps align systems for shutdown conditions.

PURPOSE: To stop equipment not needed following a reactor trip.

BASIS: Since the plant may have been operating at full power prior to the trip, certain equipment may be in operation and not needed at this time (e.g., two condensate pumps, circulating water pumps, etc.).

STEP 44 Determine if Natural Circulation cooldown is required

PURPOSE: To determine if a cooldown must be done on natural circulation.

BASIS: If the plant staff determines that a cooldown is required, then a normal cooldown should be performed if one or more NC pumps are operating. However, if no NC pumps are operating, then a natural circulation cooldown will be necessary.

If a natural circulation cooldown is required, then a transition to ES-0.2, Natural Circulation Cooldown, is made.

5.6. Final Plant Status

ES-0.1 provides the specific actions necessary to stabilize and control the plant following a reactor trip.

ES-0.1 is also used following a reactor trip combined with either a loss of offsite power or a total loss of forced NC flow.

The following table summarizes the exit guidance from ES-0.1. The left column lists each step that provides a potential exit point from ES-0.1. The right column lists the transition procedure(s). If an exit transition is necessary, the operator should transition to Step 1 of the appropriate procedure.

ES-0.1 STEP NUMBER	TRANSITION PROCEDURE(S)
Step 14	**E-0, Reactor Trip or Safety Injection**
Step 42	**OP/1/A/6100/003, Controlling Procedure for Unit Operation, Enclosure for "Shutdown Via Reactor Trip", if Reactor Trip was performed as part of a normal shutdown.**
Step 44	**OP/1/A/6100/003, Controlling Procedure for Unit Operation, if NO plant cooldown is required.**
Step 44	**ES-0.2, Natural Circulation Cooldown, if no NC pumps running.**
Step 44	**OP1/A/6100/002, Controlling Procedure for Unit Shutdown, if any NC pump is running.**

5.7. Summary/Objective Review

The objective of the recovery/restoration technique incorporated into procedure ES-0.1 is to stabilize and control the plant following a reactor trip without safety injection in operation.

The recovery/restoration technique of ES-0.1 includes the following five major action categories.

1. Ensure the primary system stabilizes at no-load conditions.
2. Ensure the secondary system stabilizes at no-load conditions.
3. Ensure necessary APs that should be run concurrently have been addressed.
4. Maintain/establish forced circulation of the NC.
5. Maintain stable plant conditions.

A. Purpose

This procedure provides actions to perform a Natural Circulation NC System cooldown and depressurization to Cold Shutdown, with no accident in progress, under requirements that will preclude any upper head void formation.

B. Symptoms or Entry Conditions

This procedure is entered from:

- EP/1/A/5000/ES-0.1 (Reactor Trip Response), Step 44, when it has been determined that a Natural Circulation cooldown is required.
- EP/1/A/5000/ES-1.1 (Safety Injection Termination), Step 31, after the plant conditions have been stabilized and no NC pumps can be started.
- EP/1/A/5000/ECA-0.1 (Loss Of All AC Power Recovery Without S/I Required), Step 29, after the plant conditions have been stabilized following the restoration of AC emergency power.

6.0 ES-0.2, NATURAL CIRCULATION COOLDOWN

6.1. Purpose

ES-0.2 provides actions to perform a natural circulation NC system cooldown and depressurization to cold shutdown, *with no accident in progress*, under requirements that will preclude any upper head void formation.

6.2. Symptoms/Conditions

Upon entry to ES-0.2, natural circulation of the NC has been established and stable plant conditions are being maintained. ES-0.2 is then entered from:

1. ES-0.1, Reactor Trip Response, when it has been determined that a natural circulation cooldown is required.
2. ES-1.1, Safety Injection Termination, after the plant conditions have been stabilized and no NC pumps can be started.
3. ECA-0.1, Loss Of All AC Power Recovery Without S/I Required, after the plant conditions have been stabilized following the restoration of AC emergency power.

There are three possible transitions out of this procedure.

1. If S/I actuation occurs, a transition to E-0, Reactor Trip or Safety Injection, should be made.
2. Since it is always desirable to have forced convection heat transfer from the core, the first step of the procedure attempts to start a NC pump. If this attempt is successful, a transition to the appropriate plant procedure is in order.
3. The third transition occurs if the plant staff determines that a natural circulation cooldown and depressurization must be performed at a rate that may form a steam void in the vessel. At that time a transition should be made to ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058 G2.2.37	
	Importance Rating		4.6

Loss of DC pwr

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

Proposed Question: SRO 81

Given the following:

- Unit 1 is at 100% power.
- A loss of Charger EVDA occurred.
- Battery EVDA voltage lowered to 109 VDC prior to restoration of a Charger to the battery.
- Battery EVDA voltage is currently 129 VDC.
- Specific gravity is 1.180 for two (2) connected cells.
- Average specific gravity is 1.202 for all connected cells.
- Electrolyte temperature is 76°F.

Which ONE of the following describes the operability status of Battery EVDA, and the TS basis for operability of the DC electrical power subsystem?

REFERENCE PROVIDED

A. The battery is considered operable but degraded; operability ensures that at least ONE DC train is available assuming a loss of off-site OR on-site power coincident with a worst case single failure.

B. The battery is considered inoperable; operability ensures that at least ONE DC train is available assuming a loss of off-site OR on-site power coincident with a worst case single failure.

C. The battery is considered operable but degraded; operability ensures that at least ONE DC channel is available assuming a loss of off-site AND on-site power.

D. The battery is considered inoperable; operability ensures that at least ONE DC channel is available assuming a loss of off-site AND on-site power.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Operable but degraded would be related to Category A or B parameter out of limits. In this case, the applicant must determine that specific gravity is out of limit for category C, making the battery inoperable

B. Correct.

C. Incorrect. See A. Also, basis plausible because it is similar to actual basis, except that Loss of off-site AND on-site power is NOT design basis for battery

D. Incorrect. Operability is correct, but basis incorrect and plausible because it is similar to actual basis, except that Loss of off-site AND on-site power is NOT design basis for battery

Technical Reference(s): TS 3.8.6 and basis; EL-EPL Rev 22 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EL-EPL # 3 (As available)

Question Source:
- Bank # ____
- Modified Bank # ____ (Note changes or attach parent)
- New X

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 2

Comments:

KA matched because the applicant must determine operability of selected equipment related to selected APE. (Loss of DC) SRO level because a determination of operability, and basis for operability, are the required knowledge items for this test item

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the channels of DC batteries shall be within the limits of Table 3.8.6-1.

APPLICABILITY: When associated channels of DC sources are required to be OPERABLE.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Category A or B limits.	A.1 Verify pilot cells electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	<u>AND</u>	
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours <u>AND</u> Once per 7 days thereafter
	<u>AND</u>	
	A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 60°F. OR One or more batteries with one or more battery cell parameters not within Category C values.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.6.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days AND Once within 7 days after a battery discharge < 110 V AND Once within 7 days after a battery overcharge > 150 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 60°F.	92 days

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	> Minimum level indication mark, and ≤ ¼ inch above maximum level indication mark[(a)]	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity[(b)(c)]	≥ 1.200	≥ 1.195 AND Average of all connected cells > 1.205	Not more than 0.020 below average of all connected cells or ≥ 1.195 AND Average of all connected cells ≥ 1.195

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND	This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources—Operating," and LCO 3.8.5, "DC Sources—Shutdown."
APPLICABLE SAFETY ANALYSES	The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation. The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one train of DC sources OPERABLE during accident conditions, in the event of: a. An assumed loss of all offsite AC power or all onsite AC power; and b. A worst case single failure. Battery cell parameters satisfy the Criterion 3 of 10 CFR 50.36 (Ref. 3).
LCO	Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.
APPLICABILITY	The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery electrolyte is only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

BASES

ACTIONS

A.1, A.2, and A.3

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met, Category B limits not met, or Category A and B limits not met) but within the Category C limits specified in Table 3.8.6-1 in the accompanying LCO, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met and operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to inspect the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery is still capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A or B limits. This periodic verification is consistent with the normal Frequency of pilot cell Surveillances.

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not

BASES

ACTIONS (continued)

completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 4), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.

SR 3.8.6.2

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 4). In addition, within 7 days of a battery discharge < 110 V or a battery overcharge > 150 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided the battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 4), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is ≥ 60°F, is consistent with a recommendation of IEEE-450 (Ref. 4), that states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

The term "representative cells" replaces the fixed number of "six connected cells", consistent with the recommendations of IEEE-450 (Ref. 4) to provide a general guidance to the number of cells adequate to

BASES

SURVEILLANCE REQUIREMENTS (continued)

monitor the temperature of the battery cells as an indicator of satisfactory performance. For some cases, the number of cells may be less than six, in other conditions, the number may be more.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 4), with the extra 1/4 inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote a to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 4) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendations of IEEE-450 (Ref. 4), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.200 (0.015 below the manufacturer fully charged nominal specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 4), the specific gravity readings are based on a temperature of 77°F (25°C).

BASES

SURVEILLANCE REQUIREMENTS (continued)

The specific gravity readings are corrected for actual electrolyte temperature and level. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.195 (0.020 below the manufacturer fully charged, nominal specific gravity) with the average of all connected cells > 1.205 (0.010 below the manufacturer fully charged, nominal specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists, and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 4), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit of average specific gravity ≥ 1.195 is based on manufacturer recommendations (0.020 below the manufacturer recommended fully charged, nominal specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The footnotes to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 4). Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days, each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

The value of 2 amps used in footnote (b) and (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition.

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
4. IEEE-450-1980.

Objective # 12

Each battery is sized to supply the continuous emergency loads and momentary loads fed from its distribution center **(two DC buses which includes the two inverters and their panelboards)**, plus supply the loads of its sister distribution center **(two DC buses which includes the two inverters and their panelboards)**, if required, for a period of one hour. The basis for selecting a one-hour capacity is a conservative time estimate for the restoration of power to the battery chargers under the most adverse credible conditions. This one-hour duty cycle capacity was assumed during the plant's safety analysis (documented in the UFSAR) and is verified every 18 months during a battery service test.

The minimum design ambient temperature in the battery room is 60 °F; hence the battery is sized based on its capacity at 60° F since the battery capacity would be greater at a higher temperature.

Since each battery is, electrically, in parallel with its battery charger, and the battery charger output voltage is slightly higher than the battery voltage, during the "floating charge"; the battery charger actually supplies power to the respective DC loads during normal operation. However, the battery will automatically assume those DC loads, without interruption, upon loss of its respective battery charger or AC power source.

Battery bus voltage is indicated by voltmeters located on the 125 VDC vital control distribution centers. The battery bus voltage is also monitored by under-voltage relays, which alarm, on Annunciator Alarm Panel 1AD-11 (Electrical), when the battery bus voltage reaches 127 volts (at this voltage the battery is still capable of performing its intended safety function).

2.3 125 VDC Vital Instrumentation and Control Power System Distribution Centers

Each of the four distribution centers (EVDA, EVDB, EVDC, and EVDD) receive power from a battery and/or a battery charger, and supplies power to two of the eight 125 VDC power panelboards (1EVDA, 1EVDB, 1EVDC, 1EVDD, 2EVDA, 2EVDB, 2EVDC, and 2EVDD), and two of the eight static inverters (1EVIA, 1EVIB, 1EVIC, 1EVID, 2EVIA, 2EVIB, 2EVIC, and 2EVID).

Objective # 13

Either of the two same train-related buses (EVDA and EVDC / Train "A" buses or EVDB and EVDD / Train "B" buses) can be tied together through their respective bus tie breakers. This will allow two distribution centers to be fed from one battery / battery charger combination.

This system is shared between the two units (Unit 1 and 2) and provides four normally independent power channels for reactor control and instrumentation. Three of the four channels will ensure that the overall system functional capability is maintained, comparable to the original design standards for safe operation. However, a loss of any two of these channel sources will result in a reactor trip or forced reactor shutdown (Technical Specifications) of both units (Unit 1 and 2).

1.0 INTRODUCTION

1.1. Purpose

Objective # 1

The 125 VDC and 120 VAC Vital Instrumentation and Control Power System provides a reliable source of continuous power for the safety related controls and instrumentation required for plant start up, normal operation, and an orderly shutdown of each unit.

1.2. General Description

125 VDC Vital Instrumentation and Control Power System

Objective # 3

The 125 VDC Vital Instrumentation and Control Power System consists of five chargers, four 125 VDC batteries, four distribution centers (with associated breakers), and eight separate panelboards. The system is designed to support a manual connection of two distribution centers (either EVDA and EVDC or EVDB and EVDD) during periods of battery maintenance.

The DC System is divided into four independent and physically separated load groups. With each load group comprised of the following: one battery, one battery charger, one DC distribution center, and two DC power panelboards.

This system is shared between the two units (Unit 1 and 2) and provides four normally independent power channels for reactor control and instrumentation. Three of the four channels will ensure that the overall system functional capability is maintained, comparable to the original design standards for safe operation. However, a loss of any two of these channel sources will result in a shutdown of both units (Unit 1 and 2).

Objective # 4

The following is a listing of typical loads that are powered from the 125 VDC Vital Instrumentation and Control Power System Distribution Centers (EVDA, EVDB, EVDC, and EVDD):

- Auxiliary Safeguards Cabinets Control Power
- Turbine Trip
- ETA and ETB Control Power
- Diesel Generator Sequencers Control Power
- Miscellaneous NV System Solenoids
- Pressurizer PORV Solenoids
- Reactor Trip Switchgear Control Power
- 600 V Load Centers ELXA, ELXB, ELXC, and ELXD Control Power
- Power supplies to the Reactor Vessel Head Vents
- Ventilation Units Shunt Trip Coils
- NCP UF-UV Monitor Panels

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
3.0	3.0	2.0	2.0	2.0

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the 125 VDC and 120 VAC Vital Instrumentation and Control Power Systems.	X	X	X	X	
2	Draw a simplified composite of the 125 VDC and 120 VAC Vital Instrumentation and Control Power Systems as provided in Training Drawing 7.2, Simplified 125 VDC and 120 VAC Vital Instrumentation and Control Power Drawing.	X	X	X	X	
3	Provide a general description of the 125 VDC Vital Instrumentation and Control Power System.	X	X	X	X	
4	List the typical loads powered from the 125 VDC Vital Instrumentation and Control Power System Distribution Centers.	X	X	X	X	
5	Provide a general description of the 120 VAC Vital Instrumentation and Control Power System.	X	X	X	X	
6	List the typical loads powered from the 120 VAC Vital Instrumentation and Control Power System Power Panelboards.	X	X	X	X	
7	Describe the basis for the sizing (loading) of the battery charger associated with the 125 VDC Vital Instrumentation and Control Power System.	X	X	X	X	
8	Discuss the normal loading demands associated with the 125 VDC Battery Chargers for the Vital Instrumentation and Control Power System.	X	X	X	X	
9	Describe any of the Kirk-Key Interlocks associated with the 125 VDC Vital Instrumentation and Control Power System and state the purpose of the Kirk-Key arrangement.	X	X	X	X	X
10	Explain how the Standby Battery Charger is used during an equalizing charge of a 125 VDC Battery for the Vital Instrumentation and Control Power System.	X	X	X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	003 AA2.02	
	Importance Rating		2.8

Ability to determine and interpret the following as they apply to the Dropped Control Rod: Signal inputs to rod control system

Proposed Question: SRO 82

Given the following Unit 1 initial conditions:

- Reactor power is at 40%
- Power range NIS indicate:
  - 40% (N41), 41% (N42), 41% (N43), 41% (N44)
- Tave for each loop indicates:
  - 567°F ('A'), 567°F ('B'), 568°F ('C'), 568°F ('D')
- Turbine power is at 481 MWe
- Rod control is in automatic
- Group demand counters and DRPI indicate Control Bank 'D' at 140 steps.

Control Bank 'D' Rod L-12 drops fully into the core and the following conditions now exist:

- Power range NIS indicate:
  - 40% (N41), 40% (N42), 42% (N43), 38% (N44)
- Tave for each loop indicates:
  - 564°F ('A'), 564°F ('B'), 563°F ('C'), 564°F ('D')
- Turbine power is 478 MWe

Assuming NO operator action, which ONE of the following describes the effect on the rod control system, and the technical specification action required?

A. Rods withdraw due to the Tave-Tref mismatch. Verify Shutdown Margin requirements are met or initiate boration to ensure Shutdown Margin is met, to ensure accident analysis assumptions remain valid.

B. Rods withdraw due to the Power Range NIS Mismatch Rate signal. Verify Shutdown Margin requirements are met or initiate boration to ensure Shutdown Margin is met, to ensure accident analysis assumptions remain valid.

C. Rods withdraw due to Power Range NIS Mismatch Rate signal. Verify AFD requirements are met to ensure that fuel design limits and hot channel factors are maintained within limits.

D. Rods withdraw due to the Tave –Tref mismatch. Verify AFD requirements are met to ensure that fuel design limits and hot channel factors are maintained within limits.

Proposed Answer: A

Explanation (Optional):

A. Correct. Tave deviation is higher than 1.5 degrees F and rods will withdraw. TS action is correct.

B. Incorrect. Power mismatch is not high enough to overcome the Tave mismatch, and power mismatch is based on rate of change with turbine power, which is minimal

C. Incorrect. Incorrect bases and also incorrect reason for rod withdrawal. Plausible because power mismatch is an input and AFD would be a concern above 50% power

D. Incorrect. Incorrect basis but AFD would be a concern at higher power, as well as action required (>50%)

Technical Reference(s)	OP-MC-IC-IRX, Rev 23	(Attach if not previously provided)
	AP/14 Rev 10	
	AP-14 Basis Document Rev 6	
	TS 3.1.4	

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-MC-IRX-Obj 5 (As available)

Question Source:	Bank #	X	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam 2002 McGuire

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

Stem not modified but distractors all different from original

KA met because inputs to rod control are the evaluated parameters.

SRO level because the effect of the failure has implications in TS basis that the applicant must determine

1.0 INTRODUCTION

1.1. Purpose

Objective #1

The Reactor Control System (IRX) allows the reactor to follow load changes automatically between 15 to 100% power without a reactor trip, steam dump actuation, or pressure relief with the following load changes:

- Step load increase or decrease of 10%
- Ramp increase or decrease of 5% per minute

1.2. General Description

The system matches reactor power to turbine load by controlling reactor coolant temperature (T_{avg}). Reference temperature (T_{ref}) is calculated as a function of turbine load from turbine impulse pressure. As turbine load changes, T_{ref} changes. When coolant temperature (T_{avg}) differs from T_{ref}, an error signal is produced.

The rate of change of the difference between reactor power and turbine power (power mismatch) is produced to provide an anticipatory signal. The power mismatch signal can generate rod movement prior to a T_{avg}/T_{ref} mismatch.

The two error signals, temperature mismatch and power mismatch are summed to yield a rod speed and direction demand signal (combined error) which is sent to the Rod Control System.

The Reactor Control System is not safety related.

2.0 COMPONENT DESCRIPTION

2.1. Loop Average Temperature (T_{avg})

T_{avg} for each of the four loops is derived from narrow range (NR) hot and cold leg Resistance Temperature Detectors (RTD's).

$$T_{avg} = \frac{T_h + T_c}{2}$$

T_h is derived by averaging the loops three hot leg RTD's. T_{avg} is used in calculating the OPΔT and OTΔT setpoints and in the Feedwater Isolation circuit (P-4 and Lo-T_{avg}). Each loop T_{avg} is indicated on the control board (530 - 630 °F).

Isolation amplifiers are used to isolate protection circuits from control circuit faults.

The T_{ref} signal is sent to the Steam Dump Control System to determine the output of the Load Rejection Controller, the T_{avg}/T_{ref} recorder on Control Board and to the Plant computer.

T_{ref} filter provides transient suppression prior to comparing with T_{avg}.

2.5. Temperature Mismatch Signal

Objective #5, 12

Auctioneered high T_{avg} is compared to T_{ref} and a temperature mismatch signal is developed. The summer output signal is then sent to the lower scale of Control Board bargraph indicator ± 15 °F.

If $T_{avg} > T_{ref}$ a positive temperature mismatch exists and rod insertion may be required. If $T_{avg} < T_{ref}$ a negative temperature mismatch exists and rod withdrawal may be required.

2.6. Power Mismatch Signal

Objective #6, 12

The auctioneered high reactor power circuit selects the highest of all power range instruments for the output signal. The auctioneered High Nuclear Power is compared to the Turbine Power (impulse pressure) in order to anticipate changes in T_{avg}.

If reactor power and turbine power are changing at different rates, there will be an output error signal. A difference between reactor power and turbine load will not generate a mismatch signal if neither signal is changing. Reactor power could be 60% and turbine load 40% steady state and there would not be a mismatch.

Any Nuclear Power Channel removed from service should be defeated using the Power Mismatch Bypass switches.

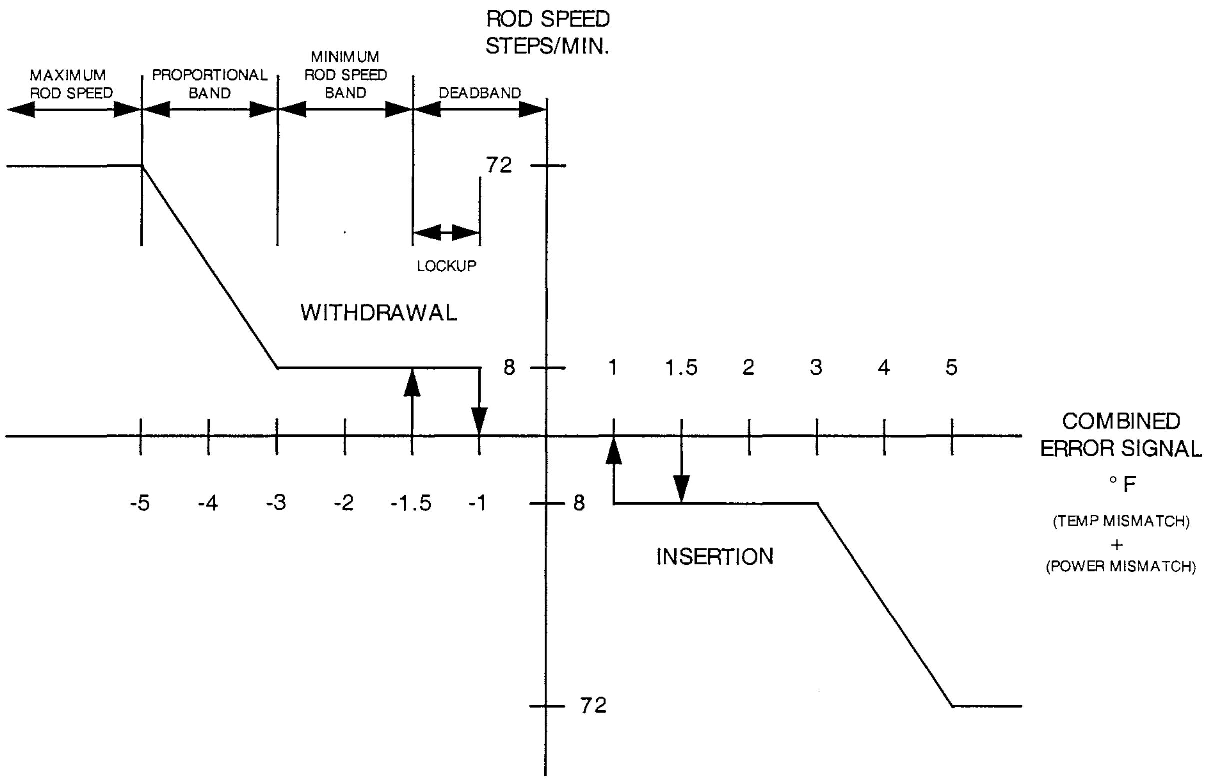
2.6.1. Derivative Circuit

This provides an output which is proportional to the rate of change of the difference in nuclear power and turbine power. The derivative is really a rate-lag unit (rate comparator). If the rate of change between nuclear power and turbine power is zero, the long term output of the derivative will be zero, even if nuclear power does not equal turbine power. When a rate of change occurs, an output results. When the rate of change returns to zero, the output will decay to zero, but it will take several minutes.

NOTE: If input error signal is not changing, derivative circuit output would be zero.

Power mismatch signal causes improved response (quicker) of output signal resulting in faster reaction of rod movement. It dominates initially on changing power mismatch signals. Temperature mismatch signal dominates during any slow load increases/or decreases.

During a rapid power mismatch transient, the temperature mismatch signal will eventually become the main or dominant rod movement signal after power mismatch change has subsided.



Polarity of the Combined Error Signal determines rod direction movement.

If the signal is positive rods step in.

- $T_{avg} > T_{ref}$
- Nuclear power increasing at a faster rate than turbine power.
- Turbine power decreasing at a faster rate than nuclear power.

If the signal is negative rods step out.

- $T_{avg} < T_{ref}$
- Nuclear power decreasing at a faster rate than turbine power.
- Turbine power increasing at a faster rate than nuclear power.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

__ 6. **Check QPTR (Tech Spec 3.2.4) - WITHIN TECH SPEC LIMITS.**

Reduce reactor power as required by Tech Specs as follows:

__ a. Do not move rods until IAE determines rod movement is available.

__ b. Borate as required during power reduction to maintain T-Ave at T-Ref.

__ c. Monitor AFD during load reduction.

d. **IF AT ANY TIME** AFD reaches Tech Spec limit **AND** reactor power is greater than 50%, **THEN**:

__ 1) Trip Reactor.

__ 2) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

e. Reduce load **PER** one of the following procedures:

__ • OP/1/A/6100/003 (Controlling Procedure For Unit Operation), Enclosure 4.2 (Power Reduction)

OR

__ • AP/1/A/5500/04 (Rapid Downpower).

7. **REFER TO Tech Specs:**

__ • Tech Spec 3.1.4 (Rod Group Alignment Limits)

__ • Ensure shutdown margin calculation performed within 1 hour.

Encl. 1 - STEP 7:

PURPOSE:

This step is an evaluation of Tech Spec requirements for Rod Group Alignment Limits Tech Spec 3.1.4 and the action requirement for determining Shutdown Margin with an untrippable or immovable control rod T.S.3.1.4, action B.2.1.1.

DISCUSSION:

These Tech Spec items are listed to ensure the Control Room SRO evaluates the requirements for Rod Group Alignment Limits and Shutdown Margin when a dropped rod has occurred and complies with the appropriate action. A SDM calculation must be completed within 1 hour since a rod that's already inserted is not available to supply shutdown margin.

Encl. 1 - STEP 8:

PURPOSE:

To perform a plant shutdown versus retrieving a dropped rod if less than 5% power.

DISCUSSION:

With the unit being in Mode 1, there is no risk of withdrawing a dropped control rod with the resulting power increase causing a mode change. With the unit in Mode 2, the risk of an increase in reactor power and mode change are possible when retrieving a dropped control rod. Other factors to consider when in Mode 2 that support a unit shutdown are:

- At power levels below 5% rated thermal power, the turbine is not on line and changes to T-Ave will be handled by steam dumps which does not allow for fine temperature control,
- With the unit in a shutdown condition, problems related to xenon and temperature changes will not have to be addressed,
- With the turbine not on line, the unit status allows for the rod control problem to be corrected without having the unit at risk.
- Prevents recriticality during dropped rod retrieval should the core become sub critical due to the rod drop.

This step is consistent with the guidance given in response to industry event OEDB 90-002761 (SER 90-15). In that event, Vogtle1 dropped several rods during physics testing, and withdrew rods to get back critical. This resulted in bypassing the carefully controlled evolution of taking the reactor critical. The appropriate response should have been to trip the reactor or drive the other

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.	A.1.1 Verify SDM is within the limit specified in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	OR	
	B.2.1.1 Verify SDM is within the limit specified in the COLR.	1 hour
	OR	
	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND	
	B.2.3 Verify SDM is within the limit specified in the COLR.	Once per 12 hours
	AND	
	B.2.4 Perform SR 3.2.1.1.	72 hours
	AND	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	AND	
	B.2.6 Re-evaluate 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 within the limit specified in the COLR. OR D.1.2 Initiate boration to restore required SDM to within limit. AND D.2 Be in MODE 3.	1 hour 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1 Verify individual rod positions within alignment limit.	12 hours AND 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 (trippability) by moving each rod not fully inserted in the core $\geq$ 10 steps in either direction.	92 days OR Prior to entering MODE 3 upon Unit 1 startup following the Unit 1 end of Cycle 13 refueling outage *
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is $\leq$ 2.2 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 551°F$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

* One time change applicable to Unit 1 only.

Response To A Dropped Rod

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Announce occurrence on paging system.**	
__ 2. **Dispatch rod control system qualified IAE to correct cause of dropped rod.**	
__ 3. **Check "ROD CONTROL URGENT FAILURE" alarm (1AD-2, A-10) - DARK.**	**Perform the following:** __ a. Do not move control rods while the "ROD CONTROL URGENT FAILURE" alarm is lit, unless instructed by IAE. __ b. **IF AT ANY TIME** IAE desires to reset "ROD CONTROL URGENT FAILURE" alarm, **THEN** depress the "ROD CONTROL ALARM RESET" pushbutton. c. **IF AT ANY TIME** while in this procedure a runback occurs **AND** no rods will move, **THEN** perform the following: __ 1) Trip Reactor. __ 2) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ 4. **Use OAC point M1P1385 (Reactor Thermal Power, Best Estimate), to determine reactor power in subsequent steps.**	
__ 5. **Check AFD (Tech Spec 3.2.3) - WITHIN TECH SPEC LIMITS.**	**IF reactor power greater than 50%, THEN:** __ a. Trip reactor. __ b. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

If the "Rod Control Urgent Failure" (1AD-2, A-10) alarm is present, the alarm is being generated by a failure in either the Logic or Power Cabinets. Control rods should not be moved until the problem has been identified and evaluated. If an attempt is made to move control rods in the individual "Bank" mode before the problem is identified, a dropped rod could result. This may occur from incorrect operation of the CRDM if the failure is in the Slave Cycler for the affected rod. If a problem has occurred in a Power Cabinet, dropped rods may result if the alarm is reset (using the "Rod Control Alarm Reset" pushbutton) before the cause of the urgent alarm is identified and repaired.

The two methods to control reactivity on a short-term (transient) basis are by adjusting turbine load or moving control rods. If a runback occurs, adjusting turbine load is not an option for the Operator. If this occurs while rods can't be moved, there remains no quick reactivity control method for the Operator to control reactor power/NC temperature, and so the conservative thing to do is trip the reactor.

Encl. 1 - STEP 4:

PURPOSE:

The step provides guidance to the operator to use the OAC point for Thermal Power Best Estimate for making procedural decisions based on power level.

DISCUSSION:

Since operators typically use the OAC program that monitors the power, AFD and QPTR parameters for each quadrant, these indications may change significantly from their normal indication with a dropped control rod. It is important that the operator monitor Thermal Power Best Estimate (OAC point M1P1385) which takes in to account all parameters of reactor power. Thermal Power Best Estimate uses heat transfer calculations and not excore nuclear instrumentation inputs. Thermal Power Best Estimate indication will be used to determine if the unit should be shutdown or remain in operation based on power level.

Encl. 1 - STEP 5:

PURPOSE:

This step is a check of AFD within Tech Spec limits since a dropped rod (especially the case where the rod is misaligned more than 50 steps below its' associated group) can affect AFD.

DISCUSSION:

Above 50% Rated Thermal Power, limits on AFD (variable from 50-100% power) are defined by Tech Specs (limits are found in Core Operating Limits Report). The limits on AFD are used to limit the amount of axial power distribution to either the top or bottom of the core. Limiting the AFD skewing over time minimizes xenon skewing and limits excessive power distributions that could potentially damage the fuel. The limit ensures power distribution remains consistent with the design values used in the safety analysis. The limit provides a margin of protection for both DNB and linear heat generation rate, which contribute to excessive power peaks. The guidance to trip the reactor if the limits are exceeded above 50% power is a conservative action to take considering that power reductions without rod movement cause a dramatic shift toward a positive AFD and so a positive shift would cause an AFD that's out of limit in the positive direction to get even more out of spec. In either case, positive or negative, trying to restore AFD within its' limits within 30 minutes could be operationally difficult without use of control rods.

Encl. 1 - STEP 6:

PURPOSE:

Ensure compliance with Tech Spec requirements for QPTR.

DISCUSSION:

A dropped control rod could significantly affect QPTR and require a unit power reduction to comply with ITS.
Above 50% Rated Thermal Power, limits on AFD (variable from 50-100% power) and QPTR ($\leq$1.02) are defined by Tech Specs (limits for AFD are found in Core Operating Limits Report).
The limit on QPTR ensures the radial power distribution remains consistent with the design values used in the safety analysis. The QPTR limit of 1.02 provides a margin of protection for both DNB and linear heat generation rate, which contribute to excessive power peaks.
The guidance to reduce reactor power is provided by the operating procedure for power reduction or by AP/4 (Rapid Downpower) per the applicable time requirements of each Tech Spec. Since "no rod motion" is directed until IAE determines it's available, direction is given to accomplish the power reduction with boron to maintain T-ave at T-ref. If a power reduction is required, direction is given to monitor AFD, since it will tend to go positive on the shutdown, and to trip the reactor if it reaches its limit before getting to 50% power, since it will only get worse before it gets better.

REFERENCES:
ITS 3.2.4

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
N/A	1.5	1.5	1.5	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose of the Reactor Control System (IRX).		X	X	X	
2	Discuss the rod speed program for both rod insertion and withdrawal as per Drawing 7.4.		X	X	X	
3	Sketch the IRX block diagram, including all input and output signals, per Drawing 7.6.		X	X	X	
4	Describe how the T_{ref} program is generated, based on turbine impulse pressure, including minimum and maximum values of T_{ref}.		X	X	X	
5	Describe how the Temperature Mismatch signal is developed and used for rod movements.		X	X	X	X
6	Describe how the Power Mismatch signal is developed and used for rod movements.		X	X	X	X
7	Explain how the Combined Error signal is used to develop rod speed and direction signals.		X	X	X	X
8	State all rod speeds for both automatic and manual operation.		X	X	X	
9	Describe all interlocks affecting rod withdrawal to include setpoints, logic and mode of operation that is affected (Automatic or Manual).		X	X	X	X
10	Describe the system operation during transients.		X	X	X	X
11	Describe the system operation and operator response to various failed input signals.		X	X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	033 AA2.05	
	Importance Rating		3.1

Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Nature of abnormality, from rapid survey of control room data

Proposed Question: SRO 83

Given the following:

- A reactor startup is in progress.
- SR Channel N-31 indicates $2X10^3$ CPS.
- SR Channel N-32 indicates $2X10^3$ CPS.
- IR Channel N-35 indicates $3.0X10^{-11}$ amps.
- IR Channel N-36 indicates $9.0X10^{-11}$ amps.

Which ONE (1) of the following describes (1) the existing plant condition, and (2) the action required in accordance with AP/16, Malfunction of Nuclear Instrumentation, and Technical Specifications?

A. (1) N-36 is undercompensated;
(2) maintain power stable until N-36 is repaired.

B. (1) N-35 is undercompensated;
(2) maintain power stable until N-35 is repaired.

C. (1) N-36 is undercompensated;
(2) Raise power to >P-10 or place the unit in Mode 3 until N-36 is repaired.

D. (1) N-35 is undercompensated;
(2) Raise power to >P-10 or place the unit in Mode 3 until N-35 is repaired.

Proposed Answer: A

Explanation (Optional):

A. Correct. N-36 is reading approximately 0.7 decades too high for the SR counts displayed, therefore undercompensated. AP/16 requires no positive reactivity additions. TS requires >P-10 or <P-6

B. Incorrect. Wrong NI is undercompensated. N-35 reads correctly. Plausible if applicant confuses overlap and indication for IR NIs

C. Incorrect. Correct NI but incorrect action taken. Mode 3 entry is not required for the given conditions, and the AP says no positive reactivity additions are allowed, so >P-10 is incorrect

D. Incorrect. Incorrect NI, Incorrect action taken. See A, B, C above

Technical Reference(s): AP/16 case 2 (Attach if not previously provided)

TS 3.3.1

IC-ENB Rev 26

Proposed references to be provided to applicants during examination: None

Learning Objective: IC-ENB-Obj 7 & 19 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2, 5

Comments:

Modified from VC Summer 2007 NRC Exam

KA is met because the applicant must determine the nature of the failure based on given indications, and SRO level because appropriate TS action for plant conditions is required knowledge at SRO level

Given the following plant conditions:

- A reactor startup is in progress.
- SR Channel N-31 indicates $7X10^3$ CPS.
- SR Channel N-32 indicates $7X10^3$ CPS.
- IR Channel N-35 indicates $8.7X10^{-6}\%$ power.
- IR Channel N-36 indicates $6.0X10^{-6}\%$ power.

Which ONE (1) of the following describes (1) the existing plant condition, (2) the status of P-6, and (3) the action required in accordance with AOP-401.8, Intermediate Range Channel Failure?

A. (1) N-36 is undercompensated;
(2) P-6 should NOT be satisfied;
(3) maintain power stable until N-36 is repaired.

B. (1) N-36 is overcompensated;
(2) P-6 should be satisfied;
(3) maintain power stable until N-36 is repaired.

C. (1) N-36 is undercompensated;
(2) P-6 should be satisfied;
(3) place the unit in Mode 3 until N-36 is repaired.

D. (1) N-36 is overcompensated;
(2) P-6 should NOT be satisfied;
(3) place the unit in Mode 3 until N-36 is repaired.

Ans. B

Control Power Fuses - Overcurrent protection for control signal circuit transformers. Control power supplies the lights on the drawer and 118 VAC to the bistable relay drivers to the plant relays. (High flux at shutdown alarm and SR high level trip). This is true for the IR and PR drawers/circuits also.

NOTE (Reference **Figure 7.21**): If either instrument or control power fuses are removed, the bistables will trip. Level Trip Bypass will prevent bistable trip for Instrument Power fuses only.

Objective # 10

Level Trip Switch - Two position switch: Normal - Switch Inactive; Bypass - Enables Operation Selector Switch for test and calibration; Provides AC signal to prevent Rx trip signal during testing.

Operation Selector Switch - Eight position switch enabled by Level Trip Switch to 'Bypass' position. Channel On Test lamp lights when not in Normal. Normal - Switch Inactive; Six Test Positions with Preset cps test values; Level Adjust - Level Adjust Potentiometer in circuit.

Level Adjust Potentiometer - Adjustable test signal into level amp. - Enables adjustment of the trip level of various bistables.

Objective # 10

High Flux at Shutdown Switch - Two position switch. Normal -allows circuit to provide "High Flux at Shutdown" and "Containment Evacuation" alarm when setpoint is exceeded; Block-used during startup - Blocks High Flux at Shutdown Alarm and Containment Evacuation Alarm.

2.2 Intermediate Range

2.2.1 Intermediate Range Detectors

Objective # 6

Reference **Figure 7.6**. Both intermediate range channels use compensated ion chambers to determine reactor power. These detectors are located just above the source range detectors in the same housing. The compensated ion chamber (CIC) uses two concentric Nitrogen gas filled, volumes: the "outer" is sensitive to both neutrons and gamma (boron lined); the "inner" sensitive only to gamma. As the two volumes are mounted concentrically in one unit, both are in essentially the same radiation field. By placing a negative potential on the inner lead, the gamma signal generated in the inner volume is made to compensate or cancel out the gamma signal generated in the outer volume. Since the two volumes can not be manufactured exactly the same size, the high voltage to the center electrode is variable to adjust the sensitivity of the inner volume. Operating in the recombination region, a change in inner volume detector voltage will vary the gamma current for a given flux level. The outer volume operates in the ion chamber region where all the ion pairs are collected.

Objective # 5

Gamma radiation becomes a smaller percentage of the detector interactions as power increases and becomes insignificant after 10^{-9} amps (first two decades). Above this power level gamma compensation is no longer required for accurate indication.

2.2.2 Over Compensation And Under Compensation

Objective # 7

Reference **Figure 7.7**. With the inner chamber voltage set properly, inner chamber gamma current will exactly match outer chamber gamma current and the two will cancel leaving only the neutron current. With inner chamber voltage set too high, inner chamber current will exceed outer chamber gamma current canceling all gamma current plus some of the neutron current. This is "over-compensation". The following are consequences of **over-compensation**:

- The indicated power level will read lower than the actual power level.
- The intermediate range instrument will "come on scale" at a higher source range level producing less overlap between the two ranges.
- During startup, the P-6 permissive will be received later, at a higher actual neutron flux level and the source range will be closer to the 10^5 cps, Hi Level Trip setpoint.
- After a Reactor Trip, power will decay to the P-6 reset sooner than normal.
- Initially, indicated SUR will be higher than actual SUR.

The effects of improper compensation are much more pronounced at low power and become a non-factor prior to taking critical data at 10^{-8} amps.

With inner chamber voltage set too low, inner chamber current will be less than outer chamber gamma current, canceling only a portion of the gamma current. This is "under-compensation". The following are consequences of **under-compensation**:

- The indicated power level will read higher than the actual power level.
- The intermediate range instrument will "come on scale" at a lower source range level producing more overlap between the two ranges.
- During startup, the P-6 permissive will be received earlier, at a lower actual neutron flux level.
- After a Reactor Trip, power will decay to the P-6 reset later than normal and may prevent automatic re-energizing of the source range detectors.
- Initially, indicated SUR will be lower than actual SUR.

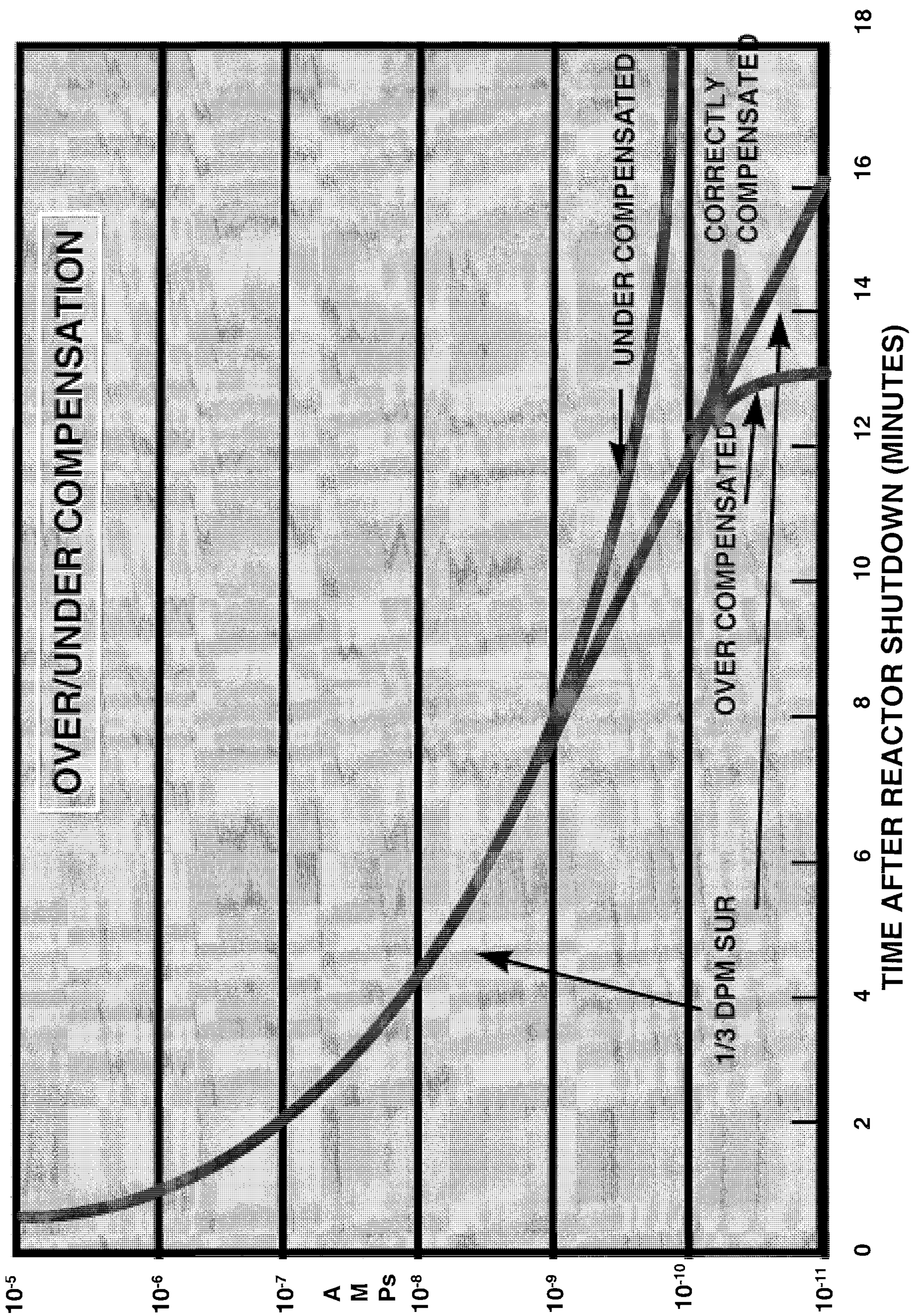
2.2.3 Intermediate Range Circuitry

Objective # 4

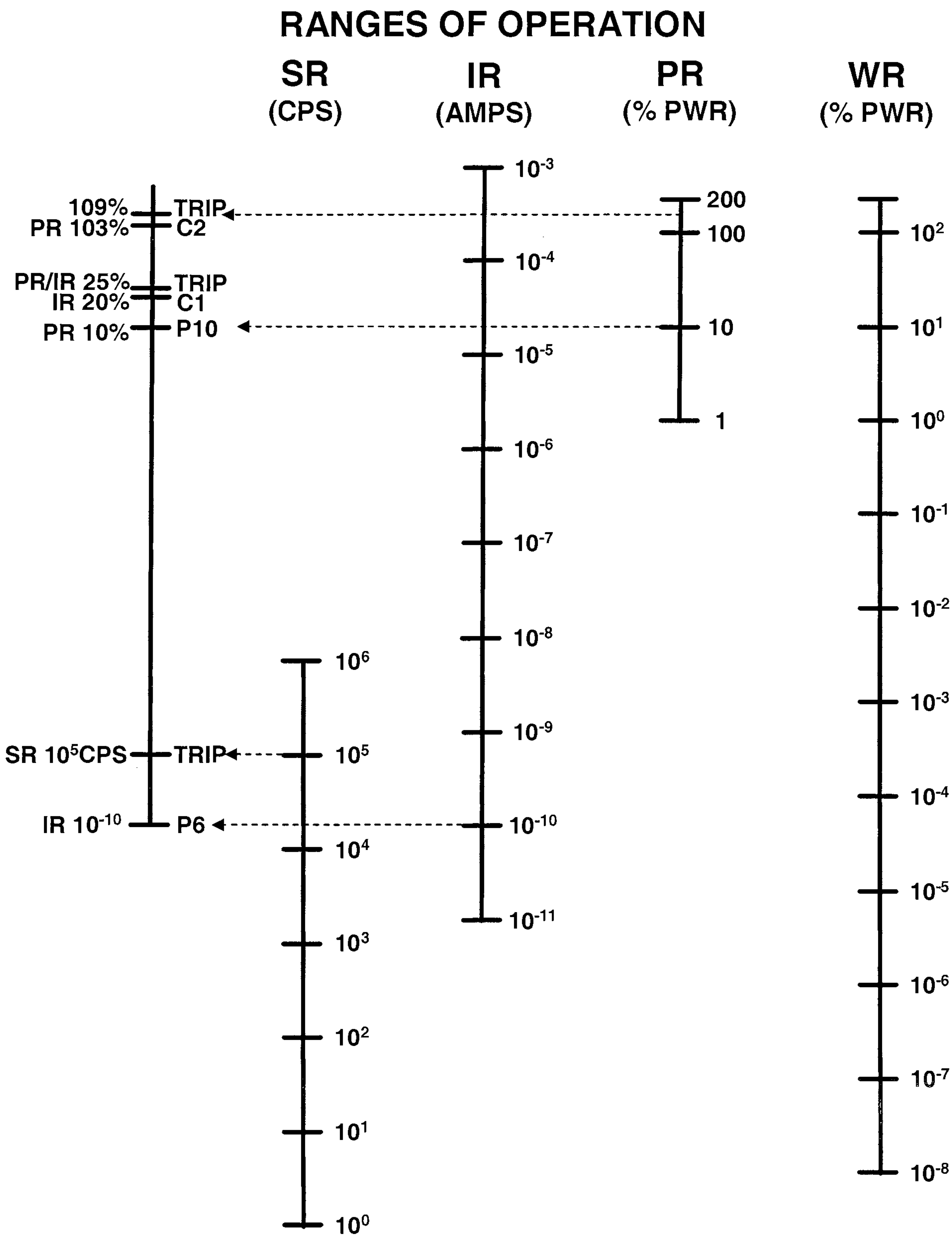
Reference **Figure 7.8**. The Intermediate Range should normally start to indicate power at a Source Range power level of 10^3 cps and the Source Range should be blocked by the time level is 10^4 cps and Intermediate level is at 10^{-10} amps. The indicating range for the Intermediate Range instrument is 10^{-11} to 10^{-3} amps, which overlaps the entire power range.

The current flow from the intermediate range detectors is too low to be used directly for control purposes so the output feeds a log level amplifier (log amp) for conversion to a usable voltage. The log level amplifier also converts the detector signal to a logarithmic output and drives the bistables, indicators and other circuits.

7.7 Over and Undercompensation (03/20/97)



7.2 Operating Ranges (01/09/02)



3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2 Be in MODE 3.	54 hours
C. One channel or train inoperable.	C.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	C.2 Open reactor trip breakers (RTBs).	49 hours

(continued)

Table 3.3.1-1 (page 1 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	3[(a)], 4[(a)], 5[(a)]	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110% RTP	109% RTP
b. Low	1[(b)],2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 26% RTP	25% RTP
3. Power Range Neutron Flux Rate						
High Positive Rate	1,2	4	D	SR 3.3.1.7 SR 3.3.1.11	≤ 5.5% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1[(b)], 2[(c)]	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30% RTP	25% RTP
	2[(d)]	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30% RTP	25% RTP
5. Source Range Neutron Flux	2[(d)]	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.3 E5 cps	1.0 E5 cps
	3[(a)], 4[(a)], 5[(a)]	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.3 E5 cps	1.0 E5 cps
	3[(e)], 4[(e)], 5[(e)]	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A

(continued)

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.
(b) Below the P-10 (Power Range Neutron Flux) interlocks.
(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------------NOTE------------------ One channel may be bypassed for up to 4 hours for surveillance testing. ---	
	E.1 Place channel in trip.	6 hours
	OR	
	E.2 Be in MODE 3.	12 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours
	OR	
	F.2 Increase THERMAL POWER to > P-10.	2 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	------------------NOTE---------------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. ---	
	G.1 Suspend operations involving positive reactivity additions.	Immediately
	AND	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. THERMAL POWER < P-6, one or two Intermediate Range Neutron Flux channels inoperable.	H.1 Restore channel(s) to OPERABLE status.	Prior to increasing THERMAL POWER to > P-6

(continued)

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
	2.0	3.0	3.0	2.0

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the Nuclear Instrumentation System.		X	X	X	
2	Explain why it is necessary to use three ranges of Excore Nuclear Instrumentation.		X	X	X	
3	Explain the operation of the detector used in each range of instrumentation.		X	X	X	
4	Sketch the outputs of each range of Nuclear Instrumentation, to include all indication, control and protective circuits.		X	X	X	
5	Explain why gamma compensation is necessary in the Source Range and Intermediate Range but not in the Power Range.		X	X	X	
6	Describe the methods of gamma compensation used by the Source and Intermediate Ranges.		X	X	X	
7	Describe the effects of 'over' and 'under' compensation on the Intermediate Range.		X	X	X	
8	Explain the functions of the control switches for each range of Nuclear Instrumentation.		X	X	X	X
9	Concerning the channel current comparator and detector current comparator:					
	• Explain the function of each.		X	X	X	
	• List the alarm setpoints for each.		X	X	X	X
10	Explain the functions of all related bypass and block switches on the Nuclear Instrumentation miscellaneous panels.		X	X	X	X
11	List the Reactor Trips associated with the Nuclear Instrumentation System. (Include setpoints, logic and interlocks)		X	X	X	X

12	List the Protection and Control Interlocks (Ps and Cs) associated with the Nuclear Instrumentation System. (Include setpoints and logic)		X	X	X	X
13	State the purpose of the Wide Range Neutron Detection System.		X	X	X	
14	Concerning the Wide Range Neutron Detection System:					
	• Describe the operation.		X	X	X	
	• Describe the indications and controls.		X	X	X	X
15	State the purpose of the Gamma-Metrics Shutdown Monitor System.		X	X	X	
16	Concerning the Gamma-Metrics Shutdown Monitor System:					
	• Describe the operation.		X	X	X	
	• Describe the alarms, indications and controls.		X	X	X	X
17	Determine the validity of indicated reactor power using alternate indications of power level.		X	X	X	X
18	Describe the Source Range instrumentation response for voiding in the core and downcomer region.		X	X	X	X
19	Concerning the Technical Specifications related to the Nuclear Instrumentation System;					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is(are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech Specs, determine required action(s).			X	X	X
	• Discuss the basis for a given Tech Spec LCO or Safety Limit.				X	*
	* SRO Only					

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	059 G2.2.38	
Importance Rating		4.5

Liq radwaste release

Equipment Control: Knowledge of conditions and limitations in the facility license.

Proposed Question: SRO 84

Given the following:

Turbine Building Sump to RC Radiation Monitor, EMF-31, is discovered to have an alarm setpoint that is set ONE decade higher than required.

Which ONE of the following describes the impact of this condition?

The dose or dose commitment to members of the public may exceed the requirements of 10CFR50 of....

A. 1.5 mrem whole body dose in a calendar quarter.

B. 5 mrem whole body dose in a calendar quarter.

C. 1.5 mrem whole body dose in a calendar year.

D. 5 mrem whole body dose in a calendar year.

Proposed Answer: A

Explanation (Optional):

A. Correct. This is a memory item. Below options are plausible because the numbers supplied are all part of the SLC

B. Incorrect. 5 mrem is Organ Dose allowed for a calendar quarter

C. Incorrect. Allowed WB dose for a calendar year is 3 mrem

D. Incorrect. 5 mrem is Organ Dose allowed for a calendar quarter.

Technical Reference(s): SLC 16.11.3, Rev 0 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: WE-RLR Obj 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 1, 2,4

Comments:

KA is matched because 10CFR50 requirements for radioactive release are limitations in the facility license. SRO knowledge because the item requires knowledge of SLC (TRM) conditions that will require action by the SRO

16.11 RADIOLOGICAL EFFLUENT CONTROLS

16.11.3 Dose - Liquid Effluents

COMMITMENT The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS (see Figure16.11.1-1) shall be limited:

a. During any calendar quarter, to $\leq$ 1.5 mrem to the total body and to $\leq$ 5 mrem to any organ, and

b. During any calendar year, to $\leq$ 3 mrem to the total body and to $\leq$ 10 mrem to any organ.

APPLICABILITY At all times.

REMEDIAL ACTIONS

---NOTES---

Enter applicable Conditions and Required Actions of SLC 16.11.12, "Total Dose," when the limits of this SLC are exceeded by twice the specified limit.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Calculated dose from release of radioactive materials in liquid effluents exceeding above limits.	---------------------NOTE-------------- The Special Report shall include the results of radiological analyses of the drinking water source, and the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act, as applicable. ---	
	A.1 Prepare and submit a Special Report to the NRC which identifies the causes for exceeding the limits, corrective actions taken to reduce releases, and actions taken to ensure that subsequent releases are within limits.	30 days

TESTING REQUIREMENTS

TEST	FREQUENCY
TR 16.11.3.1 Determine cumulative dose contributions from liquid effluents for current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM.	31 days

BASES

This commitment is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The commitment implements the guides set forth in Section II.A of Appendix I. The REMEDIAL ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. These requirements are applicable only if the drinking water supply is taken from the river 3 miles downstream of the plant discharge.

The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This commitment applies to the release of liquid effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system in accordance with the guidance given in NUREG-0133, Chapter 3.1.

REFERENCES

1. McGuire Nuclear Station, Off site Dose Calculation Manual

2. 40 CFR Part 141, Safe Drinking Water Act

3. 10 CFR Part 50, Appendix I

4. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

5. Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
6	Concerning the Selected Licensee Commitments (SLC) related to Liquid Waste Releases;					
	• Given the SLC Manual, discuss any commitments and their applicability.			X	X	X
	• For any commitments that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any commitment is (are) not met and any action(s) required within one hour.			X	X	X
	• Given the SLC Manual, discuss the basis for a given commitment.				X	*
	* **SRO only**					
	WERLR006					

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	√ core cooling E06 G2.1.20	
	Importance Rating		4.6

Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: SRO 85

Given the following:

- A LOCA has occurred on "1B" Cold Leg.
- ECCS has NOT functioned as required.
- All NC Pumps are TRIPPED.
- PZR PORVs are CLOSED and in AUTO.
- CET's indicate 692°F and rising.
- Reactor Vessel LR Level is 35% and lowering.
- Containment pressure is 3 psig and rising slowly.

Which ONE of the following procedures will the crew implement for these conditions, and the action taken if ECCS components CANNOT be restored?

A. Enter FR-C.1, Response To Inadequate Core Cooling; NC pumps are started prior to secondary depressurization to provide forced cooling of the NCS.

B. Enter FR-C.2, Response To Degraded Core Cooling; NC pumps are started prior to secondary depressurization to provide forced cooling of the NCS.

C. Enter FR-C.1, Response To Inadequate Core Cooling; secondary depressurization is initiated prior to attempting NC pump operation to depressurize the NCS and facilitate SI Accumulator injection.

D. Enter FR-C.2, Response To Degraded Core Cooling; secondary depressurization is initiated prior to attempting NC pump operation to depressurize the NCS and facilitate SI Accumulator injection.

Proposed Answer: D

Explanation (Optional):
A. Incorrect. Wrong procedure entry and also wrong action for NCP operation. A LOCA is in progress but conditions for FR-C.1 do not exist
B. Incorrect. NCP would only be operated if secondary depressurization was ineffective in achieving core cooling.
C. Incorrect. Incorrect entry but correct action with respect to secondary depressurization and NCP operation
D. Correct.

Technical Reference(s): F-0, FR-C.2 Rev 5; EP-FRC Rev 10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-FRC Obj 2 & 3 (As available)

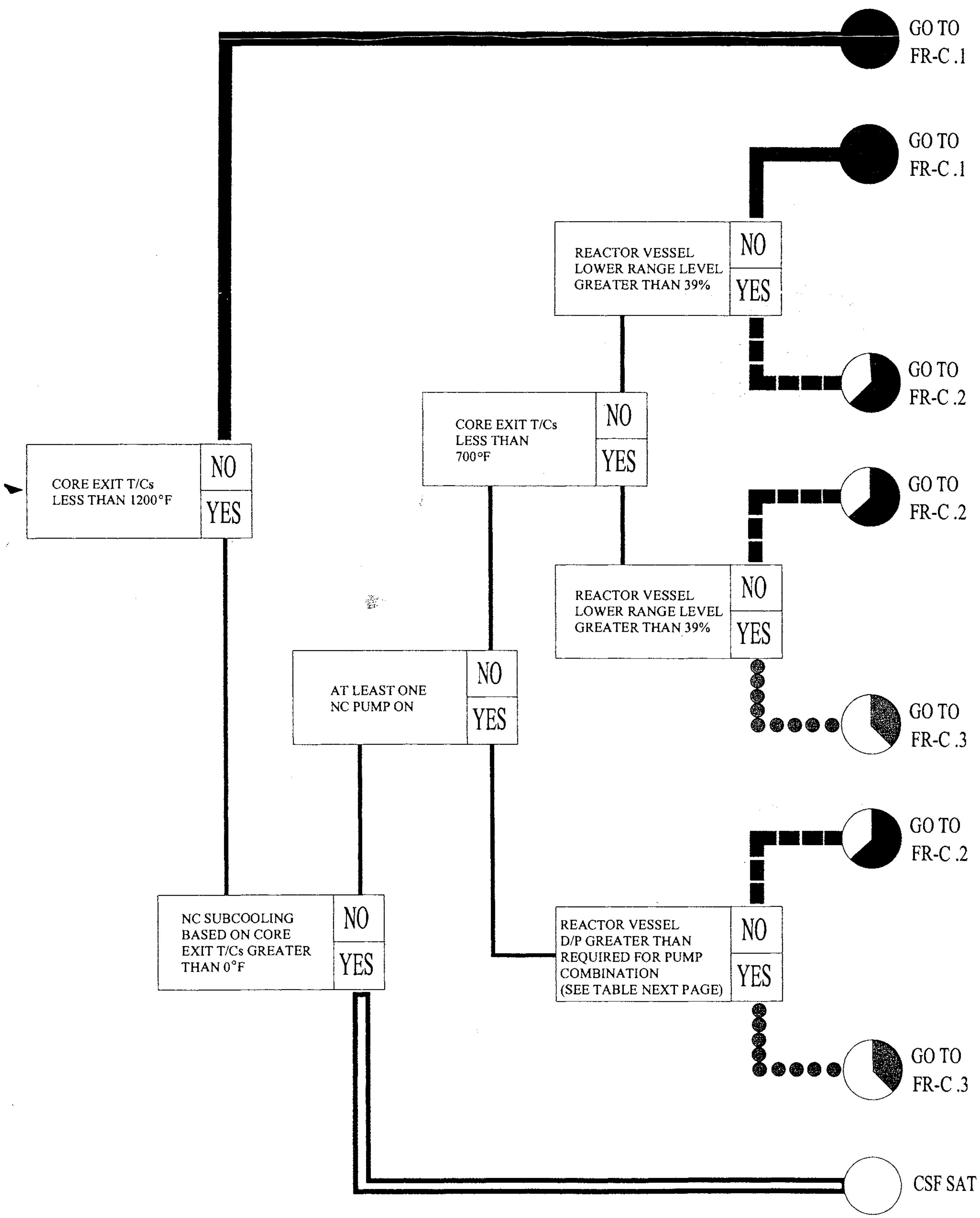
Question Source: Bank # X (WTSI)
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam BVPS-1 2007

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments:
KA is matched because item evaluates knowledge of procedure steps for degraded core cooling condition. SRO level because the applicant must assess (evaluate) plant conditions and determine procedure entry, as well as strategy for the procedure entered


CORE EXIT T/Cs LESS THAN 1200°F
NO
YES
GO TO FR-C.1
CORE EXIT T/Cs LESS THAN 700°F
NO
YES
REACTOR VESSEL LOWER RANGE LEVEL GREATER THAN 39%
NO
YES
GO TO FR-C.1
GO TO FR-C.2
REACTOR VESSEL LOWER RANGE LEVEL GREATER THAN 39%
NO
YES
GO TO FR-C.2
GO TO FR-C.3
AT LEAST ONE NC PUMP ON
NO
YES
NC SUBCOOLING BASED ON CORE EXIT T/Cs GREATER THAN 0°F
NO
YES
REACTOR VESSEL D/P GREATER THAN REQUIRED FOR PUMP COMBINATION (SEE TABLE NEXT PAGE)
NO
YES
GO TO FR-C.2
GO TO FR-C.3
CSF SAT

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE After the Low Pressure Steamline Isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation.

15. **Depressurize all intact S/Gs to 110 PSIG as follows:**

__ a. **REFER TO** Enclosure 6 (NC Cooldown Rate Monitoring) to assist in monitoring 100°F in an hour cooldown rate.

b. Check condenser available:

__ • MSIV on all intact S/Gs - OPEN

__ • "C-9 COND AVAILABLE FOR STEAM DUMP" status light (1SI-18) - LIT.

RNO: __ b. **GO TO** RNO for Step 15.e.

__ c. Check "STEAM DUMP SELECT" - IN STEAM PRESSURE MODE.

RNO: c. Perform the following to place steam dumps in steam pressure mode:

__ 1) Place "STM PRESS CONTROLLER" in manual.

__ 2) Adjust "STM PRESS CONTROLLER" output to equal "STEAM DUMP DEMAND" signal.

__ 3) Place "STEAM DUMP SELECT" in steam pressure mode.

__ d. **WHEN** "P-12 LO-LO TAVG" status light (1SI-18) lit, **THEN** place steam dumps in bypass interlock.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15. (Continued)	
__ e. Dump steam from intact S/G(s) to condenser while maintaining cooldown rate in NC T-Colds less than 100°F in an hour.	e. Dump steam using all intact S/G(s) SM PORV as follows: __ 1) Ensure Main Steam Isolation reset. __ 2) Ensure SM PORVs reset. __ 3) Dump steam using all intact S/G(s) SM PORVs while maintaining cooldown rate in NC T-Colds less than 100°F in an hour. 4) **IF** any intact S/G SM PORV closed, **THEN** dump steam using any of the following while maintaining cooldown rate in NC T-Colds less than 100°F in an hour: __ a) Dispatch operator to operate intact S/G(s) SM PORV. b) **IF** any intact S/G SM PORV is unavailable, **THEN** evaluate using the following to dump steam: __ • Reopen MSIVs and dump steam to condenser **PER** Enclosure 8 (Condenser Dumps). __ • Run TD CA pump. __ • Use steam drains **PER** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 19 (S/G Depressurization Using Steam Drains).
__ f. Check intact S/G pressures - LESS THAN 110 PSIG.	__ f. **RETURN TO** Step 11.
__ g. Check at least two NC T-Hots - LESS THAN 354°F.	__ g. **RETURN TO** Step 11.
__ h. Stop S/G depressurization and maintain S/G pressures stable.	

STEP 14 <u>WHEN</u> "P-11 PRESSURIZER S/I BLOCK PERMISSIVE" status light (1SI-18) lit, <u>THEN</u> depress "BLOCK" on Low Pressure steamline Isolation block switches.

<u>PURPOSE:</u> To prevent MSIV closure on low steamline pressure during controlled NC system cooldown.

<u>BASIS:</u> The Steamline Isolation signal on low steamline pressure can be blocked during cooldown once the P-11 Block Permissive status light is lit (approximately 1955 psig). This prevents MSIV closure, thus allowing cooldown by the preferred method of steam dump to the condenser.

NOTE After Low Pressure Steamline Isolation signal is blocked, maintaining steam pressure negative rate less than 2 PSIG per second will prevent a Main Steam Isolation..

<u>PURPOSE:</u> To warn the operator that MSIV isolation will occur if the S/G's are depressurized too quickly and provide guidance for controlling the depressurization rate.<u>BASIS:</u> N/A

STEP 15 Depressurize intact S/Gs to 110 psig as follows:

<u>PURPOSE:</u> To re-cover the core via CLA injection.

<u>BASIS:</u> The controlled secondary depressurization has been shown to be an effective way to reduce NC system pressure. Pressure must be reduced in order for the CLAs and ND pumps to inject.

To prevent nitrogen injection, the operator is directed to stop the secondary depressurization when the S/G pressure reaches 110 psig.

STEP 16 Check ND pumps – ON.

<u>PURPOSE:</u> To see if ND pumps are running.

<u>BASIS:</u> In this step the operator checks if the ND pumps are running and, if not, starts them since ND injection will be used to restore long-term core cooling. The ND pumps will inject if NC system pressure is dropped below their shutoff head.

2.1.3 NC Pump Restart and Opening Pzr PORVs (Continued)

The NC Pumps cannot be expected to run indefinitely under highly voided NC system conditions. The operator must still take action to establish a makeup source of water to the NC system to restore adequate long term cooling. NC system pressure must, therefore, be reduced in order for the CLAs and/or ND pumps to inject.

The operator should continue attempts to depressurize the S/Gs or to establish the secondary heat sink; however, if the core exit T/C temperatures remain above 1200°F and all available NC Pumps are running, the only other option is to effectively enlarge the hole in the NC system to reduce pressure. This may be achieved by opening all available NC system vent paths to containment, i.e., Pzr PORVs, head vents, etc.

It should be noted that venting the NC system to containment reduces NC system inventory and is not as effective in reducing NC system pressure as S/G depressurization. Some form of low pressure flow to the NC system must be established as soon as possible.

2.2. FR-C.2, Response to Degraded Core Cooling

Degraded core cooling is caused by a substantial loss of primary coolant. If the NC Pumps are not running, the degraded core cooling symptoms indicate the core is partially uncovered. If the NC Pumps are running, the symptoms indicate the potential for core uncovery exists if the pumps should fail or be manually tripped. Operator action is required to restore NC system inventory in either case.

Reinitiation of high pressure S/I is the most effective method to restore NC system inventory and core cooling. If some form of high pressure injection cannot be established or is ineffective in restoring core cooling, then the operator must take actions to reduce the NC system pressure in order for the S/I accumulators and ND pumps to inject. A controlled secondary depressurization is an effective method for achieving this, while at the same time avoiding a rapid NC system cooldown that could cause problems with pressurized thermal shock.

The expected system response to both of the recovery techniques is described below.

4.0 FR-C.2, RESPONSE TO DEGRADED CORE COOLING

4.1. Purpose

This procedure provides actions to restore adequate core cooling.

The major actions are to be performed sequentially. Success, as indicated by improved core cooling and increasing vessel inventory, is evaluated prior to performing the next action in the sequence.

4.2. Symptoms/Entry Conditions

This procedure is entered from EP/1/A/5000/F-0 (Critical Safety Function Status Trees) (Core Cooling), on any orange condition.

These conditions are:

1. Core exit T/Cs greater than 700°F and vessel LR level greater than 39%, or
2. Core exit T/Cs less than 700°F and vessel LR level less than 39%, or
3. Subcooling less than 0°F, at least one NC pump running, and reactor vessel D/P less than required for the NC pump combination.

4.3. Immediate/Major Actions

The recovery/restoration technique includes the following two major action categories:

1. Establish Safety Injection flow to the NC system.
2. Initiate a controlled S/G depressurization to cool down and depressurize the NC system.

The following subsections provide a more detailed discussion of each major action category:

4.3.1 Establish Safety Injection Flow to the NC System

The operator must properly align emergency S/I valves, start the S/I pumps, and then check for flow through the S/I lines to the NC system. Core exit T/Cs and the appropriate RVLIS indication are checked to determine the effectiveness of S/I in restoring core cooling and vessel inventory.

4.3.2 Initiate a Controlled S/G Depressurization to Cool Down and Depressurize the NC System

The operator must maintain a 100°F/hr cooldown of the NC system by dumping steam to the condenser or opening the S/G PORVs while maintaining adequate feedwater to the S/Gs. The CLAs must be isolated and the NC pumps tripped once the S/Gs have been depressurized to 110 psig and the NC system has been depressurized until NC T-Hots are less than 354°F. The NC system cooldown and depressurization is continued until ND flow to the NC system has been established and verified. Core exit T/Cs and the appropriate RVLIS indication are checked to determine the effectiveness of CLA and/or ND S/I in restoring core cooling and vessel inventory.

2.0 PROCEDURE SERIES BACKGROUND

2.1. FR-C.1, Response to Inadequate Core Cooling

The indication of inadequate core cooling requires prompt operator action. Inadequate core cooling is caused by a substantial loss of primary coolant resulting in a partially or fully uncovered core. Without adequate heat removal, the core decay energy will cause the fuel temperatures to rise. Severe fuel damage will occur unless core cooling is promptly restored.

Reinitiation of high pressure S/I is the most effective method to recover the core and restore adequate core cooling. If some form of high pressure injection cannot be established or is ineffective, then the operator must take actions to reduce NC system pressure in order for the CLAs and ND pumps to inject. Analyses have shown that a rapid secondary depressurization is the most effective means for achieving this. If secondary depressurization is not possible, or primary-to-secondary heat transfer is significantly degraded, then the operator must start the NC Pumps. The NC Pumps will provide forced two phase flow through the core and temporarily improve core cooling until some form of make-up flow to the NC system can be established.

The recovery techniques applied in this procedure were developed from transient analyses. The expected system response to each of the recovery techniques is described below.

2.1.1 Reinitiation of High Pressure Safety Injection

The introduction of subcooled S/I into the highly voided NC system will cause steam in the cold legs to condense. Steam flow throughout the NC system will go up because of this condensation effect. Superheated steam forced out of the core may initially cause the core exit T/C temperatures to go up. As the vessel begins to refill, heat transfer from the fuel will cause the fluid entering the core to boil vigorously. This will create a two phase mixture which will eventually re-cover the entire core and cause the core exit T/C temperatures to quickly go down to saturation temperature.

This procedure uses the trends in core exit T/C temperatures and indicated vessel level to determine appropriate operator actions. The effectiveness of S/I in restoring NC system inventory is determined by the trend in RVLIS indication. If going up, then no further action may be necessary. The effectiveness of S/I in restoring core cooling is determined by the trend in core exit T/C temperatures. If going down, no further action is necessary. Exit temperatures less than 700°F indicate success, allowing the operator to return to the procedure and step in effect.

2.1.2 Secondary Depressurization

If attempts to reinitiate high pressure S/I are unsuccessful, or are ineffective in restoring adequate core cooling, then a rapid S/G depressurization must be performed. A rapid secondary depressurization will raise primary-to-secondary heat transfer and cause steam in the primary side of the S/G U-tubes to condense. When the condensation rate exceeds the steam generation rate, the NC system will begin to depressurize. As the NC system pressure drops, voiding of the water resident in the lower plenum and downcomer will partially recover the core with a two phase mixture. The continued depressurization will eventually cause S/I accumulator injection and temporary core recovery.

The operator should check the NC hot leg temperature trend to determine the effectiveness of the S/G depressurization in reducing the NC system pressure. The hot leg temperatures may initially rise as superheated steam in the core is forced out by the advancing two phase flow, but should quickly go down to saturation and continue to go down as the NC system depressurizes.

To prevent nitrogen injection from the S/I accumulators, the operator must isolate them. NC T-Hot less than 354°F and intact S/G pressure less than 110 psig are used to determine when the S/I accumulators should be isolated.

After the CLAs have been isolated, the secondary should be depressurized to atmospheric pressure. The NC system pressure should follow secondary pressure until the ND pumps begin to inject. Adequate core cooling has been restored and preparations for long term plant recovery can be started once ND flow has been established and the core is completely covered.

2.1.3 NC Pump Restart and Opening Pzr PORVs

If some form of high pressure injection cannot be established or is ineffective in restoring adequate core cooling, and if S/G depressurization is not possible or ineffective, then starting the NC Pumps will provide forced two phase flow through the core and temporarily improve core cooling. The core exit T/C temperatures should rapidly go down and the RVLIS indication should rapidly go up as a steam/water mixture is forced through the core by the NC Pumps. Analysis has shown that with secondary heat sink available, the NC Pumps will maintain core cooling as long as they continue to run. However, it should be noted that a degraded core cooling condition still exists.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		3.0	3.0	2.0

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose of each procedure in the FR-C series. EPFRC001			X	X	
2	Discuss the entry and exit guidance for each procedure in the FR-C series. EPFRC002			X	X	
3	Discuss the mitigating strategy (major actions) of each procedure in the FR-C series. EPFRC003			X	X	X
4	Discuss the basis for any note, caution or step for each procedure in the FR-C series. EPFRC004			X	X	X
5	Given the Foldout page, discuss the actions included and the basis for these actions. EPFRC005			X	X	X
6	Given the appropriate procedure, evaluate a given scenario describing accident events and plant conditions to determine any required action and its basis. EPFRC006			X	X	X
7	Discuss the time critical task(s) associated with the FR-C series procedures including the time requirements and the basis for these requirements. EPFRC007			X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	010 A2.01	
	Importance Rating		3.6

Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures

Proposed Question: SRO 86

Given the following:

- Unit 1 is at 100% power.
- A pressurizer pressure transient has occurred, resulting in a PZR PORV momentarily opening.
- The crew has stabilized the unit.
- Actions of AP/11, Pressurizer Pressure anomalies, are being performed.
- NC pressure is 2120 psig and stable.
- PZR heater groups 1A, 1B, 1C are energized.
- PZR heater group 1D is de-energized.
- PZR Spray Valves and PORVs indicate closed.

Which ONE of the following describes the impact of the current plant conditions, and the action required in accordance with technical specifications and AP/11?

A. NC System DNB limits are exceeding TS 3.4.1 COLR limits; place group 1D PZR heater mode select switch in MANUAL and energize to raise pressure; restore NC pressure to within limits within 2 hours.

B. Pressurizer TS 3.4.9 is applicable due to de-energized backup heaters; Place PZR PRESS MASTER in MANUAL to control pressure manually; verify capacity of remaining Backup Heaters or initiate a plant shutdown to Mode 3 within the required action time.

C. Pressurizer TS 3.4.9 is applicable due to de-energized backup heaters; Place group 1D PZR heater mode select switch in MANUAL and energize to raise pressure; TS 3.4.9 no longer applies when 1D Backup Heaters are operating in MANUAL.

D. NC System DNB limits are exceeding TS 3.4.1 COLR limits; Place PZR PRESS MASTER in MANUAL to control pressure manually; restore NC pressure to within limits within 2 hours.

Proposed Answer: A

Explanation (Optional):

A. Correct. With NC pressure at 2120, DNB limits are not being met IAW COLR.

B. Incorrect. 3.4.9 not required for loss of 1D heaters. Action is plausible because it is action required if loss of 1A or 1B heaters occurs. PZR master in manual would be for 1C heaters

C. Incorrect. 3.4.9 not required for loss of 1D heaters. Action is plausible because it is action allowed for restoration of 1A or 1B heaters

D. Incorrect. Master controller will not operate bank 1D, will operate 1C. Impact is correct, however

Technical Reference(s): TS 3.4.1; COLR Rev 30 (Attach if not previously provided)

AP/11, Rev 10

TS 3.4.9 and Basis

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 2, 5

Comments:

KA is matched because the item evaluates TS impact of failure, and also requires knowledge of action required to mitigate the consequences of the event.

SRO level because item requires knowledge of TS LCOs involved, and procedure strategy required for mitigation

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in Table 3.4.1-1.

APPLICABILITY: MODE 1.

------------------------------------NOTE------------------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer pressure or RCS average temperature DNB parameters not within limits.	A.1 Restore DNB parameter(s) to within limit.	2 hours
B. RCS total flow rate $\geq$ 99%, but < 100% of the limit specified in the COLR.	B.1 Reduce THERMAL POWER to $\leq$ 98% RTP.	2 hours
	<u>AND</u>	
	B.2 Reduce the Power Range Neutron Flux - High Trip Setpoint below the nominal setpoint by 2% RTP.	6 hours

(continued)

Table 3.4.1-1 (page 1 of 1)
RCS DNB Parameters

PARAMETER	INDICATION	No. OPERABLE CHANNELS	LIMITS
1. Indicated RCS Average Temperature	meter	4	≤ The limit specified in the COLR.
	meter	3	≤ The limit specified in the COLR.
	computer	4	≤ The limit specified in the COLR.
	computer	3	≤ The limit specified in the COLR.
2. Indicated Pressurizer Pressure	meter	4	≥ The limit specified in the COLR.
	meter	3	≥ The limit specified in the COLR.
	computer	4	≥ The limit specified in the COLR.
	computer	3	≥ The limit specified in the COLR.
3. RCS Total Flow Rate			≥ 388,000 gpm and greater than or equal to the limit specified in the COLR.

McGuire 1 Cycle 19 Core Operating Limits Report

Table 4

Reactor Coolant System DNB Parameters

PARAMETER	INDICATION	No. Operable CHANNELS	LIMITS
1. Indicated RCS Average Temperature	meter	4	≤ 587.2 °F
	meter	3	≤ 586.9 °F
	computer	4	≤ 587.7 °F
	computer	3	≤ 587.5 °F
2. Indicated Pressurizer Pressure	meter	4	≥ 2219.8 psig
	meter	3	≥ 2222.1 psig
	computer	4	≥ 2215.8 psig
	computer	3	≥ 2217.5 psig
3. RCS Total Flow Rate			≥ 390,000 gpm*

*Note: The RCS minimum coolant flow rate assumed in the licensing analyses for the M1C19 core is 388,000 gpm. However, the flow is set at 390,000 gpm, which is conservative

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

11. **Check the following Pzr heaters - ON:**

    __ • 1A

    __ • 1B

    __ • 1D.

IF NC pressure below desired pressure, THEN:

__ a. Place Pzr heater mode select switches in manual.

__ b. Turn on heaters as necessary to control pressure.

__ 12. **Check 1C Pzr heaters - ON.**

IF NC pressure below desired pressure, THEN:

__ a. Place "PZR PRESS MASTER" in manual.

__ b. Control pressure.

__ c. **WHEN** Pzr pressure returns to normal **AND** automatic Pzr pressure control desired, **THEN** place "PZR PRESS MASTER" in auto.

__ 13. **Check Pzr pressure - GOING UP TO DESIRED PRESSURE.**

__ **IF pressure continues to go down, THEN REFER TO AP/1/A/5500/10 (NC System Leakage Within The Capacity Of Both NV Pumps).**

__ 14. **Check "1NC-27 PRESSURIZER SPRAY EMERGENCY CLOSE" switch - SELECTED TO "NORMAL".**

__ **Notify station management to ensure switch restored to "NORMAL" once spray valve is repaired.**

__ 15. **Check "1NC-29 PRESSURIZER SPRAY EMERGENCY CLOSE" switch - SELECTED TO "NORMAL".**

__ **Notify station management to ensure switch restored to "NORMAL" once spray valve is repaired.**

__ 16. **GO TO Step 24.**

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 92% (1600 ft^3); and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours 12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for pressurizer level to remain within the required range is consistent with the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the UFSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 3), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1600 cubic feet, which is equivalent to 92%, ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with safety analysis analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW, capable of being powered from either the offsite power source or the emergency power supply. Only heater groups A and B are capable of being powered from the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The amount needed to maintain pressure is dependent on the heat losses.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	026 A2.03	
	Importance Rating		4.4

Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of ESF

Proposed Question: SRO 87

Given the following:

- A Main Steam Break has occurred on Unit 1.
- The Train "A" Load Sequencer is de-energized.
- "B" NS Pump did NOT automatically start.
- The crew has transitioned to E-2, Faulted Steam Generator Isolation when the following conditions are observed:
    - NC SYSTEM pressure 1400 psig and lowering.
    - Containment Pressure 11 psig and rising.

Enter FR-Z.1 based on a(n)...

A. ORANGE CSF Status Tree; Ensure NC Pumps are off and start at least ONE NS Pump; procedure may subsequently be completed as time allows.

B. ORANGE CSF Status Tree; Perform all actions of FR-Z.1 and do NOT perform actions of other procedures unless a higher priority ORANGE or RED condition occurs.

C. RED CSF Status Tree; Ensure NC Pumps are off and start at least ONE NS Pump; procedure may subsequently be completed as time allows.

D. RED CSF Status Tree; Perform all actions of FR-Z.1 and do NOT perform actions of other procedures unless a higher priority RED condition occurs.

Proposed Answer: A

Explanation (Optional):

A. Correct. After step 8, procedure is treated as a yellow path for conditions such as a steam line break. This is determined by the SRO
B. Incorrect. SRO should know that a steam break is occurring and note will apply that procedure may be treated as a yellow path after initial actions are performed
C. Incorrect. Red path is 15 psig, but actions are correct
D. Incorrect. Red path is 15 psig and procedure is treated as a yellow path after step 8

Technical Reference(s): FR-Z.1 (Rev 14); EP-FRZ Rev 15; OMP 4-3 p17, 18 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-FRZ Obj 2 & 4 (As available)

Question Source:
- Bank # ____
- Modified Bank # ____ (Note changes or attach parent)
- New X

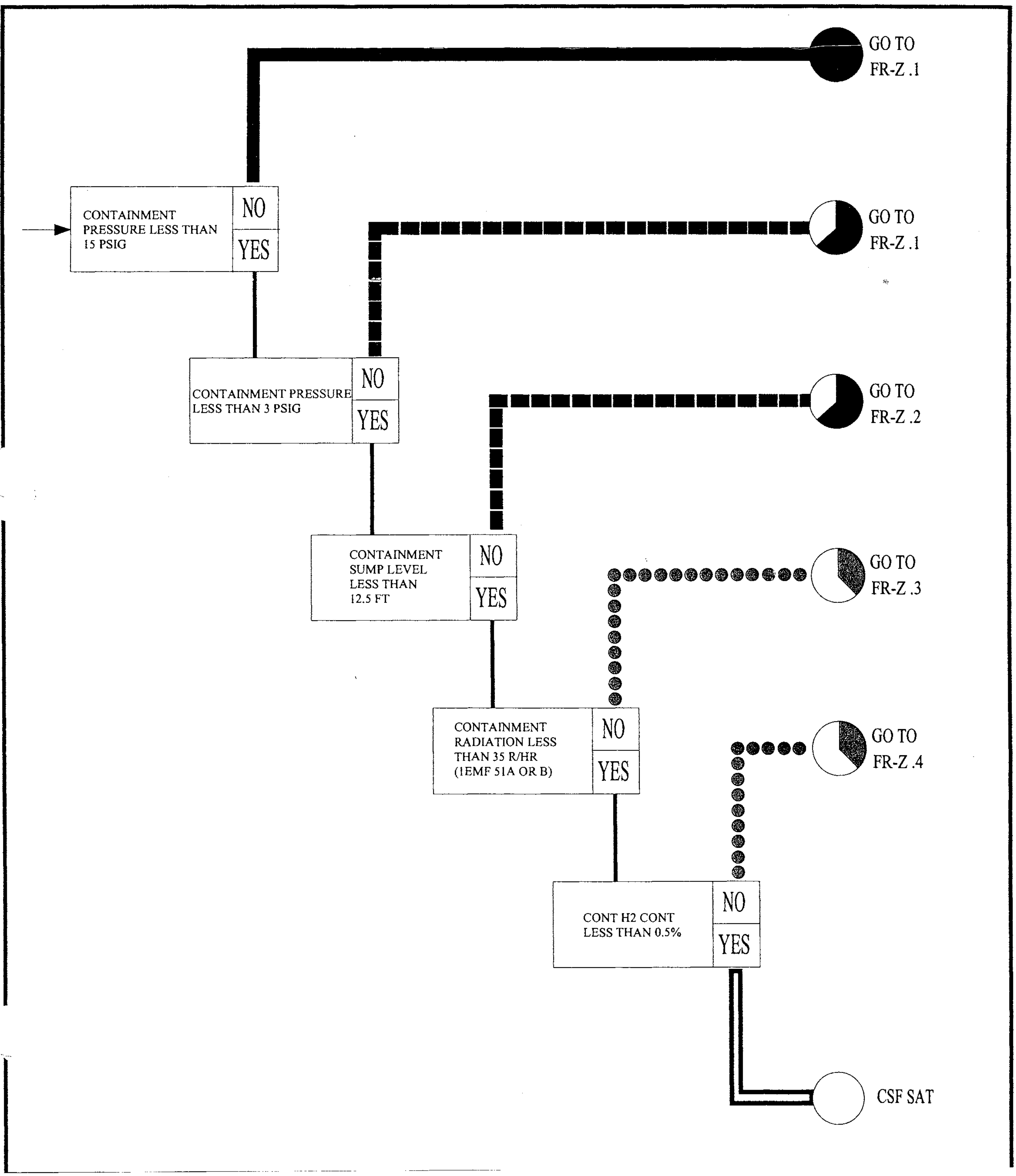
Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 5

Comments:

KA is matched because a containment spray failure has occurred. The impact is the result on CSF status, and the action required is also tested. SRO level because the SRO must select the appropriate strategy for procedure use, including a judgment of when the Containment Orange condition may be treated as a yellow condition



GO TO
FR-Z .1
CONTAINMENT PRESSURE LESS THAN 15 PSIG
NO
YES
GO TO
FR-Z .1
CONTAINMENT PRESSURE LESS THAN 3 PSIG
NO
YES
GO TO
FR-Z .2
CONTAINMENT SUMP LEVEL LESS THAN 12.5 FT
NO
YES
GO TO
FR-Z .3
CONTAINMENT RADIATION LESS THAN 35 R/HR (1EMF 51A OR B)
NO
YES
GO TO
FR-Z .4
CONT H2 CONT LESS THAN 0.5%
NO
YES
CSF SAT

A. Purpose

This procedure provides actions to respond to a high containment pressure.

B. Symptoms or Entry Conditions

This procedure is entered from EP/1/A/5000/F-0 (Critical Safety Function Status Trees) (Containment), on a red or orange condition.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **IF loss of emergency coolant recirc has occurred, THEN this procedure may be completed as time allows.**	
__ 2. **Monitor Foldout Page.**	
__ 3. **Stop all NC pumps.**	
__ 4. **Ensure all RV pumps are in manual and off.**	

CAUTION **The following breakers must be closed within 50 minutes of S/I.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **Dispatch operator to remove white tags and close the following breakers:** __ • 1EMXA-R2A (1A ND To A&B Cold Legs Cont Outside Isol Motor (1NI-173A)) (aux bldg, 750, FF-54, FF-55) __ • 1EMXB1-6B (1B ND To C&D NC Cold Leg Cont Outside Isol Motor (1NI-178B)) (aux bldg, 733, GG-55, GG-56).	
__ 6. **Check containment pressure - LESS THAN 15 PSIG.**	__ **GO TO Step 9.**
__ 7. **Check any NS pump - ON.**	__ **GO TO Step 9.**

NOTE The remainder of this EP may be completed with the priority of a yellow path EP. Completion of this EP should be delayed if faulted S/G has occurred, or other higher priority actions are required.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 8. **Perform the remainder of this EP as time allows.**	

STEP 3 Stop all NC Pumps.

STEP 4 Stop all RV Pumps.

PURPOSE: To stop all NC and RV pumps.

BASIS: The NC pumps are tripped since component cooling water to the NC pump seals and motors is isolated by the Phase B containment isolation. RV pumps are tripped because the suction is isolated by the Phase B containment isolation signal.

STEP 5 Dispatch operator to remove tags and close breakers for the following valves:

- **NI-173A (Train A ND to A&B Cold Leg)**
- **NI-178B (Train B ND to C&D Cold Leg)**

PURPOSE: To prepare the ND Aux containment spray system for use if needed.

BASIS: This step allows ND Aux containment spray to be able to be aligned in subsequent steps/procedures at the 50 minute requirement of the FSAR.

STEP 6 Check Containment Pressure - less 15 psig

STEP 7 Check any NS pump - ON

STEP 8 The remainder of this EP may be performed as time allows.

PURPOSE: Allows crew to perform other procedures in a more timely manner provided containment pressure is less than 15 psig and any NS pump is on.

BASIS: The specific scenario the WOG had in mind for this allowance is a steam break inside containment. This will aid in terminating SI prior to going solid in the pressurizer for this scenario.

If there are no other priority actions to complete, you may as well finish FR-Z.1 and get it out of the way. An example would be a large break LOCA. There is little to do until you need to transfer to Cold Leg Recirc.

7.15.1 Implementing CSF Path Procedures

7.15.1.1 CSF procedures are **NOT** to be implemented prior to transition from EP/1,2/A/5000/E-0 (Reactor Trip or Safety Injection). **IF** a CSF path is red or orange while the operating crew is in EP/1,2/A/5000/E-0, but has turned to green upon transition from E-0, the CSF procedure which was in alarm shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7. **IF** there are any valid red or orange path CSF's on transition from E-0 (unless transition is to EP/1,2/A/5000/ECA-0 (Loss of All AC Power), the associated CSF procedure shall be implemented.

7.15.1.2 **IF** a valid red or orange path flickers into alarm on SPDS but returns to green prior to the crew validating the condition and implementing the procedure (implementation of procedure being that the SRO either hands out fold-out pages or starts reading from the procedure), the CSF procedure shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7. Likewise, if a valid red path or orange path goes into alarm during performance of a higher priority CSF procedure, but returns to green prior to transition from the higher priority CSF path procedure to the lower priority CSF procedure, the associated CSF procedure shall **NOT** be implemented.

7.15.1.3 **IF** a CSF procedure directs the operator to return to the procedure and step in effect, **AND** the corresponding status tree continues to display the offnormal conditions, the corresponding CSF procedure does **NOT** have to be implemented again, since all recovery actions have been completed. However, if the same status tree subsequently changes to a valid higher priority condition, **OR** if it changes to lower condition and returns to higher priority condition again, the corresponding CSF procedure shall be implemented as required by its priority.

7.15.1.4 Red Path

IF any valid red path is encountered during monitoring, the operator is required to immediately implement the corresponding EP. Any recovery EP previously in progress shall be discontinued. **IF** during the performance of any red path procedure, a valid red condition of higher priority arises, the higher priority condition should be addressed first, and the lower priority red path procedure suspended.

7.15.1.5 Orange Path

IF any valid orange path is encountered, the operator is expected to scan all of the remaining trees, and then, if no valid red is encountered, promptly implement the corresponding EP. **IF** during the performance of an orange path procedure, any valid red condition or higher priority valid orange condition arises, the red or higher priority orange condition is to be addressed first, and the original orange path procedure suspended.

7.15.1.6 Completion of Red or Orange Path Procedure

Once procedure is entered due to a red or orange condition, that procedure should be performed to completion, unless preempted by some higher priority condition. It is expected that the actions in the procedure will clear the red or orange condition before all the operator actions are complete. However, these procedures should be performed to the point of the defined transition to a specific procedure or to the "procedure and step in effect" to ensure the condition remains clear. At this point any lower priority red or orange paths currently indicating or previously started but **NOT** completed shall be addressed.

FR-S.1, P.1 and Z.1 can be entered from either an orange or red path status. **IF** the color changes from orange to red while you are in one of these EPs, the crew should continue and complete the EP from where they are. Crew does **NOT** have to backup and restart the EP. **IF** the orange path is exited, and it subsequently turns red, the EP must be re-entered at Step 1.

Upon continuation of recovery actions in Optimal Recovery procedure, some judgment may be required by the operator to avoid inadvertent reinstatement of a Red or Orange condition by undoing some critical step in the Function Recovery procedure. The Optimal Recovery procedures are optimal assuming that safety equipment is available. The appearance of a Red or Orange condition in most cases implies that some equipment or function required for safety is **NOT** available, and by implication some adjustment may be required in the Optimal Recovery procedure.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		0.5	0.5	0.5

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose of each procedure in the FR-Z series.			X	X	
2	Discuss the entry and exit guidance for each procedure in the FR-Z series.			X	X	
3	Discuss the mitigating strategy (major actions) of each procedure in the FR-Z series.			X	X	X
4	Discuss the basis for any note, caution or step for each procedure in the FR-Z series.			X	X	X
5	Given the Foldout page, discuss the actions included and the basis for these actions.			X	X	X
6	Given the appropriate procedure, evaluate a given scenario describing accident events and plant conditions to determine any required action and its basis.			X	X	X
7	Discuss the time critical task(s) associated with the FR-Z series procedures including the time requirements and the basis for these requirements.			X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	AW 061 G2.4.4	
	Importance Rating		4.7

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: SRO 88

Given the following conditions:

- A Reactor Trip with SI occurs.
- The operators perform the immediate action steps, verify SI flow, and check CA flow in accordance with EP/1/A/5000/E-0, Reactor Trip or Safety Injection.
- The RO reports all 3 CA pumps are off.
- NCS pressure is 900 psig.
- All SG pressures are between 825 psig and 850 psig.
- All SG NR levels are off scale low.
- All SG WR levels are approximately 39%.
- E-0 directs the crew to implement EP/1/A/5000/F-0, Critical Safety Function Status Trees.

Which ONE (1) of the following actions is to be taken?

A. Transition to FR-H.1. "Response to Loss of Secondary Heat Sink," and attempt to establish CA or Feedwater flow.

B. Transition to FR-H.1, "Response to Loss of Secondary Heat Sink," and initiate NCS feed and bleed.

C. Transition to FR-H.1, "Response to Loss of Secondary Heat Sink," and then return to "procedure and step in effect" since a secondary heat sink is NOT required.

D. Remain in EP-E.0, Reactor Trip or Safety Injection, until directed to EP-E.1, Loss of Reactor or Secondary Coolant since a secondary heat sink is NOT required.

Answer: A

Explanation (Optional):
a. Correct. NC pressure is higher than SG pressure, therefore, use H.1
b. Plausible since these are actions that might be taken upon entry into FR- H.1. but SG levels do not meet the criteria. (24%, 36% ACC)
c. Incorrect. Since NCS pressure is higher than SG pressure, a secondary heat sink is required.
d. Incorrect. Plausible since a LOCA is in progress, and the only criteria making this incorrect is that NC pressure is higher than SG pressure

Technical Reference(s): FR- H.1 page 2 (Rev 1) (Attach if not previously provided)
E-0, Rev 24
EP-FRH Rev 10

references to be provided to applicants during examination: None

Learning Objective: EP-FRH Obj 2, 3, 4 (As available)

Question Source: Bank #
Modified Bank # EPFRHN011 (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

Comments:
Modified 2007 SRO Retake #80 conditions and answer. Also modified distractor D

KA is matched because a failure of AFW for these conditions results in entry to FR-H.1. SRO level because the SRO is required to evaluate procedure selection as well as strategy for the condition presented

Given the following conditions:

- A Reactor Trip with SI occurs.
- The operators perform the immediate action steps, verify SI flow, and check CA flow in accordance with EP/1/A/5000/E-0, Reactor Trip or Safety Injection.
- The RO reports all 3 CA pumps are off
- NCS pressure is 400 psig.
- All SG pressures are between 425 psig and 450 psig.
- All SG levels are 5% NR
- E-0 directs the crew to implement EP/1/A/5000/F-0, Critical Safety Function Status Trees.

Which ONE (1) of the following actions is to be taken?

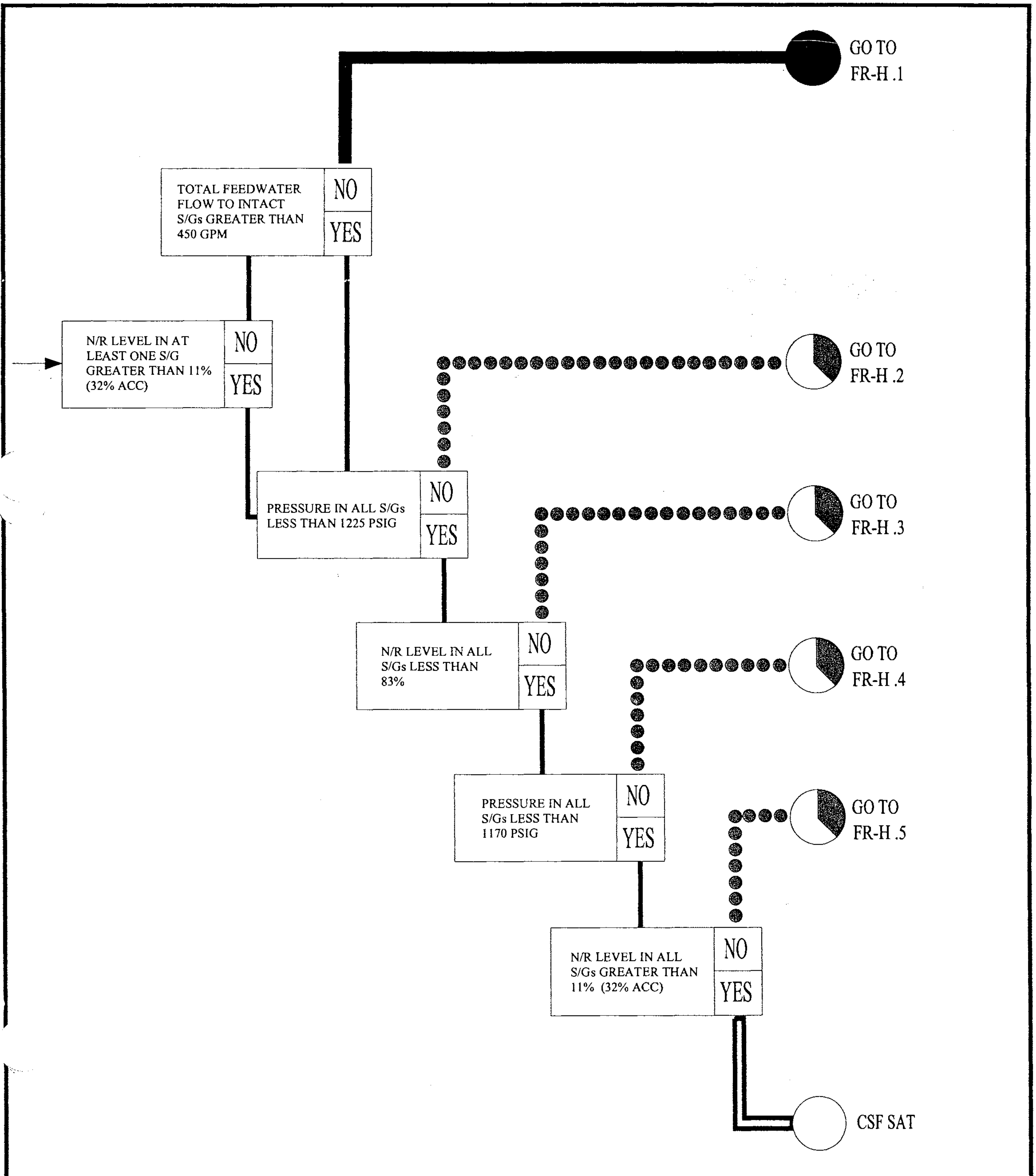
A. Transition to FR-H.1. "Response to Loss of Secondary Heat Sink," and attempt to establish CA or Feedwater flow.

B. Transition to FR-H.1, "Response to Loss of Secondary Heat Sink," and initiate NCS feed and bleed.

C. Transition to FR-H.1, "Response to Loss of Secondary Heat Sink," and then return to "procedure and step in effect" since a secondary heat sink is NOT required.

D. Remain in EP-E.0, Reactor Trip or Safety Injection since a secondary heat sink is NOT required

Ans C



TOTAL FEEDWATER FLOW TO INTACT S/Gs GREATER THAN 450 GPM
NO
YES
GO TO FR-H .1
N/R LEVEL IN AT LEAST ONE S/G GREATER THAN 11% (32% ACC)
NO
YES
GO TO FR-H .2
PRESSURE IN ALL S/Gs LESS THAN 1225 PSIG
NO
YES
GO TO FR-H .3
N/R LEVEL IN ALL S/Gs LESS THAN 83%
NO
YES
GO TO FR-H .4
PRESSURE IN ALL S/Gs LESS THAN 1170 PSIG
NO
YES
GO TO FR-H .5
N/R LEVEL IN ALL S/Gs GREATER THAN 11% (32% ACC)
NO
YES
CSF SAT

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

16. **Check CA flow:**

__ a. Total CA flow - GREATER THAN 450 GPM.

a. Perform the following:

1) **IF** N/R level in all S/Gs is less than 11% (32% ACC), **THEN**:

__ • Ensure correct valve alignment

__ • Start CA pumps.

2) **IF** N/R level in all S/Gs is less than 11% (32% ACC) **AND** feed flow greater than 450 GPM can not be established, **THEN**:

__ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees).

__ • **GO TO** EP/1/A/5000/FR-H.1 (Response To Loss Of Secondary Heat Sink).

__ b. Check VI header pressure - GREATER THAN 60 PSIG.

__ b. **IF** CA flow can not be throttled with CA control valves in subsequent steps, **THEN** control flow **PER** EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 16 (CA Flow Control With Loss of VI).

__ c. **WHEN** N/R level in any S/G greater than 11% (32% ACC), **THEN** control CA flow to maintain N/R levels between 11% (32% ACC) and 50%.

A. Purpose

This procedure provides actions to respond to a loss of secondary heat sink in all steam generators.

B. Symptoms or Entry Conditions

This procedure is entered from:

- EP/1/A/5000/E-0 (Reactor Trip Or Safety Injection), Step 16, when minimum CA flow is not verified **AND** N/R level in all S/Gs is less than 11% (32% ACC).
- EP/1/A/5000/F-0 (Critical Safety Function Status Trees) (Heat Sink), on a red condition.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. **Check at least one of the following NV pumps - AVAILABLE:** __ • 1A NV pump OR __ • 1B NV pump.	__ **GO TO Step 20.**
5. **Check if NC System feed and bleed should be initiated:**	
__ a. Check W/R level in at least 3 S/Gs - LESS THAN 24% (36% ACC).	a. Perform the following: __ 1) Monitor feed and bleed initiation criteria. __ 2) **WHEN** criteria satisfied, **THEN GO TO** Step 20. __ 3) **GO TO** Step 6.
__ b. **GO TO** Step 20.	
__ 6. **Ensure S/G BB and NM valves closed PER Enclosure 3 (S/G BB and Sampling Valve Checklist).**	
7. **Attempt to establish CA flow to at least one S/G as follows:**	
__ a. Check power to both motor driven CA pumps - AVAILABLE.	a. Perform the following: __ • **IF** essential power is not available, **THEN** restore power to the affected essential bus **PER** AP/1/A/5500/07 (Loss of Electrical Power). __ • **IF** the essential bus is energized, **THEN** dispatch operator to determine cause of breaker failure.
__ b. Ensure control room CA valves aligned **PER** Enclosure 4 (CA Valve Alignment).	
__ c. Start all available CA pumps.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11. **Check CM System in service:** __ • Hotwell pump(s) - ON __ • Condensate Booster pump(s) - ON.	**Perform the following:** __ a. IF CM System is not available to be placed in service, **THEN GO TO** Step 19. **NOTE** Hotwell and Condensate Booster pump will be started in operating procedure. Once these pumps are on, this EP provides steps to start available CF pump(s). __ b. Place CM System in service **PER** OP/1/A/6250/001 (Condensate And Feedwater System), Enclosure 4.2 (CM System Hot Restart). __ c. Do not continue until condensate booster pump is on.
__ 12. **Check CF pumps - AT LEAST ONE AVAILABLE TO START.**	__ **IF both CF pumps are known to be incapable of starting, THEN GO TO Step 15.**

NOTE If it appears that CF pump might not be restored prior to reaching feed and bleed criteria, it may be preferable to hand off Enclosure 7 (Reestablishing CF Flow) to another SRO and/or RO while continuing with subsequent steps.

__ 13. **Establish CF flow PER Enclosure 7 (Reestablishing CF Flow).**	__ **GO TO Step 15.**

1. **Cold Leg Recirc Switchover Criteria:**

   - IF FWST level reaches 180 inches ("FWST LEVEL LO" alarm), **THEN GO TO** EP/1/A/5000/ES-1.3 (Transfer To Cold Leg Recirc).

2. **CA Suction Sources:**

   - IF CA storage tank (water tower) goes below 1.5 ft, **THEN** perform EP/1/A/5000/G-1 (Generic Enclosures), Enclosure 20 (CA Suction Source Realignment).

3. **NC System Feed and Bleed Criteria (Applies after Step 2 in the body of the procedure):**

   - IF W/R level in at least 3 S/Gs goes below 24% (36% ACC), **THEN GO TO** Step 20 in the body of the procedure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

__ 1. **IF total feed flow is less than 450 GPM due to operator action, THEN RETURN TO procedure and step in effect.**

CAUTION **If a non-faulted S/G is available, then feed flow should only be established to non-faulted S/G(s) in subsequent steps.**

2. **Check if secondary heat sink is required:**

__ a. NC pressure - GREATER THAN ANY NON-FAULTED S/G PRESSURE.

RNO: __ a. **RETURN TO** procedure and step in effect.

__ b. Any NC T-Hot - GREATER THAN 350°F (347°F ACC).

RNO: b. Perform the following while continuing in this procedure:

1) Try to place ND in RHR mode:

__ a) Ensure NC pressure is less than 385 PSIG.

__ b) **IF** S/I has occurred, **THEN** place ND in RHR mode **PER** EP/1/A/5000/G-2 (Placing ND In RHR Mode).

__ c) **IF** S/I has not occurred, **THEN** place ND in RHR mode **PER** Enclosure 2 (Placing ND in RHR mode).

__ 2) **WHEN** adequate ND cooling is established, **THEN RETURN TO** procedure and step in effect.

__ 3. **Monitor Foldout Page.**

CAUTION If a non-faulted S/G is available, then feed flow should only be established to non-faulted S/Gs in subsequent steps.

PURPOSE: To alert the operator to not reestablish feed flow to a faulted S/G if an intact or ruptured S/G is available to receive the feed flow.

BASIS: Reestablishment of feed flow to a S/G may result in thermal or mechanical shocks to the S/G tubes that could result in tube leakage or tube rupture. If feed flow is reestablished to a faulted S/G and tube leakage resulted, control of the leakage would not be possible until the S/G secondary boundary was restored. Flow restoration to a non-faulted S/G will provide an effective and controllable secondary heat sink.

STEP 2 Check if a secondary heat sink is required:

PURPOSE: To check if a secondary heat sink is required for heat removal.

BASIS: Before implementing actions to restore flow to the S/Gs, the operator should check if secondary heat sink is required. For larger LOCA break sizes, the NC system will depressurize below the intact S/G pressures. The S/Gs no longer function as a heat sink and the core decay heat is removed by the break flow. For this range of LOCA break sizes, the secondary heat sink is not required and actions to restore secondary heat sink are not necessary. For these cases, the operator returns to the procedure and step in effect.

Since Step 19 directs the operator to return to Step 1 if the loss of secondary heat sink parameters are not exceeded, break sizes that take longer to depressurize the NC system will be detected on subsequent passes through Step 1.

If NC system temperature is low enough to place the ND system in service in RHR mode, then the ND system is an alternate heat sink to the secondary system. Therefore, an attempt is made to place the ND system in service (Enclosure 2, Placing ND In RHR Mode) in parallel to the attempts to reestablish feedwater flow. NC system pressure must be below normal ND system pressure limits. When adequate ND cooling is established, then the operator is directed to return to the procedure and step in effect.

Generic Enclosure G-2 (Placing ND in RHR Mode) contains guidance to align one, or both trains, of ND in RHR Mode; leaving one, or no train, available for auto swap to sump; or leaving one train on sump and one train in RHR mode.

The decision for alignment will be made with concurrence/guidance from TSC, if available.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		3	3	2.5

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose of each procedure in the FR-H series. EPFRH001			X	X	
2	Discuss the entry and exit guidance for each procedure in the FR-H series. EPFRH002			X	X	
3	Discuss the mitigating strategy (major actions) of each procedure in the FR-H series. EPFRH003			X	X	X
4	Discuss the basis for any note, caution or step for each procedure in the FR-H series. EPFRH004			X	X	X
5	Given the Foldout page, discuss the actions included and the basis for these actions. EPFRH005			X	X	X
6	Given the appropriate procedure, evaluate a given scenario describing accident events and plant conditions to determine any required action and its basis. EPFRH006			X	X	X
7	Discuss the time critical task(s) associated with the FR-H series procedures including the time requirements and the basis for these requirements. EPFRH007			X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	073 G2.2.22	
	Importance Rating		4.7

Equipment Control: Knowledge of limiting conditions for operations and safety limits

Proposed Question: SRO 89

Which ONE of the following describes (1) the MINIMUM radiation monitor requirement that provides the preferred means of NCS primary to secondary leak rate monitoring in accordance with technical specification surveillance requirements, and (2) the MINIMUM sensitivity required to ensure the monitor remains OPERABLE, in accordance with SLC and bases?

A. (1) EMF-33, Condenser Evacuation Monitor **OR** N-16 Monitors, EMF-71 – EMF-74;
(2) 75 GPD.

B. (1) EMF-33, Condenser Evacuation Monitor **OR** N-16 Monitors, EMF-71 – EMF-74;
(2) 30 GPD.

C. (1) EMF-33, Condenser Evacuation Monitor **AND** N-16 Monitors, EMF-71 – EMF-74;
(2) 75 GPD.

D. (1) EMF-33, Condenser Evacuation Monitor **AND** N-16 Monitors, EMF-71 – EMF-74;
(2) 30 GPD.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. 75 GPD is leakage defined by action in AP, but minimum monitors are correct

B. Correct.

C. Incorrect. Not all, just 'either or'. If EMF33 is operable, then the MS line monitors are not required to be operable, and vice-versa.

D. Incorrect. OR statement would make this correct, because minimum detectable requirement is correct

Technical Reference(s)	SLC 16.7.6, Rev 99	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: WE-EMF Obj 10 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

KA is matched because the SLC LCO for process radiation monitoring is being evaluated, and further SRO knowledge is evaluated because the SRO must know basis for operability of the detectors

16.7 INSTRUMENTATION

16.7.6 Radiation Monitoring for Plant Operations

COMMITMENT The radiation monitoring instrumentation channels shown in Table 16.7.6-1 shall be OPERABLE.

APPLICABILITY As shown in Table 16.7.6-1.

REMEDIAL ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more radiation monitoring channels Alarm/Trip setpoint exceeding value shown in Table 16.7.6-1.	A.1 Adjust setpoint to within the limit. OR A.2 Declare the channel inoperable.	4 hours 4 hours
B. One Containment Atmosphere Gaseous Radioactivity monitoring channel inoperable.	B.1 Verify containment purge system (VP) valves are maintained closed.	Immediately
C. One Control Room Air Intake Radioactivity monitoring channel inoperable.	C.1 Isolate the associated Control Room Ventilation System (VC) outside air intake.	1 hour

(continued)

REMEDIAL ACTIONS (continued)

CONDITION		REQUIRED ACTION		COMPLETION TIME
D.	One or more required channels for Spent Fuel Handling Area, Reactor Building Fuel Handling Area or New Fuel Vault Fuel Handling Area Radiation Monitors inoperable.	D.1	Suspend all fuel movement operations in the fuel handling area being monitored until Required Acton D.2 is completed.	Immediately
		AND		
		D.2.1	Provide a portable continuous monitor with same Alarm Setpoint.	Immediately
		OR		
		D.2.2	Provide RP continuous dose rate monitoring.	Immediately
		AND		
		D.3	Restore inoperable monitors to OPERABLE status.	30 days
E.	One Spent Fuel Pool Radioactivity monitoring channel inoperable.	E.1	Verify the Fuel Handling Ventilation System (VF) requirements in Technical Specification 3.7.12 are met.	Immediately
F.	Condenser Evacuation System Noble Gas Activity Monitor (EMF-33) inoperable.	F.1	Ensure that all N-16 Leakage Monitor (EMF-71, 72, 73, & 74) channels are OPERABLE.	Immediately
G.	One or more N-16 Leakage Monitor (EMF-71, 72, 73, & 74) channels inoperable.	G.1	Ensure that the Condenser Evacuation System Noble Gas Activity Monitor (EMF-33) is OPERABLE.	Immediately

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Condenser Evacuation System Noble Gas Activity Monitor (EMF-33) inoperable. <u>AND</u> One or more N-16 Leakage Monitor (EMF-71, 72, 73, & 74) channels inoperable.	H.1 Initiate action to restore online radiation monitor to operable. <u>AND</u> H.2 Perform TS-SR 3.4.13.2.	Immediately 72 hours

TESTING REQUIREMENTS

---NOTE---

Refer to Table 16.7.6-1 to determine which TRs apply for each Radiation Monitoring channel.

TEST	FREQUENCY
TR 16.7.6.1 Perform CHANNEL CHECK.	12 hours
TR 16.7.6.2 Perform CHANNEL OPERATIONAL TEST.	92 days
TR 16.7.6.3 Perform CHANNEL OPERATIONAL TEST.	184 days
TR 16.7.6.4 Perform a CHANNEL CALIBRATION.	18 months

TABLE 16.7.6-1

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATION

MONITOR	APPLICABLE MODES	REQUIRED CHANNELS	ALARM/TRIP SETPOINT	TESTING REQUIREMENTS
1. Containment Atmosphere Gaseous Radioactivity-High (Low Range EMF-39)	1,2,3,4,5,6	1	Must meet SLC 16.11-6 limits	TR 16.7.6.1 TR 16.7.6.2 TR 16.7.6.4
2. Spent Fuel Pool Radioactivity-High (EMF-42)	With irradiated fuel in fuel storage areas or fuel building	1	≤ 1.7 x 10-4 μCi/ml	TR 16.7.6.1 TR 16.7.6.2 TR 16.7.6.4
3. Spent Fuel Handling Area Radiation Monitor (1EMF-17, 2EMF-4)	With fuel in fuel storage areas or fuel building	1	≤ 15 mR/hr See Note (b)	TR 16.7.6.1 TR 16.7.6.3 TR 16.7.6.4
4. Reactor Building Fuel Handling Area Radiation Monitor (1EMF-16, 2EMF-3)	6	1	≤ 15 mR/hr See Note (b)	TR 16.7.6.1 TR 16.7.6.3 TR 16.7.6.4
5. New Fuel Vault Fuel Handling Area Radiation Monitors (1EMF-20, 1EMF-21, 2EMF-7, 2EMF-8)	With fuel in New Fuel Vault	1	≤ 15 mR/hr See Note (b)	TR 16.7.6.1 TR 16.7.6.3 TR 16.7.6.4
6. Control Room Air Intake Radioactivity-High (EMF-43a and 43b)	1,2,3,4,5,6	2 per station.	≤ 3.4 x 10-4 μCi/ml	TR 16.7.6.1 TR 16.7.6.2 TR 16.7.6.4
7. Condenser Evacuation System Noble Gas Activity Monitor (EMF-33)	1	1	See Note (a)	TR 16.7.6.1 TR 16.7.6.3 TR 16.7.6.4
8. N-16 Leakage Monitor (EMF-71, 72, 73 & 74)	1 (40-100% reactor power)	4 (1/steamline)	See Note (a)	TR 16.7.6.1 TR 15.7.6.3 TR 16.7.6.4

(a) The setpoint is as required by the primary to secondary leak rate monitoring program.
(b) Setpoint can be elevated above 15 mR/hr based upon direction from approved station procedures.

BASES

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

The condenser evacuation system noble gas activity monitor (EMF-33) and main steam line N-16 monitors (EMF- 71, 72, 73, & 74) are used for online monitoring of primary-to-secondary leak rate. These radiation monitors provide the preferred means to accomplish Technical Specification Surveillance SR 3.4.13.2 while in Mode 1. For the condenser evaluation system noble gas activity monitor (EMF-33) or main steam line N-16 monitor to be considered operable for primary to secondary leakage monitoring the monitor must be sensitive to a least 30 gallons per day (GPD) leakage rate.

Fuel assemblies are stored and handled in areas of the plant discussed below. Radiation monitoring is provided for these areas to detect excessive radiation levels and will provide an alarm to alert personnel if a potential radiation hazard is present.

1. Unit 1 and 2 Spent Fuel Pool; includes the cask pool area, the new fuel elevator, the fuel transfer tube area and the spent fuel storage are/racks.

2. Unit 1 and 2 Reactor Building; includes the fuel transfer tube area, the reactor core and the refueling canal.

3. Unit 1 and 2 Fuel Building; includes the new fuel vault area.

Performance of Required Acton D.1 shall not preclude completion of movement of a component to a safe position. When a fuel handling area radiation monitor channel becomes inoperable, an alternate means is required for determining dose rate and alerting individuals to excessive radiation levels. This can be accomplished by either a portable monitor with same alarm setpoint located within the area monitored by the inoperable channel or using Radiation Protection personnel performing continuous monitoring of area dose rate using a hand-held dose rate meter. This hand-held meter will not provide an alarm, but relies upon RP personnel to alert individuals of excessive radiation levels.

Certain evolutions may result in a higher gamma dose rate field, resulting in the need to adjust the alarm setpoint above the nominal alarm/trip setpoint (15 mR/hr). An approved station procedure controls adjustment of this setpont to a higher value that still ensures individuals are alerted to the presence of excessive radiation levels.

REFERENCES

1. Technical Specification 3.4.13 - RCS Operational Leakage.
2. NSD-513 - Primary to Secondary Leak Monitoring Program, Revision 5.
3. 10CFR50.68 - Criticality Accident Requirements
4. Duke letter dates July 29, 2004 - RAI Response, TS 3.7.15 and TS 4.3 Changes.
5. NRC Safety Evaluation Report dated March 17, 2005 - Amendments Nos. 225/207

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
10	Concerning the Technical Specifications / SLCs related to the EMFs:					
	• Given the LCO or SLC title, state the LCO / commitment (including any COLR values) and applicability.			X	X	X
	• For any LCOs / SLCs that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCOs or SLCs is(are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of parameter values or system conditions and the appropriate Tech Spec / SLC, determine required action(s).			X	X	X
	• Discuss the bases for a given Tech. Spec. LCO or SLC.				X	*
	*** SRO ONLY**					
	WEEMF010					

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	076 A2.02	
	Importance Rating		3.1

Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Service water header pressure

Proposed Question: SRO 90

Given the following conditions:

- A plant cooldown is in progress.
- Current conditions are:
    - NC Pressure – 1400 psig
    - NC Temperature – 440 degrees F
    - Cold Leg Accumulators have **NOT** been isolated
    - "B" Train in service

An event occurs:

- NC System pressure starts to go down at approximately 2 psi per minute.
- PZR level is going down at 5% per minute.
- Containment Pressure is rising at 0.1 psig per minute.
- Train "B" Safety Injection actuates.
- Train "A" Safety Injection did NOT actuate.

Which ONE (1) of the following describes (1) the impact on the unit, and (2) the action that must be taken?

A. (1) NC Pumps will overheat due to loss of RN cooling
(2) Enter E-0, Reactor Trip or Safety Injection, and initiate Train A Safety Injection to restore flow to Train A Essential Header and RB Non-Essential Header

B. (1) "A" EDG will overheat due to loss of RN cooling
(2) Enter E-0, Reactor Trip or Safety Injection; Reset SI Sequencers and open RN Cross-Connect valves to restore Train A Essential header

C. (1) NC Pumps will overheat due to loss of RN cooling
(2) Enter AP-34, Shutdown LOCA, and initiate Train A Safety Injection to restore flow to Train A Essential Header and RB Non-Essential Header

D. (1) "A" EDG will overheat due to loss of RN cooling
(2) Enter AP-34, Shutdown LOCA; Reset SI Sequencers and open RN Cross-Connect valves to restore Train A Essential header

Proposed Answer: A

Explanation (Optional):

A is correct.

B is incorrect. Correct procedure to enter; do not open RN cross connect valves on a valid SI signal, and even if the action was performed, one train would not operate since the sequencer has not actuated

C is incorrect. Credible because the procedure would be entered in Mode 4 if NC pressure was lower. Action to restore RN is correct though

D is incorrect. Wrong procedure as in C above. Also wrong action. If both sequencers were actuated, the action could work, but not performed for valid SI

Technical Reference(s): OMP 4-3, p8 Rev 26 (Attach if not previously provided)

E-0 step 5 Rev 24; AP-34 Rev 13

DG-EQB Rev 16

ECC-CLA Rev 28

EP-E0 Rev 12

Proposed references to be provided to applicants during examination: None

Learning Objective: DG-EQB Obj 6; PSS_RN Obj 16 (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 2007 McGuire

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

KA is matched because conditions represented by the stem indicate loss of header pressure on 1 header. SRO level because the applicant must assess plant conditions and determine procedure use based upon selected impact

Auxiliary Building RV loads:

Auxiliary Building Ventilation Units

Reactor Building RV loads:

Upper containment ventilation units

Lower containment ventilation units

2.5 Discharge Paths

Objective # 8

The normal RN system discharge path is to the RC crossover header which returns to Lake Norman. The SNSWP is also a discharge path however it is typically only used if the suction is also from the SNSWP to prevent undesirable changes to SNSWP level.

The VC/YC chillers' RN discharge headers have been modified so that they will normally discharge into the shared RN discharge headers. This will prevent having to declare a VC/YC train inoperable because the Unit 1A or 1B RN Essential header is isolated. The new flowpath will be the normally aligned path however, the old flowpath will still be available.

2.6 Valves

Objective # 12

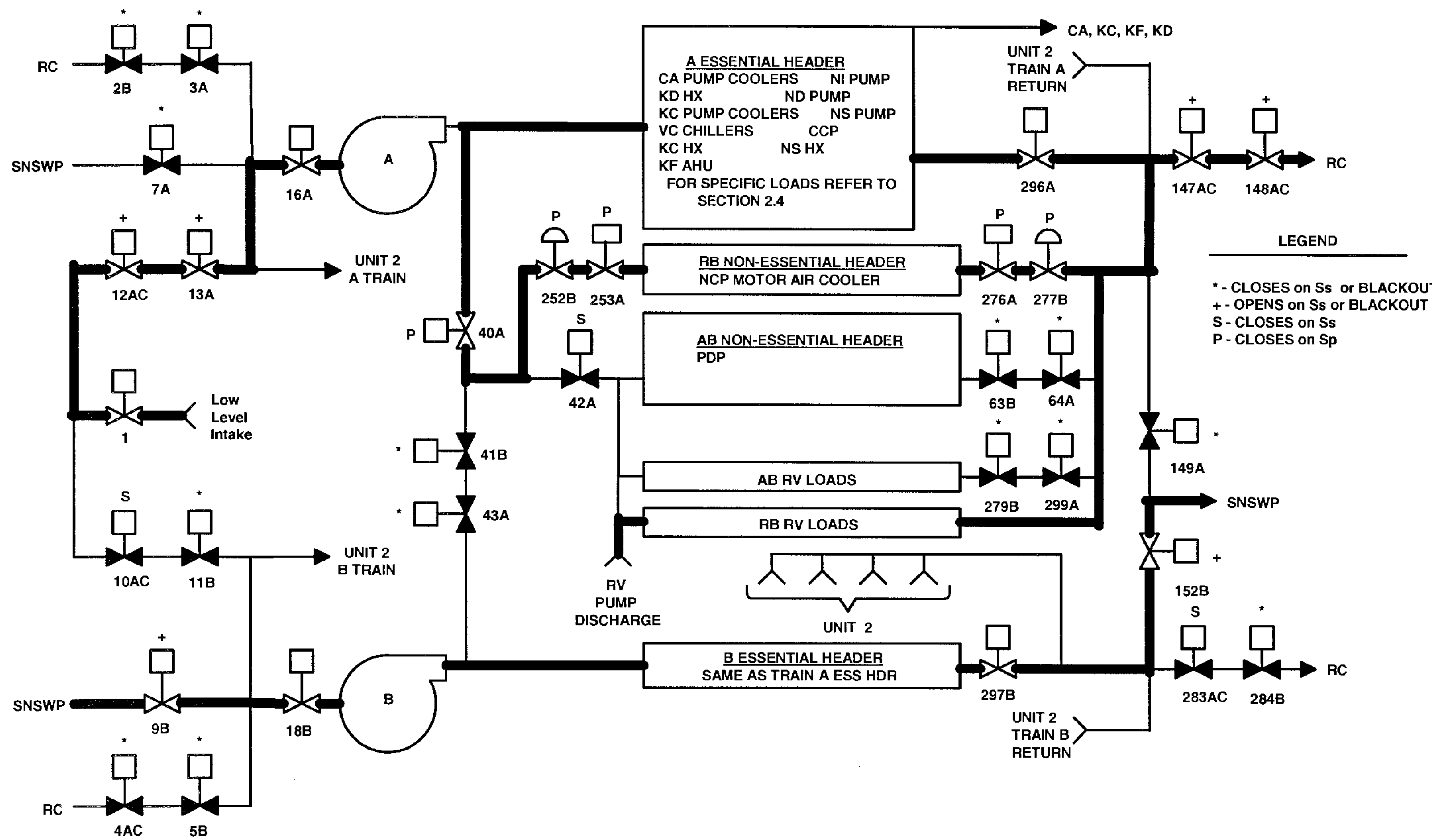
2.6.1 Blackout and Safety Injection Signals

The following is a listing of the various RN valves and how they respond to Safety Injection and/or Blackout signal(s). Valves which are shared between the units (0RN) can be powered and controlled from either unit. **(refer to Drawing 7.5)**

The following valves receive auto close signals upon receipt of either Unit 1 or 2 blackout or safety injection:

- 0RN-2B (Train 1A & 2A RC Supply)
- 0RN-3A (Train 1A & 2A RC Supply)
- 0RN- 4AC (Train 1B & 2B RC Supply)
- 0RN- 5B (Train 1B & 2B RC Supply)
- 0RN-7A (Train 1A & 2A SNSWP Supply)
- 0RN-149A (Train 1A & 2A Disch to SNSWP)
- 0RN-11B (Train 1B & 2B LLI Supply)
- 1RN-41B (Train 1B to Non-Ess Hdr Isol) Controlled only from Unit 1
- 1RN-43A (Train 1B to Non-Ess Hdr Isol) Controlled only from Unit 1
- 2RN-41B (Train 2B to Non-Ess Hdr Isol) Controlled only from Unit 2
- 2RN-43A (Train 2B to Non-Ess Hdr Isol) Controlled only from Unit 2
- 0RN-284B (Train 1B & 2B Disch to RC)

7.15, RN System Unit Safety Injection Loads and Valve Logic (03/30/06)



3.2.3 Safety Injection Alignment

On receipt of a **Safety Injection signal** basically the same automatic actuation occurs as after a blackout. The exceptions are that the supply to all nonessential equipment except the NC pump motor coolers and crossovers between essential trains are isolated. The "A" RN pump supplies Reactor Building non-essential header. The RV pumps will start automatically and supply the containment ventilation units if a blackout does not occur concurrently with the LOCA. **Drawings 7.14 and 7.15** provides the flow path for a unit safety injection.

NOTE: An S_S signal will affect both units suction, discharge and AB non-essential headers. Refer to Drawing 7.14

On receipt of a **Phase B isolation signal (S_P)** the RV pump suction is isolated to conserve water. The containment isolation valves close to isolate the NC pump motor coolers. All nonessential supply is isolated providing double isolation at this time between all essential and nonessential equipment. The NS heat exchanger inlet isolation valve is opened from the control room when required. During all modes of operation, water is available for assured makeup. **Drawings 7.16** provides the flow path following a unit safety injection with a phase B signal.

4.0 TECHNICAL SPECIFICATIONS

Objective # 17

4.1 Tech Spec 3.7.7 Nuclear Service Water System (NSWS)

4.2 Tech Spec 3.7.8 Standby Nuclear Service Water Pond (SNSWP)

3.2 Abnormal and Emergency Operation

3.2.1 Abnormal Procedure AP/1or2/A/5500/20

AP20 purpose, Cases, Symptoms, and basis for steps is covered thoroughly in the AP Lesson Plan.

Objective # 16

3.2.2 Blackout Alignment

Blackout is a loss of power to the 4160 vac bus. When the low voltage condition is detected, the D/G will start and the sequencer will load the Blackout loads onto the bus. On receipt of a **Blackout signal**, Train A valves automatically assume low level alignment; Train B assumes SNSWP alignment. Many shared valves receive signals from both units to prevent loss of water from SNSWP. Isolation valves for all heat exchangers which are needed open automatically and the train related RN pump will start. All nonessential discharge is isolated except the containment vent units and NC pump motor cooler discharge. The containment vent units and the NC pump motor coolers are supplied with cooling water from “A” RN pump. The "A" RN pumps supply the containment ventilation units with cooling water because they have more NPSH since their suction is aligned to the LLI and because the RV pumps may not have power. **Drawings 7.10 and 7.11** provides the unit blackout flow path. **Drawings 7.12 and 7.13** provides the flow path for Train A and Train B Blackout respectively.

If a Blackout occurs on the opposite unit, the non-blackout unit will have its non-essential header isolated from the B RN pump as a result of RN41B and RN43A closing **(Refer to Drawing 7.5)**. In order to supply the non-essential header on the non-blackout unit, the A Train RN pump must be started.

STEP 8 Check proper CA Pump status:

PURPOSE: To ensure proper status of the CA pumps.

BASIS: The MD CA pumps start automatically on an S/I signal to provide feed to the S/Gs for decay heat removal. If S/G levels drop below the appropriate setpoint, the TD CA pump will also automatically start to supplement the MD pumps.

STEP 9 & 10 Check all KC and both RN pumps - ON.

PURPOSE: To ensure KC and RN pumps are running.

BASIS: KC and RN pumps provide cooling to certain safeguards components.

STEP 11 Notify Unit 2 to start 2A RN Pump.

PURPOSE: To ensure required cooling.

BASIS: Both units' RN train cross ties close on a single unit S/I. If B train was previously feeding the reactor building headers on opposite unit, starting opposite unit's A RN pump ensures the reactor building headers remain cooled (only A train is aligned to reactor building headers following an S/I). Note that RV will continue to cool the opposite units reactor building headers, unless RV suction was isolated by a Phase B signal. Even if RV is supplying the reactor building headers, starting A train RN ensures desired flow rate to these headers.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. Check if S/I is actuated:	
__ a. "SAFETY INJECTION ACTUATED" status light (1SI-18)- LIT.	a. Perform the following: 1) Check if S/I is required: __ • Pzr pressure less than 1845 PSIG OR __ • Containment pressure greater than 1 PSIG. __ 2) **<u>IF</u>** S/I is required, **<u>THEN</u>** initiate S/I. 3) **<u>IF</u>** S/I is not required, **<u>THEN</u>**: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees). __ • **<u>GO TO</u>** EP/1/A/5000/ES-0.1 (Reactor Trip Response).
__ b. Both LOCA Sequencer Actuated status lights (1SI-14) - LIT.	__ b. Initiate S/I.
__ 6. **Announce "Unit 1 Safety Injection".**	

7.5 Manual Initiation of Safeguards Actions

In most scenarios, ROs and SROs are expected to manually initiate safeguards actions if an automatic action setpoint is being approached, to avoid challenging the automatic safeguards function. An example of this is to manually initiate safety injection if pressure is decreasing in an uncontrolled manner to 1845 psig.

Exceptions to this philosophy are listed below:

- Do **NOT** initiate Phase B/Containment Spray earlier than required. Early initiation of spray has the adverse affect of transferring FWST water to the containment sump and causing earlier transfer to Cold Leg Recirc. {NRC Bulletin 2003-01 response}

- During an ATWS, it is undesirable to initiate S/I in "anticipation" of an S/I signal if the reactor will **NOT** trip, since this will cause a loss of CF flow to the S/Gs. This exception is stated in the APs that manually initiate S/I in "anticipation" of an S/I signal.

The operator is expected to manually initiate any action which should have automatically occurred if the automatic function fails, such as the Safety Injection fails to initiate during an uncontrolled Reactor Coolant depressurization at 1845 psig (even during an ATWS) or an ECCS pump fails to start on a Safety Injection signal.

IF directed to initiate a signal, initiate both trains unless otherwise specified.

7.6 Resetting Safety Systems

IF directed to reset a signal, reset both trains unless otherwise specified.

IF a procedure directs resetting a signal that has **NOT** been received or that has been previously reset, the reset pushbuttons do **NOT** have to be depressed since the intent of the step has been met. Likewise, if a procedure directs the operator to stop, start or reposition a component which is already in the desired position; the component's control switch does **NOT** have to be depressed.

1.0 INTRODUCTION

1.1 Purpose

Objective # 1

The Diesel Generator Load Sequencing System (EQB) functions to energize the necessary Blackout and/or Safety Injection loads in such a manner that the diesel generator or auxiliary transformer (ATC, ATD, SATA, SATB) is not momentarily overloaded.

Objective # 2

A power loss to the 4160 Volt Bus or a Safety Injection Actuation Signal from the Solid State Protection System (SSPS) actuates the Load Sequencer.

1.2 General Description

The sequencer has basically two modes of operation: priority and secondary. The priority mode is actuated by a safety injection (SI) signal from the Solid State Protection System (SSPS). The secondary mode is actuated by a loss of voltage (LOV) on the 4160 volt essential bus.

The Sequencing System is designed to be actuated automatically without any operator action and to initiate loading of the Engineered Safeguard bus as rapidly as loading transients permit without overloading the normal transformer or diesel generator.

The controlling parameters of sequencer logic are the ESF signal from SSPS, the time from initial actuation, the voltage on the ESF Bus and the Diesel Generator frequency (speed).

1.3 Redundancy requirements

There are two identical systems, one associated with each diesel. They are independent of each other and in no way can the failure of one affect the other. The single failure is considered to be the entire loss of one system.

1.4 Sequencer Actuation Signals

Signal	Setpoint	Coincidence	Interlock	Protection
Manual Safety Injection		1/2 Switches		Operator Judgment
Low Pressurizer Pressure	1845 psig	2/4 Channels	P-11	LOCA
High Containment Pressure	1.0 psig	2/3 Pressure Switches		Steam Break LOCA

2/3 Under-voltage on affected 4160 Volt Bus (Blackout)

2.0 SYSTEM DESCRIPTION

2.1 Sequencer Modes of Operation

Objective # 3

The Sequencer has basically two modes of operation;

The **Priority Mode** of operation is actuated by a Safety Injection signal from the SSPS. When Safety Injection is actuated, the signal seals in and sequencing begins immediately.

The **Secondary Mode** of operation is actuated by a 2/3 phase Loss of Voltage (LOV) on the 4160 Volt Essential Bus. Upon actuation, the sequencer starts the diesel and goes through an 8 second test for verification of a Blackout. If a Blackout does not exist, the Sequencer will automatically reset to its initial operating state and the Diesel Generator must be manually shut down. For an actual Blackout, the signal is sealed in, the 4160V bus normal and alternate incoming breaker is tripped, the 4160 Volt Essential Bus is load shed, and the Diesel Generator Breaker is closed provided the Diesel Generating unit has attained 95% speed.

Objective # 4

When both actuation signals (LOV and SI) are present simultaneously, the Sequencer will select the SI logic and perform those functions necessary to sequence that mode (i.e., load shed, sequencer reset, removing blackout logic, and energizing SI loads). This is also true when the Loss of Voltage condition was initiated by the Degraded Voltage relaying. If an SI signal were present following the completion of the 9.7 second alarm timer cycle, the 4160 Volt Normal and Standby incoming circuit breakers would trip immediately. This causes the SI loads to be connected to the Diesel Generator initially, therefore ensures a reliable power supply for the Essential Auxiliary loads.

The Sequencer is designed to initiate loading of the 4160 Volt Essential Bus as rapidly as loading transients permit without overloading the Normal Transformer or Diesel Generator.

2.2 System Protection

Each unit is protected from abnormal voltage conditions by two levels of voltage protection, Loss of Voltage and Degraded Voltage. For each train, there are three Loss of Voltage and three Degraded Voltage relays connected in 2/3 logic. Another relay is provided which is used as a permissive for the Load Sequencer Accelerated Sequence Mode (127AX Special). All relays affect Sequencer operation.

The Degraded Voltage relays are set to operate at 89% of nominal bus voltage on U2 which is 3703 Volts. On U1 the Degraded Voltage relays are set to operate at 88.4% or 3678.5 Volts. The relays are a high accuracy type with a small reset dead band. These relays use time delays before initiating any actions. With a 4160 Volt bus de-energized, the Degraded Voltage relays must be placed in **TEST** before the Normal and Standby circuit breakers can be operated.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

A. Purpose

Provide actions for protecting the reactor core in the event of a LOCA that occurs during either Mode 3 after the Cold Leg Accumulators are isolated or Mode 4.

2.2. Discharge Piping

A 10 inch line connects each Accumulator to a cold leg. Each line contains two (2) check valves in series, one (1) normally open isolation valve and a flow restrictor. The flow restrictor is installed on the outlet of each accumulator and ensures accumulator discharge line resistance is within ECCS analysis tolerance band.

2.3. Isolation Valves

One (1) motor operated valve per accumulator provides isolation of CLA. The valves are normally opened with power removed prior to exceeding 1000 psig, and with NCS temperature between 400 and 425 degrees.

Objective #4

Alarms on the Control Board alert the operator when a valve is less than fully open (ACCUM ISOL NOT FULLY OPEN). This alarm is not active < P-11. The control circuitry for each valve is equipped with a disconnect/enable switch which allows isolation of the motor from the power source to prevent inadvertent operation. Removal of power to the valves is required by Tech Specs because the valves fail to meet single failure criteria. OPEN/CLOSE pushbuttons are located on the Control Board.

Objective #3

The valves are designed to automatically open at > P-11 setpoint (1955 psig) or on a S_S signal if closed and power is available to the valve. The valve motors are powered from EMXA and EMXB. Power supplies are as follows:

Valve Number	Unit 1	Unit 2
NI-54A	1EMXA-2 Comp. 3A	2EMXA-2 Comp. 3A
NI-65B	1EMXB-4 Comp. 2C	2EMXB-4 Comp. 2C
NI-76A	1EMXA-2 Comp. 3B	2EMXA-2 Comp. 3B
NI-88B	1EMXB-4 Comp. 3D	2EMXB-4 Comp. 3D

2.4. Check Valves

Swing check valves are installed in the discharge line to prevent flow from the Reactor Coolant System to the accumulator. These valves open at ΔP of 0.5 psi (upstream to downstream)

2.5. Relief Valves

Relief valves are installed on each accumulator to prevent over-pressurization. Sized to pass more than makeup capability, these valves are designed to pass N_2 or water. Relief valves are set at 700 psig.

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
16	Explain how a Safety Injection or Blackout on either unit affects that and the other unit during normal operations and what action the operator must perform.	X	X	X	X	X
17	Concerning the Technical Specifications related to the RN System:					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is(are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of parameter values or system conditions and the appropriate Tech Spec, determine required action(s).			X	X	X
	• Discuss the bases for a given Tech. Spec. LCO or Safety Limit.				X	*
	*** SRO ONLY**					

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
1.0	1.5	1.5	1.5	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of the Diesel Generator Load Sequencing System.	X	X	X	X	
2	List the Sequencer Automatic Actuation Signals.	X	X	X	X	X
3	List the two Sequencer Modes of Operation and give a brief explanation of each mode.	X	X	X	X	X
4	State which of the Sequencer Modes has priority.	X	X	X	X	X
5	Describe the sequence of events which occur during the Blackout Mode of Sequencer Operation.			X	X	X
6	Describe the sequence of events which occur during the Safety Injection Mode of Sequencer Operation.			X	X	X
7	Describe the sequence of events which occur during a Blackout followed by a Safety Injection.			X	X	X
8	Describe the sequence of events which occur during a Safety Injection Actuation followed by a Blackout. (NOTE: with S_s reset and with S_s not reset).			X	X	X
9	Describe the sequence of events required to be done in order to return the 4.16 KV bus back to normal following a: • Safety Injection • Blackout • Safety Injection followed by a Blackout • Blackout followed by a Safety Injection			X	X	X
10	Given a Limit and/or Precaution associated with an operating procedure, discuss its bases and when the it applies.	X	X	X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	015 G2.1.7	
	Importance Rating		4.7

Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question: SRO 91

Given the following plant conditions:

- A reactor startup is in progress.
- Control Bank "C" rods are at 130 steps.
- Both Intermediate Range channels indicate approximately $5X10^{-11}$ amps.
- Source Range Channel N-31 drifts DOWNSCALE and now indicates $5X10^0$ CPS.

Which ONE (1) of the following describes the correct action for the plant conditions and the technical specification basis for the action?

A. Continue the reactor startup; with only one source range channel operable; 48 hours is allowed to restore two channels to service.

B. Suspend the reactor startup; source range channels are not currently required to trip the reactor; however, the source range monitoring functions must be available.

C. Continue the reactor startup; only one source range channel is required for Source Range High Flux Trip protection.

D. Suspend the reactor startup; with only one source range channel operable, the minimum required Source Range High Flux Trip protection is not met.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Cannot continue the startup, although 48 hours is the LCO action time for 3.3.1

B. Incorrect. Correct that startup is suspended, but power is below P-6, and SR high flux trip is required
C. Incorrect. Cannot continue and Power <P-6/P-10, SR high flux is not enabled. 1 channel required for trip, but 2 required to be operable
D. Correct. TS basis discusses minimum protection for current conditions requiring 2 SR NIs

Technical Reference(s): TS 3.3.1 Basis (Attach if not previously provided)
AP/16 Rev 10
AP-16 Basis Document
IC-ENB Rev 26

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 2

Comments:
WTSI Bank, previous NRC – Unit not identified

KA matched because an evaluation must be made based upon the NI failure presented. SRO level because they must know the TS action required for startup, and also the basis for the action

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2 Be in MODE 3.	54 hours
C. One channel or train inoperable.	C.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	C.2 Open reactor trip breakers (RTBs).	49 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	-----------------NOTE----------------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed.	
I. One Source Range Neutron Flux channel inoperable.	I.1 Suspend operations involving positive reactivity additions.	Immediately
J. Two Source Range Neutron Flux channels inoperable.	J.1 Open RTBs.	Immediately
K. One Source Range Neutron Flux channel inoperable.	K.1 Restore channel to OPERABLE status. <u>OR</u> K.2 Open RTBs.	48 hours 49 hours
	-----------------NOTE----------------- Plant temperature changes are allowed provided that SDM is maintained and Keff remains < 0.99.	
L. Required Source Range Neutron Flux channel inoperable.	L.1 Suspend operations involving positive reactivity additions. <u>AND</u> L.2 Close unborated water source isolation valves. <u>AND</u> L.3 Perform SR 3.1.1.1.	Immediately 1 hour 1 hour <u>AND</u> Once per 12 hours thereafter

(continued)

Table 3.3.1-1 (page 1 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110% RTP	109% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 26% RTP	25% RTP
3. Power Range Neutron Flux Rate						
High Positive Rate	1,2	4	D	SR 3.3.1.7 SR 3.3.1.11	≤ 5.5% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30% RTP	25% RTP
	2(d)	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30% RTP	25% RTP
5. Source Range Neutron Flux	2(d)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.3 E5 cps	1.0 E5 cps
	3(a), 4(a), 5(a)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.3 E5 cps	1.0 E5 cps
	3(e), 4(e), 5(e)	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A

(continued)

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.
(b) Below the P-10 (Power Range Neutron Flux) interlocks.
(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10. **Check at least one of the following S/R Channels - OPERABLE:** __ • N-31 OR __ • N-32.	**Perform the following:** __ a. **IF** in Mode 1 **OR** 6, **THEN GO TO** Step 12. __ b. **IF** in Mode 2 above P-6, **THEN GO TO** Step 12. __ c. **IF** a loss of SG feedwater occurs, **THEN REFER TO** AP/1/A/5500/06 (S/G Feedwater Malfunction) as required. __ d. Suspend operations involving positive reactivity additions. **NOTE** The following step will cause a loss of normal "blended" VCT M/U. __ e. Dispatch an operator to close and lock 1NV-250 (Rx Makeup Water Supply to Unit 1 NV Isol) (aux bldg, 733, JJ-54, 25 ft north of KC pumps) within one hour. __ f. **IF AT ANY TIME** 1NV-250 is required to be open for make-up to FWST, **THEN REFER TO** Tech Spec Bases 3.9.2 for list of unborated water source isolation valves required to be locked closed. __ g. Monitor available I/R channels and W/R Neutron Flux Monitors. __ h. Ensure adequate shutdown margin **PER** OP/0/A/6100/006 (Reactivity Balance Calculation) within one hour. __ i. **GO TO** Step 12.
__ 11. **Check plant status - IN MODE 3 OR BELOW.**	__ **IF in Mode 2 below P-6, THEN suspend operations involving positive reactivity additions.**

Ensures compliance with Technical Specifications if **NO** S/R instruments are OPERABLE for reactor trip instrumentation for Mode's 5, 4, 3, and 2 (below P-6).

DISCUSSION:

Note this is only for the case where step 1 has not sent the operator to E-0.

STEP 11:

PURPOSE:

Ensures compliance with Technical Specification for **ONE** S/R instrument inoperable under reactor trip instrumentation for MODE 2, below P-6.

DISCUSSION:

The AP doesn't address the situation of **ONE** S/R instrument inoperable for Modes 3, 4, &5 (breakers closed) because sufficient time is allowed to refer to and comply with T.S. in this case.

STEP 12:

PURPOSE:

With only **ONE** S/R instrument inoperable, per Technical Specifications, operation can continue with the reactor trip breakers closed. This step is intended to block an inadvertent reactor trip signal from the failed channel. Another action taken in this step is to block an inadvertent High Flux at Shutdown alarm from coming in.

DISCUSSION:

The S/R reactor trip signal logic is one out of two channels. If the failed channel reaches 10^5 cps or instrument power is lost, a reactor trip signal will result. Placing a channel's "LEVEL TRIP" switch in "BYPASS" will prevent this as long as control power is available on that channel. The channel is placed in "BYPASS" at this time even though these conditions may not yet be true, because before the process is complete to restore the channel, these conditions may occur (i.e., IAE troubleshooting and/or calibration may bring in trip bistable). This same logic applies to blocking the High Flux at Shutdown, typically set at ½ decade above background counts. These actions are grouped in this step, along with verifying the level trip bypass light lit, since these actions complete the ones required at the failed S/R drawer. The next step has the actions grouped together that are performed at the MCB.

STEP 13:

PURPOSE:

Verify the actions taken in a previous step are successful.

12	List the Protection and Control Interlocks (Ps and Cs) associated with the Nuclear Instrumentation System. (Include setpoints and logic)		X	X	X	X
13	State the purpose of the Wide Range Neutron Detection System.		X	X	X	
14	Concerning the Wide Range Neutron Detection System:					
	• Describe the operation.		X	X	X	
	• Describe the indications and controls.		X	X	X	X
15	State the purpose of the Gamma-Metrics Shutdown Monitor System.		X	X	X	
16	Concerning the Gamma-Metrics Shutdown Monitor System:					
	• Describe the operation.		X	X	X	
	• Describe the alarms, indications and controls.		X	X	X	X
17	Determine the validity of indicated reactor power using alternate indications of power level.		X	X	X	X
18	Describe the Source Range instrumentation response for voiding in the core and downcomer region.		X	X	X	X
19	Concerning the Technical Specifications related to the Nuclear Instrumentation System;					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is(are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech Specs, determine required action(s).			X	X	X
	• Discuss the basis for a given Tech Spec LCO or Safety Limit.				X	*
	* SRO Only					

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	028 A2.03	
	Importance Rating		4.0

Ability to (a) predict the impacts of the following malfunctions or operations on the HRPS; and (b) based on those predictions, use Procedures to correct, control, or mitigate the consequences of those malfunctions or operations: The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment dam-age in containment

Proposed Question: SRO 92

Given the following:

- A LOCA has occurred on Unit 2.
- Due to subsequent failures, the crew is performing actions contained in FR-C.1, Response to Inadequate Core Cooling.
- Hydrogen Analyzers are in service.
- Hydrogen igniters are OFF.
- NF AHUs are OFF.
- Containment Hydrogen Concentration is currently 3% and rising slowly.

Which ONE of the following describes the action required, and the reason for the action, in accordance with FR-C.1?

A. Place hydrogen igniters in service; do NOT operate Hydrogen recombiners; recombiner operating temperatures may cause a challenge to containment integrity due to hydrogen flammability.

B. Place hydrogen igniters and hydrogen recombiners in service; containment hydrogen concentration is below the limit causing concern for containment integrity violations due to hydrogen ignition.

C. Do NOT place hydrogen igniters OR hydrogen recombiners in service; consult management for recommendation related to hydrogen reduction. Operation of either component may result in a challenge to containment integrity.

D. Place hydrogen recombiners in service; do NOT operate Hydrogen igniters; igniters must be placed in service prior to hydrogen concentration reaching 0.5%, because ignition above that concentration may cause a challenge to containment integrity.

Proposed Answer: B

Explanation (Optional):

Per the basis document of FR-C.1, step 4, if hydrogen concentration is between 0.5% and 6%, there is limited burn potential. Therefore, both the igniters and the recombiners are placed in service. If hydrogen is less than 0.5%, a flammable situation is not imminent, so the igniters are placed in service. If hydrogen is greater than 6% there is a potential explosive mixture. Hydrogen concentration must be reduced in other ways before starting the recombiners or igniters.

A. Incorrect. Recombiners are allowed to be started below 6%
B. Correct.
C. Incorrect. Action is correct for >6% concentration
D. Incorrect. Igniters will be placed in service as well as recombiners if concentration is below 6%

Technical Reference(s): FR-C.1 step 4 Rev 5 and basis; EP-FRC Rev 10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-FRC Obj 6 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 1, 5

Comments:

KA is matched because it evaluates operation of equipment to keep hydrogen concentration below the explosive limit. SRO only because the applicant must know the design and procedural basis for operation of hydrogen igniters and recombiners

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4. **Check containment H_2 concentration:**	
__ a. Ensure Operator dispatched to stop Unit 2 NF AHUs.	
__ b. Check H_2 analyzers - IN SERVICE.	b. Perform the following:
	__ 1) Dispatch operator to place H_2 analyzers in service **PER** EP/2/A/5000/G-1 (Generic Enclosures), Enclosure 5 (Placing H_2 Analyzers In Service).
	__ 2) **WHEN** H_2 analyzers in service, **THEN** complete Steps 4.c through 4.e.
	__ 3) Observe Note prior to Step 5 and **GO TO** Step 5.
__ c. Check H_2 concentration - LESS THAN 6%.	c. Perform the following:
	__ 1) Obtain recommendation from station management to reduce H_2 concentration.
	__ 2) Observe Note prior to Step 5 and **GO TO** Step 5.
__ d. Check H_2 igniters - ON.	d. **WHEN** the following conditions met, **THEN** place H_2 igniters in service:
	__ • NF AHUs off
	__ • H_2 concentration less than 6%.
__ e. Check H_2 concentration - LESS THAN 0.5%.	__ e. Dispatch operator to place H_2 recombiners in service **PER** EP/2/A/5000/G-1 (Generic Enclosures), Enclosure 4 (Placing H_2 Recombiners In Service).

STEP 4 Check containment H_2 concentration:

PURPOSE: To check if an excessive containment hydrogen concentration is present.

BASIS: This step instructs the operator to obtain a current hydrogen concentration measurement. Dependent upon the magnitude of the hydrogen concentration, the operator will either obtain a recommendation from management to reduce hydrogen concentration or place the recombiners in service. Once the concentration has been obtained, the operator is directed to continue in this procedure.

When inadequate core cooling has occurred, the containment hydrogen concentration may be as much as 10 to 12 volume percent, depending on the amount of metal-water reaction that has occurred in the core. The hydrogen concentration is of concern since a flammable mixture can cause a sudden rise in containment pressure which may challenge containment integrity. Note that in order to have the potential for flammable hydrogen concentrations, an inadequate core cooling situation must have already existed. Without this situation, sufficient hydrogen would not be expected to have been produced to cause flammable mixtures.

If the hydrogen mixture is between 0.5 volume percent and 6.0 volume percent in dry air, either no hydrogen burn is possible or a limited burn may occur which does not produce a significant pressure rise. In this case the operator is instructed to start the hydrogen recombiner and igniters system to reduce containment hydrogen concentration. If the hydrogen concentration is less than 0.5 volume percent in dry air, a flammable situation is not imminent, the hydrogen igniters are placed in service. If the concentration is greater than 6.0 volume percent in dry air, station management is notified for additional recovery actions while proceeding with the procedure.

All hydrogen measurements are referenced to concentrations in dry air even though the actual containment environment may contain significant steam concentrations. There are two reasons for this - first, most hydrogen measurement systems remove moisture from the sample thus approximating a dry air condition, and second, the amount of hydrogen to reach flammability is greater when steam is present, thus the volume percent used in the procedure is conservative.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		3.0	3.0	2.0

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Explain the purpose of each procedure in the FR-C series. EPFRC001			X	X	
2	Discuss the entry and exit guidance for each procedure in the FR-C series. EPFRC002			X	X	
3	Discuss the mitigating strategy (major actions) of each procedure in the FR-C series. EPFRC003			X	X	X
4	Discuss the basis for any note, caution or step for each procedure in the FR-C series. EPFRC004			X	X	X
5	Given the Foldout page, discuss the actions included and the basis for these actions. EPFRC005			X	X	X
6	Given the appropriate procedure, evaluate a given scenario describing accident events and plant conditions to determine any required action and its basis. EPFRC006			X	X	X
7	Discuss the time critical task(s) associated with the FR-C series procedures including the time requirements and the basis for these requirements. EPFRC007			X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	079 A2.01	
	Importance Rating		3.2

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

Proposed Question: SRO 93

Given the following conditions:

- Unit 1 is at 100% power.
- The following alarm is received:
    - VI/VS LO PRESS.
- VI pressure is 80 psig and lowering slowly.
- The crew is performing actions of AP/1/A/5500/022, Loss of VI.

Which ONE (1) of the following describes the action required for current plant conditions in accordance with AP-22, and the position of 1VI-820, VI Supply to VS Control valve?

A. Perform Enclosure 4, Diesel VI Operation ONLY; 1VI-820 is currently closed to prevent a fault in the VS system from causing a loss of VI.

B. Perform Enclosure 4, Diesel VI Operation ONLY; 1VI-820 remains open because VI pressure has not decreased to the point where automatic isolation is required.

C. Perform Enclosure 4, Diesel VI Operation AND Enclosure 5, VI Dryer and VI to VS System Isolation; 1VI-820 is currently closed to prevent a fault in the VS system from causing a loss of VI.

D. Perform Enclosure 4, Diesel VI Operation AND Enclosure 5, VI Dryer and VI to VS System Isolation; 1VI-820 remains open because VI pressure has not decreased to the point where automatic isolation is required.

Proposed Answer: C

Explanation (Optional):

A is incorrect. Because pressure is 80 psig (below 82) BOTH enclosures must be performed

B is incorrect, as the valve will auto close at 90, and pressure is currently 80

C is Correct. AP/22 requires performance of BOTH enclosures at this air pressure

D is incorrect. Credible because the action is correct and will be performed when LO-LO PRESS alarm is received at 82 psig, but valve is already closed

Technical Reference(s): SS-VI, rev 32 (Attach if not previously provided)
AP/1/A/5500/022 Rev 27
AP-22 Basis Document

Proposed references to be provided to applicants during examination: None

Learning Objective: SS-VI Obj 2 (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments:

KA is matched because the item evaluates knowledge of plant conditions that will result in isolation of VI/VS cross-tie (Instrument Air to Service Air) SRO level because plant conditions are presented where the applicant must know which procedure sections will be required to mitigate the consequences and restore pressure

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

4. **Restore VI as follows:**

__ a. Check VI header pressure - GREATER THAN 70 PSIG.

a. Perform the following:

1) Dispatch operator to perform the following enclosures using copy located beside Service Bldg Lube Oil Station door:

__ • Enclosure 3 (Startup of A, B, and C VI Compressors)

__ • Enclosure 6 (D, E and F VI Compressor Operation with Low Control Air)

__ 2) **GO TO** Step 4.c.

b. Dispatch operator(s) to start VI compressors as necessary to restore VI using copy of enclosures located beside Service Bldg Lube Oil Station door:

__ • Enclosure 2 (Startup of D, E and F VI Compressors)

__ • Enclosure 3 (Startup of A, B, and C VI Compressors).

__ c. Dispatch operator to search for possible leaks.

__ d. **IF AT ANY TIME** VI header pressure is 90 PSIG and going down ("VI/VS LO PRESS" alarm), **THEN** dispatch operator to ensure Diesel VI compressors running **PER** Enclosure 4 (Diesel VI Compressor Operation), using copy of procedure located on column 2B32, Unit 2 TB, near exit door leading to Diesel VI compressors.

__ e. **IF AT ANY TIME** VI header pressure goes below 82 PSIG ("VI/VS LO-LO PRESS" alarm), **THEN** dispatch operator to bypass VI dryers and isolate VS **PER** Enclosure 5 (VI Dryer and VI to VS System Isolation) using copy of procedure located beside Service Bldg Lube Oil Station door.

STEP 3:

PURPOSE:

Avoid KC pump runout.

DISCUSSION:

KC flow control valves on most components fail open on a loss of instrument air. A loss of VI could significantly increase KC flow, possibly beyond the 4000 gpm/ pump limit for runout for continued operation. If only one KC pump is running, then starting a second one could prevent severe runout. A pump operating well over its runout point can sustain damage in very little time. This step is early in the AP because this action doesn't take much time for the benefit that may be gained. A loss of VI KC flow model was performed under Plan 95-0791. The assumption for the model was 2 KC pumps running. The model indicated 4700gpm/pump flow as a result of loss of VI, with the 2 pumps running. Based on discussions with the vendor, this would present no pump problems for a short duration of approx. 3 hrs. A step later in the AP addresses getting KC flow to less than 4000gpm/pump for sustained losses of VI.

REFERENCES:
Plan 95-0791, Loss of VI KC flow modeling.

STEP 4:

PURPOSE:

Restore VI pressure.

DISCUSSION:

This is the only step in the body of the AP that restores VI pressure. It attempts this by getting VI compressors running, bypassing the VI dryers/isolating VS, and stopping VI leaks (if applicable).

The compressors of choice are D, E, and F because of their capacity, quality of air, and reliability. An operator is dispatched to perform an enclosure to get these three compressors running. Also, as necessary, an operator is dispatched to perform an enclosure to start A, B, and C compressors.

There are two basic requirements in the enclosures for starting D, E, and F compressors. First, KR must be available (> 50 psig should be sufficient cooling water). KR is interlocked such that the compressors will not start without KR.

Second, sufficient control air pressure must be available for proper operation. "70 PSIG" VI header pressure determines which enclosure the operator will use to start and monitor D, E, and F compressor operation (Enclosure 2 or 6). Note that if VI pressure is less than 70 PSIG, the AP requires implementation of Enclosure 6, even if all three compressors are running. This is because D, E, and F VI compressors may be running in a degraded condition. (In 2003, Grand Gulf Nuclear Station had an event where their VI compressors began failing similar to what is described below.)

D, E, and F VI compressors respond to low control air pressure as follows:

1. VI compressor inlet and bypass valves require control air to operate. As control air pressure drops below 70 PSIG, these valves begin moving toward their fail safe position, causing compressor output to go down. If control air pressure continues to drop, these valves eventually reach their fail safe position, and the compressor will be fully unloaded.

2. Compressor "surge" is another possible outcome of low control air pressure. If the VI compressor inlet valve fails closed faster than the bypass valve fails open, a reversal of flow through the compressor may occur, resulting in "surge". If surge occurs, the compressor will automatically unload, and control air must be restored to re-load the compressor.

3. If control air pressure drops low enough, the compressor will trip on low seal air pressure.

Enclosure 6 (D, E, and F VI Compressor Operation with Low Control Air) uses the following strategies to restore VI:

1. If running compressors are able to restore VI to normal, Enclosure 6 is exited.

2. If running compressors cannot restore pressure above 70 PSIG, VB will be aligned to the control air header. This should restore normal control air pressure to the compressor control valves so that proper compressor operation is possible.

3. Station management may decide to isolate the Unit 1/ Unit 2 Turbine bldg headers to reduce VI system demand, and thus restore normal VI pressure. This option may be worthwhile if running compressors are unable restore VI to normal and VI header pressure has dropped to the point that air operated valves have begun to fail. Note, however, that isolating either Turbine Bldg header too soon could end up causing a transient (i.e. uncontrolled cooldown due to steam drains failing open, CF reg valves failing closed, etc).

4. If VB is unavailable, it may be necessary to align nitrogen cylinders to the VI compressor control air header.

If VI header pressure is 90 psig and going down, this step dispatches an operator to ensure G and H diesel compressors are running. G and H get an auto start signal when 0VIPS5070 senses VI pressure at 90 psig, or when there is a loss of KR (as sensed on 3/3 pressure switches). This step uses "90 psig and going down" as a setpoint, because the pressure indicated in the C/R is sensed downstream of the VI Dryers, while 0VIPS5070 is sensed upstream of the dryers at the receiver tanks. Therefore, C/R indication will read less than 0VIPS5070, depending on the DP across the VI dryers. Since the VI Dryer D/P is typically less than 2 psig, G and H VI compressors should start when the C/R reading is slightly less than 90 psig. Also, "90 PSIG" coincides with the "VI/VS LO PRESS" annunciator setpoint.

At VI header pressure of 82 PSIG, an operator is dispatched to ensure the VI dryers are removed from service. VI dryer malfunctions are the primary contributor to loss of VI event probability. Enclosure 5 verifies several auto actions occur - 1VI-1812 (VI Air Dryer Bypass Filter Isol) auto opens at 85 PSIG, VI dryer purge exhaust valves auto close at 90 PSIG, and VI-820 (VI to VS) auto closes at 90 psig. Once the operator verifies that the dryers are properly bypassed, the inlet and outlet valves for each dryer are closed to completely isolate the dryers. The bypass line contains a filter to ensure liquid water, oil aerosols, and corrosion products are removed prior to entering VI headers. The "82 PSIG" setpoint for performing Enclosure 5 coincides with the "VI/VS LO LO PRESS" annunciator setpoint.

This step also cues the operator to start looking for leaks (with the assumption if a leak is located, it will be isolated per management expectations).

<u>REFERENCES:</u>
VI System Engineer letter dated 4/8/99
VI DBD Section 30.3.3.5.2
Grand Gulf Nuclear Station - NRC Special Team Inspection Report 50-416/03-07

1.2.12 1VI-820 VI to VS Control Valve

The VI System normally supplies the Low Pressure VS System through control valve 1VI-820.

Controls and indication for 1VI-820 are located at the VI Sequencer Control Panel. The valve control switch is a three position switch:

- Close
- Auto
- Open

Objective # 7

Indication provided at the VI Sequencer Control Panel consists of the following:

- 1VI-820 Close (green light)
- 1VI-820 Open (red light)

This valve is normally in the AUTO position and will automatically close should VI System Pressure decrease to <90 psig. Upon valve closure 1VI-820 can be reopened once VI System Pressure has increased >90 psig by placing the valve to the OPEN position. After opening the valve 1VI-820, the switch should be returned to the AUTO position. If not, the valve will reopen without operator action, after closure, as soon as pressure has increased above 90 psig.

1.2.13 VI System Air Dryers A, B, and C

Objective # 9

VI Dryers A, B, and C (AMLOC-CHA Dryers) are fully automatic, desiccant-type air dryers designed to remove vaporous moisture from the Instrument Air System.

Generally, two of the three desiccant air dryers (A, B, and C) are in-service while one remains in standby, ready and available for service when needed. Each in-service dryer will alternately cycle air through one of the two desiccant chambers for moisture removal, while the other chamber is regenerated (removal of previously adsorbed moisture) and re-pressurized.

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
6	Explain the control function associated with each of the following VI Air Compressor (A, B, and C) pushbuttons: • Start/Stop pushbutton • Reset pushbutton	X	X	X	X	
7	List the interlocks / trips associated with operation of the following plant air system components: • VI Air Compressors • VI-820 (VI to VS Supply Valve) • VS Low Pressure Air Compressor • VB Air Compressor	X	X	X	X	X
8	Describe the following controls and/or indications associated with operation of VI Air Compressors D, E, and F: • On/Off switch and indication • Start/Stop pushbuttons • Pre-lube pump status • Acknowledge/Reset pushbutton	X	X	X	X	
9	Describe how the following VI System components function to provide a continuous supply of clean dry air: • Service Building Air Receiver Tanks (and drains) • Air Dryers • Auxiliary Building Instrument Air Tanks	X	X	X	X	
10	Explain each one of the following controls and /or indications, associated with the Breathing Air Compressors: • Start/Stop Pushbutton • "Power ON" Light • "RUN" Light • Discharge Air Over-Temperature Light • Rotor Oil Filter Service Light • Bearing Oil Filter Service Light • Air/Oil Separator Service Light • Service Air Filter ΔP Gauge • Purification Filter ΔP Gauge • Rotor Coolant Temperature Gauge • Discharge Air Pressure Gauge • Discharge Air Temperature Gauge	X	X	X	X	
11	Describe normal operation of the Breathing Air Compressor(s).	X	X	X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.35	
	Importance Rating		3.9

Knowledge of the fuel-handling responsibilities of SRO's.

Proposed Question: SRO 94

Unit 1 is in Mode 6, core alterations are in progress.

Which ONE of the following, by title, must approve bypass of a Fuel Handling interlock not specified in accordance with procedures for routine fuel handling activities?

A. Shift Manager

B. Fuel Handling SRO

C. Refueling Supervisor

D. Reactor Engineer

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Shift Manager responsible for unit, but FH SRO is responsible for all refueling activities

B. Correct.

C. Incorrect. Administrative oversight required, but not approval for FH bypass

D. Incorrect. Nuclear Engineers will be involved in the core alterations, but are not part of approval for FH bypass; they are only approval authority during physics testing

Technical Reference(s): NSD-414 Rev 2; FH-FC Rev 18 (Attach if not previously provided)

Proposed references to be provided to applicants during None examination: ____________

Learning Objective:	FH-FC, 1 and 5	(As available)
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	6, 7

Comments:

KA is matched because the item evaluates a decision by refueling SROs. SRO level because knowledge of SRO responsibilities during refueling is 10CFR55.43 (b) item 6/7 specific

9. Responsible for PMs/PTs on all Fueling Handling equipment.
10. Responsible for maintaining and troubleshooting all Fuel Handling equipment.
11. Responsible for preparing, loading, and transporting of Dry Storage Canisters.
12. Qualified / certified to Fuel Handling procedures for assigned equipment.
13. Perform procedures related to SNM (Special Nuclear Material) inventory control related to fuel.
14. Install and maintain communications systems required for refueling activities (installation and checkout).
15. Maintain underwater lights.
16. Support special projects as needed.
17. Perform all fuel handling activities.
18. Operate overhead cranes and hoists as necessary during fuel handling activities
19. Establish and maintain housekeeping, material condition, and FME controls of all fuel handling areas. This includes Upper Containment Refueling Canal Area, Spent Fuel Pools and Fuel Receiving Areas.

414.2.6 FUEL HANDLING ADVISORS (VENDOR)

A. Provide expertise for fuel handling activities (cranes, hoists, tooling, including industry knowledge, etc.).

B. Participate as an active member of the Fuel Handling Team.

C. Can perform the following:

- Review procedures.
- Provide "hands on" work as requested and approved by the Job Sponsor.

414.2.7 OPERATIONS SHIFT MANAGER (OPS)

NOTE: Operations is responsible for performing the SOER 91-01 Briefing for core reload.

A. During fuel movement, fuel receipt, special projects, and dry cask storage:

1. Ensure SRO's/RO's are cognizant of all fuel handling activities in progress or planned.
2. Maintain awareness of any activities that could impact fuel handling activities and ensure appropriate fuel handling personnel are aware of these activities.
3. Ensure appropriate response and notifications to any abnormal fuel handling event and verify any Technical Specification implications.
4. Has ultimate responsibility for the safety of the reactor core and fuel stored on site.
5. Ensure the 91-01 Briefing is performed prior to core reload.

414.2.8 CONTROL ROOM SRO AND RO (OPS)

A. During fuel movement, fuel receipt, special projects, and dry cask storage:

1. Monitor the Nuclear Instrumentation during core alterations.
2. Implement any responses required by Abnormal Procedures.

3. Log, verify, and maintain Technical Specification for Mode 6, Core Alterations, and other Technical Specifications for Spent Fuel Building activities.
4. Maintain awareness of fuel handling and Spent Fuel Building activities (i.e. - logging, turnover, etc.).
5. Maintain awareness of core configuration during core alterations.
6. Ensure reactivity monitoring is performed during refueling.

414.2.9 REFUELING SRO RESPONSIBLE FOR FUEL HANDLING

A. During core alterations:

1. Shall be present in the Reactor Building to observe and provide oversight of fuel handling activities anytime Core Alterations are being performed.
2. Shall have an SRO License or a SRO license limited to fuel handling.
3. Maintain a working knowledge of procedures and Technical Specifications associated with fuel handling and command immediate action as required.
4. Approve use of fuel handling bypass interlocks as necessary when not specified by an approved procedure.
5. Approve alternate fuel assembly moves as recommended by Reactor Engineering.
6. The Refueling SRO should be stationed on the refueling bridge any time Fuel Assemblies are being moved in the Reactor.
7. The Refueling SRO will ensure the following:
    a) Fuel Handling Procedures are performed as written.
    b) All refueling personnel adhere to STAR Self-checking techniques, procedure use and adherence, communication standards and independent verification.
    c) Understands the need for and approve all contingency actions which may be required, in accordance with Maintenance procedures for operating the Reactor Building Manipulator Crane.
    d) Direct Reactor building Activities during performance of Abnormal Procedures.
    e) No activities occur that adversely affect reactivity control.
    f) Foreign Material Exclusion controls are implemented per NSD 104 in the Refueling Canal area and that all housekeeping standards are maintained.
    g) Assure approved safety practices are followed during operation of the Manipulator Crane.
    h) Suspend all refueling operations anytime he/she thinks refueling operations are not being performed correctly or safely.

414.2.10 TRAINING

A. Develop and maintain initial training for designated Maintenance, Operations, and contract personnel on fuel handling topics.

B. Assist in the development of Just in Time (JITT) on relevant fuel handling topics using the systematic approach to training (SAT) process. Provide this training for the above designated personnel to maintain a well qualified work force for safe and efficient fuel handling operations and to maintain awareness of NSD 414 and fuel handling related issues.

- Provide periodic oversight as required.
- Be present at the 91-01 briefing.

414.2.4 REACTOR SERVICES (MNT) SUPERVISOR

A. During fuel movement, fuel receipt, special projects, and dry cask storage:

1. Maintain responsibility and control of all activities in the fuel handling areas. This includes the Spent Fuel Pools and the Fuel Receiving Areas.
2. Ensure housekeeping and material condition, and FME controls of all fuel handling areas is maintained. This includes Upper Containment Refueling Canal, Spent Fuel Pools, and Fuel Receiving Areas.
3. Provide work direction of Fuel Handling Team in support of fuel handling activities.
4. Coordinate interface with other groups during fuel handling activities.
5. Maintain ownership of fuel handling maintenance procedures. Ensure fuel handling maintenance procedures are developed and enhanced.
6. Act as a contact for scheduling all fuel handling PM's. Ensure all PM's are scheduled in a coordinated manner.
7. Performance management of the Fuel Handling Team.
8. Ensures qualifications and training requirements are met.
9. Maintain list and location of handling tools and equipment.
10. Perform all required fuel handling equipment PM's.
11. Provide ownership for Fuel Handling ETQS tasks.
12. Communicate any Fuel Handling issues to Operations in a timely manner.
13. Assist in providing the following SOER 91-01 oversight for core reloading:
    - Ensure Management's expectations are met.
    - Provide periodic oversight as required.
    - Be present at the 91-01 briefing.

414.2.5 MAINTENANCE FUEL HANDLING TECHNICIANS

A. During fuel movement, fuel receipt, special projects, and dry cask storage:

1. Operate fuel handling equipment in accordance with approved procedures.
2. Operate the fuel transfer system.
3. Operate all Fuel Handling tools.
4. Operate the Spent Fuel Pool bridge during fuel handling activities.
5. Operate the Reactor Building main crane during fuel handling activities.
6. Provide Spotter for the Spent Fuel Pool bridge during fuel handling activities.
7. Provide Spotter in the Reactor Building during Fuel Handling activities.
8. Monitor the camera installed at the spent fuel pool up-enders to ensure that the correct fuel assembly is in transit to the reactor building during refueling.

414. FUEL HANDLING

414.1 INTRODUCTION

The purpose of this NSD is to identify the roles and responsibilities for Fuel Handling activities at the three nuclear sites. Mechanical Maintenance (Reactor Services) is the owner of Fuel Handling and performs operation of all the tools and equipment used to move and manipulate fuel and components. Reactor Engineering provides the core designs, configuration control, and assists with the coordination and oversight of fuel handling activities. Operations provide oversight and ensure reactivity management during fuel manipulations.

414.2 ROLES AND RESPONSIBILITIES

414.2.1 REACTOR ENGINEERING (RES)

A. During Fuel Movement:

1. Determine fuel movement sequence.
2. Determine acceptable storage locations per Tech Specs.
3. Ensure reactivity monitoring is performed during refueling.
4. Provide technical oversight of controlling procedure during unload and reload.
5. Assist with pre-job briefings as required.
6. Provide instructions for alternate moves as required. (must be approved by FH SRO).
7. Provide technical assistance during foreign object retrieval.
8. Perform plant engineering roles (i.e., core verification, gap alignments, ensure SNM database is updated, etc.) in accordance with Equipment Reliability Program (NSD120)

NOTE: Operations is responsible for performing the SOER 91-01 Briefing for core reload.

9. Assist in providing the following SOER 91-01 oversight for core reload:
   - Ensure Management's expectations are met.
   - Provide periodic oversight as required.
   - Be present at the 91-01 briefing.

B. During Fuel Receipt:

1. Serve as point contact for scheduling fuel and/or component receipt (i.e. - vendor interface). The dates are established and provided to the FH Supervisor for further notification and follow up.
2. Prepare documentation for new fuel receipt.
3. Ensure QA inspection has been performed.
4. Interface with GO/vendor for evaluation of any defects found.
5. Responsible for loading patterns in the Spent Fuel Pool.
6. Ensure the SNM (Special Nuclear Materials) database is updated

C. During Special Projects:

1.0 INTRODUCTION

1.1. Fuel Handling Overview

Movement of Nuclear Fuel during core offload and core reload is a significant plant evolution. The fuel assemblies and inserts are discharged from the core into the spent fuel pool (core offload). Control rods, burnable poisons, source rods and thimble plugs are shuffled. Fuel rods are examined for leakers, leakers are reconstituted. Fresh fuel assemblies, along with once and twice burned fuel are reloaded into the core (core reload).

Several important issues need to be considered during the performance of fuel handling operations:

- Roles and Responsibilities
- Controlling Core Reactivity
- Foreign Material Exclusion
- Bypassing Fuel Handling Interlocks
- Abnormal Procedures

1.2. Roles and Responsibilities

Operations Shift Manager

Responsible for the safe operation of the plant. Supervises all of the licensed and unlicensed Operators. Is responsible for responding to any abnormal plant response including refueling problems.

Objective #2

Fuel Handling SRO

An SRO with no other concurrent responsibilities and shall direct supervision of core alterations. No reactivity additions or core alterations can be made without the direct supervision of the Fuel Handling SRO. The fuel handling SRO should be notified of any indications of fuel damage, unexpected reactivity changes or changes in refueling or spent fuel pool water levels. Core alterations include: 1) Fuel Movement 2) Control Rod Movement (including latching and unlatching control rods) 3) Neutron Source manipulation 4) Removal of Reactor Vessel Internals.

Who is the Fuel Handling SRO?

The SRO actively in charge on the reactor building operating deck during core alteration activities. Although the relief SRO may be on site, **all approvals shall be through the SRO actively in charge.**

The following is a specific list of Fuel Handling SRO responsibilities:

1. Ensure all fuel handling activities are performed in a safe and efficient manner.
2. Securing fuel handling operations as required by Tech Specs, Plant conditions, Safety concerns, or during times of uncertainty.
3. Should monitor refueling cavity to insure FME is being maintained.
4. Maintain constant communications with the control room during core alterations.
5. Assist the control room in monitoring refueling canal level, audible count rate and EMF or containment evacuation alarms.
6. Assist fuel handling crew in visually verifying fuel assemblies are lowered and raised safely. Gives hoist operator clearance to engage or disengage on fuel assemblies. Verifies assemblies are aligned properly and down on core plate prior to giving concurrence to disengage gripper.
7. Gives verbal clearance prior to pulling control rods during control rod latching, unlatching, and drag testing activities.
8. During core alterations, approve use of fuel handling bypass interlocks as necessary when not specified by an approved procedure (NSD 414).

Objective #1

Control Room Operators

Direct monitoring and manipulation of plant and reactor controls. Including monitoring of subcritical multiplication from nuclear instruments during core alterations. Responsible for implementing any necessary responses required by Abnormal Procedures. Logging and verifying technical specifications for MODE 6 and for core alterations. The Reactor Operator on the headset in the back of the control room communicates with the refueling crew. The Reactor Operator on the headset will get permission from the "Operator At The Controls" prior to unloading each fuel assembly. **The Operator at the Controls may stop fuel handling operations if, in his/her judgement, control room indication or communications show warranting conditions.**

Nuclear/Reactor Engineering

One responsibility is coordination of fuel movements during core loading operation by use of controlling procedure. Another is monitoring nuclear instrumentation to verify appropriate subcritical behavior and shutdown margin.

Reactor Services Technicians

One responsibility is operation of Fuel Handling Equipment in a safe manner moving fuel to locations recommended by reactor engineers by procedure. Another is the ability to recognize and properly respond to abnormal conditions.

2.2 Bypassing Fuel Handling Interlocks

Objective #5

Fuel handling procedures direct bypassing an interlock when required by known specific operations. During core alterations, the Licensed SRO for Fuel Handling is tasked with approving the use of bypasses for fuel handling interlocks as necessary when not specified by an approved procedure (NSD 414).

3.0 OPERATION

3.1. Normal Operation

Refer to OP/0/A/6550 Series Procedures
Refer to Drawings 7.1 and 7.2

3.1.1 Fuel and Component Handling

Fuel and Component Handling is covered by the above procedures and includes:

- Transfer of New Fuel from the Storage Vault to the Spent Fuel Pool
- Transfer of New Fuel from the Spent Fuel Pool to the Storage Vault
- Spent Fuel Pool Manipulator Crane Operation
- Reactor Building Manipulator Crane Operation
- Fuel Transfer System Operations
- Fuel Handling Tool Operations

3.1.2 Sequence of Refueling Operations

The first major step in refueling operations concerns preparation. The Reactor is shutdown and brought to COLD SHUTDOWN. RCS inventory is lowered to the vessel flange. The Fuel Handling Equipment is checked out.

Next is Reactor Disassembly. All connections are removed from the head. The Refueling Cavity is prepared for flooding (checkout underwater lights, tools and Fuel Transfer equipment; close the refueling canal drain valves, and remove blind flange from the transfer tube). Then the vessel head bolts are removed. The head is raised as the canal is flooded by FWST Pumps. The head is taken to it's storage location. Next the control rod drive shafts are disconnected and with the upper internals are removed from the vessel and stored. Now the core is free from obstructions and the core is ready for refueling.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		1.5	1.5	1.5

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Describe the roles and responsibilities of Control Room Operators during Fuel Handling operations.			X	X	X
2	Describe the roles and responsibilities of Fuel Handling SRO's during Fuel Handling operations.				X	X
3	Describe how monitoring of core reactivity is accomplished during Fuel Handling.			X	X	X
4	Deleted					
5	Describe the requirements that must be met before **bypassing** a Fuel Handling Interlock.			X	X	X
6	Concerning AP-25, Spent Fuel Damage; AP-40, Loss of Refueling Canal; and AP-41, Loss of Spent Fuel Cooling or Level: • State the purpose of the AP • Given symptoms, state the AP and Case (if applicable)			X	X	X
7	Concerning the Technical Specifications related to the FC System;					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech. Spec. is (are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters values or system conditions and the appropriate Tech Specs, determine required action(s).				X	*
	• Discuss the basis for a given Tech. Spec. LCO or Safety Limit.					
	* **SRO only**					

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.2.7	
	Importance Rating		3.6

Knowledge of the process for conducting special or infrequent tests.

Proposed Question: SRO 95

Given the following:

- Unit 1 is in Mode 1 on night shift.
- The Work Window Manager and Site Risk Expert are unavailable.
- A temporary test (TT) procedure is being performed on RN.
- During performance of the TT, an equipment failure occurred, resulting in a condition not evaluated during planning of the test.

In accordance with SOMP 2-2, Operations Roles in the Risk Management Process, who is responsible for determining the risk level, and what action is required if the risk level becomes ORANGE?

A. WCC SRO; OSM must evaluate the restoration plan and provide final authority on whether the plan is implemented.

B. WCC SRO; On-Shift CRS must evaluate the restoration plan and provide final authority on whether the plan is implemented.

C. On-Shift CRS; OSM must evaluate the restoration plan and provides final authority on whether the plan is implemented.

D. On-Shift CRS; On-Shift CRS must evaluate the restoration plan and provide final authority on whether the plan is implemented.

Proposed Answer: A

Explanation (Optional): SOMP 02-02 summarizes the responsibilities of individuals in the Operations (OPS) organization in the processes used to assess and manage risk significant activities at Duke nuclear sites.

A. Correct. During non-core business hours when the WWM or Site Risk Expert are not available, it is the responsibility of the WCC SRO to evaluate the current risk when existing conditions do not match those evaluated based on a planned schedule due to emergent work. (SOMP 02-02 Section 5.6.2) **WHEN** entering an orange or red condition from emergent work, the OSM will evaluate the restoration plan and have final authority on whether the plan is implemented. (SOMP 02-02 Section 5.5.3)
B. Incorrect. While it is the responsibility of the WCC SRO to determine the risk level as mentioned above it is the responsibility of the OSM to evaluate the restoration plan and who has the final authority to implement.
C. Incorrect. On-shift CRS is not responsible for determining risk. On-Shift CRS responsibility in the risk management process is to maintain awareness of current electronic Risk Assessment color conditions for each Unit. He is to immediately notify the WCC SRO or any emergent equipment problems but is not responsible for determining the change is risk status for the affected Unit. (SOMP 02-02 Section 5.7)
D. Incorrect. On-shift CRS is not responsible for either task

Technical Reference(s)	SOMP 02-02 p 7	(Attach if not previously provided)
	OP-MC-ADM-MRA, p49, 51 (Rev 9)	

Proposed references to be provided to applicants during examination: None

Learning Objective: ADM-MRA, Obj # 7 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content: 55.41

55.43 3

Comments:
KA and SRO level is matched because the item evaluates SRO responsibilities during equipment or procedure step failure during a temporary test procedure

5.5 Operations Shift Manager (OSM):

5.5.1 The OSM maintains an awareness of current Electronic Risk Assessment color conditions for each unit.

5.5.2 The OSM provides guidance and direction during resolution of any scheduling conflicts identified by the Electronic Risk Assessment tool.

5.5.3 **WHEN** entering an orange or red condition from emergent work, the OSM will evaluate the restoration plan and have final authority on whether the plan is implemented. Additionally, at their discretion, the OSM may require development of a written risk management plan for actions to be taken in the event of further degradations.

5.5.4 The OSM is responsible for communicating the risk assessment results to OPS Shift personnel at the beginning of each shift.

5.6 Work Control Center SRO (WCC SRO):

5.6.1 Prior to releasing on-line work, an SRO assigned to the WCC will verify the work is part of the committed schedule, has the correct PRA code for the current plant configuration, and is being performed at the scheduled time.

5.6.2 During non-core business hours when the Work Window Manager (WWM) or Risk Site Expert is unavailable, it is the responsibility of the WCC SRO to evaluate the current risk when the existing conditions do **NOT** match those evaluated based on the schedule due to emergent work or schedule carry-overs.

5.7 Control Room Supervisor (CRS):

5.7.1 The CRS maintains an awareness of current Electronic Risk Assessment color conditions for each unit.

5.7.2 Immediately notifies the WCC SRO of any emergent equipment problems.

5.7.3 Remains cognizant of all unavailable equipment and any required contingency plans.

6. Reporting Requirements

None

- Operations Superintendent:
  - Has the final responsibility to ensure risk assessment has been performed in accordance with WPM 609 and 608.
- Operations Work Process Manager (OWPM):
  - Has overall responsibility for providing operation focus into the site work scheduling plan.
- OWPM Group:
  - Will perform a detailed schedule review utilizing the Electronic Risk Assessment Tool Results to ensure compliance with Tech Spec's, SLC, and Probabilistic Risk Assessment (PRA) concerns. The group will provide guidance and assistance in creating the schedule for the execution week. Also the group is responsible for assisting the Work Control Center (WCC) Supervisor with the final risk assessment, and assisting in conflict resolution.

Objective # 7

- **Operation Shift Manager (OSM):**
  - **Maintains the role of command and control of the plant. Maintains an awareness of current Electronic Risk color conditions or overall shutdown risk level. Provide guidance and direction during resolution of conflicts. WHEN entering an Orange or Red condition from emergent work, the OSM will evaluate the restoration plan and have the final authority whether the plan is implemented. Additionally, OSM may require development of a written risk management plan for actions to be taken in the event of further degradations.**
- **WCC Supervisor:**
  - **Will utilize the Electronic Risk results as an aid to ensure minimal risk consequences occur from scheduled work.**
  - **Prior to releasing work, the WCC Supervisor will verify the work is part of the committed schedule, and is being performed at the scheduled time.**
  - **For emergent work, the supervisor will review work order activities and assign an appropriate PRA code so that activities are appropriately included in the Electronic Risk analysis. Assist the Work Window Manager (WWM) in evaluating emergent work against the current schedule utilizing Electronic Risk.**
  - **IF the R&R requirement is changed from planned work or if the R&R itself is revised, evaluate any potential change in risk.**
  - **WHEN performing any procedure, ensure the planned configuration is evaluated for risk. IF the component is rendered "Unavailable", then perform a "What-If" scenario in the risk assessment tool.**

- **During non-core business hours when the WWM or Risk Site Expert is unavailable, it is the responsibility of the WCC Supervisor to evaluate the current risk when the existing SSC's do NOT match those evaluated based on the schedule.**
- **WHEN there is any doubt concerning the applicability of any PRA Code, the conservative choice is to apply the code for that SSC.**

Objective # 7

- **CONTROL ROOM SRO:**
  - **The Control Room SRO is responsible to maintain an awareness of current Electronic Risk Assessment Tool color conditions on his/her Unit. This includes an awareness of the work causing the increased level of risk along with contingency plans for system restoration.**

3.1.2 Items To Consider:

- All work order tasks and/or maintenance activities must be risk assessed against actual plant configuration as required by 10CFR50.65.
- Variances from the established schedule require re-evaluation. Changing the plant configuration or the work sequence may invalidate the risk assessment.
- High Safety Significant SSCs are **NOT** always the same as TS/SLCs. For example, Instrument Air or Spent Fuel Cooling is **NOT** a TS/SLC item, however is in Electronic Risk Assessment Tool.
- Inoperable items should be considered "Unavailable" unless an evaluation to determine its' availability has been performed (e.g., *NO_CODE on work order task).
- All risk evaluations can **NOT** be predetermined. Persons performing the risk evaluation should use additional sources as necessary to perform the evaluation (i.e., plant drawings, procedures, previous evaluations, knowledge, & training).
- Electronic Risk determination is evaluated by two distinct methods;
  - Deterministic and Probabilistic.
    - ➢ Probabilistic Risk Assessment (PRA) is based on the adverse affect on Core Damage Frequency when a SSC is determined to be "Unavailable".
    - ➢ Deterministic Risk Assessment is based on the determined risk (expert judgment) when a SSC is determined to be "Unavailable".
- As with all aspects of nuclear power operation, if in doubt, be conservative.
- **WHEN** coding, realize that a specific code may **NOT** be available for the SSC(s) which are unavailable. In this case, if warranted, use of an alternate code that removes the same function may be necessary.

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
7	Explain the roles and responsibilities associated with the Electronic Risk Assessment Tool work release process, for the following Operations personnel: • OSM (Operations Shift Manager) • Control Room SRO • WCC (Work Control Center) SRO ADMMRA007		X	X	X	X
8	Deleted ADMMRA008		X	X	X	X
9	Deleted ADMMRA0010		X	X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.2.22	
	Importance Rating		4.7

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: SRO 96

Given the following:

- Unit 1 is in Mode 3.
- Shutdown Banks are withdrawn.
- NC system pressure has increased to 2772 psig.

In accordance with Tech Spec Bases, which **ONE** of the following **CORRECTLY** describes all the components that are assumed to operate at their setpoints to ensure that NC pressure remains below the Technical Specification Safety Limit, and **THE MAXIMUM TIME** allowed to reduce NC pressure to below the Safety Limit?

A. Pressurizer Code Safeties, Main Steam Code Safeties, High Pressure Rx. Trip; 1 hour.

B. Pressurizer Code Safeties, Main Steam Code Safeties, High Pressure Rx. Trip; 5 minutes.

C. Pressurizer PORVs, Main Steam PORVs, High Pressurizer Level Rx. Trip; 1 hour.

D. Pressurizer PORVs, Main Steam PORVs, High Pressurizer Level Rx. Trip; 5 minutes.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. 1 hour allowed for Modes 1 and 2, but Mode 3 and below require pressure to be reduced below the SL within 5 minutes

B. Correct.

C. Incorrect. High PZR level trip is a backup and is not considered for safety limit protection. Plausible because it is a valid reactor trip. PORVs are not credited in the accident analysis for High RCS pressure. Plausible because they will operate and do perform a safety related function

D. Incorrect. High PZR level trip is a backup and is not considered for safety limit protection, but time is correct. PORVs are not credited in the accident analysis for High RCS pressure

Technical Reference(s)	TS 2.1.1 and basis; TS 3.3.1 and basis	(Attach if not previously provided)
	PS-NC Rev 30	
	IC-IPE Rev 28	

Proposed references to be provided to applicants during examination: None

Learning Objective: IC-IPE Obj 10, 14; PC-NC Obj 17 (As available)

Question Source:	Bank #	X	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam Vogtle (year?)

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	
	55.43	1, 2

Comments:

NRC developed test item for Vogtle exam

KA is matched because item requires knowledge of LCOs, NSSS setpoints and basis for setpoints, and action in a lower mode specific to protection of a safety limit

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 limits specified in the COLR for four loop operation; and the following SLs shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥1.14 for the WRB-2M CHF correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained <5080 degrees F, decreasing 58 degrees F for every 10,000 MWd/mtU of fuel burnup.

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.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the transient peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, RCS Flow Rate, ΔI, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the UFSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The Figure provided in the COLR shows the loci of points of Fraction of Rated Thermal power, RCS Pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, and that the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95 / 95 DNB criteria) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal

Objective # 15

The common discharge line from the PORVs has a temperature element which provides indication for PORV discharge temperature via meter located on 1(2)MC10 and an alarm on **1(2)AD6 " Pzr PORV Disch Hi Temp" (setpoint 140^0 F).** This indication is used to assist in identifying if a PORV is leaking which has Tech Spec implications.

Objective # 16

Each PORV has a loop seal between the PORV and its electric isolation. These loop seals were designed to assist in preventing the leakage of H_2 through the PORV valve seat. Industry concerns were raised over potential water slug acceleration and subsequent piping damage when a PORV or safety was opened.). It was determined, as documented in PIP 1-M94-1470 that in this application a water slug would not damage the piping to the extent that the PORVs would become inoperable. However, each loop seal between the PORV block valve and PORV has a drain line which normally drains the condensate back to the pressurizer **(Refer to Drawing 7.10).** These drain valves do not have to be open for the PORVs to be operable. Each drain line has normally open isolation valve (NC269, 270, 271). Each valve is solenoid actuated and can be operated from the control room on 1(2)MC10. The drain lines join to a common line which can be isolated by manual valve NC 61. Sample valve NM6A,C, and NM7B provide the flow path for this line. If a PORV is leaking, its associated block valve and loop seal drain isolation valve will be closed to prevent bypass of the block valve function.

2.8 Pressurizer Code Safety Valves

Objective #15, 17,18

The purpose of the **safety valves** (NC1,2 and 3) is to prevent the NCS from being pressurized above its safety limit of 2735 psig. Each unit has three totally enclosed pop-type, spring loaded, self-actuated safety valves set at 2485 psig. The combined capacity of the three valves is greater than or equal to the maximum surge capacity following a complete loss of load without a reactor trip. The 6 inch pipes connecting the pressurizer nozzles to their respective code safety valves are shaped in the form of a loop seal. Originally, the loops seals were designed to collect condensate, as a result of normal heat losses to the containment atmosphere. The condensate was to prevent any leakage of hydrogen gas or steam through the safety valve seats. However, a concern was raised that if a water slug were to be accelerated when the safety valve opened, the resultant water hammer could result in severe damage to the valve and/or downstream piping which could result in an unisolable leak from the steam space of the pressurizer. Therefore the safety valve internals were replaced with a design that could seal on steam and drains for the loops seals were added to continuously drain condensate back to the pressurizer via one of the upper pressurizer level detector penetrations. Each of these drain lines has a strap on RTD which provides temperature indication on the OAC. LO (approx. 110 degrees) and LO LO (approx. 100 degrees) OAC alarms are provided to notify Engineering to assess operability of the Safety Valves at low temperatures.

Objective # 10

Power Range NIS Low Setpoint (2/4 channels = 25%) - Protects against startup accidents. The trip can be manually blocked when 2/4 PR channels > 10% (P-10) by using the two control board switches, one per train. The control board provides indication of the bistable block. This trip is auto-reinstated when 3/4 PR channels < 10% (P-10).

Power Range NIS High setpoint (2/4 channels = 109%) - protects against an overpower condition which could lead to a DNB concern. This circuit also provides a rod withdrawal stop when 1/4 channels > 103% power (C-2).

Power Range Positive (+) Rate (2/4 channels + 5% in 2 sec) - protects against an ejected rod accident for DNB concerns.

Pressurizer High Pressure (2/4 channels = 2385 psig) - Protects against losing NC system integrity.

Pressurizer Low Pressure (2/4 channels = 1945 psig) - protects against DNB due depressurization. This "at-power" trip protection is auto-blocked < 10% power (P-7) and is automatically reinstated > P-7.

Pressurizer High Level (2/3 channels = 92%) - protects system integrity by preventing the passage of water through the safeties. This "at-power" trip protection is auto-blocked < 10% power (P-7) and is automatically reinstated > P-7.

OTΔT (2/4 channels = variable) - provides DNB protection. DNB causes a large decrease in the heat transfer coefficient between the fuel surface and the coolant, resulting in high fuel clad temperature. The setpoint is a function of the 120% full power ΔT, Tavg, Pressurizer Pressure, and Δ Flux. Pressures below 2235 psig cause the setpoint to decrease while pressures above 2235 psig cause an increase in the setpoint. Tavg above 585 °F causes the setpoint to decrease while Tavg below 585 °F causes an increase in the setpoint. A Δ Flux more positive than the limit in the COLR (positive breakpoint) causes the setpoint to decrease. This circuit also provides a rod withdrawal stop and Turbine Runback 2% (C-3) below the trip setpoint.

OPΔT (2/4 channels = variable) - protects against excessive fuel centerline temperature due to high fuel rod power density (kW/ft). The setpoint is a function of the 109% full power ΔT, Tavg, Rate of Tavg increase, and Δ Flux. Tavg above 585 °F cause the setpoint to decrease with no credit for Tavg below 585 °F. A Δ Flux more positive than the limit in the COLR (positive breakpoint) or more negative than the limit in the COLR (negative breakpoint) causes the setpoint to decrease. This circuit also provides a rod withdrawal stop and Turbine Runback 2% (C-4) below the trip setpoint.

NC Pump Bus Low Voltage (2/4 busses = 74%) - this anticipatory loss of coolant flow trip protects against DNB. This "at-power" trip protection is auto-blocked < 10% power (P-7) and is automatically reinstated > P-7.

7.5 Reactor Trips (3/27/01)

REACTOR TRIP	SETPOINT	LOGIC	PERMISSIVES	BASES
MANUAL	Sw. turned 45°	1/2 sw.		operator judgment
S.R. NI HIGH	10^5 CPS	1/2 ch.	P6, P10	uncontrolled rod withdrawal/ startup accidents
I.R. NI HIGH	amps-25% power	1/2 ch.	P10	uncontrolled rod withdrawal/ startup accidents
P.R. NI LOW	25% power	2/4 ch.	P10	reactivity excursion from low powers
P.R. NI HIGH	109% power	2/4 ch.		reactivity excursion from all powers DNB
P.R. POS RATE	+5%/2 sec	2/4 ch.		DNB (rod ejection)
PZR HIGH PRESS	2385 psig	2/4 ch.		coolant system integrity
PZR LOW PRESS	1945 psig	2/4 ch.	P7	DNB
PZR HIGH LEVEL	92%	2/3 ch.	P7	water through safeties (system integrity)
OTΔT	$\Delta T \geq OT\Delta Tsp$	2/4/ ch.		DNB
OPΔT	$\Delta T \geq OP\Delta Tsp$	2/4 ch.		KW/FT
NCP BUS LOW VOLT	74% of normal	2/4 ch.	P7	DNB (anticipatory loss of flow)
NCP BUS LOW FREQ	56 Hz	2/4 ch.	P7	DNB (anticipatory loss of flow)
S/G LO-LO LVL	17%	2/4 in 1/4 s/g		loss of heat sink
1 LOOP LOSS OF FLOW	88%	2/3 in 1/4 loops	P8	DNB
2 LOOP LOSS OF FLOW	88%	2/3 in 2/4 loops	P7	DNB
SAFETY INJECTION	any S/I signal actuated	1/2 S/I trains		trip reactor if trip not generated by trip instrumentation
GENERAL WARNING ALARM	loose card, loss of voltage, train in test, by-pass bkr connected/closed, logic ground return fuse blown	2/2 alarms		loss of protection
TURBINE TRIP	low Auto-stop oil press <45 psig or all 4 stop valves closed	2/3 ASO Press switches 4/4 valves	P8	trip reactor on turbine trip

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Pressurizer Pressure—Low

The Pressurizer Pressure—Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels of Pressurizer Pressure—Low to be OPERABLE.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure—Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent (P-13)). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, power distributions that would cause DNB concerns are unlikely.

b. Pressurizer Pressure-High

The Pressurizer Pressure-High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires four channels of the Pressurizer Pressure—High to be OPERABLE.

The Pressurizer Pressure-High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trips 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 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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

9. Pressurizer Water Level-High

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 setpoints are based on percent of instrument span. 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 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 overfilling 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

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. The setpoints are based on the minimum flow specified in the

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
12	Concerning the Pzr cold and hot calibrated level indication:					
	• state the purpose of this indication		X	X	X	
	• describe how the operator corrects the indicated level for temperature		X	X	X	X
	• state the problems which can occur if the level is not corrected for temperature		X	X	X	X
13	State the purpose of the pressurizer power operated relief valves.	X	X	X	X	
14	List the parameters and setpoints associated with the NCS relief valves.	X	X	X	X	
15	Describe the indications which would be used to identify a leaking Pzr PORV or safety.		X	X	X	X
16	Concerning the Pzr PORV loop seals:					
	• what was their original purpose		X	X	X	
	• why are they continuously drained during operation		X	X	X	
	• describe the operational concern of leaving the drain valve open while its associated PORV is leaking		X	X	X	X
	• state from where the loop seal drain valves are operated.		X	X	X	X
17	State the purpose of the Pzr Code safety valve.	X	X	X	X	
18	Concerning the Pzr Code safety valves loop seals:					
	• what was their original purpose		X	X	X	
	• why are they continuously drained during operation		X	X	X	X
19	State the purpose of the pressurizer relief tank and the design features which accomplish the purpose.	X	X	X	X	

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
8	Describe the function of the First-Out annunciator panel. ICIPE008			X	X	
9	Given a Limit and/or Precaution associated with an operating procedure, discuss its basis and applicability. ICIPE009		X	X	X	X
10	List all the Reactor Trip Signals including the setpoints, logic permissives and bases/protection afforded by each. ICIPE010		X	X	X	X
11	List all the protective system permissive ("P" signal) interlocks to include input parameter(s), logic and function. For interlocks which provide Trip block, state the Trips affected and whether Auto or Manual block. ICIPE011			X	X	X
12	List all the protection system control ("C" signal) interlocks including logic and functions. ICIPE012			X	X	X
13	Briefly describe the incident that occurred at Salem Nuclear Plant and how this event affected McGuire Reactor Trip Breaker operation. ICIPE013			X	X	X

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
14	Concerning the Technical Specifications related to the Reactor Protection System;					
	• Given the LCO title, state the LCO (including any COLR values) and applicability.			X	X	X
	• For any LCO's that have action required within one hour, state the action.			X	X	X
	• Given a set of parameter values or system conditions, determine if any Tech Spec LCO's is (are) not met and any action(s) required within one hour.			X	X	X
	• Given a set of plant parameters or system conditions and the appropriate Tech Specs, determine required action(s).			X	X	X
	• Discuss the basis for a given Tech Spec LCO or Safety Limit.				X	*
	* SRO Only ICIPE014					

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	2.3.14	
	Importance Rating		3.8

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: SRO 97

Given the following:

- A load reduction from 100% to 60% was performed on Unit 1 in the last 30 minutes due to a Feedwater Control problem.
- The following alarms are received:
  - 1EMF-48 REACTOR COOLANT HIGH RAD
  - 1EMF-18, REACTOR COOLANT FILTER 1A
- Chemistry sample indicates that the high activity is due to failed fuel.
- Dose-Equivalent Iodine-131 is approximately 5 microcuries per gram.
- The crew enters AP/18, High Activity in Reactor Coolant.

Which ONE of the following actions will be performed in accordance with AP/18, and which ONE of the following describes the technical specification implications of this condition?

REFERENCE PROVIDED

A. Raise Letdown flow to 120 GPM; plant shutdown and cooldown to less than 500°F must be performed.

B. Raise Letdown flow to 120 GPM; plant operation may continue with increased NC SYSTEM sampling frequency.

C. Ensure Mixed Bed Demin is in service and evaluate use of Cation Bed Demin; plant shutdown and cooldown to less than 500°F must be performed.

D. Ensure Mixed Bed Demin is in service and evaluate use of Cation Bed Demin; plant operation may continue with increased NC SYSTEM sampling frequency.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Letdown flow is raised only for crud burst. Failed Fuel is indicated by iodine activity. TS shutdown required only after being above 3.4.16-1 acceptable operation.
B. Incorrect. Letdown flow is raised only for crud burst. Failed Fuel is indicated by iodine activity, as decribed by conditions presented.
C. Incorrect. TS shutdown required only after being above 3.4.16-1 acceptable operation. This condition is above TS steady state limit but below the transient limit on the curve
D. Correct.

Technical Reference(s): AP/18 Rev 2 and Basis Document (Attach if not previously provided)
TS 3.4.16

Proposed references to be provided to applicants during examination: TS Figure 3.4.16-1

Learning Objective: ____ (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 2, 4

Comments:

KA is matched because the item evaluates understanding of a fuel failure vs a crud burst. SRO level because the SRO must determine appropriate action based upon evaluation of this condition. The action taken is required by technical specifications

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	--------------------Note-------------------- LCO 3.0.4.c is applicable. --	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ $100/\bar{E}$ μCi/gm.	7 days
SR 3.4.16.2	-----------------------------NOTE----------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.3	-----------------------------NOTE----------------------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours. ---	
	Determine $\bar{E}$ from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours.	184 days

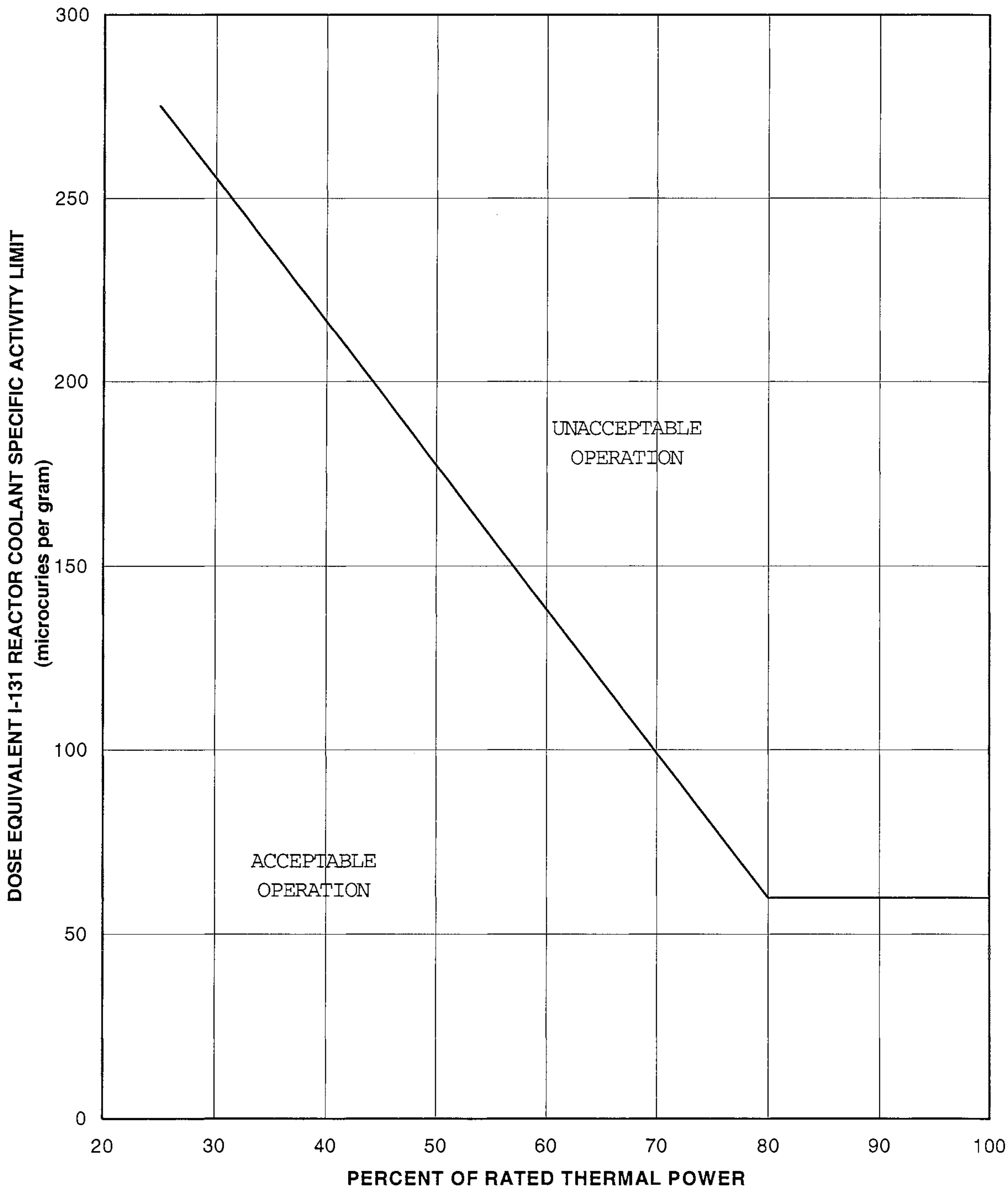


Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. Symptoms

- **"1EMF-48 REACTOR COOLANT HI RAD" alarm**
- **"1EMF-18 REACTOR COOLANT FILTER 1A" alarm**
- **"1EMF-19 REACTOR COOLANT FILTER 1B" alarm**
- **Chemistry sample results indicate an unexpected increase in NC System activity.**

C. Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 1. **Check 1NV-127A (L/D Hx Outlet 3-Way Temp Cntrl) - ALIGNED TO DEMIN.**	__ **Align valve to "DEMIN" position.**
2. **Determine cause of high activity:**	
__ a. Request Chemistry to check decontamination factor of mixed bed demineralizer.	
__ b. Notify Chemistry to perform an NC System isotopic analysis to determine if high activity is from a crud burst or failed fuel.	
__ 3. **IF AT ANY TIME it is determined that high activity is from crud burst, THEN raise letdown flow to 120 GPM.**	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

4. **IF AT ANY TIME it is determined that high activity is from failed fuel, THEN:**

   __ a. Ensure mixed bed demineralizer in service.

   __ b. Notify Chemistry to consult with Reactor Group and RP to determine if the cation bed demineralizer should be placed in service.

   __ c. **IF AT ANY TIME** Chemistry requests cation bed demineralizer be placed in service, **THEN** place in service **PER** OP/1/A/6200/001D (Chemical and Volume Control System Demineralizers), Enclosure 4.3 (Removing/Returning the Cation Bed Demineralizer from/to Service).

   __ d. **REFER TO** RP/0/A/5700/000 (Classification of Emergency).

   __ e. Notify Reactor Group to perform OP/0/A/6550/017 (Estimate of Failed Fuel Based on Iodine-131 Concentration).

__ 5. **Notify Radwaste to ensure VCT H_2 purge flow is established.**

__ 6. **REFER TO Tech Spec 3.4.16 (RCS Specific Activity).**

END

STEP 3:

PURPOSE:

Reduce redeposition of crud throughout the plant.

DISCUSSION:

At the normal letdown flow rate of 75 gpm, it takes almost 21 hours to pass one entire volume of reactor coolant through the NV System. But a letdown flow of 120 gpm will circulate one entire volume of reactor coolant in approximately 12 hours (at 120 gpm letdown flow, 50% of the crud is removed every 12 hours).

REFERENCES:
Primary Chemistry Lesson Plan OP-MC-CH-PC

STEP 4:

PURPOSE:

To clean up high activity associated with failed fuel and ensure appropriate classification of emergency is made.

DISCUSSION:

Step 4.a ensures mixed bed demin in service to facilitate removal of both the ion types produced by failed fuel (halogens and soluble metal ions).

Step 4.b notifies Chemistry to determine if the cation bed should be placed in service so they can get with Reactor Group, RP, and themselves to weigh the pros and cons of placing the cation bed in service. While the cation will remove the soluble metal ions like Cesium, in doing so it will also remove the Lithium ion that is used for PH control. Operating with PH out of spec must be weighed against the urgency of removing the failed fuel ions (dose control, etc.)

Step 4.c gets the cation bed in serivce, if requested using the OP.

Step 4.d is a reference to RP/0/A/5700/000 (Classification of Emergency) to ensure the proper declaration is made. If a plant shutdown required by T.S. 3.4.16 (RCS Specific Activity) is commenced, a Notification of Unusual Event is declared based on failed fuel. For grosser failures beyond the T.S. limits, other classification levels may be reached.

Step 4.e is a quick gross guess at the extent of the failed fuel. The Reactor Group has more qualitative tools that they'll implement as warranted, but this is a quick estimate. Basically, this procedure takes the I-131 concentration in uCI/ml and divides by a number depending on initial conditions (normal, clad damage, severe fuel overtemperature, or fuel melting), with correction factors for sampling temperature and power history:

A. Normal:	I-131 uCI/ml ÷ 1.8 uCI/ml = Percent failed fuel
B. Clad damage	I-131 uCI/ml ÷ 83.7 uCI/ml = Percent failed fuel
C. Severe Fuel Overtemperature	I-131 uCI/ml ÷ 1535 uCI/ml = Percent failed fuel
D. Fuel Melting	I-131 uCI/ml ÷ 2790 uCI/ml = Percent failed fuel

As seen above, the more the fuel cladding is stressed by the failed fuel mechanism, the more activity is expected for a given percentage of failed fuel.

REFERENCES:
RP/0/A/5700/000 (Classification of Emergency), OP/0/A/6550/17 (Estimate of Failed Fuel Based on Iodine-131)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	2.3.6	
	Importance Rating		3.8

Ability to approve release permits

Proposed Question: SRO 98

Unit 1 is shutdown in mode 6 refueling.
A Radwaste Operator brings a liquid radiological release permit to the SRO for approval.
Given the following information on the permit:

- Release ID = WMT - B
- RC Pumps running = 4
- RC Pumps assigned to release = 3
- Total RC Pumps required = 1
- Allowable release rate = 1.61E+05 gpm
- Recommended release rate = 6.00E+01 gpm
- EMF-49 (L) (LIQUID WASTE DISCH) in service = yes
- EMF background = 4.49E+03
- Trip 1 setpoint = 8.97E+03
- Trip 2 setpoint = 1.34E+04

If no other releases are in progress, which one of the following actions is correct for approval of this release permit?

A. The release may not be approved because there is an error in the number of RC pumps required

B. The release may not be approved because the EMF-49(L) trip setpoints are not correct

C. The release may not be approved because the release rate is not correct

D. The release may be approved as presented if a source check of EMF-49(L) is performed successfully.

Proposed Answer: A

Explanation (Optional):
A. Correct the remarks section states 4 RC pumps are required but the number of RC pumps required is listed as 3 in the RC pump data section
B. Incorrect: - nothing wrong with EMF-49L trip setpoints
Plausible: - background < trip 1 < trip 2
C. Incorrect: - allowable release rate < recommended release rate
Plausible: - if candidate does not understand this requirement
D. Incorrect: - the RC pumps required is not correct, but otherwise this is correct

Technical Reference(s): OP-MC-WE-RLR, Rev 13 (Attach if not previously provided)
OP/0/B/6200/107 p 6

Proposed references to be provided to applicants during examination: None

Learning Objective: OP-MC-WE-RLR obj 3 (As available)

Question Source: Bank # X
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 4

Comments:
KA is matched because the item evaluates requirements for issuing a radioactive liquid waste release permit. SRO level because the SRO is responsible for authorizing the release based on given conditions

3.16 CR SRO performs the following steps: {PIP M-03-01124} {PIP M-04-03470}

_____ SRO 3.16.1 Determine operability status of the following components and circle "Yes" or "No" to so indicate:

0WMLP5140 (B WMT Pump Disch Flow) [i.e., 0WMCR5130 (Waste Mon Tank Pumps Disch Flow) or 0WMFT5140 (B Waste Monitor Tank Pump Disch Flow)]	**(Yes/No)**
1WP-35 (WMT & VUCDT to RC Cntrl)	**(Yes/No)**
1WP-37 (Liquid Waste to RC Cntrl)	**(Yes/No)**
0EMF49 (Liquid Waste Disch Radiation Monitor)	**(Yes/No)**
0WMFS5440 (0EMF49 Outlet Flow) {PIP M-03-02673}	**(Yes/No)**

_____ SRO 3.16.2 **IF** any component listed in Step 3.16.1 is inoperable, notify Radwaste Chemistry and return LWR Document.

_____ SRO 3.16.3 Ensure the following items on LWR Document are complete:

- Number of "RC Pumps Running" is greater than or equal to "RC Pumps Assigned to this Release".
- Number of "RC Pumps Running" is greater than "Total RC Pumps Required (all concurrent Releases)".
- "Recommended Release Rate (gpm)" is less than "Allowable Release Rate (gpm)".
- 0EMF49L is operable and in service.
- 0EMF49 source check performed.
- "Expected CPM" is less than "Trip 1 Setpoint" and "Trip 2 Setpoint".

_____ SRO 3.16.4 **WHEN** approved for release, place signature, date, and time of SRO authorization on LWR Document.

- If a site assembly occurs during a release, Chemistry will secure the release.
- In the event of any problem with an EMF that would require a work request, contact RP for initiation of the work request.

2.2 Releasing a WMT

Refer to Drawing 7.2, WMT Subsystem. Radwaste initiates the procedure. They select the tank to be discharged, recirculate it for mixing, and obtain a sample. Next, the sample is analyzed. Radwaste delivers the sample to RP for isotopic analysis.

RP then generates the Release Discharge Document using the RETDAS Computer Program. RP assigns the next sequential LWR number and calculates the recommended release rates.

Objective #2

The **Recommended Release Rate** is the lesser of:

- Maximum System Release Rate for WMT = 120 gpm, OR
- Allowable Release Rate.

The "Allowable Release Rate" is determined by the amount of activity present in the tank.

RP indicates the "EMF Utilized", which is 0EMF-49L for WMT releases. RP next indicates the EMF background cpm, expected cpm, trip 1, and trip 2 setpoints.

Objective #3

RP then takes the release procedure and the discharge document to the control room. The SRO authorizes the release by signing the release document. The SRO authorizing the release ensures the following:

- Ensures the LWR document agrees with the Radwaste procedure (i.e., the procedure directs releasing the same tank that is listed on the LWR.)

- Operability of EMF 49 and the discharge release valves (1WP-35 & 37). The pump discharge flow meter and the EMF outlet flow meter also needs to be operable.
  - If any of these are inoperable, then the LWR document is returned to Radwaste.

Prior to signing the LWR document, the SRO should review the following:

- The required number of RC pumps are in operation

NOTE: **The RC minimum flow interlock is set to the minimum # of pumps required for the release. If the total # RC pumps running is less than the selected number, 1WP-35 and 1WP-37 will close.**

- The "Recommended Release Rate" is less than or equal to the "Allowable Release Rate".

Objective #4

- The proper EMF is utilized. (For a WMT release, this is EMF-49)
- A source check has been performed on EMF-49.
- The "Expected CPM of the EMF" and the "EMF Trip I Setpoint" are less than the "EMF Trip II Setpoint"
- Any special instructions

The RO ensures the LWR number is in autolog (normally logged by Chemistry). The purpose of the log is to maintain an account, in the control room, of all LWR/GWR releases. The information contained in the log is:

- Release #
- Start Time & Date
- Stop Time & Date
- Volume Released
- Any unusual events encountered during the release

Now the release is ready to be started. Radwaste notifies the SRO the discharge is initiated. The Radwaste technician aligns the WMT to be discharged to RC and commences the release.

Objective #5

Based on an agreement between MNS RP, GO RP, and MNS Radwaste, releases that are interrupted by a Trip 2 on EMF49 may be reinitiated up to a maximum of two times without resampling before terminating the release procedure. **Specifically, 3 release attempts are allowed. If EMF-49 Trip 2 occurs on the third release attempt, the LWR must be terminated, the WMT must be re-sampled and new LWR paperwork must be generated.** When the release is terminated, the SRO is notified. Autolog is updated, and the Release document is closed out, with the SRO signing the Release document acknowledging the completion.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		2	2	2

OBJECTIVES

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the systems that are used to release radioactive liquids to the environment. WERLR001			X	X	X
2	Given a completed LWR, state the recommended release rate. WERLR002			X	X	X
3	Given the applicable procedure and LWR paperwork, review the LWR and determine if a release can be initiated. WERLR003			X	X	X
4	Given a completed LWR, state the proper EMF to be used for the release. WERLR004			X	X	X
5	Evaluate plant parameters to determine any abnormal system conditions that may exist. WERLR005			X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.46	
	Importance Rating		4.2

Ability to verify that the alarms are consistent with the plant conditions.

Proposed Question: SRO 99

Given the following conditions:

A transient has occurred on Unit 2 resulting in the following alarms:

- OTDT RUNBACK/ROD STOP ALERT
- TREF/T-AUCT ABNORMAL

Reactor power indicates the following:

- N41 – 104.1%
- N42 – 103.2%
- N43 – 104.3%
- N44 – 102.9%
- Tavg is 590 degrees F

Which ONE (1) of the following has occurred, and what is the technical specification implication of the event?

A. Uncontrolled Rod Withdrawal; Linear Heat Rate and Hot Channel Factors may be challenged.

B. Uncontrolled Rod Withdrawal; Shutdown Margin assumptions for anticipated operational transients may be invalid.

C. Secondary Steam Leak; Linear Heat Rate and Hot Channel Factors may be challenged.

D. Secondary Steam Leak; Shutdown Margin assumptions for anticipated operational transients may be invalid.

Proposed Answer: A

Explanation (Optional):

A is correct because core power is increasing and LHR is a function of rods.

B is incorrect because SDM is a function of several parameters, and the positive reactivity added by rod withdrawal is cancelled by the negative reactivity from power defect and MTC.

C is incorrect because a steam leak would result in a higher power, but Tavg would be lower, not higher. Tave is currently about 4-5 degrees above program

D is incorrect for same reason as C, and basis is incorrect, but plausible because shutdown margin would be the concern if a steam leak were occurring

Technical Reference(s): AP-01 (Rev 14) and Basis Document (Rev 5) (Attach if not previously provided)

TS 3.2.1 Basis

CTH-CP Rev 9

Proposed references to be provided to applicants during examination: None

Learning Objective: CTH-CP Obj 1 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam 2006 Exam 100 Modified

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

KA matched because the item evaluates understanding for the cause of alarms, related to current plant conditions. SRO level because the item evaluates knowledge of accident analysis assumptions and core operating limits as stated in TS basis

Given the following conditions:

A transient has occurred on Unit 1 resulting in the following alarms:

- OTDT RUNBACK/ROD STOP ALERT
- ROD CONTROL URGENT FAILURE
- OPDT REACTOR TRIP

Reactor power indicates the following:

- N41 – 105.2%
- N42 – 106.2%
- N43 – 105.9%
- N44 – 106.1%
- Tavg is 581 degrees F

Which ONE (1) of the following has occurred, and which procedure(s) is/are required to be implemented?

A. Uncontrolled Rod Withdrawal; E-0, Reactor Trip or Safety Injection.

B. Uncontrolled Rod Withdrawal; AP-14, Rod Control Malfunctions.

C. SG Safety Valve opened coincident with a rod control failure; E-0, Reactor Trip or Safety Injection.

D. SG Safety Valve opened coincident with a rod control failure; AP-01, Steam Leak and AP-14, Rod Control Malfunctions.

Answer: C

In some applications, heat transfer is discussed in relationship to a heat transfer rate per unit area............... $\frac{BTU / hr}{ft^2} = \frac{BTU}{hr - ft^2} = \frac{\dot{Q}}{A}$ = q" ≡ HEAT FLUX.

Therefore, if .. $\dot{Q}$ = UA (ΔT), then.................. $\frac{\dot{Q}}{A}$ = U (ΔT) = U (T_{clad} - $T_{coolant}$).

NOTE: Average HEAT FLUX, at RATED THERMAL POWER (RTP), is 189,800 BTU / hr-ft^2 while MAXIMUM HEAT FLUX, at RTP, is 440,300 BTU / hr-ft^2.

Objective # 5

One of the variables discussed above, local heat generation rate, is synonymous with another term, local power density. Local power density or power density is the term used to describe variations in power distribution throughout the reactor core.

Power density, quite simply, is the amount of power being produced per unit volume of the reactor. Therefore, one would expect the units of power density to be some power related term divided by some volume related term. Such as....

Power Density = $^{Watts}/_{cm^3}$

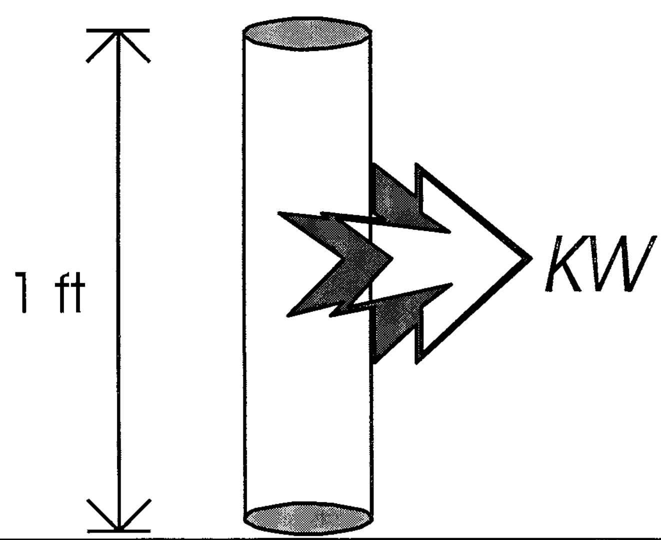
The average power density for either McGuire Unit at full power *(100% RTP) is approximately 340 watts / cm^3.*

Since reactor power production occurs solely within the fuel, power density is *power production per unit volume of fuel.* Ideally, if the power produced from the reactor was evenly distributed, every fuel assembly would contribute an equal amount of the total power, and therefore, every foot of fuel would be producing the average power density. Thus, the power distribution term, Average Power Density.

Objective # 1

Since the reactor fuel rods, within tolerances, are dimensionally identical to one another. A <u>unit length of fuel rod</u>, then, represents a certain volume of fuel. Therefore, **we also define "Linear Heat Generation Rate" *(a power density term)* as the power produced per linear foot of fuel rod (KW / ft).**

See if you can determine **average power density** by performing the example problem below.



3.3 Hot Channel Factors

Two hot channel factors are specified in our Technical Specifications as core limits; the *Heat Flux Hot Channel Factor* and the *Nuclear Enthalpy Rise Hot Channel Factor.* Excessive fuel and cladding temperatures must be avoided during reactor operation to prevent fuel rod burnout. This not only applies during normal operation but also during accident conditions, as well.

Theoretically a *Hot Channel Factor* represents the specific core location with the worst possible performance characteristics. By controlling this location such that the limits on core performance are not exceeded, we are somewhat assured that the entire core is operating within limits. These Hot Channel Factors are calculated by analyzing core data obtained during core *(flux)* mapping.

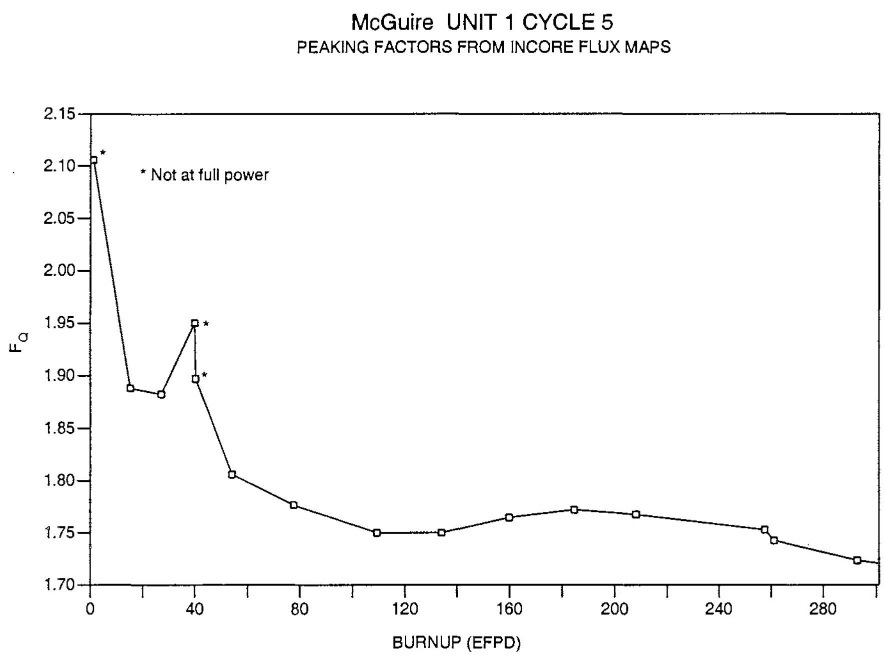
Objective # 1

HEAT FLUX HOT CHANNEL FACTOR

The *Heat Flux Hot Channel Factor, F_Q (X,Y,Z),* is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming normal fuel pellet and fuel rod dimensions:

$$F_Q = \frac{peak\frac{KW}{ft}}{average\frac{KW}{ft}}$$

Therefore, F_Q (X,Y,Z), *Heat Flux Hot Channel Factor,* is calculated based on the data obtained during core or flux mapping with the incore detector system.



F_Q (X,Y,Z) varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution and to a lesser extent, with moderator temperature.

An example of how it can change with fuel burnup is illustrated on Training Drawing 7. 17, F_Q (X,Y,Z) versus Burnup Graph.

F_Q (X,Y,Z) is measured periodically using the incore detector system. Approximately 620 different sets of data are taken along the length of each fuel assembly that is mapped. Each "mapped" fuel assembly will provide data from the same fuel elevation (z). This then provides a representative slice of power distribution throughout the core, at various core elevations (z). These measurements are generally taken with the core at, or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for F_Q (X,Y,Z). However, because this value represents a steady state condition, it does not include the variations in the value of F_Q (X,Y,Z) that are present during non-equilibrium situations. To account for these possible variations, F_Q (X,Y,Z) is limited by pre-calculated factors to account for perturbations from the steady state condition.

Objective # 18

These pre-calculated factors include:

- ***Measurement Uncertainty Factor (F_Q^U)***

Accounts for uncertainties in the flux mapping process and variations in fuel rod dimensions.

- ***Engineering Hot Channel Factor (F_Q^E)***

Provides additional conservatism in the hot channel estimate.

Typically, a five percent conservatism is applied to UMT *(Measurement Uncertainty Factor);* UMT = 1.05 *(1.04 Westinghouse Fuel)*, and a three percent conservatism is applied to *MT (Engineering Hot Channel Factor)*; MT = 1.03 *(1.033 Westinghouse Fuel).*

The *Measured Nuclear Heat Flux Hot Channel Factor* is multiplied by the *Engineering Hot Channel Factor* and the *Measurement Uncertainty Factor* to provide the *Nuclear Heat Flux Hot Channel Factor* at core elevation *z*.

$$F_Q(z) = F_Q^M \cdot UMT \cdot MT$$

Limits for the *Nuclear Heat Flux Hot Channel Factor* as specified within the COLR (Core Operating Limit Report) are related to the *Rated Thermal Power Nuclear Heat Flux Hot Channel Factor,* F_Q^{RTP}.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$)

BASES

BACKGROUND

The purpose of the limits on the values of $F_Q(X,Y,Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_Q(X,Y,Z)$ varies axially (Z) and radially (X,Y) in the core.

$F_Q(X,Y,Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(X,Y,Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_Q(X,Y,Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution and to a lesser extent, with boron concentration and moderator temperature.

$F_Q(X,Y,Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at, or near steady state conditions.

Using the measured three dimensional power distributions, it is possible to derive a measured value for $F_Q(X,Y,Z)$. However, because this value represents a steady state condition, it does not include the variations in the value of $F_Q(X,Y,Z)$ that are present during nonequilibrium situations.

To account for these possible variations, the $F_Q(X,Y,Z)$ limit is reduced by precalculated factors to account for perturbations from steady state conditions to the operating limits.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F for small breaks and there is a high level of probability that the peak cladding temperature does not exceed 2200°F for large breaks (Ref. 1);

b. The DNBR calculated for the hottest fuel rod in the core must be above the approved DNBR limit. (The LCO alone is not sufficient to preclude DNB criteria violations for certain accidents, i.e., accidents in which the event itself changes the core power distribution. For these events, additional checks are made in the core reload design process against the permissible statepoint power distributions.);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_Q(X,Y,Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other Reference 1 criteria must also be met in LOCAs (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, transient strain, and long term cooling). However, the peak cladding temperature is typically most limiting.

$F_Q(X,Y,Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_Q(X,Y,Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_Q(X,Y,Z)$ satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

The Heat Flux Hot Channel Factor, $F_Q(X,Y,Z)$, shall be limited by the following relationships:

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{P} K(Z) \qquad \text{for } P > 0.5$$

$$F_Q^M(X,Y,Z) \le \frac{F_Q^{RTP}}{0.5} K(Z) \qquad \text{for } P \le 0.5$$

LCO (continued)

where: F^{RTP}_Q is the $F_Q(X,Y,Z)$ limit at RTP provided in the COLR, and is reduced by measurement uncertainty, K(BU), and manufacturing tolerances provided in the COLR,

K(Z) is the normalized $F_Q(X,Y,Z)$ as a function of core height provided in the COLR, and

$$P = \frac{THERMAL\ POWER}{RTP}$$

The actual values of F^{RTP}_Q, K(BU), and K(Z) are given in the COLR.

For relaxed AFD limit operation, $F^M_Q(X,Y,Z)$(measured $F_Q(X,Y,Z)$) is compared against three limits:

- Steady state limit, $(F^{RTP}_Q/P) * K(Z)$,
- Transient operational limit, $F^L_Q(X,Y,Z)^{OP}$, and
- Transient RPS limit, $F^L_Q(X,Y,Z)^{RPS}$.

A steady state evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value $F^M_Q(X,Y,Z)$ of $F_Q(X,Y,Z)$. Then, $F^M_Q(X,Y,Z)$ is adjusted by a radial local peaking factor and compared to F^{RTP}_Q which has been reduced by manufacturing tolerances, K(BU), and flux map measurement uncertainty.

K(BU) is the normalized $F^L_Q(X,Y,Z)$ as a function of burnup and is provided in the COLR.

$F^L_Q(X,Y,Z)^{OP}$ and $F^L_Q(X,Y,Z)^{RPS}$ are cycle dependent design limits to ensure the $F_Q(X,Y,Z)$ is met during transients. The expression for $F^L_Q(X,Y,Z)^{OP}$ is:

$$F_Q^L(X,Y,Z)^{OP} = F_Q^D(X,Y,Z) * M_Q(X,Y,Z)/(UMT * MT * TILT)$$

LCO (continued)

where: $F^L_Q(X,Y,Z)^{OP}$ is the cycle dependent maximum allowable design peaking factor which ensures that the $F_Q(X,Y,Z)$ limit will be preserved for operation within the LCO limits. $F^L_Q(X,Y,Z)^{OP}$ includes allowances for calculational and measurement uncertainties.

$F^D_Q(X,Y,Z)$ is the design power distribution for F_Q provided in the COLR.

$M_Q(X,Y,Z)$ is the margin remaining in core location X,Y,Z to the LOCA limit in the transient power distribution and is provided in the COLR for normal operating conditions and power escalation testing during startup operations. UMT and MT are only included in the calculation of $F^L_Q(X,Y,Z)^{OP}$ if these factors were not included in the LOCA limit.

UMT is the measurement uncertainty.

MT is the engineering hot channel factor.

TILT is the peaking penalty that accounts for allowable quadrant power tilt ratio of 1.02 and is specified in the COLR.

The expression for $F^L_Q(X,Y,Z)^{RPS}$ is:

$$F_Q^L(X,Y,Z)^{RPS} = F_Q^D(X,Y,Z) * M_C(X,Y,Z)/(UMT * MT * TILT)$$

where: $F^L_Q(X,Y,Z)^{RPS}$ is the cycle dependent maximum allowable design peaking factor which ensures that the center line fuel melt limit will be preserved for operation within the LCO limits. $F^L_Q(X,Y,Z)^{RPS}$ includes allowances for calculational and measurement uncertainties.

$M_C(X,Y,Z)$ is the margin remaining to the center line fuel melt limit in core location X,Y,Z from the transient power distribution and is provided in the COLR for normal operating conditions and power escalation testing during startup operations. UMT and MT are only included in the calculation of $F^L_Q(X,Y,Z)^{RPS}$ if these factors were not included in the fuel melt limit.

LCO (continued)

The $F_Q(X,Y,Z)$ limits typically define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during a small break LOCA and a high level of probability that the peak cladding temperature does not exceed 2200°F for a large break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the $F_Q(X,Y,Z)$ limits. If $F_Q(X,Y,Z)$ cannot be maintained within the steady state LOCA limits, reduction of the core power is required.

Violating the steady state LOCA limits for $F_Q(X,Y,Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(X,Y,Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(X,Y,Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power. The exception to this is the steam line break event, which is assumed for analysis purposes to occur from very low power levels. At these low power levels, measurements of $F_Q(X,Y,Z)$ are not sufficiently reliable. Operation within analysis limits at these conditions is inferred from startup physics testing verification of design predictions of core parameters in general.

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F^M_Q(X,Y,Z)$ exceeds its steady state limit, maintains an acceptable absolute power density. $F^M_Q(X,Y,Z)$ is the measured value of $F_Q(X,Y,Z)$ and the steady state limit includes factors accounting for measurement uncertainty and manufacturing tolerances. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

ACTIONS (continued)

A.2

A reduction of the Power Range Neutron Flux—High trip setpoints by $\geq$1% for each 1% by which $F^M_Q(X,Y,Z)$ exceeds its steady state limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

Reduction in the Overpower ΔT trip setpoints (value of K_4) by $\geq$ 1% (in ΔT span) for each 1% by which $F^M_Q(X,Y,Z)$ exceeds its steady state limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions since the transient response is limited by the setpoint reduction. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Verification that $F^M_Q(X,Y,Z)$ has been restored to within its steady state and transient limits, by performing SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.1.3 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions. Since $F^M_Q(X,Y,Z)$ exceeds the steady state limit, the transient operational limit and possibly the transient RPS limit may be exceeded. By performing SR 3.2.1.2 and SR 3.2.1.3, appropriate actions with respect to reductions in AFD limits and OTΔT trip setpoints will be performed ensuring that core conditions during operational and Condition 2 transients are maintained within the assumptions of the safety analysis.

B.1 and B.2

The operational margin during transient operations is based on the relationship between $F^M_Q(X,Y,Z)$ and the transient operational limit, $F^L_Q(X,Y,Z)^{OP}$, as follows:

ACTIONS (continued)

$$\% \text{ Operational Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{F_Q^L(X,Y,Z)^{OP}}\right) * 100\%$$

If the operational margin is less than zero, then $F^M_Q(X,Y,Z)$ is greater than $F^L_Q(X,Y,Z)^{OP}$ and there exists a potential for exceeding the peak local power assumed in the core in a LOCA or in the loss of flow accidents. Reducing the AFD by ≥ 1% from the COLR limit for each 1% by which $F^M_Q(X,Y,Z)$ exceeds the operational limit within the allowed Completion Time of 4 hours restricts the axial flux distribution such that even if a transient occurred, core peaking factors are not exceeded. Adjusting the transient operational limit by the equivalent change in AFD limits establishes the appropriate revised surveillance limits.

<u>C.1 and C.2</u>

The margin contained within the reactor protection system (RPS) Overtemperature ΔT setpoints during transient operations is based on the relationship between $F^M_Q(X,Y,Z)$ and the RPS limit, $F^L_Q(X,Y,Z)^{RPS}$, as follows:

$$\% \text{ RPS Margin} = \left(1 - \frac{F_Q^M(X,Y,Z)}{F_Q^L(X,Y,Z)^{RPS}}\right) * 100\%$$

If the RPS margin is less than zero, then $F^M_Q(X,Y,Z)$ is greater than $F^L_Q(X,Y,Z)^{RPS}$ and there exists a potential for $F^M_Q(X,Y,Z)$ to exceed peak clad temperature limits during certain Condition 2 transients. The Overtemperature ΔT K1 value is required to be reduced as follows:

$K1_{ADJUSTED} = K1 - |KSLOPE * \% \text{ RPS Margin}|$

Where $K1_{ADJUSTED}$ is the reduced Overtemperature ΔT K1 value

KSLOPE is a penalty factor used to reduce K1 and is defined in the COLR

% RPS Margin is the most negative margin determined above.

ACTIONS (continued)

Reducing the Overtemperature ΔT trip setpoint from the COLR limit is a conservative action for protection against the consequences of transients since this adjustment limits the peak transient power level which can be achieved during an anticipated operational occurrence. Once the OTΔT trip setpoint is reduced, the available margin is increased. An adjustment is then necessary in the $F^L_Q(X,Y,Z)^{RPS}$ limit, using the increased margin, in order to restore compliance with the LCO and exit the condition. These adjustments maintain a constant margin and ensure that centerline fuel melt does not occur. The Completion Time of 72 hours is sufficient considering the small likelihood of a limiting transient in this time period. Adjusting the transient RPS limit by the equivalent change in OTΔT trip setpoint establishes the appropriate revised surveillance limit.

D.1

If Required Actions A.1 through A.4, B.1, or C.1 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.1.3 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F^M_Q(X,Y,Z)$ is within the specified limits after a power rise of $\geq$ 10% RTP over the THERMAL POWER at which it was last verified to be within specified limits. Because $F^M_Q(X,Y,Z)$ could not have previously been measured in this reload core, power may be increased to RTP prior to an equilibrium verification of $F^M_Q(X,Y,Z)$ provided nonequilibrium measurements of $F^M_Q(X,Y,Z)$ are performed at various power levels during startup physics testing. This ensures that some determination of $F^M_Q(X,Y,Z)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last

SURVEILLANCE REQUIREMENTS (continued)

verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

SR 3.2.1.1

Verification that $F^M_Q(X,Y,Z)$ is within its specified steady state limits involves either increasing $F^M_Q(X,Y,Z)$ to allow for manufacturing tolerance, K(BU), and measurement uncertainties for the case where these factors are not included in the F_Q limit. For the case where these factors are included, a direct comparison of $F^M_Q(X,Y,Z)$ to the F_Q limit can be performed. Specifically, $F^M_Q(X,Y,Z)$ is the measured value of $F_Q(X,Y,Z)$ obtained from incore flux map results. Values for the manufacturing tolerance, K(BU), and measurement uncertainty are specified in the COLR.

The limit with which $F^M_Q(X,Y,Z)$ is compared varies inversely with power above 50% RTP and directly with functions called K(Z) and K(BU) provided in the COLR.

If THERMAL POWER has been increased by ≥ 10% RTP since the last determination of $F^M_Q(X,Y,Z)$, another evaluation of this factor is required 12 hours after achieving equilibrium conditions at this higher power level (to ensure that $F^M_Q(X,Y,Z)$ values have decreased sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2 and 3.2.1.3

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(X,Y,Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, is determined by a maneuvering analysis (Ref. 5).

SURVEILLANCE REQUIREMENTS (continued)

The limit with which $F^M_Q(X,Y,Z)$ is compared varies and is provided in the COLR. No additional uncertainties are applied to the measured $F_Q(X,Y,Z)$ because the limits already include uncertainties.

$F^L_Q(X,Y,Z)^{OP}$ and $F^L_Q(X,Y,Z)^{RPS}$ limits are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and

b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F^M_Q(X,Y,Z)$ is evaluated and found to be within the applicable transient limit, an evaluation is required to account for any increase to $F^M_Q(X,Y,Z)$ that may occur and cause the $F_Q(X,Y,Z)$ limit to be exceeded before the next required $F_Q(X,Y,Z)$ evaluation.

In addition to ensuring via surveillance that the heat flux hot channel factor is within its limits when a measurement is taken, there are also requirements to extrapolate trends in both the measured hot channel factor and in its operational and RPS limits. Two extrapolations are performed for each of these two limits:

1. The first extrapolation determines whether the measured heat flux hot channel factor is likely to exceed its limit prior to the next performance of the SR.

2. The second extrapolation determines whether, prior to the next performance of the SR, the ratio of the measured heat flux hot channel factor to the limit is likely to decrease below the value of that ratio when the measurement was taken.

Each of these extrapolations is applied separately to each of the operational and RPS heat flux hot channel factor limits. If both of the extrapolations for a given limit are unfavorable, i.e., if the extrapolated factor is expected to exceed the extrapolated limit and the extrapolated factor is expected to become a larger fraction of the extrapolated limit

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

B. Symptoms

- **Reactor power greater than turbine power**
- **Reactor power greater than 100%**
- **"P/R OVER POWER ROD STOP" alarm**
- **NC T-Ave going down in an uncontrolled manner**
- **High containment pressure, temperature, humidity, or sump level without abnormal radiation**
- **Loss of secondary inventory**
- **Observed secondary steam leak.**

INTRODUCTION

This procedure directs the required Operator action to be taken for a steam leak. It is written for all modes of operation, but the plant response and Operator actions are largely dependent on the mode of operation and the severity of the leak.

Summary

For relatively small steam breaks, normal plant control systems are capable of maintaining nominal or near nominal operating conditions. For a small steamline break upstream of the turbine stop valves, the system transient response would be similar to a step load increase. The secondary system would indicate an increase in load with a resultant decrease in primary system average temperature and pressure. The control rods would withdraw from the core in an effort to restore the primary average temperature if the rod control system was in an automatic mode of operation. Due to the apparent increased load, the steam flow from the steam generators would be increasing in at least one loop, depending upon the location of the break. If the break occurred in the steam header, all loops would experience increased steam flow. Due to the increased steam flow, the feedwater control valves would modulate to a more open position in an attempt to maintain steam generator water level. As a result, the main feed flow in at least one loop (all loops if break is in steam header) would be increased. Another indication of this type of break would be a decreasing water level in the condenser hotwell. A containment temperature and/or pressure increase may be observed if the break occurred inside containment. If the break was outside containment, an audible or visual confirmation of the break may be possible. A drop in generator MW output may also be observed. Larger size breaks may require reactor trip and/or safety injection.

A different set of symptoms might be encountered for steam leaks that occur downstream of the turbine (on extraction lines, MSR's, and feedwater heaters). For these locations, it may be possible to observe a change in plant efficiency; however, an audible or visual indication may be the first symptom encountered.

ENTRY CONDITIONS

This procedure can be entered any time the listed symptoms are encountered. It should be noted that the symptom “Observed secondary steam leak” is the only symptom that definitively identifies a steam leak (and even then the magnitude of the leak may be considered for entry conditions). The other symptoms could indicate a steam leak, or some other event. In some cases the combination of symptoms can be the best indication the event is a steam leak and not some other event.

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
	8	16	16	16

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	Define the following terms associated with core performance: • Fuel rod burnout • Heat flux • Departure from nucleate boiling (DNB) • Critical heat flux (CHF) • Departure from nucleate boiling ratio (DNBR) • Linear heat generation rate • Average power density • Local power density • Axial flux difference (AFD) • AFD Target • Quadrant power tilt ratio (QPTR) • Heat flux hot channel factor (F_Q) • Enthalpy rise hot channel factor ($F_{\Delta H}$) CTHCP001		X	X	X	
2	Using a diagram of heat flux versus differential temperature between the cladding surface and the reactor coolant, identify and explain how the following affect fuel rod heat transfer, fuel and cladding temperature: *(Refer to Training Drawing 7.1, Nucleate Boiling Curve).* • Convective heat transfer region • Nucleate boiling region • Departure from nucleate boiling • Transition (partial film) boiling region • Film boiling region • Critical heat flux CTHCP002		X	X	X	X
3	Describe how the Critical Heat Flux (CHF) changes with changes in reactor coolant flow, average reactor coolant temperature, and reactor coolant pressure. CTHCP003		X	X	X	X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.8	
	Importance Rating		4.5

Knowledge of how abnormal operating procedures are used in conjunction with EOP's.

Proposed Question: SRO 100

Given the following:

- Unit 1 was at 100% power.
- A complete loss of RN occurred.
- The crew entered AP/20, Loss of RN.
- The operators attempted to manually trip the reactor but the trip breakers failed to open.

Which ONE of the following statements correctly describes the proper procedural flow path for these conditions?

A. Go directly to FR-S.1, Response to Nuclear Power Generation/ATWS, and perform concurrently with AP/20. Go to E-0, Reactor Trip or Safety Injection, as directed by FR-S.1.

B. Enter E-0 and immediately transition to FR-S.1; continuing in AP/20 only after exit from the EOP network.

C. Enter E-0, continuing in AP/20 until transition to FR-S.1. AP/20 may only be performed when FR-S.1 is complete.

D. Enter E-0 and immediately transition to FR-S.1 while continuing on in AP/20 as time and conditions permit.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. No direct EOP entry to FR-S.1. Performance of these 2 procedures is opposite of what would be performed

B. Incorrect. AP/20 may be performed concurrently because it provides support for EOP use

C. Incorrect. Use of AP/20 may be restricted when ECCS is actuated, but not by use of FR unless it clashes with steps in FR. Generally, AP use is not advisable in EPs, but may be used if required to support performance of EPs

D. Correct.

Technical Reference(s): OMP 4-3 p16, 17. 22 (Attach if not previously provided)

AP-20, Rev 23 and Basis Document (Rev 4)

EP-F) Rev 7

E-0 Rev 24

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-F0 Obj 3 (As available)

Question Source:
Bank # X
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam Wolf Creek 2007

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41
55.43 5

Comments:

Same basic question but applied to McGuire, so distractors and procedures are different

KA is matched because the item evaluates use of an AOP with the EOPs. SRO level because the applicant must determine procedure usage requirements for the given plant conditions.

7.15.1 Implementing CSF Path Procedures

7.15.1.1 CSF procedures are **NOT** to be implemented prior to transition from EP/1,2/A/5000/E-0 (Reactor Trip or Safety Injection). **IF** a CSF path is red or orange while the operating crew is in EP/1,2/A/5000/E-0, but has turned to green upon transition from E-0, the CSF procedure which was in alarm shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7. **IF** there are any valid red or orange path CSF's on transition from E-0 (unless transition is to EP/1,2/A/5000/ECA-0 (Loss of All AC Power), the associated CSF procedure shall be implemented.

7.15.1.2 **IF** a valid red or orange path flickers into alarm on SPDS but returns to green prior to the crew validating the condition and implementing the procedure (implementation of procedure being that the SRO either hands out fold-out pages or starts reading from the procedure), the CSF procedure shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7. Likewise, if a valid red path or orange path goes into alarm during performance of a higher priority CSF procedure, but returns to green prior to transition from the higher priority CSF path procedure to the lower priority CSF procedure, the associated CSF procedure shall **NOT** be implemented.

7.15.1.3 **IF** a CSF procedure directs the operator to return to the procedure and step in effect, **AND** the corresponding status tree continues to display the offnormal conditions, the corresponding CSF procedure does **NOT** have to be implemented again, since all recovery actions have been completed. However, if the same status tree subsequently changes to a valid higher priority condition, **OR** if it changes to lower condition and returns to higher priority condition again, the corresponding CSF procedure shall be implemented as required by its priority.

7.15.1.4 Red Path

IF any valid red path is encountered during monitoring, the operator is required to immediately implement the corresponding EP. Any recovery EP previously in progress shall be discontinued. **IF** during the performance of any red path procedure, a valid red condition of higher priority arises, the higher priority condition should be addressed first, and the lower priority red path procedure suspended.

2.0 PROCEDURE SERIES BACKGROUND (continued)

Once the Status Trees are being monitored, the following rules of usage apply:

1. The Status Trees should be continuously monitored in order of Critical Safety Function priority.
2. CSF procedures are not to be implemented prior to transition from E-0, Reactor Trip or Safety Injection. If a CSF path is red or orange while the operating crew is in E-0, but has turned to green upon transition from E-0, the CSF procedure, which was in alarm, shall not be implemented. If the CSF path is yellow, it shall be handled as any other yellow path procedure. If there are any valid red or orange path CSFs on transition from E-0 (unless the transition is to ECA-0 (Loss of All AC Power), the associated CSF procedure shall be implemented.
3. If a valid red or orange path flickers into alarm on SPDS but returns to green prior to the crew validating the condition and implementing the procedure (implementation of procedure being that the SRO either hands out fold-out pages or starts reading from the procedure), the CSF procedure shall not be implemented. If the CSF path is yellow, it shall be handled as any other yellow path procedure. Likewise, if a valid red path or orange path goes into alarm during performance of a higher priority CSF procedure, but returns to green prior to transition from the higher priority CSF path procedure to the lower priority CSF procedure, the associated CSF procedure shall not be implemented. If the CSF path is yellow, it shall be handled as any other yellow path procedure.
4. If a CSF procedure directs the operator to return to the procedure and step in effect, AND the corresponding status tree continues to display the off-normal conditions, THEN the corresponding CSF procedure doesn't have to be implemented again, since all recovery actions have been completed. However, if the same status tree subsequently changes to a valid higher priority condition, (OR if it changes to lower condition and returns to higher priority condition again), THEN the corresponding CSF procedure shall be implemented as required by its priority.
5. Once status tree monitoring is initiated, the STA should monitor status tree continuously if an orange or red path condition exists. If no condition more serious than yellow is found, monitoring frequency may be reduced to 10 – 20 minutes unless some significant change in plant status occurs. Status tree monitoring may be performed using the OAC SPDS display or F-0 (Critical Safety Function Status Trees). If the OAC SPDS display is being used, the STA will validate the OAC SPDS status every 10 – 20 minutes using control board indications. If the STA is not available, the OSM shall assume the STA responsibilities or delegate the STA responsibilities to another licensed operator.

7.14.2 The configuration control cards filled out in Step 7.14.1 shall be handled per the following two situations:

- Without Operations Support Center (OSC) activation

The configuration control card will be handled by OPS shift per SOMP 02-01 (Safety Tagging and Configuration Control).

- With OSC activation

WHEN the OSC is activated, OPS will report to the OSC and shall bring with them all configuration control cards that have been filled out. The cards taken to the OSC shall be given to the OPS SRO in the OSC. For handling cards in the OSC, refer to RP/0/A/5700/020 (Activation of the Operations Support Center (OSC)).

7.15 Usage of Status Trees

There are six different trees, each one evaluating a separate Critical Safety Function (CSF) of the plant. Color-coding of the status tree end points will be either red, orange, yellow, or green, with green representing a "satisfied" safety status. Each non-green color represents an action level that should be addressed according to the Rules of Priority as discussed below.

The six Status Trees are always evaluated in the sequence:

- Subcriticality
- Core Cooling
- Heat Sink
- Integrity
- Containment
- Inventory

IF identical color priorities are found on different trees during monitoring, the required action priority is determined by this sequence.

Initial monitoring of the status trees should begin on either of the following conditions:

- As directed by an action step in EP/1,2/A/5000/E-0 (Reactor Trip or Safety Injection).

- **WHEN** a transfer is made out of the Safety Injection procedure to another EP.

  An exception to this is that CSF procedures are **NOT** required to be implemented during the Loss of All AC Power EP since none of the electrically powered safeguards equipment can be used. **WHEN** power is subsequently restored, EP/1,2/A/5000/ECA-0.1 or 0.2 (Loss of All AC Power Recovery procedures) will direct the operator when implementing CSF procedures is required.

7.18 Multiple Use of EPs and APs.

The Control Room SRO will determine how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress. More than one EP shall **NOT** be run concurrently unless directed by the procedure. Generally the use of APs in conjunction with EPs should be avoided. In some instances it would be proper to use an AP concurrently during a major accident which is being addressed by the EPs. An example of this is upon loss of all Nuclear Service Water in the middle of an accident, the operators would need to utilize the AP for Loss of RN also. **IF** an AP is used during an S/I event, USE CAUTION. APs are generally written assuming an S/I has **NOT** occurred (exception - AP/35, ECCS Actuation During Plant Shutdown). Evaluate any AP steps in post S/I events to ensure the steps do **NOT** conflict with any EP in effect. **NOT** all AP actions would be appropriate if an S/I occurred. (Enclosures in EP/G-1 (Generic Enclosures) may be used when reference by EPs or APs.)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. Operator Actions

__ 1. **Monitor Foldout page.**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2. **Check Reactor Trip:** __ • All rod bottom lights - LIT __ • Reactor trip and bypass breakers - OPEN __ • I/R amps - GOING DOWN.	**Perform the following:** __ a. Trip reactor. b. **IF** reactor will not trip, **THEN**: __ • Implement EP/1/A/5000/F-0 (Critical Safety Function Status Trees). __ • **GO TO** EP/1/A/5000/FR-S.1 (Response To Nuclear Power Generation/ATWS).
3. **Check Turbine Trip:** __ • All throttle valves - CLOSED.	**Perform the following:** __ a. Trip turbine. b. **IF** turbine will not trip, **THEN**: __ 1) Place turbine in manual. __ 2) Close governor valves in fast action. 3) **IF** governor valves will not close, **THEN** close: __ • All MSIVs __ • All MSIV bypass valves.
__ 4. **Check 1ETA and 1ETB - ENERGIZED.**	**Perform the following:** __ a. **IF** both busses de-energized, **THEN GO TO** EP/1/A/5000/ECA-0.0 (Loss Of All AC Power). __ b. **WHEN** time allows, **THEN** try to restore power to de-energized bus **PER** AP/1/A/5500/07 (Loss of Electrical Power) while continuing with this procedure.

A. Purpose

This procedure provides actions to add negative reactivity to a core which is observed to be critical when expected to be shut down.

B. Symptoms or Entry Conditions

This procedure is entered from:

- EP/1/A/5000/E-0 (Reactor Trip Or Safety Injection), Step 2, when reactor trip is not verified and manual trip is not effective.
- EP/1/A/5000/F-0 (Critical Safety Function Status Trees) (Subcriticality), on either a red or orange condition.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21. **Check NC pumps as follows:**	
__ a. Any NC pump - ON.	__ a. **GO TO** Step 22.
__ b. NC pump stator winding temperature - LESS THAN 311°F.	b. Perform the following: __ 1) Secure any dilution in progress. 2) Open the following: __ • 1NV-221A (NV Pumps Suct From FWST) __ • 1NV-222B (NV Pumps Suct From FWST). 3) Close the following: __ • 1NV-141A (VCT Outlet Isol) __ • 1NV-142B (VCT Outlet Isol). __ 4) Start TD CA pump. __ 5) Maintain S/G NR levels greater than 17% to avoid auto start of MD CA pumps. __ 6) Trip reactor. __ 7) **WHEN** reactor is tripped, **THEN** trip all NC pumps. __ 8) Have available operator continue to monitor bearing temperatures on running pumps. __ 9) **WHEN** time allows, **THEN** continue with Case I, starting at Step 22. __ 10) **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).
__ c. Monitor stator winding temperatures.	
__ d. **IF AT ANY TIME** any NC pump stator winding temperature reaches 311°F, **THEN** perform Step 21.	

CASE I STEP 21:

PURPOSE:

Ensure protection for NCPs without RN cooling.

DISCUSSION:

Without RN cooling to the NCP motor coolers, the stator temperatures will increase to the trip criteria (311°F) in about 20 minutes. According to engineering, if temps go up a couple more degrees while actions in the RNO are performed, that's ok. Keep in mind that the thermocouple location is probably not measuring the hottest spot in the NCP stator. When the OAC reaches 311, there are probably areas that are 10 degrees hotter. That's ok as long as we get the pumps off within a couple minutes. The danger zone for the hottest spot starts around 330 deg.

Securing dilution prior to tripping NCPs should reduce the risks of highly diluted pockets of water from forming in the NC System (PIP M-99-0222).

Swapping charging pump suction to the FWST ensures the VCT will not heat up excessively for NCP seal injection and NV Pump NPSH concerns. Note that as the KC System temperature heats up, letdown and NV Pump recirc back to the VCT would cause it to heat up.

The TD CA Pump is manually started prior to tripping the reactor to avoid the auto start of the MD CA Pumps. This will also avoid the subsequent auto start on the RN pumps off the MD CA Pumps. The TD CA Pumps do not have RN cooling and will not overheat like the MD CA Pumps. Therefore, it is the preferred CA pump to run in this scenario. There is a trade-off with running the TD CA Pump in that it may contribute to a post-trip cooldown.

Waiting for the reactor to trip prior to tripping NCPs avoids loss of NC flow during an ATWS.

Before going to E-0, direction is given to have another operator continue with this AP. This is as high or higher priority than many of the EP actions, since the equipment assumed available in the EPs is cooled by RN, which is not available at this point in the AP. The highest priority in this scenario is the maintenance of NCP seal cooling and the restoration of RN (the actions of this AP). For this reason, direction is given to continue with Case I of the AP, as a higher priority than continuing with Case II at this point.

REFERENCES:

NC Pump manual (MCM-1201.01-193)
PIP M-99-0222

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
		2	2	2

OBJECTIVES

SEQ	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the purpose of each of the six CSF Status Trees.			X	X	
2	Explain the priority system associated with the CSF status trees.			X	X	X
3	Explain the "Rules of Usage" for Critical Safety Function status trees.			X	X	X
4	Explain the bases for all blocks in the six Status Trees.			X	X	X